Columbia is at 70% power when a jet pump fails due to a loss of the nozzle (rams head) on Reactor Recirculation Loop A.

Based on this failure, the reactor operator would expect:

A. Reactor recirculation total flow input to APRM channels A, C, and E to decrease.

B. Indicated core flow to decrease.

C. Indicated flow for Recirculation Loop A to increase.

D. The failed jet pump's differential pressure indication to be more noisy.

ANSWER: C

QUESTION TYPE: Closed Reference

KA # & KA VALUE: 295001 AA2.06 Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION: Nuclear Boiler Instrumentation (3.2)

REFERENCE: CGS Systems Description, "Reactor Recirculation System", Rev. 13, pg. 21-22; SD000178

SOURCE: New

LO: 5023 a. Predict the impacts of the RRC system of each of the following: Jet Pump Failure

RATING: Knowledge: Analysis Difficulty: 3

ATTACHMENT: None

JUSTIFICATION:

A. Incorrect - Loop flow will increase.
B. Incorrect - Indicated flow will increase.
C. Correct - Flow will increase due to less resistance from the nozzle.
D. Incorrect - Jet pump d/p indication will be less noisy on the failed jet pump.

COMMENTS: Ref. 10CFR55.41(5)

Columbia is in MODE 5 and the Benton substation 115 kV feeder is unavailable due to maintenance.

If bus SM-1 trips on undervoltage, what is the expected sequence of events if no operator action is taken?

A. Diesel Generator 1 automatically starts and reenergizes SM-7.

B. Diesel Generator 1 automatically starts and SM-7 remains deenergized..

C. Feeder breaker 7-1 opens and breaker B-7 closes powering SM-7 from TR-B.

D. Feeder breaker 7-1 opens and the bus remains deenergized.

ANSWER: A

QUESTION TYPE: Closed Reference

KA # & KA VALUE: 295003 AA1.03 Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER: Systems necessary to assure safe plant shutdown (4.4)

REFERENCE: CGS System Description, AC Distribution, Rev. 12, pg. 30; SD000182

SOURCE: New

LO: 5051d. Explain or identify the system interlock or response: Identify SM7 response to primary and secondary undervoltage.

RATING: Knowledge: Fundamental Difficulty: 2

ATTACHMENT: None

JUSTIFICATION: A Correct - On a primary undervoltage condition, the EDG will automatically start and the output breaker will close when the generator is up to rated speed and voltage.
B Incorrect – DG-2 cannot energize SM-7.
C Incorrect - Transformer TR-B is deenergized as given in the stem.
D Incorrect - The lockout relay does not trip.

COMMENTS: 10CFR55.41(7)
Changed distractor B

Given the following:

- An Extended Station Blackout is in progress
- RCIC is currently injecting to the RPV

If all DC power will soon be lost, powering which of the following DC buses from DG-4, the Alternate AC Station Battery Charger, will prevent a RCIC trip and allow CONTINUED RCIC injection?

A. S1-1

B. S1-2

C. S2-1

D. S1-7

ANSWER: A

QUESTION TYPE: Closed Reference

KA # & KA VALUE: 295004 AA2.04 Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER: System lineups (3.2)

REFERENCE: CGS System Description, DC Distribution, rev. 7, pg. 23 – 30; SD000188
CGS System Description, AC Distribution, rev. 13, pg. 29; SD000182

SOURCE: New Question
LO:
RATING: Knowledge: Fundamental Difficulty: 4

ATTACHMENT: None

JUSTIFICATION:
A. Correct – Bus S1-1 by itself will allow RCIC to continue to function, without this power, RCIC will trip.
B. Incorrect – There is minimal effect on RCIC from loss of this bus.
C. Incorrect – RCIC continues to run without indication on most valves.
D. Incorrect – This bus does not supply RCIC loads.

COMMENTS: Ref: 10 CFR 55.41 (7)

The purpose of the Feedwater Level Control System Setpoint Setdown is to:

A. Maintain reactor water level below level 8 following a SCRAM.

B. Lower the reactor water level setpoint when there is only one feedpump running.

C. Ensure the feedpump turbines do not overspeed following a SCRAM.

D. Lower the reactor water level setpoint when in single element control.

ANSWER: A

QUESTION TYPE: Closed Reference

KA # & KA VALUE: 295006 AK3.04 Knowledge of the reasons for the following responses as they apply to SCRAM: Reactor water level setpoint Setdown: Plant specific (3.1)

REFERENCE: CGS System Description, Feedwater Level Control System, Rev. 13, Section V. A. ; SD000157

SOURCE: New Question

LO: 5397 State the purpose of Setpoint Setdown, when it initiates, how it is reset, and how it affects the FWLC system.

RATING: Knowledge: Fundamental Difficulty: 2

ATTACHMENT: None

JUSTIFICATION:

A. Correct - The signal happens upon a scram, and it is to prevent reaching level 8.
B. Incorrect - The signal only occurs on a scram.
C. Incorrect - The feedpumps will slow down automatically following a scram due to less feedwater demand.
D. Incorrect - Setpoint set down has no function associated with being in single element control.

COMMENTS: 10 CFR 55.41 (7)

ABN-RCC is being performed due to a complete loss of RCC.

The bases for this procedure requiring the operating crew to place all the RCC pump switches in the pull to lock position is to:

A. minimize the potential for damaging the pumps and/or motors.

B. minimize break flow in the event of a piping failure.

C. allow the system to be returned to service in a controlled manner.

D. maintain RCC inventory until the system is restored.

ANSWER: C

QUESTION TYPE: Closed Reference

KA # & KA VALUE: 295018 / 2.1.28 Knowledge of the purpose and function of major components and controls.
2.1.20 Ability to execute procedure steps (4.3) [Deleted]

REFERENCE: ABN-RCC, Rev. 3, bases for step 4.1.4

SOURCE: New

LO:

RATING: Knowledge: Fundamental Difficulty: 3

ATTACHMENT: None

JUSTIFICATION: A. Incorrect – see C.
B. Incorrect – see C.
C. Correct- The bases for placing the pumps in PTL is that in the event of a loss of power to busses SL-71 and SL-81 the pump breakers remain closed due to there not being an undervoltage breaker trip. Placing the switches in PTL ensures an orderly return to service.
D. Incorrect – see C.

COMMENTS: Ref: 10 CFR 50.41 (4) & (10)

Which of the following will fail in the closed direction following a complete loss of Control Air System pressure?

A. CRD flow control valves (CRD-FCV-2A/2B).

B. Intertie between the Containment Nitrogen Inerting System and the Containment Instrument Air System (CN-V-65).

C. Inboard Main Steam Isolation Valves (MS-V-22A/22B/22C/22D).

D. Reactor Closed Cooling Water heat exchanger discharge valves (RCC-V-2A/2B/2C).

ANSWER: A

QUESTION TYPE: Closed Reference

KA # & KA VALUE: 295019 AA2.02 (AK2.03 RFW) Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR: Status of safety related instrument air system loads (see AK2.1– AK2-19) (3.6)

REFERENCE: CGS System Description, "Control Rod Hydraulic System", Rev. 12, Section IV. K; SD000142

SOURCE: New Question

LO: 7605. Describe the effect of a CAS failure on system loads.

RATING: Knowledge: Fundamental Difficulty: 2

ATTACHMENT: None

JUSTIFICATION:

A. Correct - The CRD flow control valve fails closed on a loss of CAS. They have a stop that prevents them from fully closing.
B. Incorrect – The intertie valve is gagged open, so that it will stay open with a loss of CAS.
C. Incorrect - The feedpump governor valves are electro-hydraulically operated.
D. Incorrect - The RCCW heat exchanger discharge valves are motor operated.

COMMENTS: Ref. 10 CFR 55.41 (7)

Columbia is in MODE 4. An inadvertent drain-down of the reactor vessel has reduced level to -55 inches.

Which of the following methods of decay heat removal from the RPV is available?

A. RHR A or B - Shutdown Cooling.

B. Injection with HPCS AND opening some SRVs.

C. Reject heat through the RWCU non-regenerative heat exchanger.

D. Run at least one Reactor Recirc Pump.

ANSWER: B

QUESTION TYPE: Closed Reference

KA # & KA VALUE: 295021 AK1.03 Knowledge of the operational implications of the following concepts as they apply to LOSS OF SHUTDOWN COOLING: Adequate core cooling (3.9)

REFERENCE: CGS System Description, RHR, Rev. 11, pg. 20; SD000198

SOURCE: New Question

LO: 5781 c. List the interlocks and trips associated with the following RHR system components: RHR-V-6A/B

RATING: Knowledge: Analysis Difficulty: 2

ATTACHMENT: None

JUSTIFICATION:

A. Incorrect - V-8 closes at level 3.
B. Correct - HPCS is available, as well as SRVs in this mode.
C. Incorrect - RWCU isolates at level 2.
D. Incorrect – This provides forced core cooling, but not decay heat removal.

COMMENTS: Ref : 10 CFR 50.41 (7) & (8)

Given the following conditions :

- Columbia was operating at 100% power when a LOCA occurred.
- Drywell pressure is 15 psig.
- Reactor pressure is 400 psig.

For the above conditions, which of the following operator actions will clear the interlocks for RHR-V-24A, Suppression Pool cooling, so that it can be opened?

A. Place switch RHR-RMS-S105, RHR-V-42A Valve Logic Override, in OVERRIDE.

B. Place switch RHR-RMS-S101A, RHR-V-42A Permissive Override, in TEST.

C. ONLY CLOSE RHR-V-42A, LPCI injection.

D. CONTINUALLY hold switch for RHR-V-24A in the OPEN position.

ANSWER: C

QUESTION TYPE: Closed Reference

KA # & KA VALUE: 295024 EK2.12 Knowledge of the interrelations between HIGH DRYWELL PRESSURE and the following: Suppression pool cooling (3.5)

REFERENCE: CGS System Description, RHR, Rev. 11, pg. 15-16

SOURCE: New Question

LO: 5781 List the interlocks and trips associated with the following RHR system components: f. RHR-V-24A/B and RHR-V-27A/B

RATING: Knowledge: Fundamental Difficulty: 3
ATTACHMENT: None
JUSTIFICATION:
A. Incorrect – This will allow throttling of 42a, but is not necessary below 470.
B. Incorrect – This will allow 42A to be tested during normal ops.
C. Correct – This will clear the interlock, allow the valve to be opened and stay open as long as pressure stays below 470.
D. Incorrect – This only overrides the logic if a LPCI initiation has not already occurred.

COMMENTS: Ref : 10 CFR 55.41 (7) & (9) & (10)

The EOPs require an Emergency Depressurization be performed prior to exceeding the high wetwell temperature limit.

The bases for this requirement is to protect the:

A. Fuel Cladding.

B. Reactor Pressure Vessel.

C. Primary Containment.

D. Reactor Building.

ANSWER: C

QUESTION TYPE: Closed Reference

KA # & KA VALUE: 295026 EK3.01 Knowledge of the reasons for the following responses as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE: Emergency/normal depressurization (3.8)

REFERENCE: 5.0.10, EOP Flowchart Training Manual, Rev. 7, page 200

SOURCE: New Question

LO: 5629 State the three (3) purposes of the suppression chamber.

RATING: Knowledge: Analysis Difficulty: 2

ATTACHMENT: None

JUSTIFICATION:

A. Incorrect - ECCS suction would be adversely affected by an ED, but even if primary containment was lost, CST is still available.
B. Incorrect – ED is to mitigate the possibility of a LOCA, not to prevent one.
C. Correct – The basis is to maintain the pressure suppression capability of the containment.
D. Incorrect – Secondary containment would not be damaged by this problem.

COMMENTS: Ref: 10 CFR 55.41 (9) & (10)

PPM 5.2.1, Primary Containment Control, has been entered and it has been determined that WW level CANNOT be maintained above 19 ft 2 in. According to the procedure, an Emergency Depressurization is required.

If the Emergency Depressurization is not started until level has lowered to 18 ft 6 in, what are the consequences?

A. Vortexes at the suction ECCS pumps can begin and result in air binding of the pumps.

B. Suppression pool temperature indication becomes invalid.

C. Condensation of steam from the SRV downcomers cannot be assured.

D. The SRV Tail Pipe Level Limit (SRVTPLL) will be exceeded.

ANSWER: C Post Exam Comment – The licensee recommended this question be deleted because the term “SRV downcomers” is confusing. They contend the correct term is “quencher”. This comment was rejected because the steam reaches the quencher via the downcomers and the SRV downcomers are the only ones located in the suppression pool. Based on this, distractor C is correct as stated.

QUESTION TYPE: Closed Reference

KA # & KA VALUE: [New KA] 295030 EK1.01 Knowledge of the operational implications of the following concepts as they apply to LOW SUPPRESSION POOL WATER LEVEL: Steam Condensation (3.8) [New KA]
[KA Deleted] 295030 EK1.03 Knowledge of the operational implications of the following concepts as they apply to LOW SUPPRESSION POOL WATER LEVEL: Heat capacity (3.8) [KA Deleted]

REFERENCE: PPM 5.2.1, Primary Containment Control, Rev. 15 and 5.0.10, Flowchart Training Manual, Rev. 7, pg. 262

SOURCE: 2003 CGS Initial Licensing Exam – Modified

RATING: Knowledge: Analysis Difficulty: 3

ATTACHMENT: None

JUSTIFICATION:
A. Incorrect - Vortexing does not begin until 17.5 ft for the RCIC system.
B. Incorrect - Temperature indication is still valid at this level.
C. Correct - The downcomer vents will be exposed, and steam will not be properly condensed.
D. Incorrect - The SRVTPLL will be exceeded by raising level.

COMMENTS: Ref : 10 CFR 50.41 (5) & (8)

Procedure 5.1.2, RPV Control – ATWS, has been entered. Boron Injection is required and the SLC system keylock switches have been taken to operate.

Which of the following conditions would prevent boron injection if no other operator action is taken?

A. The SLC Test Tank Outlet Valve SLC-V-31 is OPEN.

B. The RWCU Outboard Isolation Valve RWCU-V-4 is OPEN.

C. Storage Tank Outlet Valve SLC-V-1A OR SLC-V-1B is CLOSED.

D. Loss of continuity to squib valves SLC-V-4A & 4B AFTER keylock switches are taken to OPER.

ANSWER: A

QUESTION TYPE: Closed Reference

KA # & KA VALUE: 295037 EA1.04 Ability to operate and/or monitor the following as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN: SBLC (4.5)

REFERENCE: CGS System Description, SLC, Rev. 11, pg. 9; SD000172

SOURCE: Clinton 1 7/23/2001 – Modified

LO: 5925 Describe the expected response to placing the SLC system A or B keylock switch in the operate position.

RATING: Knowledge: Analysis Difficulty: 3

ATTACHMENT: None

JUSTIFICATION:

A. Correct - The pump suction valves cannot open if the test valve is open. The pump suction valves not fully open prevents boron injection if no other operator action is taken.
B. Incorrect - The only input to SLC pump start circuitry is the suction valve position.
C. Incorrect - Flow is still possible through one train.
D. Incorrect - This is the indication expected for the squib valves after firing (opening).

COMMENTS: 10 CFR 50.41 (6) (7)

Which of the following describes the Halon system's discharge rate in the Control Room floor modules?

Does the Halon discharge, by itself, make the Control Room uninhabitable?

A. Discharges fully immediately. Control Room is still habitable.

B. Discharges fully immediately. Control Room must be evacuated.

C. Discharges over an extended period of time. Control Room is still habitable.

D. Discharges over an extended period of time. Control Room must be evacuated.

ANSWER: C Post Exam Comment – During the exam, one applicant asked if the question involved one bottle of Halon or the Halon System. The applicant was told to consider the question from the system perspective.

QUESTION TYPE: Closed Reference

KA # & KA VALUE: 600000 AK1.02 Knowledge of the operation applications of the following concepts as they apply to Plant Fire On Site: Fire fighting (2.9)

REFERENCE: CGS System Description, Fire Protection, Rev. 11, Pg. 11-13; SD000177

SOURCE: New Question

LO: 5376 Briefly explain the operation of the following types of fire suppression systems: e. Control Room Halon 1301 Floor Modules

RATING: Knowledge: Fundamental Difficulty: 3

ATTACHMENT: None

JUSTIFICATION:
A. Incorrect – the discharge is in three stages over 12 minutes.
B. Incorrect – the discharge is in three stages over 12 minutes.
C. Correct – The discharge is extended, and the Halon is designed so that the CR will still be habitable.
D. Incorrect – The Halon is designed to leave the CR habitable.

COMMENTS: Ref : 10 CFR 55.41 (4)

Columbia was operating at full power with RCIC tagged out. A trip of both Reactor Feedwater Pumps caused RPV level to drop and a Reactor scram to occur on low RPV water level. HPCS recovered RPV level and HPCS-V-4 was closed when RPV water level reached +35". Conditions are currently:

-Reactor Pressure 700 psig (rising at 10 psig per minute)
-Time after scram 5 minutes
-CRD pumps both tripped
-Drywell pressure 0.3 psig

If no additional operator actions are taken, what is the expected RPV water level response over the next 10 minutes and why?

RPV water level will…

A. rise above the high RPV water level trip setpoint due to heatup.

B. rise above the high RPV water level trip setpoint due to Startup Valve leakage exceeding decay heat requirements.

C. lower below the low level alarm point due to cooldown.

D. lower below the low RPV level trip setpoint due to steam loads reducing RPV water inventory.

ANSWER: A

QUESTION TYPE: Closed Reference
KA # & KA VALUE: 295008 AA2.05 Ability to determine and/or interpret the following as they apply to HIGH REACTOR WATER LEVEL: Swell (2.9)
REFERENCE:
SOURCE: Cooper Exam 8/2/2002
LO:
RATING: Knowledge: Analysis Difficulty: 3
ATTACHMENT: None
JUSTIFICATION:
A. Correct – The specific volume change from the CST water to saturated liquid at 700 psig results in ~40% increase in gallons/inches of RPV level. 80" of cold water added = 112".
B. Incorrect – Rx pressure is above Condensate Booster pressure.
C. Incorrect - RPV water level will rise.
D. Incorrect – RPV level will rise. BPVs are closed and other steam loads will not lower level under these conditions.

COMMENTS: Ref: 10 CFR 55.41 (5)

Procedure ABN-CRD-MAXFLOW has been entered due to a low RPV level condition. This procedure contains a caution that states "Do not lower drive water pressure to LT 260 psid".

The bases for maintaining drive water pressure greater than 260 psid is to:

A. Maintain seal water to the reactor recirc pump seals.

B. Support control rod insertion per the EOP's.

C. Prevent CRD pump runout at low RPV pressures.

D. Maintain cooling water to the Control Rod Drive Mechanisms.

ANSWER: B

QUESTION TYPE: Closed Reference

KA # & KA VALUE: 295009 Low Reactor Water Level 2.1.32 Ability to explain and apply system limits and precautions (3.4)

REFERENCE: ABN-CRD-MAXFLOW, Revision 1, Bases section
CGS System Description, Control Rod Drive Hydraulic System, Rev. 12

SOURCE: New Question

LO: 5186. a. Describe the impact of CRD pressure controller operation on the following CRD System parameters: Drive water differential pressure

RATING: Knowledge: Fundamental / Difficulty: 3

ATTACHMENT: None

JUSTIFICATION:
A. Incorrect – Drive water DP and suction pressure of the pump are not directly related.
B. Correct – Dp is necessary to drive rods.
C. Incorrect - This does not prevent runout, because the pump could still runout through the scram valves if it was going to.
D. Incorrect – The FCV maintains cooling flow.

COMMENTS: Ref. 10 CFR 55.41 (6)

After a 370 day run at full power, a Group 1 isolation resulted in a reactor scram. EOP 5.1.1, RPV Control, and EOP 5.2.1, Primary Containment Control have been entered.

Given the following conditions:

- RPV Pressure is being maintained 800 to 1000 psig using SRVs.
- RPV Level is +30 inches and stable.
- Suppression Pool cooling is unavailable.
- Wetwell temperature is 106°F and rising.

Which of the following injection lineups will result in the LEAST amount of heat added to the Wetwell assuming this lineup will be maintained for several hours?

A. HPCS flow from the CST.

B. RCIC flow from the CST.

C. HPCS flow from the Wetwell.

D. RCIC flow from the Wetwell.

ANSWER: B Post Exam Comment - The NRC does not agree with the licensee's recommendation to delete Question #15. The HPCS system is a larger pump that will add more heat to the suppression pool via the minimum flow line than RCIC. Secondly, RCIC is drawing 800-1000 psig steam off the reactor and exhausting it to the suppression pool at a much lower enthalpy. Otherwise, the steam would be exhausted directly from the reactor to the suppression pool via the relief valves. Both these factors would reduce the amount of heat being added to the suppression pool making RCIC the only correct answer.

QUESTION TYPE: Closed Reference
KA # & KA VALUE: 295013 AA1.02 Ability to operate and/or monitor the following as they apply to HIGH SUPPRESSION POOL TEMPERATURE: Systems that add heat to the suppression pool (3.6)
REFERENCE: CGS System Description, RCIC, Rev. 12, Procedure 5.2.1, Primary Containment Control, Rev. 15, and Procedure 5.1.1, RPV Control, Rev. 16
SOURCE: New Question
LO:
RATING: Knowledge: Analysis, Difficulty: 4
ATTACHMENT: None
JUSTIFICATION:

A. Incorrect – The injection flow into the RPV would be the same as RCIC, but none of the steam sent to the WW would go through RCIC, so it would have more overall enthalpy.
B. Correct – Some of the steam would exhaust through RCIC vice the SRVs lowering the overall heat being added to the wetwell.
C. Incorrect - using the warmer water as a suction requires less decay

heat from the reactor to produce the same amount of steam. It is also not using the steam for work.

D. Incorrect - using the warmer water as a suction requires less decay heat from the reactor to produce the same amount of steam.

COMMENTS: 10 CFR 55.41 (5), (8), (14)

Due to special test requirements, Columbia Generating Station is in MODE 2 being shut down by control rod insertion instead of a manual scram. Reactor pressure is steady at 900 psig and reactor power is at 12 on IRM range 7.

Due to a plant problem control rod insertion has been stopped.

With no operator action taken, which of the following will result?

A. Reactor Scram due to hi neutron flux.

B. Reactor Scram due to MSIV closure.

C. Rod block due to SRM upscale.

D. Rod block due to IRM downscale.

ANSWER: A Post Exam Comment - The NRC does not agree with the licensee's recommendation to delete Question #16. With the power in intermediate range 7, the reactor is critical and the turbine is off-line. In addition to the steaming, the makeup water to the reactor it will be relatively cold and moderator temperature will be driven down. This adds positive reactivity thereby increasing power until the scram setpoint is reached making A the only correct answer.

QUESTION TYPE: RO

KA # & KA VALUE: 295014 AK2.06 Knowledge of the interrelations between Inadvertent Reactivity Addition and the following: Moderator temperature (3.4 / 3.5)

REFERENCE: SER 24-91; SD000161

SOURCE: Bank – Modified; Analysis Difficulty: 3

LO: 5192

RATING: H3

ATTACHMENT: None

JUSTIFICATION:

A. Correct – Reactor power will increase due to the effects of the cooldown causing a hi flux scram.
B. Incorrect – RPV Pressure is on DEH in automatic. MSIV close at 831 in RUN.
C. Incorrect – SRM rod block is bypassed.
D. Incorrect – Power increases not decreases.

COMMENTS: POAH is 25 on IRM Range 8

Columbia is operating at 100% power. Radiation readings for the Reactor Building Exhaust Air Plenum begin rising on REA-RIS-609A, B, C, and D.

If the radiation source continues to give off INCREASING amounts of radioactive gas, the Reactor Building Exhaust Plenum Radiation Monitoring System recorder outputs will rise to the 'Z' signal setpoint of _________ and then the Reactor Building Exhaust Air Plenum Radiation Monitoring System recorder output readings will _________ .

A. 13 mr/hr, continue to RISE.

B. 13mr/hr, STABILIZE OR DROP.

C. 15mr/hr, continue to RISE.

D. 15mr/hr, STABILIZE OR DROP.

ANSWER: B

QUESTION TYPE: Closed Reference

KA # & KA VALUE: 295034 EA1.02 Ability to operate and/or monitor the following as they apply to SECONDARY CONTAINMENT VENTILATION HIGH RADIATION: Process Radiation monitoring system (3.9)

REFERENCE: CGS System Description, PRM, rev. 11, pg. 19-20; SD000147; SD000173

SOURCE: New Question

LO: 5647 State the automatic actions associated with each of the following gaseous and liquid stream Process Radiation Monitors upon sensing high radiation levels: g. Reactor Building Exhaust Plenum RMS

RATING: Knowledge: Analysis Difficulty: 2

ATTACHMENT: None

JUSTIFICATION:

A. Incorrect – Z signal isolates flow through plenum at 13mr/hr, therefore, no more radioactive gases would be drawn in, and what gas was already in the plenum would decay.
B. Correct – Z signal isolates flow through plenum at 13mr/hr.
C. Incorrect – The Z signal setpoint is 13mr/hr.
D. Incorrect – The Z signal setpoint is 13mr/hr.

COMMENTS: Ref : 10 CFR 55.41 (4) & (7) & (11) & (13)

Columbia is in MODE 4 twelve hours after a SCRAM with Shutdown Cooling in service on RHR A. Reactor water level is 55 inches and pressure is 75 psig. The CRO mistakenly throttles down on RHR-V-3A, Heat Exchanger Shell Side Outlet, causing RHR flow to DECREASE to 700 gpm.

Over the next 30 minutes, if no other operator actions are taken, RPV :

A. temperature will remain STABLE.

B. level will INCREASE.

C. temperature will DECREASE.

D. level will DECREASE.

ANSWER: B

QUESTION TYPE: Closed Reference

KA # & KA VALUE: 205000 A1.05 Ability to predict and/or monitor changes in parameters associated with operating the SHUTDOWN COOLING SYSTEM/MODE controls including: Reactor water level (3.4)

REFERENCE: CGS System Description, RHR, rev. 11, pg. 22; SD000198

SOURCE: Fermi 2, 12/11/1995 – Modified

LO: 7728 Describe the physical connection and / or the cause-and-effect relationships between the RHR system and the following: g. RPV

RATING: Knowledge: Analysis Difficulty: 3

ATTACHMENT: None

JUSTIFICATION:

A. Incorrect – The reactor will heat up the RPV because of lower SDC flow.
B. Correct – Flow through the RHR HX drops causing less cooling resulting in a temp rise causing RPV level to rise.
C. Incorrect –The reactor will heat up the RPV because of lower SDC flow.
D. Incorrect – The reactor will heat up the RPV because of lower SDC flow.

COMMENTS: 10 CFR 55.41 (7), (14)

Which of the following combinations of RPV Wide Range Level (MS-LIS-37A/B) and Drywell Pressure Switches (MS-PS-48A/B/C/D) will cause the Low Pressure Core Spray System to automatically initiate?

A. MS-LIS-37B and MS-PS-48A

B. MS-LIS-37A and MS-PS-48B

C. MS-LIS-37A and MS-PS-48C

D. MS-LIS-37B and MS-PS-48D

ANSWER: C

QUESTION TYPE: Closed Reference

KA # & KA VALUE: 209001 K1.09 Knowledge of the physical connections and/or cause- effect relationships between LOW PRESSURE CORE SPRAY SYSTEM and the following: Nuclear boiler instrumentation (3.2)

REFERENCE: CGS System Description, LPCS, Rev. 10, pg. 7; SD000192

SOURCE: New Question

LO: LO-5481 List the signals and setpoints, which cause a LPCS initiation.

RATING: Knowledge: Fundamental Difficulty: 4

ATTACHMENT: None

JUSTIFICATION:
A. Incorrect – this is not a combination that will auto start.
B. Incorrect – this is not a combination that will auto start.
C. Correct – see reference.
D. Incorrect – this will start LPCI B & C

COMMENTS: 10 CFR 55.41 (7)

A Large Break LOCA has occurred and the RPV is depressurized. HPCS is injecting from the CST, and LPCS is injecting. There are NO other injection lineups available.

To prevent Wetwell level from reaching the SRV Tail Pipe Level Limit of _______, the _______ pump should be stopped.

A. 51 ft, LPCS

B. 41 ft, LPCS

C. 51 ft, HPCS

D. 41 ft, HPCS

ANSWER: C

QUESTION TYPE: Closed Reference

KA # & KA VALUE: 209002 A2.12 Ability to (a) predict the impacts of the following on the HIGH PRESSURE CORE SPRAY SYSTEM (HPCS); and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: High suppression pool level (3.3)

REFERENCE: Procedure 5.2.1, Primary Containment Control, rev. 15

SOURCE: New Question

LO: 8384 Given a list, identify the statement that describes the purpose of terminating injection into the primary containment if wetwell level and RPV pressure cannot be maintained below the SRVTPLL.

RATING: Knowledge: Analysis Difficulty: 2

ATTACHMENT: None

JUSTIFICATION:
A. Incorrect – this is not an injection source external to the PC.
B. Incorrect – this is not an injection source external to the PC.
C. Correct – per the EOP, stop injection source external to the PC, the SRVTPLL is 51 ft.
D. Incorrect – this is the incorrect SRVTPLL.

COMMENTS: 10 CFR 55.41 (8)

Columbia is operating at 90% power in three-element control when an inadvertent actuation of HPCS occurs.

If all systems respond as expected, with no operator action, what will indicated RPV water level do?

Indicated RPV water level will….

A. rise, then stabilize lower than the original level.

B. rise, then stabilize higher than the original level.

C. lower, then stabilize lower than the original level.

D. lower, then stabilize higher than the original level.

ANSWER: B

QUESTION TYPE: Closed Reference

KA # & KA VALUE: 209002 A1.03 Ability to predict and/or monitor changes in parameters associated with operating the HIGH PRESSURE CORE SPRAY SYSTEM (HPCS) controls including: Reactor water level (3.7)

REFERENCE: CGS FSAR, Section 15.5.1, Inadvertent High-Pressure Core Spray Startup, Amendment 57
CGS System Description, Feedwater Level Control System, Rev. 13

SOURCE: New Question
LO:
RATING: Knowledge: Analysis Difficulty: 4

ATTACHMENT: None

JUSTIFICATION:

A. Incorrect – Level will initially swell, but the higher steam flow than feed flow will produce a level error due to higher steam flow than feed flow, this will speed up the FP until the level error goes away in the FWLC system.
B. Correct – Level will initially swell. The FWLC system will have an error due to steam flow > feed flow. It will bias stabilized level higher.
C. Incorrect – Level will initially swell due to pressure decrease from HPCS spray in shroud.
D. Incorrect – Level will initially swell.

COMMENTS: 10 CFR 55.41 (5)

When the SLC System A keylock switch is taken to the 'OPER' position, what is the overall expected RWCU System response?

A. RWCU-V-4 closes AND RWCU-FCV-33 closes if open

B. RWCU-V-1 closes AND RWCU-FCV-33 closes if open

C. ONLY RWCU-V-4 closes

D. ONLY RWCU-V-1 closes

ANSWER: A

QUESTION TYPE: Closed Reference

KA # & KA VALUE: 211000 A4.06 Ability to manually operate and/or monitor in the control room: RWCU system isolation (3.9)

REFERENCE: CGS System Description, RWCU, Rev. 10; SD000190

SOURCE: New Question

LO: 5931 Given one or more systems that interrelate to SLC, state the importance or function of that relationship

RATING: Knowledge: Fundamental Difficulty: 2

ATTACHMENT: None

JUSTIFICATION:
A. Correct, SLC operation causes V-4 to close, V-4 closed causes FCV-33 to close. It is important that FCV-33 close to remain an option for boron injection.
B. Incorrect, V-4 closes.
C. Incorrect, FCV-33 closes on a V-4 closure signal.
D. Incorrect, V-4 closes.

COMMENTS: Reference : 10 CFR 55.41 (6), (9) Both SLC system keylock switches have been taken to OPERATE. ONE of the squib valves failed to open.

What should the operator expect the APPROXIMATE boron injection flowrate to the RPV to be if all other components operate as expected?

A. 30 gpm

B. 45 gpm

C. 60 gpm

D. 90 gpm

ANSWER: D Post Exam Comment – The licensee recommended deleting this question because the design pumping rate of each SLC pump is 43 gpm. This makes the total system flow 86 gpm and not 90 gpm as distractor D contains. This comment was rejected because the stem of the question asks what is the “APPROXIMATE” flowrate. This was used because the actual flowrate will vary from 85 to 90 GPM due to variances in pump construction and 86 gpm would logically be approximately 90 gpm compared to any of the other distractors.

QUESTION TYPE: Closed Reference

KA # & KA VALUE: 211000 A1.04 Ability to predict and/or monitor changes in parameters associated with operating the STANDBY LIQUID CONTROL SYSTEM controls including: Valve operations (3.6)

REFERENCE: CGS System Description, SLC, Rev. 11; SD000172

SOURCE: New Question

LO: 5922 Describe the following SLC system flowpaths: a. Normal Injection

RATING: Knowledge: Analysis Difficulty: 3

ATTACHMENT: None

JUSTIFICATION:

A. Incorrect – dual pump flow is approximately 90.
B. Incorrect – single pump output is 45 gpm, but the pumps are cross-tied.
C. Incorrect - dual pump flow is approximately 90.
D. Correct – the pumps are cross-tied, and they are positive displacement pumps. Pump output will be 90 gpm.

COMMENTS: Reference : 10 CFR 55.41 (6)

Changed B to 45 from 40 and D to 90 from 80

Columbia is operating at 99% power with no equipment out of service. I&C technicians are preparing to start a test on RPS trip system A that will produce a half-scram. You notice one of the four RPS Scram Group lights is NOT lit on RPS trip system B.

What are your immediate actions per procedure ABN-RPS, and what would the consequences be if the RPS A half scram is initiated?

A. Replace fuse, half-scram with no rod movement.

B. Immediately stop the work on RPS A, half-scram with no rod movement.

C. Replace fuse, one quarter of the rods scram.

D. Immediately stop the work on RPS A, one quarter of the rods scram.

ANSWER: D

QUESTION TYPE: Closed Reference

KA # & KA VALUE: 212000 A2.19 Ability to (a) predict the impacts of the following on the REACTOR PROTECTION SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Partial system activation (Half-SCRAM) (3.8)

REFERENCE: Procedure ABN-RPS, Rev. 2
CGS System Description, RPS, rev. 12; SD000161

SOURCE: New Question

LO: 7683 Predict the effect(s) that a failure of the RPS system will have on: a. Scram and Backup Scram valves

RATING: Knowledge: Analysis Difficulty: 3
ATTACHMENT: None
JUSTIFICATION:
A. Incorrect – One quarter of the B scram solenoids are de-energized. When the A scram solenoids are de-energized, those rods will scram.
B. Incorrect- One quarter of the rods will move.
C. Incorrect – Procedure ABN-RPS directs you to stop work on the other trip system.
D. Correct – One quarter of the rods would scram, procedure directs you to stop work on the A trip system.

COMMENTS: Reference 10 CFR 55.41 (7) & (10)

IRM channel E is on Range 3 and is reading 10/40 scale. Over the next few minutes, reactor power doubles about 3.5 times.

Based on this change in reactor power, the CRO should expect IRM Channel E to now indicate approximately……

A. 25/40 on Range 3

B. 78/125 on Range 4

C. 12/40 on Range 5

D. 95/125 on Range 6

ANSWER: C

QUESTION TYPE: Closed Reference

KA # & KA VALUE: 215003 A4.03 Ability to manually operate and/or monitor in the control room: IRM range switches (3.6)

REFERENCE: CGS System Description, Intermediate Range Monitor, Rev. 8; SD000138

SOURCE: Bank – Slightly Modified

LO: 5461 Describe the correlation between Reactor Period and IRM indication.

RATING: Knowledge: Analysis Difficulty: 3

ATTACHMENT: None

JUSTIFICATION: 10 to 20 on Range 3 is one doubling; 20 to 40 on Range 3 is another doubling; 40 to 80 on Range 4 is another doubling (3 total); .57 doublings more is about 120 Range 4 or 12 Range 5.

COMMENTS: Ref : 10 CFR 55.41 (7) & (1)
Revised stem

Which of the following circumstances for the Source Range Monitors will generate a rod block?

A. 5×10^4 counts per second and all IRMs on range 3.

B. 0.9 counts per second.

C. SRM channel A mode switch in standby.

D. One detector retracted with 165 counts per second.

ANSWER: C

QUESTION TYPE: Closed Reference

KA # & KA VALUE: 215004 K1.03 Knowledge of the physical connections and/or cause- effect relationships between SOURCE RANGE MONITOR (SRM) SYSTEM and the following: Rod control and information system: Plant specific (3.0)

REFERENCE: CGS System Description, Source Range Monitor, Rev. 10; SD000132

SOURCE: New Question

LO: 5943 List the scrams and the rod blocks generated by the SRM system. Include the setpoints for each and when they are bypassed.

RATING: Knowledge: Fundamental Difficulty: 2

ATTACHMENT: None

JUSTIFICATION:
A. Incorrect - 10^5 counts per second is the rod block.
B. Incorrect – The setpoint is 0.7 cps.
C. Correct – One channel being out of operate provides a rod block.
D. Incorrect – This rod block is set at 100cps.

COMMENTS: Ref : 10 CFR 55.41 (7)

With Columbia operating at full power, annunciator CIA DIV 1 OUT OF SERVICE annunciates. You then receive a call reporting that CIA-RV-5A, which is located downstream of Nitrogen backup bank bottle pressure control valve, CIA-PCV-2A, and between CIA-V-30A and CIA-V-39A, is stuck open. There is a large amount of nitrogen escaping depressurizing the line.

What supply(ies) is(are) still available to operate ADS valves in addition to the accumulators?

A. ONLY the Main Header for ALL SRVs.

B. ONLY ADS Accumulator Header B for Division II ADS SRVs.

C. The Main Header for ALL SRVs, and ADS Accumulator Header B for ALL SRVs.

D. The Main Header for ALL SRVs, and ADS Accumulator Header B for Division II ADS SRVs.

ANSWER: D

QUESTION TYPE: Closed Reference

KA # & KA VALUE: 218000 K6.04 Knowledge of the effect that a loss or malfunction of the following will have on the AUTOMATIC DEPRESSURIZATION SYSTEM: Air supply to ADS valves: Plant specific (3.6)

REFERENCE: CGS System Description, Containment Instrument Air, Rev. 8; SD000156

SOURCE: New Question

LO: 7748 Determine the effect a CIA malfunction has on: b. SRVs

RATING: Knowledge: Analysis Difficulty: 3

ATTACHMENT: None

JUSTIFICATION:

A. Incorrect – ADS Accumulator Header B is still available.
B. Incorrect – the Main header is still available.
C. Incorrect – the ADS Accumulator Header B cannot backfeed into A.
D. Correct – A header isolates, main header and B header are still available.

COMMENTS: Reference : 10 CFR 55.41 (7), (8), & (10)

Given the following conditions :

- Drywell Pressure Instrument MS-PS-48D in TRIP.
- Drywell Pressure Instrument MS-PS-48B failed HIGH.
- Procedure ABN-FAZ has been entered.

Which of the following describes the automatic actuations of the NSSSS system AND describes the operator action necessary to mitigate that impact?

A. ALL RCC to the Drywell is ISOLATED, VENT the Drywell.

B. ALL Circulating Water Pumps TRIP, Place RHR in Suppression Pool Cooling.

C. ALL TSW Pumps TRIP, Ensure a TSW pump STARTS after DG starts.

D. RWCU to the Drywell is ISOLATED, OPEN RWCU-FCV-33, Blowdown Control Valve, to prevent RWCU relief valves from lifting.

ANSWER: A

QUESTION TYPE: Closed Reference

KA # & KA VALUE: 223002 A2.06 Ability to (a) predict the impacts of the following on the PCIS/NSSSS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Containment instrumentation failures (3.0)

REFERENCE: ABN-RCC, Rev. 3, pg. 3
CGS System Description, NS4, Rev. 10, pg. 5; SD000173

SOURCE: New Question

LO:

RATING: Knowledge: Analysis Difficulty: 3

ATTACHMENT: None

JUSTIFICATION:
A. Correct – RCC isolated can cause drywell pressure to exceed F signal, venting may be necessary.
B. Incorrect – Circ water pump A does not trip.
C. Incorrect – NO pumps will trip without a LOOP signal.
D. Incorrect - RWCU does NOT isolate on drywell pressure or plant trip or ECCS initiation.

COMMENTS: Ref : 10 CFR 55.41 (5) & (7) & (10)

The Control Room has been evacuated due to a noxious chemical. The following conditions exist:

RPV Level is BELOW Level 1.
RHR-P-2A AND RHR-P-2B are running.
DP-S1-2A is de-energized.
All Normal/Emergency switches at the Remote Shutdown Panel (RSP) and the Alternate Remote. Shutdown Panel (ASP) have been taken to Emergency.

Based on the given conditions, which of the following describes operation of the SRVs?

A. ADS CAN automatically initiate AND ADS SRVs can be manually operated from the RSP.

B. ADS CAN automatically initiate BUT ADS SRVs CAN NOT be manually opened from the RSP.

C. ADS CAN NOT automatically initiate BUT ADS SRVs can be manually opened from the RSP.

D. ADS CAN NOT automatically initiate AND ADS SRVs CAN NOT be manually opened from the RSP.

ANSWER: C

QUESTION TYPE: Closed Reference
KA # & KA VALUE: 239002 K4.05 Knowledge of RELIEF/SAFETY VALVES design feature(s) and/or interlocks which provide for the following: Allows for SRV operation from more than one location: Plant specific (3.6)
REFERENCE: CGS System Description, ADS, Rev. 10, pg. 8; SD000186 and RSP, Rev. 6, pg. 8 ; SD000210
SOURCE: New Question
LO: 5077 List the power supplies to the ADS solenoids
5886 State the effects to associated component controls and alarms when their Power Transfer Switches are placed to the EMERGENCY position.
RATING: Knowledge: Analysis Difficulty: 4
ATTACHMENT: None
JUSTIFICATION:

A. Incorrect – ADS B logic has lost power. ADS A has logic, but will be ‘blocked’ by the ARS emergency switches. Therefore – not ALL ADS valves will open, but some will.
B. Incorrect – ADS B logic has lost power. ADS A has logic, but will be ‘blocked’ by the ARS emergency switches. Therefore – not ALL ADS valves will open, but some will.
C. Correct – ADS valves will still operate from the RSP because they are powered from DP-S1-2D after the transfer.
D. Incorrect - ADS valves will still operate from the RSP because they are powered from DP-S1-2D after the transfer.

COMMENTS:

Columbia is operating at 95% power with the Feedwater Level Control System in three-element control. RPV level is 36".

If the controlling Narrow Range Level instrument fails high, what will be the immediate trend of the ACTUAL RPV level, AND, what is the MAXIMUM / MINIMUM level WHEN RPV level stabilizes?

A. Up, BELOW Level 7

B. Up, ABOVE Level 7

C. Down, BELOW Level 4

D. Down, ABOVE Level 4

ANSWER: D

QUESTION TYPE: Closed Reference

KA # & KA VALUE: [New KA] 259002 K5.03 Knowledge of the operational implications of the following concepts as they apply to REACTOR WATER LEVEL CONTROL SYSTEM: Water level measurement (3.1) [New KA]
[KA Deleted] 259002 K5.09 Knowledge of the operational implications of the following concepts as they apply to REACTOR WATER LEVEL CONTROL SYSTEM: Adequate core cooling: FWCI (3.8) [KA Deleted]

REFERENCE: CGS System Description, FWLC, Rev. 13, pg. 6,16; SD000157

SOURCE: New Question

LO: 5400 Predict the expected response of the feedwater level control system in both Single and Three Element Control, to a failure or malfunction of the following: Loss of the selected RPV Level Channel
9711 Describes the FWLC system malfunctions, which will initiate a RFW CONTR SYSTEM TROUBLE alarm.

RATING: Knowledge: Analysis Difficulty: 2

ATTACHMENT: None

JUSTIFICATION:
A. Incorrect – If controlling level fails high, RFPT will slow down, causing level to go down.
B. Incorrect – If controlling level fails high, RFPT will slow down, causing level to go down.
C. Incorrect – Level should deviate a max of 3 inches before recovering.
D. Correct – Level should deviate a max of 3 inches before recovering.

COMMENTS: Ref : 10 CFR 55.41 (3) & (7)

An accident occurred allowing radioactive gas to be released into the Reactor Building.

If a SGT pressure controller fails, and Reactor Building pressure increases above zero, this would cause the offsite release rate of halogens to ___________. The operator can REDUCE the offsite release of halogens by manually___________ Standby Gas Treatment flow to the elevated release.

A. Increase, Increasing

B. Increase, Decreasing

C. Decrease, Increasing

D. Decrease, Decreasing

ANSWER: A

QUESTION TYPE: Closed Reference

KA # & KA VALUE: 261000 K3.02 Knowledge of the effect that a loss or malfunction of the STANDBY GAS TREATMENT SYSTEM will have on following: Off-site release rate (3.6)

REFERENCE: CGS System Description, SGT, Rev. 12 SD0900144

SOURCE: New Question

LO: 5822 State the Reactor Building pressure the SGT system is designed to maintain, as well as the pressure its DPIC is set to maintain and why it is at that setting.

RATING: Knowledge: Fundamental Difficulty: 2

ATTACHMENT: None

JUSTIFICATION:
A. Correct, unfiltered release increases. More flow means less pressure.
B. Incorrect – less flow will relatively increase pressure.
C. Incorrect – positive pressure creates outflow, which will increase release rate.
D. Incorrect – less flow will relatively increase pressure.

COMMENTS: Ref : 10 CFR 55.41 (13)

Columbia was operating at full power with Division 1 Standby Gas Treatment train tagged out for charcoal replacement when a Large Break Loss of Coolant Accident occurred.

Flow to the elevated release point from the Standby Gas Treatment system will be ________ with only one train operating instead of two, and the offsite release will be _____________ 10 CFR Part 100, Reactor Site Criteria, limits during the accident.

A. the same, above

B. the same, below

C. lower, above

D. lower, below

ANSWER: B

QUESTION TYPE: Closed Reference

KA # & KA VALUE: 261000 K6.01 Knowledge of the effect that a loss or malfunction of the following will have on the STANDBY GAS TREATMENT SYSTEM: A. C. Electrical distribution (2.9)

REFERENCE: CGS System Description, SGT, Rev. 12, pg. 3; SD000144

SOURCE: New Question

LO: 5821 State the purpose of the Standby Gas Treatment system.

RATING: Knowledge: Analysis Difficulty: 3

ATTACHMENT: None

JUSTIFICATION:

A. Incorrect - The design bases of the SGT is to maintain releases within part 100 during DBAs.
B. Correct – Flow is controlled to pressure, and the design of SGT is to maintain within part 100 limits.
C. Incorrect - The flow will be the same because the controller controls to a certain pressure which can be maintained with a single train.
D. Incorrect - The flow will be the same because the controller controls to a certain pressure which can be maintained with a single train.

COMMENTS: Ref : 10 CFR 55.41

On 4160 volt AC breakers NOT in the HPCS system which of the following is correct?

The charging motor charges the __________ spring.
If DC control power is lost, __________ breaker trip(s) is / are still active.

A. opening; no

B. opening; the overcurrent

C. closing; no

D. closing; the overcurrent

ANSWER: C

QUESTION TYPE: Closed Reference

KA # & KA VALUE: 262001 K5.02 Knowledge of the operational implications of the following concepts as they apply to A. C. ELECTRICAL DISTRIBUTION: Breaker control (2.6)

REFERENCE: CGS System Description, AC Distribution, Rev. 13, Pg. 20-23; SD000182

SOURCE: New Question

LO: 5065, 5051

RATING: Knowledge: Fundamental Difficulty: 3

ATTACHMENT: None

JUSTIFICATION:
A. Incorrect – The closing springs are charged by motor.
B. Incorrect – The closing springs are charged by motor.
C. Correct.
D. Incorrect – The overcurrent trip is powered from DC and is NOT active when DC is lost.

COMMENTS: Ref : 10 CFR 55.41 (7)

During a Station Blackout, as the battery discharges over time loads such as a motor will draw ___________ current while running. This is because battery voltage ___________ over time.

A. less, decreases.

B. less, increases.

C. more, decreases.

D. more, increases.

ANSWER: C

QUESTION TYPE: Closed Reference

KA # & KA VALUE: 263000 A1.01 Ability to predict and / or monitor changes in parameters associated with operating the D.C. ELECTRICAL DISTRIBUTION controls including: Battery charging/discharging rate (2.5)

REFERENCE: CGS Procedure 5.6.1, Rev. 12, pg. 5

SOURCE: New Question

LO:

RATING: Knowledge: Analysis/ Difficulty: 3

ATTACHMENT: None

JUSTIFICATION:

A. Incorrect – As voltage drops, the DC motor will demand more current.
B. Incorrect – As voltage drops, the DC motor will demand more current.
C. Correct – The battery voltage will decrease and more current is drawn from the battery as it discharges
D. Incorrect – Motor current goes up, and battery voltage decreases over time.

COMMENTS: Ref : 10 CFR 55.41 (8)

The plant has experienced a small LOCA. Drywell pressure is 6 psig. All systems functioned as designed. Ten minutes after the accident, all off-site power is lost.

Which of the following automatic responses would you expect following the loss of off-site power?

A. SW-P-1A/1B will start after its 20 second time delay.

B. DG-3 will trip on high jacket water temperature.

C. A Failure to Auto Start alarm will annunciate for all DGs.

D. HPCS-P-2 will start regardless of its discharge valve position.

ANSWER: D Post Exam Comment – Answer A was also accepted as correct based on a post exam comment. The reason distractor A was thought to be incorrect is the pump would not start until the discharge valve closed following the restoration of power to the bus. This would take longer than the 20 second time delay. However, the wording of the distractor states the pump will start "after its 20 second time delay." This is a true statement and is therefore accepted as correct.

QUESTION TYPE: Closed Reference

KA # & KA VALUE: 264000 A3.06 Ability to monitor automatic operations of the EMERGENCY GENERATORS (DIESEL/JET) including: Cooling water system operation (3.1)

REFERENCE: CGS System Description, Standby Service Water, Rev. 14; SD000204

SOURCE: New Question

LO: 7744 Describe the physical connection and/or cause-and-effect relationship between Service Water and: b. Diesel Generators

RATING: Knowledge: Analysis Difficulty: 3

ATTACHMENT: None

JUSTIFICATION:

A. Incorrect – the discharge valve must stroke closed after the sequencer sequences SW on, and then start to reopen. This adds to the normal start time.
B. Incorrect – this trip will still be bypassed after the LOOP.
C. Incorrect – this would trip the DG.
D. Correct – this pump auto starts regardless of its discharge valve position.

COMMENTS: Ref : 10 CFR 55.41 (8)

Columbia was operating at full power with TR-S tagged out of service to facilitate BPA work. A leak in containment caused drywell pressure to rise to 4 psig. All systems operated as designed except that breaker CB-B-8 failed to auto close.

If, coincident with the start of DG-2, a Generator Overcurrent condition were to occur, which of the following is correct?

A. DG-2 would start and trip due to the overcurrent condition. RHR-P-2B and RHR-P-2C would lose power.

B. DG-2 would tie onto and re-energize SM-8. RHR-P-2C starts and 5 seconds later, RHR-P-2B starts.

C. DG-2 would start but not tie onto SM-8 due to the overcurrent condition. RHR-P-2B and RHR-P-2C would lose power.

D. DG-2 would tie onto and re-energize SM-8. RHR-P-2B starts and 10 seconds later, RHR-P-2C would start.

ANSWER: B

QUESTION TYPE: Closed Reference

KA # & KA VALUE: 264000 K3.01 Knowledge of the effect that a loss or malfunction of the EMERGENCY GENERATORS (DIESEL/JET) will have on following: Emergency core cooling systems (4.2)

REFERENCE: CGS System Description, DG Pg 18, 19, 52; SD000200

SOURCE: New Question

LO: 5313 (DG) 7772 (AC)

RATING: Knowledge: Analysis Difficulty: 2

ATTACHMENT: None

JUSTIFICATION: A & C are incorrect because this trip is bypassed with a LOCA signal present. D is incorrect because the loading sequence is not correct.

COMMENTS: Ref : 10 CFR 55.41 (7) & (8)

Which of the following are available to cool the Control Air Compressors during a Loss of Offsite Power?

A. ONLY CJW-P-1A

B. ONLY CJW-P-1B

C. Fire Water AND CJW-P-1A

D. Fire Water AND CJW-P-1B

ANSWER: D

QUESTION TYPE: Closed Reference

KA # & KA VALUE: 300000 K1.04 Knowledge of the connections and/or cause effect relationships between INSTRUMENT AIR SYSTEM and the following: Cooling water to compressor (2.8)

REFERENCE: CGS System Description, Control and Service Air System, Rev. 9; SD000205

SOURCE: New Question

LO: LO 7606 Determine the affect on the CAS from the following events: b. Loss of Offsite Power

RATING: Knowledge: Fundamental Difficulty: 3

ATTACHMENT: None

JUSTIFICATION:

A. Incorrect, this pump is shed during a LOOP.
B. Incorrect, Fire Water is available due to the connections and the DG fire pump.
C. Incorrect, this CJW pump is shed during a LOOP.
D. Correct , the CJW pump is powered by the diesel on a vital load center, and fire water is available through the DG fire pump.

COMMENTS: Ref : 10 CFR 55.41 (4)

Which of the following conditions will prevent the manual insertion of any control rod using the Reactor Manual Control System?

A. RMCS Activity Control disagree.

B. IRM downscale at 14% power.

C. RPIS malfunction at 24% power.

D. RDCS Rod bypassed.

ANSWER: A

QUESTION TYPE: Closed Reference

KA # & KA VALUE: 201002 K4.04 Knowledge of REACTOR MANUAL CONTROL SYSTEM design feature(s) and/or interlocks which provide for the following: Single notch rod withdrawal and insertion (3.3)

REFERENCE: CGS System Description, RMCS, Rev. 11, pg. 15; SD000148

SOURCE: New Question

LO: 5799 State the function of the following rod motion indicators: b. Activity control disagree.

RATING: Knowledge: Fundamental Difficulty: 2

ATTACHMENT: None

JUSTIFICATION:

A. Correct – See reference.
B. Incorrect – The IRM downscale will only prevent rod withdrawal.
C. Incorrect – RPIS will only cause a rod block through RWM or RSCS below 20% power.
D. Incorrect - This only prevents a single rod from inserting using RMCS.

COMMENTS: Ref : 10 CFR 55.41 (6)

If power is lost to bus SH-5, which of the following components will lose power?

A. RRC-P-1A

B. COND-P-5

C. TSW-P-1A

D. CRD-P-1B

ANSWER: A

QUESTION TYPE: Closed Reference

KA # & KA VALUE: 202001 K2.01 Knowledge of electrical power supplies to the following: Recirculation Pumps Plant specific (3.2)

REFERENCE: CGS System Description, AC Distribution, Rev. 13; SD000182

SOURCE: New Question

LO: 5058 Identify the loads on the following buses: e. SH5, SH6

RATING: Knowledge: Fundamental Difficulty: 2

ATTACHMENT: None

JUSTIFICATION:
A. Correct – RRC A is powered from SH-5.
B. Incorrect – It is powered from SI-63 via SH-6.
C. Incorrect – Not powered from SH-5.
D. Incorrect – Not powered from SH-5.

COMMENTS: Ref : 10 CFR 55.41 (6)

The reactor is at 75% power when a voltage transient results in the loss of one Reactor Recirculation Pump. The transient has resulted in the reactor now operating in the AIA.

Based on this event, the operator should first:

A. prevent an ASD over-frequency pump trip by placing the operating loop controller in manual.

B. place the operating loop controller in manual to minimize the potential for vibration induced jet pump damage.

C. manually adjust the operating loop flow controller to exit the AIA and preclude uncontrolled power oscillations.

D. manually adjust the master flow controller to exit the AIA and avoid exceeding the power to flow scram setpoint.

ANSWER: C

QUESTION TYPE: Closed Reference

KA # & KA VALUE: 202002 A2.01 Ability to (a) predict the impacts of the following on the RECIRCULATION FLOW CONTROL SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Recirculation Pump Trip (3.4)

REFERENCE: SD000184, RRFC, Rev 14, page 12 of 43

SOURCE: NEW

LO: 9687 – State the conditions that will cause an individual ASD controller to automatically shift from AUTO to MANUAL.

RATING: Knowledge: Fundamental Difficulty: 2
ATTACHMENT: None

JUSTIFICATION:
A. Incorrect – The loop flow controller automatically shifts to manual on the trip of one of the running pumps.
B. Incorrect – The same reason as A.
C. Correct – Given the reactor is operating in the AIA, the first action should be to exit the AIA using flow control.
D. Incorrect – The master flow controller cannot be used with only one reactor recirc pump running (by interlock).

COMMENTS: 10CFR55.41 (5)

With the reactor stable at 920 psig and 36 inches, if containment temperature were to increase from 90 to 350 degrees, the indicated level on the Upset Level Range would:

A. increase as temperature increases.

B. decrease as temperature increases.

C. increase ONLY after the calibration temperature of 135 degrees is exceeded.

D. decrease ONLY after the calibration temperature of 135 degrees is exceeded.

ANSWER: A

6QUESTION TYPE: Closed Reference

KA # & KA VALUE: 216000 K5.07 Knowledge of the operational implications of the following concepts as they apply to NUCLEAR BOILER INSTRUMENTATION: Elevated temperature effects on level indication (3.6)

REFERENCE: CGS System Description, Nuclear Boiler Instrumentation, Rev. 9; SD000126
SOURCE: New Question
LO:

RATING: Knowledge: Analysis Difficulty: 3

ATTACHMENT: None

JUSTIFICATION: With the reactor stable, an increase in containment temperature would cause an increase in the temperature of the reference leg. This would lower the density making the reference leg “lighter”. Because level is derived by a dp cell measuring the difference in weight, the decrease in the weight of the reference leg would cause indicated level to increase making A the correct answer.

COMMENTS: Ref : 10 CFR 55.41 (5)
Added ‘as temperature increases’ to A and B.

The plant is at 99% power when condenser vacuum begins to lower (trending to no vacuum). If this trend continues the MSIVs will close at _____________ . Per ABN-VACUUM, the operator is required to _____________ .

A. 7” Hg; ONLY TRIP the Main Turbine

B. 7” Hg; SCRAM the Reactor and then trip the Main Turbine

C. 8.3” Hg; ONLY TRIP the Main Turbine

D. 8.3” Hg; SCRAM the Reactor and then trip the Main Turbine

ANSWER: D

QUESTION TYPE: Closed Reference

KA # & KA VALUE: 239001 A2.08 Ability to (a) predict the impacts of the following on the MAIN AND REHEAT STEAM SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Low condenser vacuum (3.6)

REFERENCE: ABN-VACUUM

SOURCE: New Question

LO:

RATING: Knowledge: Fundamental Difficulty: 2

ATTACHMENT: None

JUSTIFICATION:
A. Incorrect – MSIVs close at 8.3"
B. Incorrect – MSIVs close at 8.3"
C. Incorrect – If you trip the turbine first, then the Reactor might scram on high pressure.
D. Correct- MSIVs close at 8.3, tripping reactor first helps limit SRV usage and automatic trips.

COMMENTS: Ref : 10 CFR 55.41 (7) & (10)
Reworded stem

Columbia is in the process of a start up following a refueling outage. The following are some of the normal steps in placing the turbine generator on line:

1. Throttle valve / governor valve transfer
2. Bypass valves control pressure at 920 psig
3. Bypass valves close

Which of the following gives the correct sequence for these activities?

A. 1, 2, then 3

B. 2, 1, then 3

C. 3, 1, then 2

D. 1, 3, then 2

ANSWER: B

QUESTION TYPE: Closed Reference

KA # & KA VALUE: 241000 K1.28 Knowledge of the physical connections and/or cause- effect relationships between REACTOR/TURBINE PRESSURE REGULATING SYSTEM and the following: Reactor startup (3.2)

REFERENCE: CGS System Description, Main Turbine, Rev. 9, pg. 39-41; SD000129

SOURCE: New Question

LO:

RATING: Knowledge: Fundamental Difficulty: 2

ATTACHMENT: None

JUSTIFICATION:
A. Incorrect
B. Correct
C. Incorrect
D. Incorrect

COMMENTS: Ref : 10 CFR 55.41 (4) & (10)

Columbia was at 90% power when a malfunction of the A Moisture Separator Reheater 2nd stage temperature controller caused the Temperature Control Valves to go closed.

Due to the above, final steady state reactor thermal power will be ________ the original thermal power, and the main turbine governor valves will travel in the _________ direction.

A. the same as, open

B. the same as, closed

C. lower than, open

D. lower than, closed

ANSWER: A

QUESTION TYPE:	Closed Reference
KA # & KA VALUE:	245000 K3.03 Knowledge of the effect that a loss or malfunction of the MAIN TURBINE GENERATOR AND AUXILIARY SYSTEMS will have on following: reactor power (3.9)
REFERENCE:	CGS System Description, Columbia Simulator
SOURCE:	New Question
LO:	7747 Determine the affects of MSR Second Stage Reheater operations on the LP Main Turbine
RATING:	Knowledge: Analysis Difficulty: 4
ATTACHMENT:	None
JUSTIFICATION:	A. Correct – Rx power will be the same because there has been no change in the net reactivity for the reactor. Pressure goes up due to more steam flow through governor valves, which drives governor valves slightly open. B. Incorrect – See A. C. Incorrect – Power stays the same – colder feedwater offsets higher pressure. D. Incorrect – See C.
COMMENTS:	Ref : 10 CFR 55.41 (1) & (4) & (5) & (14)

Which of the following responses is expected for the Reactor Feedwater System following a complete loss of Plant Service Water (TSW)?

A. The feedpumps will eventually auto trip on high vibration.

B. The bearing temperatures will rise on the feedpumps.

C. The feedpumps will eventually auto trip on high lube oil temperature.

D. The feedpump auxiliary oil pump will auto start.

ANSWER: B

QUESTION TYPE: Closed Reference

KA # & KA VALUE: 259001 K6.06 Knowledge of the effect that a loss or malfunction of the following will have on the REACTOR FEEDWATER SYSTEM: Plant service water (2.7)

REFERENCE: CGS System Description, Feedwater, Rev. 9; SD000151

SOURCE: New Question

LO: 5768 Describe how the following systems interrelate with the Feedwater system. B. TSW

RATING: Knowledge: Fundamental Difficulty: 2

ATTACHMENT: None

JUSTIFICATION:

A. Incorrect – There is no automatic trip on high vibration.
B. Correct – The oil is no longer being cooled, so the bearing temperature will rise.
C. Incorrect – There is no automatic trip on high lube oil temperature
D. Incorrect - This pump has a start signal on a loss of pressure, not high temperature.

COMMENTS: Ref. 10 CFR 55.41 (4)

Columbia is in the process of a plant startup after a refueling outage. The Offgas and Air Removal systems are in the process of being placed in service.

Which of the following statements correctly describes the operation of the Steam Jet Air Ejector 1st Stage Pressure Control Valve, MS-PCV-16A, when its control switch is placed in the 'AUTO' position?

MS-PCV-16A opens when….

A. downstream pressure is LT 50 psig.

B. 1st stage steam flow is LT 9200 lbm/hr.

C. MS-V-12A, 2nd stage startup steam supply closes.

D. upstream steam supply pressure is GT 120 psig.

ANSWER: D Post Exam Comment - Answer A was also determined to be correct. The distractor was believed to be incorrect because the LT 50 psig causes the valve to open if the valve controller is in standby and upstream pressure is GT 120 psig. However, MS-PCV-16A, regulates downstream pressure at 120 psig so if downstream header pressure is LT 50 psig and upstream pressure is GT 120 psig, the valve should be open. Therefore, both A and D are correct.

QUESTION TYPE: Closed Reference

KA # & KA VALUE: 271000 A1.15 Ability to predict and/or monitor changes in parameters associated with operating the OFFGAS SYSTEM controls including: Steam supply pressures (2.7)

REFERENCE: CGS System Description, Air Removal System, Rev. 10; SD000181

SOURCE: New Question

LO: 5621 Describe the physical connection and/or the cause-and-effect relationship between the Offgas Processing system and the following: c. Control and Service Air system AND d. Main Steam System

RATING: Knowledge: Fundamental Difficulty: 3

ATTACHMENT: None

JUSTIFICATION:
A. Incorrect – In STDBY it opens if LT 50 psig and upstream press GT 120#.
B. Incorrect – Steam flow closes AR-V-2A/B/C

C. Incorrect – 2^{nd} stage operates independently from 1^{st} stage valves.
D. Correct – In AUTO valve opens when upstream pressure GT 120#.

COMMENTS: Ref : 10 CFR 55.41 (13)

Columbia was operating at 99% power with #2 Remote Intake valves, WOA-V-51B/52B, CLOSED.

If an event occurs that causes the Reactor Building Exhaust Plenum Radiation indication to raise to 16 mr/hr and stabilize, which of the following is the resultant Control Room HVAC lineup?

A. Normal Supply Plenum valves, WOA-V-51C/52C, OPEN; #1 Remote Air Intake valves, WOA-V-51A/52A, OPEN

B. Normal Supply Plenum valves, WOA-V-51C/52C, OPEN; #1 Remote Air Intake valves, WOA-V-51A/52A, CLOSED

C. Normal Supply Plenum valves, WOA-V-51C/52C, CLOSED; #1 Remote Air Intake valves, WOA-V-51A/52A, OPEN

D. Normal Supply Plenum valves, WOA-V-51C/52C, CLOSED; #1 Remote Air Intake valves, WOA-V-51A/52A, CLOSED

ANSWER: C

QUESTION TYPE: Closed Reference

KA # & KA VALUE: 290003 K1.03 Knowledge of the physical connections and/or cause- effect relationships between CONTROL ROOM HVAC and the following: Remote air intakes: Plant Specific (2.8)

REFERENCE: CGS System Description, Control Room HVAC, Rev. 10; SD000201

SOURCE: New Question

LO: 7649 Describe the CR HVAC response system response to a FAZ signal

RATING: Knowledge: Analysis Difficulty: 3

ATTACHMENT: None

JUSTIFICATION:

A. Incorrect – The normal supply plenum closes on a Z signal (13 mr/hr on RB exhaust).
B. Incorrect – The normal supply plenum closes on a Z signal.
C. Correct – The remote air intake (manual valves) must be open for the Control Room to pressurize.
D. Incorrect – With all intakes closed, the CR will not pressurize.

COMMENTS: Ref. 10 CFR 55.41 (7)
Reworded stem

With the mode switch in shutdown and reactor pressure at 135 psig, the reactor would be in Mode:

A. 2

B. 3

C. 4

D. 5

ANSWER: B

QUESTION TYPE: Closed Reference

KA # & KA VALUE: 2.1.22 Ability to determine the Mode of Operation (2.8)

REFERENCE: CGS Technical Specifications Table 1.1-1

SOURCE: New Question

LO:

RATING: Knowledge: Fundamental Difficulty: 2

ATTACHMENT: Steam Table

JUSTIFICATION:

A. Incorrect because mode switch is not in startup.
B. Correct because mode switch in shutdown, and pressure indicates temperature above 200F.
C. Incorrect because pressure indicates temperature above 200F.
D. Incorrect because pressure indicates that vessel head is off.

COMMENTS: Ref : 10 CFR 55.41 (5) & (10)

With Columbia operating at 100% power, a lower seal cavity (Seal No. 1) pressure of ____ psig and an upper seal cavity (Seal No. 2) pressure of ____ psig would be indicative of a degraded upper seal on a Reactor Recirculation Pump?

A. 310; 910

B. 510; 710

C. 710; 510

D. 910; 310

ANSWER: D

QUESTION TYPE: Closed Reference

KA # & KA VALUE: 2.1.7 Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretations (3.7)

REFERENCE: CGS Procedure ABN-RRC-SEAL, Rev 4, Step 1.2.

SOURCE: New Question

LO:

RATING: Analysis/Difficulty: 3

ATTACHMENT: None

JUSTIFICATION:

A. Incorrect – The indications would be the failure of the lower seal.
B. Incorrect – The indications would be the failure of the lower seal.
C. Incorrect – The indications would be the failure of the lower seal
D. Correct – Upper seal pressure would drop below 510 psig for a failure of upper seal.

COMMENTS: Ref : 10 CFR 55.41 (3)

Procedure 6.3.2, Fuel Shuffling and/or Offloading and Reloading states, "Total core flow is restricted to LE 10,000 GPM drive flow via RHR and/or RRC." This applies with fuel bundles removed from the core.

The reason for this precaution is to prevent damaging the:

A. Control Rods.

B. LPRMs.

C. Jet Pumps.

D. Refueling Equipment.

ANSWER: B

QUESTION TYPE: Closed Reference

KA # & KA VALUE: 2.2.26 Knowledge of refueling administrative requirements (2.5)

REFERENCE: CGS Procedure 6.3.2, Fuel Shuffling and/or Offloading and Reloading, Rev. 16, pg. 12

SOURCE: New Question

LO:

RATING: Knowledge: Analysis Difficulty: 3

ATTACHMENT: None

JUSTIFICATION:

A. Incorrect – The Control Rods are unaffected due to support from fuel.
B. Correct – See reference.
C. Incorrect – Jet pumps are adequately supported during refueling.
D. Incorrect – Cross flow of concern should only occur in between the fuel assemblies.

COMMENTS: Ref : 10 CFR 55.41 (2) & (10)

Which of the following is the criteria for stopping the fuel shuffle process in procedure 6.3.2, Fuel Shuffling and/or Offloading and Reloading?

A. Doubling in the SRM period.

B. Doubling in the average SRM count rate.

C. Doubling of any single SRM period.

D. Doubling of any single SRM count rate.

ANSWER: B

QUESTION TYPE: Closed Reference

KA # & KA VALUE: 2.2.28 Knowledge of new and spent fuel movement procedures (2.6)

REFERENCE: CGS Procedure 6.3.2, Fuel Shuffling and/or Offloading and Reloading, Rev. 16, Section 2.2

SOURCE: New Question

LO:

RATING: Knowledge: Fundamental Difficulty: 3

ATTACHMENT: None

JUSTIFICATION:
A. Incorrect – The reference says count rate.
B. Correct – See reference.
C. Incorrect – Reference says TWO doublings of any single SRM count rate.
D. Incorrect - Reference says TWO doublings of any single SRM count rate.

COMMENTS: Ref: 10 CFR 55.41 (10)

The reason for the automatic reactor scram associated with a main turbine trip is to:

A. limit cycling of the SRVs.

B. mitigate the reactor power increase.

C. prevent a main steam line rupture.

D. minimize the wetwell heatup.

ANSWER: B

QUESTION TYPE: RO

KA # & KA VALUE: 295005AK1.01 4.0/4.1 10CFR55.41 Knowledge of the operational implications of the following concepts as they apply to Main Turbine Generator Trip: Pressure effects on reactor power. (4.0)

REFERENCE: SD000161; TS Bases 3.3.1.1

SOURCE: Modified

LO: 5949

RATING: Knowledge: Fundamental Difficulty: 2

ATTACHMENT: None

JUSTIFICATION: As stated in the system description, the basis for the automatic scram is to limit the pressure and corresponding power increase following the closure of the throttle valves. Additionally, the basis for Turbine Trip LCO states the pressure and power effects on the reactor following a trip of the Main Turbine must be limited. B is correct.

COMMENTS:

The reactor was operating at 99% power when the reactor scrammed. The following conditions exist:

HPCS and RCIC auto started
RCIC is maintaining level in the normal band
Drywell pressure is .92 psig
Both Reactor Recirculation Pumps have tripped
CB-RPT-3A/4A and CB-RPT-3B/4B are open
Both Reactor Feed Pumps have tripped

Which of the following caused the scram?

A. Main turbine trip

B. MSIV isolation

C. Reactor level + 13 inches

D. Reactor pressure 1060 psig

ANSWER: A

QUESTION TYPE: RO/SRO

KA # & KA VALUE: 295005 AK2.03 Knowledge of the interrelations between MAIN TURBINE GENERATOR TRIP and the following: Recirculation system (3.2)

REFERENCE: SD000178 RRC Systems text pages 10, 23 and 24

SOURCE: Bank

LO: 5023

RATING: Knowledge: Analysis Difficulty: 2

ATTACHMENT: None

JUSTIFICATION: Only a Main Turbine trip opens both CB-RPT-3A/4A and CB-RPT 3B/4B. A is correct. The other three choices are scram signals but would only open CB-RPT-3A and 3B.

COMMENTS:

The Control Room has been abandoned and all immediate actions have been completed.

According to ABN-CR-EVAC, the RPV must be Emergency Depressurized from the Remote Shutdown Panel when indicated RPV level reaches:

A. -147”.

B. -150".

C. -161”.

D. -183”.

ANSWER: A

QUESTION TYPE: RO/SRO

KA # & KA VALUE: 295016 AA1.06 Ability to operate and/or monitor the following as they apply to CONTROL ROOM ABANDONMENT: Reactor water level (4.0)

REFERENCE: ABN-CR-EVAC; SD000126

SOURCE: NEW

LO: 11401

RATING: Knowledge: Fundamental Difficulty: 3

ATTACHMENT: None

JUSTIFICATION: ABN-CR-EVAC states ED is required at -147”. ED is performed when RPV level drops below lowest usable indicated RPV level which is -147” on the wide range. A is correct. Emergency Depressurization, without control room evacuation would occur at an RPV level of -161” (non-ATWS) and -183” (ATWS) thus C and D are incorrect. -150” is lowest meter indication for a wide range instrument, is not usable, and therefore D is incorrect.

COMMENTS: Reworded stem

Columbia is in a refueling outage. Fuel shuffle evolutions are on-going on the Reactor Building 606' elevation. Due to an error associated with the placement of spent fuel bundles in the Spent Fuel Pool the 606' Fuel Pool area criticality Monitor, ARM-RIS-2, alarms in the Control Room.

What indications of the alarming radiation monitor are available on the Reactor Building 606' Refueling Floor?

A. A rotating amber light only.

B. A pulsing red light only.

C. A rotating amber light and an klaxon alarm.

D. A pulsing red light and an klaxon alarm.

ANSWER: C

QUESTION TYPE: RO/SRO

KA # & KA VALUE: 295023 AA1.04 Ability to operate and/or monitor the following as they apply to REFUELING ACCIDENTS: Radiation Monitoring equipment. (3.4 3.7)

REFERENCE: SD000141

SOURCE: NEW

LO: 5114

RATING: Knowledge: Fundamental Difficulty: 2

ATTACHMENT: None

JUSTIFICATION: Per the systems text all Area Radiation Monitors have a rotating beacon. Additionally, ARM-RIS-2 has a klaxon horn associated with its alarm condition. C is correct.

COMMENTS:

A scram occurs with Columbia at 90% power and near the end of the operating cycle. RPV level has been recovered from -65 inches and the scram has been reset.

After resetting the scram, the reactor operator notes all RPS Group white lights are illuminated but the scram discharge volume vents and drains did not open. The reactor operator also notes the scram accumulators are not recharging.

These indications would be expected if:

A. APRM power peaked at 120 percent.

B. RPV pressure peaked at 1138 psig.

C. Drywell pressure peaked at 1.9 psig.

D. Scram Discharge Volume level peaked at the 530' elevation.

ANSWER: B

QUESTION TYPE: RO/SRO

KA # & KA VALUE: 295025 EK2.04 Knowledge of the interrelations between HIGH REACTOR PRESSURE and the following: ARI/RPT/ATWS: Plant specific (3.9)

REFERENCE: SD000142

SOURCE: Bank Slightly Modified

LO: 5189

RATING: Knowledge: Analysis Difficulty: 3

ATTACHMENT: None

JUSTIFICATION: ATWS/ARI Logic is initiated at a RPV pressure of 1120 and a RPV level of -50". B is correct as it is GT 1120 psig. A, B, and C would cause a scram but not the initiation of ATWS ARI.

COMMENTS:

Which of the following would preclude the use of a wide range level instrument to report RPV level while operating in the EOPs?

A. RPV Pressure of 25 psig; Drywell Temperature of 300°F; no erratic indications observed

B. RPV Pressure of 50 psig; Drywell Temperature of 285°F; erratic indication observed

C. RPV Pressure of 75 psig; Drywell Temperature of 330°F; erratic indication observed

D. RPV Pressure of 100 psig; Drywell Temperature of 325°F; no erratic indication observed

ANSWER: C

QUESTION TYPE: RO/SRO

KA # & KA VALUE: 295028 EK2.03 Knowledge of the interrelations between HIGH DRYWELL TEMPERATURE and the following: Reactor Water level indication (3.6)

REFERENCE: PPM 5.0.10 RPV Saturation Temperature Curve

SOURCE: New

LO: 8488

RATING: Knowledge: Analysis Difficulty: 2

ATTACHMENT: SATURATION TEMPERATURE CURVE from PPM 5.0.10

JUSTIFICATION: Per PPM 5.0.10 and the RPV Saturation Temperature Curve, the answer is C as the parameters are within the unsafe region of figure A and erratic indications are observed

COMMENTS:

The plant was operating at rated power when a trip of both Reactor Feed Pumps occurred. RPV level dropped to -75 inches before returning to the normal operating band. All systems operated as designed except that coincident with the reactor scram, the feeder breaker to MC-4A tripped open.

Given these conditions, the operator should trip DG-3….

A. by locally closing the engine fuel oil supply valve.

B. by placing the Unit Mode Selector Switch in the “MAINT” position.

C. from the control room within 6 minutes.

D. at the local control panel immediately.

ANSWER: D

QUESTION TYPE: RO/SRO

KA # & KA VALUE: 295031 Reactor Low Water Level. 2.4.24 Knowledge of loss of cooling water procedures. (3.3 / 3.7)

REFERENCE: ABN-SW; SD000204; SD000200

SOURCE: NEW

LO: 6760, 5835, 5837

RATING: Knowledge: Analysis Difficulty: 3

ATTACHMENT: None

JUSTIFICATION: A RPV level of -50 inches causes DG-3 to start. The HPCS service water pump, HPCS-P-2, is powered from MC-4A. With the feeder opening to MC-4A, this prevents HPCS-P-2 from starting. This means DG-3 is running without service water. Per ABN-SW, DG-3 is immediately tripped. D is correct. C is incorrect because DG-3 cannot be tripped from the control room. Additionally the time is incorrect for DG-3 but is correct for DG-1 and DG-2 if they were running without service water. A is incorrect because it is not per procedure. B is incorrect as it would not stop local starts of the DG.

COMMENTS: Reworded stem

The plant was operating at 97% power when a transient occurred resulting in a gaseous release. QEDPS indicates a TEDE (Whole Body) dose that requires a General Emergency classification. The CDE (Thyroid/Iodine) dose is only 20% of the required General Emergency dose threshold.

Based on these conditions, the operating crew should conclude that the release is from the:

A. Reactor Building with SGT in service.

B. Reactor Building with SGT not in service.

C. Turbine Building with Turbine Building HVAC in service.

D. Turbine Building with Turbine Building HVAC not in service.

ANSWER: A Post Exam Comment – The licensee recommended this question be deleted because it is beyond the scope of knowledge for an RO. This recommendation is based on the interpretation of QEDPS not being an RO task. This recommendation was rejected because the RO was not asked to interpret the QEDPS data as this was done in the stem of the question. This data was provided so the RO could determine the release was being filtered thereby eliminating distractors C and D. The only gaseous filtration system is the SGT system and the RO should be able to observe from the information provided it must be in service. Because the filtration function is one of the SGT system's primary purposes, this is testable RO knowledge.

QUESTION TYPE: RO/SRO

KA # & KA VALUE: 295038 High Off-site Release Rate EA2.04 Ability to determine and/or interpret the following as they apply to HIGH OFF-SITE RELEASE RATE: Source of off-site release (4.1 4.5)

REFERENCE: SD000144

SOURCE: Bank

LO: 5821

RATING: Knowledge: Analysis Difficulty: 3

ATTACHMENT: None

JUSTIFICATION: A filtered release, i.e. with SGT in operation, results in a relatively low CDE (Thyroid from Iodine) dose. A projected dose at the site boundary high enough for a General Emergency, but with a relatively low Thyroid dose can only be the result of a release through SGT. A is correct.

COMMENTS:

The plant was operating at 99% power when a Main Turbine Trip occurred but the reactor did not scram. Direction in the EOPs is given that if SRVs are cycling, manually open SRVs until pressure drops to 945 psig.

Which of the following describes the basis for this direction?

A. Maintains reactor water inventory in the Containment.

B. Maximizes the amount of steam condensed in the wetwell.

C. Maximizes the amount of energy directed to the main condenser.

D. Maintains pressure below the scram setpoint and allows resetting of the scram.

ANSWER: C

QUESTION TYPE: RO/SRO

KA # & KA VALUE: 295007 AK3.04 Knowledge of the reasons for the following responses as they apply to HIGH REACTOR PRESSURE: Safety/relief valve operation: Plant specific (4.0 4.1)

REFERENCE: 5.0.10

SOURCE: Bank (slightly modified stem and modified answer to be consistent with distractors)

LO: 8053

RATING: Knowledge: Analysis Difficulty: 3

ATTACHMENT: None

JUSTIFICATION: SRVs are opened to stop SRVs from cycling and pressure is reduced to 945 psig which is the pressure at which steam flow through the BPVs is at 100%. C is correct.

COMMENTS:

The plant was operating at 89% power when a Recirculation Suction Line break caused a High Drywell Pressure reactor scram. The High Drywell signal has cleared and ONLY the scram has just been reset.

Which of the following is correct concerning these conditions?

A. EDR-R-5 (Sump in the CRD Pump room) is filling from the scram discharge header, and pumps down based on the operation of the Fill/Pump out Timer.

B. EDR-R-5 (Sump in the CRD Pump room) is filling from the scram discharge header, but does not pump down due to the isolation of the outlet discharge valve EDR-V-395.

C. FDR-R-3 (Sump in the HPCS Pump room) is filling from the broken RRC Suction line and pumps down based on the operation of the Fill/Pumpout Timer.

D. FDR-R-3 (Sump in the HPCS Pump room) is filling from the broken RRC Suction line, but does not pump down due to the isolation of the outlet discharge valve FDR-V-220.

ANSWER: B

QUESTION TYPE: RO/SRO

KA # & KA VALUE: 295036 EA2.03 Ability to determine and/or interpret the following as they apply to SECONDARY CONTAINMENT HIGH SUMP/AREA WATER LEVEL: Cause of the high water level (3.4 3.8)

REFERENCE: SD000142; SD000167; SD000173

SOURCE: Bank

LO: 5475

RATING: Knowledge: Analysis Difficulty: 3

ATTACHMENT: None

JUSTIFICATION: C and D are incorrect because the drywell outlet valves for the floor drains isolate on the high drywell pressure and does not reopen based on the scram being reset. A is incorrect because the sump outlet isolates on the high drywell pressure and does not reopen based on the scram being reset. B is correct because the water in the sump comes from the SDV and it does not pump down until the “F” signal is reset.

COMMENTS:

The following plant conditions exist following an extended run at rated conditions:

Reactor level was -137 inches for the last 3 minutes and is now trending up slow
SM-7 is out of service due to ongoing maintenance
HPCS-P-1 is injecting into the core
RHR-P-2B and RHR-P-2C are not running
ADS is NOT inhibited

Which of the following describes the response to a manual start of RHR-P-2C?

A. When RHR-P-2C discharge pressure is GE 125 psig, all ADS SRVs will open immediately.

B. When RHR-P-2C discharge pressure is GE 125 psig for 105 seconds, all ADS SRVs will open.

C. When the breaker for RHR-P-2C closes, all ADS SRVs will open immediately.

D. When the breaker for RHR-P-2C closes, all ADS SRVs will open 105 seconds later.

ANSWER: A

QUESTION TYPE: RO/SRO

KA # & KA VALUE: 203000 K3.03 Knowledge of the effect that a loss or malfunction of the RHR/LPCI INJECTION MODE will have on following: Automatic depressurization logic (4.2 4.3)

REFERENCE: SD000186

SOURCE: Bank – Modified stem and distractors

LO: 5070

RATING: Knowledge: Analysis Difficulty: 3

ATTACHMENT: None

JUSTIFICATION: ADS will initiate when both of the following conditions are met: 105 seconds after -129” and RHR pressure GE 125 psig. A is correct. B is not correct because it includes the 105 seconds that have already timed out. C and D are incorrect because they are based on breaker closure not system pressure.

COMMENTS:

Average Power Range Monitor (APRM) channel "C" is powered from:

A. Critical Instrument Power Inverter IN-1.

B. 125 VDC Distribution Panel DP-S1-1A.

C. 24 VDC Distribution Panel DP-SO-A.

D. Reactor Protection System Bus "A".

ANSWER: D

QUESTION TYPE: RO/SRO

KA # & KA VALUE: 215005 K2.02 Knowledge of electrical power supplies to the following: APRM channels (2.6 2.8)

REFERENCE: SD000149

SOURCE: NEW

LO: 5096

RATING: Knowledge: Fundamental Difficulty: 3

ATTACHMENT: None

JUSTIFICATION: COMMENTS: As stated in SD000149 – APRM ‘C’ is powered from RPS A thus D is correct.

RCIC has been manually started for RPV pressure control and is operating in the CST to CST mode.

If CST level decreases to 1' 6", then:

A. a RCIC turbine trip will occur on low RCIC pump suction pressure causing RCIC-V-1 to close.

B. RCIC will take a suction from the Suppression Pool and discharge back to the Suppression Pool through the full flow test line.

C. RCIC will take a suction from the Suppression Pool and discharge back to the Suppression Pool through RCIC-V-19.

D. RCIC will take a suction from the Suppression Pool and transfer water to the CSTs through RCIC-V-59.

ANSWER: C

QUESTION TYPE: RO/SRO

KA # & KA VALUE: 217000 K4.07 Knowledge of REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) design feature(s) and/or interlocks which provide for the following: Alternate supplies of water (3.6 3.6)

REFERENCE: PPM 4.601.A4-3.4

SOURCE: Bank – Slightly modified

LO: 5724

RATING: Knowledge: Fundamental Difficulty: 2

ATTACHMENT: None

JUSTIFICATION: At a CST level of 1’10” RCIC-V-31 (SP suction) opens and RCIC-V-10 (CST suction) closes. When RCIC-V-31 is full open RCIC-V-22 and V-59 (full flow test line to CST) close thus C is correct. A is not correct because one suction valve does not close until the other is full open. B is incorrect because RCIC-V-22 and V-59 close. D is incorrect because RCIC-V-19 discharges to the SP.

COMMENTS:

The Control Room Operator (CRO) manually aligned the valves for the Reactor Core Isolation Cooling (RCIC) system from the control room and started the RCIC pump during the performance of a pump operability surveillance. RCIC is now operating in the CST to CST mode.

An Equipment Operator just reported the presence of a small steam leak on the RCIC turbine. The CRS has directed the CRO to secure the RCIC turbine.

The CRO depresses the RCIC “MANUAL ISOLATION” pushbutton.

In response to this action, the RCIC turbine will:

A. trip and both the inboard and outboard RCIC steam supply line isolation valves (RCIC-V-63 and RCIC-V-8) will close.

B. continue to operate normally.

C. trip and ONLY RCIC-V-63 (steam supply line inboard isolation valve) will close.

D. trip and ONLY RCIC-V-8 (steam supply line outboard isolation valve) will close.

ANSWER: B

QUESTION TYPE: RO/SRO

KA # & KA VALUE: 217000 A3.01 Ability to monitor automatic operations of the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) including: Valve operation (3.5 3.5)

REFERENCE: SD000180

SOURCE: Bank – modified slightly

LO: 5723

RATING: Knowledge: Analysis Difficulty: 3

ATTACHMENT: None

JUSTIFICATION: If an initiation signal were present, the isolation P/B would close RCIC-V-8 and trip the turbine. In the stem, it is clear an initiation signal is not present thus B is the correct answer and other answers are incorrect.

COMMENTS: Stem revised

Which of the following describes the effect of a loss of normal 480 VAC power to UPS inverter E-IN-1?

A. The Static Switch provides a make before break forward transfer of loads from the normal AC source, MC-7A, to the 250 VDC battery.

B. The 250 VDC battery, which supplies the inverter in parallel with the output of the rectifier fed from MC-7A, assumes the load.

C. A break before make transfer to the Kirk Key Bypass Source, MC-7F, results in a momentary (4 millisecond) loss of power to inverter E-IN-1 loads.

D. The Static Switch provides a bumpless transfer of the critical UPS loads from the inverter output, to the Bypass AC source, fed from MC-7F.

ANSWER: B

QUESTION TYPE: RO/SRO

KA # & KA VALUE: 262002 K4.01 Knowledge of UNINTERRUPTIBLE POWER SUPPLY (A. C./D. C.) design feature(s) and/or interlocks which provide for the following: Transfer from preferred power to alternate power supplies (3.1 3.4)

REFERENCE: ABN-INV

SOURCE: Bank – Slightly modified

LO: 5891

RATING: Knowledge: Fundamental Difficulty: 3

ATTACHMENT: None

JUSTIFICATION: The battery is the first to assume load on a loss of normal AC as stated in the answer B. A is incorrect because there is no static switch involved. C is incorrect because the Kirk Key swaps power between normal and bypass source MC-7A. D is incorrect because static switch no involved and bypass source is MC-7F.

COMMENTS:

Using the attached Electrical Wire Diagram of the Plant Service Water System, which of the following statements is correct?

A. Contact S21 / S21T is closed when the breaker for TSW-P-1A is open and would be opened when the breaker for TSW-P-1A is closed.

B. When emergency power is restored, TSW-P-1A auto starts after a 10 second time delay if Standby Pump Selector Switch is in TSW-P-1A position.

C. An undervoltage on SM-85 causes TSW-P-1A to start regardless of lube water flow if start occurs within 60 seconds of undervoltage on SM-85.

D. TSW-P-1A trips when TSW-V-53A is 15% open in the closed direction regardless of TSW-P-1A control switch position.

ANSWER: B

QUESTION TYPE: RO/SRO

KA # & KA VALUE: 400000 2.1.24 Ability to obtain and interpret station electrical and mechanical drawings (2.8 3.1)

REFERENCE: EWD – 57E – 002

SOURCE: NEW

LO: 4047

RATING: Knowledge: Analysis Difficulty: 2

ATTACHMENT: EWD-57E-002

JUSTIFICATION: A is incorrect because Contact is an ‘a’ contact and follows breaker position. C is incorrect because lube water flow has to be normal on any pump start regardless of any time considerations. D is incorrect because the contacts above valve at 15% contacts requires the switch to be in any position other than Auto after start. B is correct as the pump selector switch needs to be in the TSW-P-1A position and on the pump start a 10 second time delay is enforced by TSW-RLY-62/TSW1A.

COMMENTS:

A few minutes after resetting a valid reactor scram signal, the CRO notes that the red and green lights for the Scram Discharge Volume Drain Valves, are both illuminated.

Based on these indications, the CRO should conclude:

A. One drain valve is intermediate and the other drain valve is either intermediate or full open.

B. One drain valve is intermediate and the other drain valve is either intermediate or full closed.

C. The outboard drain valve is full closed and the inboard drain valve is full open.

D. The inboard drain valve is full closed and the outboard drain valve is full open.

ANSWER: A

QUESTION TYPE: RO/SRO

KA # & KA VALUE: 201001 A3.10 Ability to monitor automatic operations of the CONTROL ROD DRIVE HYDRAULIC SYSTEM including: Lights and Alarms (3.0 2.9)

REFERENCE: SD000142

SOURCE: Bank – modified stem and distractor wording

LO: 5198

RATING: Knowledge: Fundamental Difficulty: 2

ATTACHMENT: None

JUSTIFICATION: As per SD000142; both SDV drain valve lights illuminated indicate both drain valves are at least in the intermediate position. A is correct.

COMMENTS:

The control room switch for Fuel Pool Circulating Pump A (FPC-P-1A) is in the IR-71 position and the control switch for FPC-P-1B is in the IR-69 position.

With this switch alignment:

A. taking the local control switch for the standby FPC pump to the START position will start the pump without enforcing the start permissives.

B. and both local control switches in the NEUTRAL position, the standby pump only auto start if the operating FPC pump has a low discharge pressure.

C. and the local control switch for both FPC pumps in the START position, neither pump will start if there is an 'F' or 'A' signal.

D. and the local control switch for both FPC pumps in the NEUTRAL position, if the operating FPC pump trips, the standby FPC pump will not auto start.

ANSWER: D

QUESTION TYPE: RO/SRO

KA # & KA VALUE: 233000 2.1.30 Ability to locate and operate components / including local controls (3.9 3.4)

REFERENCE: SD000202

SOURCE: NEW

LO: 15308

RATING: Knowledge: Analysis Difficulty: 3

ATTACHMENT: None

JUSTIFICATION: A is incorrect because start permissives are still enforced with operation from the local control panel. B is incorrect because there is no auto feature associated with the standby pumps with C/S in NEUTRAL. C is incorrect because the FPC start logic does not look at an F or A signal. D is correct per systems text.

COMMENTS:

Columbia was operating at rated power when a manual scram was initiated but control rods did not insert. PPM 5.1.1 was entered and exited to PPM 5.1.2 due to the ATWS condition. RPV level is currently -20" and reactor power is approximately 20%. All three Circ Water pumps are in operation and the Main Turbine is on line.

In accordance with the EOPs, PPM 5.5.6, Bypassing MSIV Low RPV Level and High Steam Tunnel Temperature Isolation Interlocks, has been performed and RCIC-V-1 has been manually closed.

The EOPs now direct that RPV level be lowered in an effort to reduce reactor power. RPV level is now -60 inches and trending down slowly.

Which of the following choices indicates the correct lineup for the above conditions?

A. CW-P-1B, CW-P-1C, and the Main Turbine tripped at -50 inches. CW-P-1A will continued to operate.

B. CW-P-1B and CW-P-1C tripped at -50 inches. CW-P-1A continued to operate. The Main Turbine stayed on line.

C. All three Circ Water pumps tripped at -50 inches. The Main Turbine will trip on loss of Main Condenser Vacuum.

D. All three Circ Water pumps continued to operate. The Main Turbine stayed on line.

ANSWER: B

QUESTION TYPE: RO/SRO

KA # & KA VALUE: 2.1.31 Ability to locate control room switches, controls and indications and to determine that they are correctly reflecting the desired plant lineup (4.2 3.9)

REFERENCE: SD000180; SD000193

SOURCE: NEW

LO: 11241

RATING: Knowledge: Analysis Difficulty: 3

ATTACHMENT: None

JUSTIFICATION: Closing RCIC-V-1 prevents RCIC start at Level 2 and tripping off the Main Turbine. PPM 5.5.6 does nothing to the logic for the CW Pumps. CW-P-1B and CW-P-1C will trip at Level 2. CW-P-1A does not trip on a Level 2 signal. A is incorrect as it indicates the MT will trip. C is incorrect because it states all 3 CW pumps will trip at -50”. D is incorrect because B and C CW pumps do trip at -50”. B is correct.

COMMENTS:

If stroke time testing a valve closed, the stopwatch is started when:

A. both red and green lights are illuminated and is stopped when the red light extinguishes.

B. the control switch is rotated to the closed position and is stopped 10 seconds after the red light extinguishes.

C. the control switch begins to be rotated towards the closed position and is stopped when the red light extinguishes.

D. both red and green lights are illuminated and is stopped when the red light extinguishes.

ANSWER: C

QUESTION TYPE: RO/SRO

KA # & KA VALUE: 2.2.12 Knowledge of surveillance procedures (3.0 3.4)

REFERENCE: OSP-RHR/IST-Q702 precaution and limitation 4.7

SOURCE: NEW

LO: 10776

RATING: Knowledge: Fundamental Difficulty: 2

ATTACHMENT: None

JUSTIFICATION: Per the surveillance procedures, the stopwatch is started when the control switch is turned and stopped when the valve indicates full open or closed. C is correct.

COMMENTS: Removed wording in stem

You have been directed to perform a task in an area of the plant where the radiation level is 80 mr/hr. You expect the task to last 30 minutes. This quarter you have received 987 mrem TEDE.

Based on the dose you will receive performing this task…….

A. an Administrative Dose Hold Point will become effective when you receive an additional 13 mrem TEDE.

B. an Administrative Dose Extension shall be approval by the Plant General Manager PRIOR to beginning the task.

C. an Administrative Dose Extension shall be approval by the Radiation Protection Manager PRIOR to beginning the task.

D. there are no Administrative Dose Hold Points associated with the completion of this task.

ANSWER: D

QUESTION TYPE: RO

KA # & KA VALUE: 2.3.4 Knowledge of the radiation exposure limits and contamination control / including permissible levels in excess of those authorized. (2.5 3.1)

REFERENCE: GEN-RPP-06

SOURCE: NEW

LO: 11257

RATING: Knowledge: Fundamental Difficulty: 3

ATTACHMENT: None

JUSTIFICATION: As stated in GEN-RPP-06 attachment 8.1, an administrative dose hold point occurs for a TEDE of 2 Rem. The question will have the individual dose exceeding 1 Rem therefore no dose hold point is applicable. D is correct. If 2 rem were exceeded an Administrative Dose Hold Point would occur. RPM approval is required. If 4 rem TEDE were to be exceeded, then the Plant General Manager's approval would be required.

COMMENTS: Reword stem

Your electronic dosimeter reads 0 mrem when you entered a posted radiological area. Now, after spending 10 minutes in the area, it reads 20 mrem.

Based on this information, this area should be posted as a:

A. Radiation Area.

B. High Radiation Area.

C. High High Radiation Area.

D. Locked High Radiation Area.

ANSWER: B

QUESTION TYPE: RO

KA # & KA VALUE: 2.3.1 Knowledge of 10 CFR 20 and related facility radiation control requirements. (2.6 3.0)

REFERENCE: PPM 11.2.7.1; SWP-RPP-01

SOURCE: NEW

LO: 11257

RATING: Knowledge: Fundamental Difficulty: 2

ATTACHMENT: None

JUSTIFICATION: 20 mrem times 6 (6 – 10 minute periods in an hour) is 120 mrem which is a High Radiation Area. A high Radiation Area is posted with a sign and the words CONTACT HP PRIOR TO ENTRY. B is correct.

COMMENTS:

With Columbia operating at power, heavy smoke is quickly filling the Control Room.

According to ABN-CR-EVAC, which of the following actions must be completed prior to evacuating the Control Room?

A. If Control Rods failed to insert, initiate ARI.

B. Arm and Depress MSIV Isolation Logic Pushbuttons.

C. Have the Safe Shutdown Operator perform Attachment 7.1.

D. Place the Mode Switch in the 'Refuel' position.

ANSWER: B

QUESTION TYPE: RO

KA # & KA VALUE: 2.4.11 Knowledge of Abnormal Condition procedures. (3.4 3.6)

REFERENCE: ABN-CR-EVAC

SOURCE: NEW

LO: 6889

RATING: Knowledge: Analysis Difficulty: 3

ATTACHMENT: None

JUSTIFICATION: Per ABN-CR-EVAC none are immediate operator actions except closing the MSIVs. B is correct.

COMMENTS:

Events occur which require the Shift Manager to declare an Unusual Event.

Which of the following is correct concerning emergency notifications for this event classification?

A. The designated individual will initiate the NRC Event Notification System within thirty minutes after the emergency event declaration.

B. The designated Equipment Operator will initiate the NRC Event Notification System within one hour after State and Local authorities are notified.

C. The designated Reactor Operator will initiate the NRC Event Notification System within one hour after the emergency event declaration.

D. The NRC Event Notification System is not required to be activated at this event classification level.

ANSWER: C

QUESTION TYPE: RO

KA # & KA VALUE: 2.4.29 Knowledge of the Emergency Plan (2.6 4.0)

REFERENCE: PPM 13.4.1

SOURCE: NEW

LO: 6176

RATING: Knowledge: Fundamental Difficulty: 2

ATTACHMENT: None

JUSTIFICATION: Per PPM 13.4.1, the NRC is notified within one hour of event notification. C is correct.

COMMENTS:

Columbia is operating with the following conditions given:

The reactor is operating at 100% power
Rod line is 100%
OPRMs are inoperable

Due to an ASD fault, both RRC-P-1A and RRC-P-1B run back to 15 Hz.

Based on the given conditions, which is correct?

The plant would be in:

A. region A of the power to flow map. ABN-POWER would be entered and the reactor would be manually scrammed.

B. both the OPRM enabled region and the Area of Increased Awareness. ABN-POWER would be entered and control rods would be inserted per the fast shutdown sequence.

C. the OPRM enabled region but no other on the power to flow map. ABN-CORE would be entered and exit from the region would be accomplished by inserting control rods per the fast shutdown sequence.

D. region A of the power to flow map. ABN-CORE would be entered and the reactor would be manually scrammed.

ANSWER: D

QUESTION TYPE: SRO

KA # & KA VALUE: 295001AA2.01 – Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION: Power/flow map (3.5 3.8) 10CFR55.43.5

REFERENCE: SOP-RRC-START & ABN-CORE

SOURCE: **NEW**

LO: 5022

RATING: L3

ATTACHMENT: YES - SOP-RRC-START Attachment 6.1 - Two Loop Power/Flow Map

JUSTIFICATION: A and B are incorrect because ABN-POWER does not give any direction for RRC pump runback. B is also incorrect because you are in region A. C is incorrect because ABN-CORE does not direct exiting the region by inserting rods. Also you are not just in the OPRM region. D is correct. The conditions given would leave the plant in region A. ABN-CORE would be entered and a manual scram would be inserted because the OPRMs were inoperable.

Columbia Generating Station is in Hot Shutdown. All systems are operational. The feeder breaker to the HPCS Battery Charger, HPCS-C1-1, then trips open.

Based on these conditions, the CRS should declare…..

A. affected required features inoperable immediately and initiate actions to restore required DC electrical power subsystem to operable status immediately.

B. HPCS system inoperable immediately, verify RCIC operable by administrative means immediately, and restore HPCS system to operable status in 14 days.

C. HPCS inoperable within 2 hours or cooldown to LE 200 °F within 36 hours.

D. HPCS system inoperable immediately and restore HPCS system to operable status within 4 hours.

ANSWER: B

QUESTION TYPE: SRO

KA # & KA VALUE: 295004AA2.04 – Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER: System lineups (3.2 3.3) 10CFR55.43.5

REFERENCE: TS 3.8.4B

SOURCE: **NEW**

LO: 7657

RATING: H2

ATTACHMENT: YES - TS 3.8.4 ; TS 3.8.5; TS 3.5.1; TS 3.5.2

JUSTIFICATION: The plant is in Mode 3. B is correct as it uses TS 3.8.4B for Mode 1, 2 or 3 as basis for answer. B is incorrect because it uses the DC shutdown TS 3.8.5. C is incorrect because it uses the completion time for Div 1 and 2 DC systems from TS 3.8.4. D is incorrect because it uses ECCS Shutdown and HPCS is not a required operable system.

COMMENTS: Revised stem

The plant experienced a transient that has resulted in the following plant conditions:

Drywell temperature is 310 degrees
All control rods are fully inserted
Reactor pressure is 50 psig
Narrow Range indicates +7 inches and has been erratic for the last 30 minutes
Shutdown Flooding Range indicates +7 inches
Upset Range indicates +7 inches
All other level indications have been judged unreliable

Based on the above, determine the correct level indication and procedure action for these conditions.

A. RPV Level should be considered unknown and PPM 5.1.4, RPV Flooding, should be entered.

B. The Upset Range indication should be used and level should be recovered using PPM 5.1.1, RPV Control.

C. The Shutdown Flooding Range indication should be used and RPV Level should be recovered using PPM 5.1.1, RPV Control.

D. Narrow Range indication should be used and RPV level should be recovered using PPM 5.1.1, RPV Control.

ANSWER: B Post Exam Comment – Upon grading the exams it was determined A is the correct answer instead of B. This was an administrative error made during the construction of the final exam.

QUESTION TYPE: SRO

KA # & KA VALUE: 295028 EA2.03 Ability to determine and/or interpret the following as they apply to HIGH DRYWELL TEMPERATURE: Reactor water level (3.9) 10CFR55.43.5

REFERENCE: Reference: PPM 5.0.10 EOP Figure A

SOURCE: Bank

LO: 8456

RATING: H3

ATTACHMENT: EOP Caution 1 and Figure A

JUSTIFICATION: RPV Pressure vs Drywell temperature is outside the saturation limit shown in Figure A making any erratic indicator unusable (Answer D) Answer B and C are incorrect because the indications do not meet the minimum usable levels per Caution 1. Therefore RPV Level cannot be determined and RPV

Flooding is required,. Answer A.

COMMENTS:

The plant is in MODE 3 with the scram reset and RHR-P-2B in Shutdown Cooling with the following conditions:

RHR-P-2A	inoperable
Reactor water level	+65 inches and stable
RRC-P-1A	in operation at 15 Hz

SW-P-1B then trips and will not restart.

If this condition exists for an extended period of time, which of the following statements is correct?

A. Due to lowering RPV level, PPM 5.1.1 RPV Control would be entered to re-establish adequate core cooling.

B. Due to rising drywell temperature, PPM 5.2.1 Primary Containment Control would be entered to lower drywell temperature.

C. Due to rising reactor pressure, ABN-RHR-SDC-LOSS would be entered to re-establish shutdown cooling.

D. Due to trip of RRC-P-1A on high motor temperature, ABN-RRC-LOSS would be entered to re-establish forced core flow.

ANSWER: C

QUESTION TYPE: SRO

KA # & KA VALUE: 295021AA2.06 – Ability to determine and /or interpret the following as they apply to LOSS OF SHUTDOWN COOLING: Reactor pressure (3.2 3.3) 10CFR55.43.5

REFERENCE: ABN-RHR-SDC-LOSS

SOURCE: **NEW**

LO: 5780 b

RATING: H3

ATTACHMENT: None

JUSTIFICATION: A is incorrect because the level will go up due of heat up. B is incorrect because the loss of SW has no effect on PC temperature. C is the correct answer because the loss of cooling will cause reactor temperature and pressure to increase until Shutdown Cooling would isolate at 125 psig. D is incorrect because the RRC pumps do not trip on high temperature.

COMMENTS:

The plant was operating at 98% power when a transient occurred that resulted in a Drywell Floor Downcomer sheared off 6 inches below the Drywell Floor. The following conditions exist:

Drywell pressure	32 psig and stable
RPV Level	-155 inches and down slow
Wetwell Level	29 feet and down slow
2 Control Rods	Not fully inserted
ARM-RIS-13 HPCS Pump Room	Pegged high at 10E4

An Emergency Depressurization shall be directed per…

A. PPM 5.2.1, Primary Containment Control

B. PPM 5.1.1, RPV Control

C. PPM 5.1.2, RPV Control - ATWS

D. PPM 5.3.1, Secondary Containment Control

ANSWER: A

QUESTION TYPE: SRO

KA # & KA VALUE: 295024EA2.04 – Ability to determine and/or interpret the following as they apply to HIGH DRYWELL PRESSURE: Suppression Chamber Pressure. (3.9 3.9)
10CFR55.43.5

REFERENCE: PPM 5.2.1

SOURCE: **NEW**

LO: 8341 8340

RATING: H4

ATTACHMENT: YES - PSP Curve, PPM 5.3.1 table 24 and section S

JUSTIFICATION: Due to the downcomer failure, Suppression Chamber Pressure and Drywell Pressure are equal and an ED is required because the PSP curve has been exceeded. This makes A correct. B and C are both incorrect because neither of these procedures requires an ED above TAF. D is incorrect because 5.3.1 requires that there be 2 areas above MSOV prior to ED.

COMMENTS:

Columbia was operating at 96% power when an unplanned Reactor Feedwater transient occurred. The following conditions now exist:

Reactor level	+21 inches and up slow
Reactor pressure	1048 psig and up fast
Reactor power	27% and stable
All white scram group lights	**NOT** illuminated
MSIVs	Closed
Drywell pressure	1.58 psig and up
Suppression Pool temperature	85°F and up
Reactor Building pressure	-.11 inches of water

Which of the following procedures should be entered first?

A. ABN-LEVEL

B. ABN-PRESSURE

C. PPM 5.1.1. RPV Control

D. PPM 5.1.2, RPV Control ATWS

ANSWER: C

QUESTION TYPE: SRO
KA # & KA VALUE: 295037 2.4.6 SCRAM Condition Present and Power above APRM Downscale or Unknown: Knowledge of symptom based EOP mitigation strategies. (3.1 / 4.0) 10CFR55.43.6
REFERENCE: PPM 5.1.1 RPV Control, PPM 5.0.10 page 100
SOURCE: **NEW**
LO: 8017
RATING: L3
ATTACHMENT: None
JUSTIFICATION: C is the correct answer because with the white scram group lights out, there is a scram signal present. With power at 27%, not all control rods inserted. This requires an entry into PPM 5.1.1 RPV Control prior to the entry into PPM 5.1.2 RPV Control ATWS. The SRO must make a choice under these conditions as to which procedure to enter. Since EOPs take precedence over ABNs, the correct choice would be to enter the correct EOP even though both ABN-LEVEL and ABN-PRESSURE have entry conditions..
COMMENTS:

Columbia is operating at 99% power. A failure occurs in the Radwaste Building resulting in a spill of a large amount of resin from a RWCU Demineralizer 45 minutes ago.

Reactor Power	99%
WEA-RIS-14 Rad Waste Bldg. Exhaust, Low	1.8E6 cpm

Based on the above conditions, the CRS should enter…

A. PPM 13.1.1 and PPM 5.4.1, perform a Site Evacuation, and evacuate the Columbia River, Horn Rapids ORV Park, Ringold Fishing Area, Wahluke Hunting Area, and Schools in EPZ.

B. PPM 13.1.1 and PPM 5.4.1, perform a Site Evacuation, and evacuate all sections 0-2 miles and 10 miles downwind, and shelter remaining sections.

C. PPM 5.4.1 concurrently with PPM 5.1.1 and manually scram the reactor.

D. PPM 5.4.1 and Emergency Depressurize the reactor.

ANSWER: A

QUESTION TYPE: SRO

KA # & KA VALUE: 295038 2.4.44 – High Offsite release rate: Knowledge of the Emergency Plan Protective Action Recommendations. (2.1 / 4.0) 10CFR55.43.5

REFERENCE: PPM 13.2.2 rev. 15, PPM 13.1.1, rev. 34

SOURCE: **NEW**

LO: 8893

RATING: H4

ATTACHMENT: YES - PPM 5.4.1, rev. 12 with entry conditions, and PPM 13.1.1, rev. 34. table 3

JUSTIFICATION: A is correct because the conditions given meet the requirements for a SAE and the actions are the automatic PARS for that EAL. B is incorrect because these actions are for a GE, which has not been reached. C and D are both incorrect because there is no primary system discharging outside of the plant.

COMMENTS:

Following an Emergency Depressurization due to a coolant leak, the following conditions exist:

Drywell Temperature	275 °F	Reactor Pressure	25 psig
Drywell Pressure	27 psig	RPV Level	–162 inches and stable
Wetwell Pressure	22 psig	RHR-P-2A and LPCS-P-1	Injecting
Wetwell Level	35 ft	SM-8	Locked out

Based on the above plant parameters, the CRS should enter……

A. PPM 5.2.1, Primary Containment Control, and spray the Drywell regardless of adequate core cooling.

B. PPM 5.1.1, RPV Control, determine PC Flooding is required and exit to Severe Action Guidelines (SAGs).

C. PPM 5.1.1, RPV Control, and monitor Reactor Level instruments for erroneous/erratic indications.

D. PPM 5.2.1, Primary Containment Control, and lower Suppression Pool Level to LT +2 inches utilizing SOP-RHR-SPC.

ANSWER: C

QUESTION TYPE: SRO

KA # & KA VALUE: 295012 2.1.25 High Drywell Temperature: Ability to obtain and interpret station reference materials such as graphs, monographs, and tables with contain performance data (2.8 / 3.1) 10CFR55.43.5

REFERENCE: PPM 5.0.10 rev. 9, PPM 5.2.1 rev. 16
SOURCE: **NEW**
LO: 4104
RATING: H3

ATTACHMENT: YES - PPM 5.2.1 – Primary Containment Control EOP Flowchart, RPV Saturation Temperature Curve A, PCPL Curve B, P-8, P-9, P-11, P-13 and P-14 of the PC Pressure Leg, L1 on WW level leg.

JUSTIFICATION: A incorrect - conditions do not exist which require DW sprays regardless of adequate core cooling. B incorrect - conditions do not exist which require venting the Primary Containment. C correct - the combination of DW pressure and low reactor pressure have resulted in an entry into the Sat Curve. D incorrect because the valve lineup for lowering suppression pool isolated at 1.68 psig DW pressure.
COMMENTS: Revised stem

During a reactor startup with power at 28% a rod drop accident causes a power spike and has resulted in the following plant parameters:

Reactor pressure	990 psig	Reactor power	31%
Reactor level	36 inches	MAPRAT	0.68
MCPR	1.01	LHGR	0.27

Based on given conditions, which of the following is correct?

A. Insert all operable control rods within two hours.

B. Adjust the APRM gain within six hours.

C. Verify control rod separation criteria are met and disarm the associated Control Rod drive within two hours.

D. Restore MCPR to within the limits in two hours and reduce thermal power to LT 25% RTP within 4 hours.

ANSWER: A

QUESTION TYPE: SRO

KA # & KA VALUE: 295014 AA2.05 ability to determine and/or interpret the following as they apply to INADVERTENT REACTIVITY ADDITION: Violation of a Safety Limit IMP 4.6
10CFR55.43.2

REFERENCE: Tech Spec 2.1, 3.1.3, 3.2.2, 3.2.4

SOURCE: **NEW**

LO: 10304

RATING: H2

ATTACHMENT: YES - TS 3.1.3, 3.2.2, 3.2.4

JUSTIFICATION: A is correct because the MCPR safety limit has been violated. B is incorrect because LHGR (MFLPD) is LT the FRTP. C is incorrect because the action is for a stuck rod. D is incorrect because one or the other conditions would be performed, not both.

COMMENTS:

Columbia was operating at full power when a transient occurred. All required EOP actions have been completed up to this point in the event. The following plant conditions now exist:

RPV Water Level is –140 inches and steady
RPV Pressure is 200 psig and down slow
Drywell Pressure is 95 psig and up slow
Wetwell Pressure is 91 psig and up slow
Wetwell Level is 46 feet and steady
Wetwell Temperature is 230°F and down slow

Which of the following describes the next action the CRS should take based on the above conditions?

A. Direct performance of PPM 5.5.14 which would preclude failure of the containment and subsequent loss of systems required to maintain adequate core cooling.

B. Direct performance of PPM 5.5.15 which prevents exceeding 1 rem TEDE at the site boundary during the release.

C. Direct Emergency Depressurization per PPM 5.1.3 which would preclude the failure of the SRV Tailpipe and subsequent loss of Pressure Suppression function of the wetwell.

D. Direct Emergency Depressurization per PPM 5.1.3 which would preclude failure of containment by assuring that RPV blowdown does not cause PCPL to be exceeded.

ANSWER: A

QUESTION TYPE: SRO

KA # & KA VALUE: 295029 EA2.01 Ability to determine and/or interpret the following as they apply to HIGH SUPPRESSION POOL WATER LEVEL: Suppression pool water level (3.9/3.9) 10CFR55.43.5

REFERENCE: PPM 5.0.10

SOURCE: **NEW**

LO: 8040

RATING: H3

ATTACHMENT: PPM 5.2.1 PC Pressure leg steps P12, P13, P14 & PCPL Curve; L-13 & PSP Curve, WT-5 & HCTL Curve

JUSTIFICATION: PPM 5.5.14 would be performed due to WW Level. 5.0.10 defines the PCPL as the limit used to preclude containment failure and subsequent loss of the ability to maintain adequate core cooling. A is correct. B is incorrect because 5.5.15 is performed if WW/L is GT 51’. C is incorrect because emergency depressurization would have already been performed. D is incorrect because ED due to HCTL is not required.

COMMENTS:

The reactor was operating at 92% power with HPCS-P-1 in operation in Full Flow Test mode, Suppression Pool to Suppression Pool. A transient has occurred which resulted in a scram and the following conditions:

Reactor Building Exhaust Plenum	12 mr/hr and stable
Wetwell Level	-3 inches and down slow
Reactor Level	22 inches and down slow
Reactor Pressure	1048 psig and up slow
Control Rod 30-31	Position 24
Control Rod 15-47	Position 08

Which of the following is correct concerning these conditions?

A. HPCS-V-15 remains open, PPM 5.3.1 Secondary Containment Control and PPM 5.1.2 RPV Control ATWS are entered.

B. HPCS-V-15 closes, PPM 5.2.1 Primary Containment Control is entered, and SOP-HPCS-CST/SP is utilized for Suppression Pool level control.

C. HPCS-V-15 closes, PPM 5.3.1 Secondary Containment Control is entered and PPM 5.1.2 RPV Control ATWS are entered.

D. HPCS-V-15 remains open, PPM 5.2.1 Primary Containment Control is entered, and PPM 5.5.23 Emergency Suppression Pool Makeup is utilized for Suppression Pool level control.

ANSWER: D

QUESTION TYPE: SRO

KA # & KA VALUE: 209002A2.11 Ability to (a) predict the impacts of the following on the HIGH PRESSURE CORE SPRAY SYSTEM (HPCS); and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Low suppression pool level (3.3 3.5) 10CFR55.43.5

REFERENCE: SD000174 rev. 10 page 10 and PPM 5.2.1

SOURCE: **NEW**

LO: 8017, 5429
RATING: H3

ATTACHMENT: NONE

JUSTIFICATION: A is incorrect because there are no entry conditions for PPM 5.3.1. B is incorrect because HPCS-V-15 remains open and SOP-HPCS-CST/SP is incorrect. C is incorrect because HPCS-V-15 remains open and there are no entry conditions for PPM5.3.1. D is correct because there is no low level interlock to close HPCS-V-15 and the entry for PPM 5.2.1 on SP level is given. PPM 5.5.23 is used to refill the SP per PPM 5.2.1.

COMMENTS:

Columbia is operating at 65% power. The last performance of the weekly RPS Manual Scram Channel Functional Test Surveillance was completed at 1200 on October 21st. It was discovered at 1000 on October 30th that the next performance of this surveillance had not yet been completed.

Select the statement below which correctly describes the actions which must be taken based on the above condition.

A. The missed surveillance must be completed by 0600 on October 31st or be in MODE 3 within 12 hours.

B. The completion of the surveillance, if started immediately, is within Technical Specification time requirements.

C. Manage the risk impact and complete the missed surveillance by 1000 on November 6th.

D. The missed surveillance has resulted in Columbia having to be in MODE 3 within 12 hours from the time of discovery.

ANSWER: C

QUESTION TYPE: SRO

KA # & KA VALUE: 212000 A2.03 Ability to (a) predict the impacts of the following on the REACTOR PROTECTION SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Surveillance Testing (3.3 3.5) 10CFR55.43.3

REFERENCE: TS 3.3.1.1

SOURCE: **NEW**

LO: 10301

RATING: H3

ATTACHMENT: TS 3.3.1.1 including table ``3.3.1.1-1 and SR 3.0.3

JUSTIFICATION: B is incorrect – SR 3.0.2 allows 1.25 times 7 days from last performance which would be 0600 on October 30 (8 days and 18 hours) – surveillance is late. A is based on the 24 hours from recognition of a missed surveillance and is incorrect because if the risk is managed the surveillance can go longer than Oct. 31st . C is correct per SR 3.0.3 which has been changed to allow 24 hours or surveillance frequency if an risk impact is performed. D is incorrect as it does not take into account TS 3.0.3.

COMMENTS:

The plant was operating at 90% power when a seismic event caused the following:

Reactor level	-25 inches and stable
Scram Group Lights	not illuminated
Blue Scram lights	27 are illuminated
SLC-P-1A and SLC-P-1B	loss of power indicated
Main Generator	undergoing oscillations from 450 Mwe to 1100 Mwe

Based on these conditions, the CRS enters…

A. PPM 5.1.2 and directs boron injection with RCIC.

B. PPM 5.1.2 and directs the closure of RCIC-V-1 to prevent a Main Turbine trip.

C. ABN-POWER and directs the start of both RRC pumps at 15 Hz to stop the Main Generator oscillations.

D. ABN-POWER and directs that control rods be inserted in reverse order of the fast shutdown sequence.

ANSWER: A

QUESTION TYPE: SRO

KA # & KA VALUE: 217000 2.1.20 – RCIC - Ability to execute procedural steps (4.3 4.2)
10CFR55.43.5

REFERENCE: PPM 5.1.2, rev. 17, step Q-11.

SOURCE: **NEW**

LO: 11145

RATING: H2

ATTACHMENT: YES - Q10 through Q14 of PPM 5.1.2

JUSTIFICATION: A is correct as required by PPM 5.1.2 step Q-14. MG Oscillations are in excess of 25% thermal power. B is incorrect because PPM 5.1.2 directs the use of RCIC for boron injection. C and D are both incorrect because PPM 5.1.2 takes precedent over any direction in ABN-POWER under these conditions.

COMMENTS:

The plant was operating at 95% power when MS-RV-5B failed in the open position and could not be closed. Suppression Pool temperature has reached 108°F and is trending up slowly.

Based on the above conditions, the CRS should enter ___________ in order to _________ .

A. ABN-SRV and immediately reduce RRC flow to 60 mlbm/hr; limit the reactor pressure/power transient associated with the SRV closure.

B. ABN-SPC and place two loops of RHR Suppression Pool Cooling in service; limit the rate of Suppression Pool heatup.

C. PPM 5.1.1 RPV Control and place the Mode Switch in SHUTDOWN; comply with Technical Specifications.

D. PPM 5.2.1 Primary Containment Control and initiate an Emergency Depressurization; prevent exceeding the Heat Capacity Temperature Limit.

ANSWER: C

QUESTION TYPE: SRO

KA # & KA VALUE: 239002A2.03 Ability to (a) predict the impacts of the following on the RELIEF/SAFETY VALVES; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: stuck open relief valve (4.1 4.2) 10CFR55.43.5

REFERENCE: PPM 5.2.1 rev. 16 WW temp leg; ABN-SRV; PPM 5.0.10

SOURCE: **NEW**

LO: 8300

RATING: L2

ATTACHMENT: NONE

JUSTIFICATION: C is correct because PPM 5.2.1 block WT-4 requires entry into PPM 5.1.1 before WW temp reaches 110°F. A is incorrect because ABN-SRV directs that action as subsequent actions, not immediate actions and power is reduced to 90% not flow to 60 Mlbm/hr. B is incorrect because there is no procedure ABN-SPC. D is incorrect because there is no ED required until HCTL is exceeded for temperature.

COMMENTS: Revised stem and distractors

The plant is operating at 96% power when a broken coupling is discovered on SW-P-1B.

The CRS is required to declare SW-P-1B inoperable and…

A. DG-2 inoperable immediately.

B. prevent DG-2 start immediately.

C. its associated ECCS pumps inoperable immediately.

D. run SW-P-1A immediately to determine its operability.

ANSWER: A

QUESTION TYPE: SRO

KA # & KA VALUE: 262001 2.1.11 Knowledge of less than one hour technical specification actions statements for systems: AC Electrical Distribution (3.0 3.8) 10CFR55.43.2

REFERENCE: TS 3.7.1 and TS 3.8.1

SOURCE: **NEW**

LO: 9414

RATING: H2

ATTACHMENT: NONE

JUSTIFICATION: A is correct because TS 3.7.1 directs the cascade and TS 3.8.1 applicability requires the DG inoperability. B is incorrect because DG-3 is not associated with SW-P-1B. C is incorrect because TS 3.7.1 make no direction for considering the ECCS pumps. D is incorrect because, while a “common cause” determination is required there is no immediate requirement for a SW-P-1A run.

COMMENTS:

The plant was operating at 99% power when a failure of TR-N2 caused the loss of both SH-5 and SH-6.

Which of the following actions is correct for this condition?

Enter…

A. ABN-POWER, verifies operation in Region A prior to scramming the reactor.

B. ABN-RRC-LOSS, verifies operation in Region A prior to scramming the reactor.

C. ABN-POWER and immediately scram the reactor.

D. ABN-RRC-LOSS and immediately scram the reactor.

ANSWER: D

QUESTION TYPE: SRO

KA # & KA VALUE: 202001A2.04 A Ability to (a) predict the impacts of the following on the RECIRCULATION SYSTEM (HPCS); and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Multiple Recirculation Pump trips. (3.7 3.8) 10CFR55.43.5

REFERENCE: ABN-RRC-LOSS rev. 1, immediate actions

SOURCE: **NEW**

LO: 6733

RATING: H2

ATTACHMENT: NONE

JUSTIFICATION: The immediate actions for ABN-RRC-LOSS state that the plant must be scrammed if both RRC pumps trip in Modes 1 or 2. D is the only correct answer.

COMMENTS:

The plant is at 32% power with APRM F out of service and a peripheral control rod selected on the rod select matrix. APRM B then fails upscale.

Which of the following is correct?

RBM-B is inoperable,…

A. and must be restored to operable status within 24 hours.

B. and must be placed in the trip condition within 1 hour.

C. but is not required by Tech Specs until reactor power exceeds 35%.

D. but is not required by Tech Specs because a peripheral control rod is selected.

ANSWER: D

QUESTION TYPE: SRO

KA # & KA VALUE: 215002 2.1.12 – Ability to apply Tech Specs for a system. (2.9 4.0) 10CFR55.43.2

REFERENCE: Tech Spec 3.3.2.1 Am 169

SOURCE: **NEW**

LO: 5701

RATING: H2

ATTACHMENT: YES – Tech Spec 3.3.2.1 and table 3.3.2.1-1

JUSTIFICATION: RBM operability is required by TS anytime rector power is GE 30% unless a peripheral control rod is selected. As stated in the stem, a peripheral control rod is selected which does not require RBM operability. D is correct.

COMMENTS:

With Columbia operating at 100% power, a leak in the Main Condenser has caused a reactor water chlorides to reach 250 ppb (.25 ppm).

Select the statement that correctly describes the actions to be taken for the above condition.

A. Restore conductivity to within limits within 72 hours.

B. Be in MODE 3 in 12 hours and MODE 4 in 36 hours.

C. Perform an orderly unit shutdown and be in cold shutdown as rapidly as operating conditions permit.

D. If chlorides not below 200 ppb within 6 hours, reduce core flow to 60 Mlbm/hr and SCRAM the reactor per PPM 3.3.1.

ANSWER: D

QUESTION TYPE: SRO

KA # & KA VALUE: 256000 2.1.34 Ability to maintain primary and secondary plant chemistry within allowable limits (2.3 2.9) 10CFR55.43.5

REFERENCE: SWP-CHE-02 Rev.11 Page 6 and Page 10

SOURCE: **NEW**

LO: 5013

RATING: H3

ATTACHMENT: SWP-CHE-02 Rev.11 Page 1, 6, 7, 10; LCS 1.4.1 Rev. 28 pages 1 thru 4

JUSTIFICATION : If only TS was referenced, A would be correct. A is incorrect but a viable action per LCS 1.4.1 Table 1.4.1-1. B is incorrect but an action per LCS 1.4.1 if the required completion time for condition A is not met. C is incorrect as this action would be required if Action Level 2 was exceeded. Conductivity exceeds Action Level 3 value which require a flow reduction and scram if not below 200 ppb within 6 hours. D is correct.

COMMENTS:

Columbia is operating at 99% power when several crew members become sick and go home four hours prior to the end of their shift. The remaining shift complement consists of 1 Senior Reactor Operator, 2 Reactor Operator's, and 1 Equipment Operator.

Which of the following describes the Technical Specification requirements concerning this situation?

A. The required Senior Reactor Operator, Reactor Operator, and Equipment Operator positions may be vacant for a period not to exceed 4 hours provided action is taken to replace these positions within 2 hours.

B. With less than the required shift complement, action must be taken within 2 hours to replace the required position or be in Mode 2 within 12 hours and Mode 3 within the following 12 hours.

C. The required Senior Reactor Operator, Reactor Operator, and Equipment Operator positions may be vacant for a period not to exceed 2 hours provided immediate action is taken to replace these positions.

D. With less than the required shift complement, action must be taken within 2 hours to replace the required position or immediately take actions to place the reactor in Mode 3.

ANSWER: C

QUESTION TYPE: SRO

KA # & KA VALUE: 2.1.4 Knowledge of shift staffing requirements (2.3 3.4) 10CFR50.43.2

REFERENCE: Tech Spec 5.2.2b

SOURCE: **NEW**

LO: 6071, 6933

RATING: H2

ATTACHMENT: NONE

JUSTIFICATION : Per TS 5.2.2b, C is correct.

COMMENTS:

The plant is in MODE 5 with Refueling activities in progress on the Refuel Floor.

Which of the following is considered a core alteration which would require an SRO on the Refuel Floor?

A. Withdrawal of one SRM with the control switch from the control room.

B. Withdrawal of a control rod from a cell with no fuel.

C. Movement of an irradiated fuel bundle in the Fuel Pool.

D. Reseating of a fuel bundle in the core with the refuel mast.

ANSWER: D

QUESTION TYPE:	SRO
KA # & KA VALUE:	2.2.29 Knowledge of SRO fuel handling responsibilities. (1.6 3.8) 10CFR55.43.7
REFERENCE:	PPM 6.3.5 rev. 10, page 3
SOURCE:	**Bank**, 2002 NRC Exam – slightly changed.
LO:	7699 – For a given refueling operation, determine if the evolution is a Core Alteration.
RATING:	L3
ATTACHMENT:	NONE
JUSTIFICATION:	A , B, and C are all incorrect because they do not meet the Tech Spec/Columbia Procedural definition of a core alteration. D is correct because PPM 6.3.5 specifically states the reseating of a fuel bundle during core verification is a core alteration.
COMMENTS:	Revised stem

A Temporary Modification has just been installed in the plant.

Who signs and dates the "Installation Complete" block on the TMR?

A. Operations Manager

B. Minor Modifications Group Supervisor

C. Design Engineer

D. CRS/Shift Manager

ANSWER: D

QUESTION TYPE: SRO

KA # & KA VALUE: 2.2.11 Knowledge of the process for controlling temporary changes. (2.5 3.4)
10CFR55.43.3

REFERENCE: PPM 1.3.9 Rev. 39 Step 3.2.4

SOURCE: **NEW**

LO: 8628 SRO only

RATING: L3

ATTACHMENT: NONE

JUSTIFICATION: PPM 1.3.9 Temporary Modifications states the CRS/Shift Manager signs the "Installation Complete" block. D is the correct answer.

COMMENTS:

Who is responsible to perform the final review and approval of a Planned Special Exposure.

A. Radiation Protection Manager

B. Plant General Manager

C. Operations Manager

D. Shift Manager

ANSWER: B

QUESTION TYPE: SRO

KA # & KA VALUE: 2.3.2 Knowledge of facility ALARA program. (2.5 2.9) 10CFR55.43.4

REFERENCE: GEN-RPP-08 Rev. 1 page 3

SOURCE: **BANK** LO00257 – 2000 NRC exam slightly modified

LO: 11258

RATING: H2

ATTACHMENT: None

JUSTIFICATION: Per GEN-RPP-08 the Plant General Manager has final review/approval. B is correct.

COMMENTS: Revised stem

The plant is operating at 50% power following a forced outage. A batch of nonradioactive RCC water has to be discharged following maintenance on the system. Sample results confirmed no identifiable activity other than naturally occurring isotopes.

Who authorizes the release of this RCC water?

A. Radiation Protection Manager

B. Operations Manager

C. Chemistry Manager

D. CRS/Shift Manager

ANSWER: D

QUESTION TYPE: SRO

KA # & KA VALUE: 2.3.6 Knowledge of the requirements for reviewing and approving release permits.
(2.1 3.1) 10CFR50.43.4

REFERENCE: PPM 12.2.14 R4 Page 4

SOURCE: **Bank** – 2001 NRC Exam – slightly modified

LO: 11260

RATING: L4

ATTACHMENT: NONE

JUSTIFICATION: Per PPM 12.2.14, the CRS/Shift Manager approves the release.
COMMENTS:

Columbia is operating at 80% power. A surveillance concurrent with an instrument failure causes the HPCS system to inject to the RPV. Injection is secured by overriding HPCS-V-4, the HPCS injection valve, closed and stopping HPCS-P-1.

Which of the following is true in regards to NRC reportability?

This would be a/an…

A. 4 hour report for ECCS injection into the RPV.

B. 4 hour report for Tech Spec required shutdown.

C. 8 hour report for valid actuation of a system.

D. 8 hour report for single train inoperable.

ANSWER: D

QUESTION TYPE: SRO

KA # & KA VALUE: 2.4.30 Knowledge of which events related to system operation/status should be reported to outside agencies. (2.2 3.6) 10CFR55.43.5

REFERENCE: PPM 1.10.1 Rev. 27 Pages 9 - 12, NUREG 1022 3.2.6

SOURCE: **NEW**

LO: 6011

RATING: H3

ATTACHMENT: PPM 1.10.1 rev. 27, page 9 – 12 ; NUREG-1022 Page 45 for 3.2.6

JUSTIFICATION: A and C are incorrect because this condition is not a valid initiation signal. B is incorrect because this situation does not require a TS shutdown. D is correct because HPCS is a single train which is now unable to perform its safety function.

COMMENTS:

A LOCA has occurred that resulted in the following conditions:

Reactor level	-138 inches and stable on the Compensated Fuel Zone
Reactor level	off scale low on the Wide Range
Reactor Pressure	105 psig and stable
Wetwell temperature	199°F and up slow
Wetwell level	GT 51 feet
Wetwell pressure	91 psig and up fast
Offsite dose rate	9 mrem/hr TEDE and 5 mrem/hr CEDE

Which of the following is correct concerning these conditions?

A. Enter PPM 5.4.1, Radioactivity Release Control. The Reactor should be emergency depressurized because the Offsite Release has exceeded the Alert Classification.

B. Enter PPM 5.1.1, RPV Control. The Reactor should be emergency depressurized because the HCTL has been exceeded.

C. Enter PPM 5.2.1, Primary Containment Control. Containment should be vented through the drywell, regardless of offsite release rate, to prevent the loss of systems required for adequate core cooling.

D. Enter PPM 5.2.1, Primary Containment Control. Containment should be vented through the wetwell, regardless of offsite release rate, to prevent the loss of systems required for adequate core cooling.

ANSWER: C

QUESTION TYPE: SRO

KA # & KA VALUE: 2.4.22 Knowledge of the bases for prioritizing safety functions during abnormal/emergency conditions. (3.0 4.0) 10CFR55.43.5

REFERENCE: PPM 5.0.10 rev. 9, pages 89, 268, & 269

SOURCE: **NEW**

LO: 11229

RATING: H3

ATTACHMENT: Yes – PCPL Curve; PPM 5.2.1 P-13 & P-14; HCTL Curve; PPM 5.4.1 with entry conditions

JUSTIFICATION: A and B are both incorrect because neither has exceeded the limits. C is incorrect because you are directed to vent the drywell with wetwell level GT 51 feet.