Question Number: 001
Exam Date 20090413
Exam Level R
K/A 007EA1.04
Question
The plant was operating at 100%, MOL when RXCP 'A' #1 seal leakoff increased to 9 gpm. The reactor was subsequently tripped and RXCP 'A' was stopped and placed in PULLOUT

Which of the following best reflects the expected values for Loop 'A' Flow, Loop 'B' Flow, and Loop 'B' Bypass Flow, respectively, ten minutes later?

a. 18% 108% 300 gpm

b. 40% 78% 175 gpm

c. 0% 98% 250 gpm

d. 25% 85% 150 gpm

Answer
a.
Reference:
AOP-RC-005, Abnormal RXCP Operation, NOTE 2 Step 1
System Description 36, Reactor Coolant System (RC), 3.5.4, page 25
Accident and Transient Analysis, Chapter VI Decreased Reactor Coolant Flow Events, Partial loss of forced flow. Figure VI.13 & VI.14.
NEW
HIGHER
K/A: Ability to operate and/or monitor the following as they apply to a reactor trip: RCP operation and flow rates. RO Imp / SRO Imp - 3.6 / 3.7
JUSTIFICATION:
a. Correct. Flow in the idle loop will reverse direction with the operating loop providing the driving force. The RCS loop flow sensors work on the principle of DP between the inside and outside diameters. With flow in the reverse direction the DP between inside and outside diameters will remain in the same direction but the absolute value will be less. Flow in the operating loop will increase above normal due to the additional flow being provided in the reverse direction through the idle loop. Flow increases in the operating loop because discharge head decreases on the operating RXCP. Therefore both Loop B and Loop B Bypass flow indications will increase above normal full power values.
b. Wrong. The RCS loop flow sensors work on the principle of DP between the inside and outside diameters. With flow in the reverse direction the DP between inside and outside diameters will remain in the same direction but the absolute value will be less. Loop A flow at 40% is a plausible distractor because with RXCP A idled in the loop one might incorrectly assume that Loop A flow would decrease to zero than reverse flow would be close to half of normal flow. Flow in the operating loop will increase above normal due to the additional flow being provided in the reverse direction through the idle loop. Flow increases in the operating loop because discharge head decreases on the operating RXCP. Therefore both Loop B and Loop B Bypass flow indications will increase above normal full power values. Loop B and Loop B Bypass flow indications are as expected given the specified plant condition.
c. Wrong. The RCS loop flow sensors work on the principle of DP between the inside and outside diameters. With flow in the reverse direction the DP between inside and outside

diameters will remain in the same direction but the absolute value will be less. Loop A flow at 0% is a plausible distractor because with RXCP A idled in the loop one might incorrectly assume that Loop A flow would decrease to zero. Loop B and Loop B Bypass flows are plausible distractors because these are values are near normal (slightly less) for full power operations.

d. Wrong. The RCS loop flow sensors work on the principle of DP between the inside and outside diameters. With flow in the reverse direction the DP between inside and outside diameters will remain in the same direction but the absolute value will be less. With flow in the reverse direction the DP between inside and outside diameters will remain in the same direction but the absolute value will be less.

Question Number: 002
Exam Date 20090413
Exam Level R
K/A 008AA1.03
Question
Given the following:

- The plant has just tripped from 100% power.
- Safety Injection has automatically initiated due to a failed OPEN Pressurizer PORV.
- Tavg is 543°F and stable.
- Steam Dumps are in Steam Pressure Mode.
- Condenser backpressure is 2.3" HgA.

Assuming NO operator action, how will steam dump operation be affected by PT-484, Main Steam Line Pressure Transmitter, failing high?

a. All steam dump valves will remain closed.

b. All condenser steam dump valves arm until reset.

c. All condenser steam dump valves open, but reclose when RCS Tavg falls to 540°F.

d. All condenser steam dump valves open, until low steam line pressure causes an automatic Main Steam Line Isolation.

Answer
c.
Reference
E-1626, Integrated Logic Diagram, Main Steam & Steam Dump System
E-1627, Integrated Logic Diagram, Main Steam & Steam Dump System
NEW
HIGHER
K/A: Ability to operate and/or monitor the following as they apply to the Pressurizer Vapor Space Accident: Turbine bypass in manual control to maintain header pressure.
JUSTIFICATION:
a. Wrong. With PT-484 failed high, in the steam pressure mode of operation, MS header pressure will appear to be higher than that manually established using HC-484. It matters not whether HC-484 has been adjusted to maintain 543 ºF or is in its original position as it is the steam pressure signal which controls the position of the steam dump valves in this mode. All condenser steam dump valves will open in response to the failed steam pressure signal. This is a plausible distractor in that, apart from PT-484 having failed high, the steam dumps would most likely be closed under these plant conditions. Particularly if HC-484 hadn't been adjusted to maintain Tavg constant.
b. Wrong. The question specifies that the steam dump affect will be as a result of PT-484 failing high. That said, PT-484 has nothing to do with arming or disarming steam dumps. The arming signal is derived from the rate of change sensed by PT-486. This is a plausible distractor in that steam dump disarming is accomplished by going to reset.
c. Correct. With PT-484 failed high, in the steam pressure mode of operation, MS header pressure will appear to be higher than that manually established using HC-484. It matters not whether HC-484 has been adjusted to maintain 543 ºF or is in its original

position as it is the steam pressure signal which controls the position of the steam dump valves in this mode. All condenser steam dump valves will open in response to the failed steam pressure signal. Once the condenser steam dump valves open, the RCS will begin to cooldown due to the increased steam demand. The steam dumps will remain open until the Lo Lo Tavg setpoint of 450 ºF is reached at which time the arming signal is removed from all condenser steam dumps.

d. Wrong. The steam dumps will remain open until the Lo Lo Tavg setpoint of 540 ºF is reached at which time the arming signal is removed from all condenser steam dumps. This is a plausible distractor because there is an automatic MSIV closure related to Lo-Lo Tavg coincident with SI but it is in conjunction with high steam flow NOT low steam line pressure.

Question Number: 003
Exam Date 20090413
Exam Level R
K/A 009E 2.4.6
Question
Given the following:

- A Small Break LOCA has occurred.
- The crew has transitioned to ES-1.2, "Post LOCA Cooldown and Depressurization".
- Both RXCPs are stopped in PULLOUT.
- A cooldown to Cold Shutdown is in progress.
- Charging Pumps 'A' and 'B' are running at maximum output taking a suction from the RWST.
- Both RHR pumps are stopped and in AUTO.
- SG 'A' Narrow Range level is 13% and slowly increasing.
- SG 'B' Narrow Range level is 21% and slowly increasing.
- Subcooling is 62°F and stable.
- Pressurizer level is 31% and slowly increasing.
- Conditions have been established for starting RXCPs 'A' and 'B'

Starting RXCP(s) …

a. 'B' is preferred to maximize pressurizer spray flow.

b. 'A' and 'B' are preferred to equalize RCS cold leg temperatures.

c. 'A' and 'B' are preferred to maximize forced circulation in the RCS.

d. 'A' is preferred to minimize pressure transients when placing RHR in service.

Answer
a.
Reference
RO4-04-LP019.002 - Summarize the purpose or basis of the following items as they relate to ES-1.2, "Post LOCA Cooldown and Depressurization": All procedure steps.
NEW
FUNDAMENTAL
K/A: Small Break LOCA: Knowledge of EOP mitigation strategies.
Justification:
a. Correct: RXCP 'B' is in the same loop as the pressurizer surge line and spray. It provides the best spray with the heat input of only pump.
b. Wrong: Two RXCPs are not preferred because of the additional heat input.
c. Wrong: Two RXCPs are not preferred because of the additional heat input.
d. Wrong: 'A' is preferred only if RXCP 'B' is not available.

Question Number: 004
Exam Date 20090413
Exam Level R
K/A 011EK2.02
Question
Given the following:

- A Large Break LOCA has occurred.
- The reactor has tripped.
- Safety Injection has initiated.
- RCS pressure is 200 psig and decreasing.

What is the reason for stopping the RxCPs?

a. To maximize safety injection flow.

b. To prevent two phase flow in core.

c. To minimize RCS inventory depletion.

d. To prevent RxCP degradation and/or damage.

Answer
d.
Reference
BKG E-0, Foldout 1, Knowledge/Abilities
BKG E-1, Foldout 1, Knowledge/Abilities
AOP-RC-005, Abnormal RxCP Operation, step 1 #1 RXCP Seal D/P
Fundamental
BANK
K/A Knowledge of the interrelations between the Large Break LOCA and the Following: Pumps.
JUSTIFICATION:
a. Wrong. Stopping RxCPs will have no effect on safety injection flow.
b. Wrong. Development of two phase flow in the core is not a concern for a Large Break LOCA.
c. Wrong. Stopping RxCPs to minimize RCS inventory depletion is applicable for Small Break LOCAs not Large Break LOCAs.
d. Correct. Continuing to run the RxCPs with out of spec parameters (#1 seal D/P) will cause damage to the pump. Wrong. Stopping RxCPs will have no effect on safety injection flow.

Question Number: 005
Exam Date 20090413
Exam Level R
K/A 015/017AK3.02
Question
Thermal barrier flow control valves (CC-610A and CC-610B) automatically close when Reactor Coolant Pump Component Cooling flow rate reaches 260 gpm.

What describes the reason for this?

a. Limit the loss of Component Cooling system inventory.

b. Limit the leakage of reactor coolant into the Component Cooling system.

c. Ensure a continued supply of cooling water to the Excess Letdown heat exchanger.

d. Ensure a continued supply of cooling water to the Reactor Coolant Pump upper oil pot oil cooler.

Answer
b.
Reference
AOP-RC-005, Abnormal RXCP Operation, NOTES prior to Step 10
System Description 31, Component Cooling Water System (CC), 3.6.6, page 16
Fundamental
NEW
K/A Knowledge of the reasons for the following responses as they apply to the Reactor Coolant Pump Malfunctions: CCW lineup and flow paths to RCP oil coolers.
JUSTIFICATION:
a. Wrong. In the event that a RXCP thermal barrier tube ruptures, the much higher pressure of the reactor coolant will flow into the lower pressure CC system. This is a plausible distractor because it addresses inventory control which is ultimately the reason answer C is correct however, this distractor incorrectly reverses the direction of flow.
b. Correct. In the event that a RXCP thermal barrier tube ruptures, the much higher pressure of the reactor coolant will flow into the lower pressure CC system. This will result in a dramatic increase in the flow rate returning from the affected RXCP. This increased flow rate is used as the initiating parameter to close CC-610A/B as appropriate therefore, minimizing the amount of reactor coolant entering the CC system.
c. Wrong. The CC return from the Excess Letdown Heat Exchanger is separate from the CC return from the RXCPs. This is a plausible distractor in that the RXCPs and the Letdown Heat Exchangers are the only two CC loads inside Containment and share a common supply and return header penetration.
d. Wrong. The CC return from the RXCP upper oil pot oil cooler is a separate line from the CC return from the RXCP thermal barrier. If the RXCP thermal barrier return line is isolated by the closing of CC-610A and/or CC-610B the RXCP upper oil pot oil cooler return remains unaffected in any significant way. A minor affect may be a small increase in flow through the RXCP upper oil pot oil cooler. This is a plausible distractor in that CC provides cooling to both of these RXCP components.

Question Number: 006
Exam Date 20090413
Exam Level R
K/A 022AA2.03
Question
Given the following:

- The plant is at full power, steady state, Middle of Life (MOL).
- Tavg is stable.
- Normal charging and letdown are in service.
- The Reactor Makeup Control System is energized and in AUTO.
- VCT level is 20% and increasing with Auto Makeup in progress.

Assuming NO operator action, if power is lost to CVC-403/CV-31092, Boric Acid To Blender, . . .

a. CVC-403/CV-31092, Boric Acid To Blender, will cycle from THROTTLED to CLOSED.

b. Annunciator 47045-L, RX MAKEUP WATER FLOW DEVIATION, will alarm after 10 seconds.

c. Annunciator 47044-L, RX MAKEUP BORIC ACID FLOW DEVIATION, will alarm after 10 seconds.

d. CVC-406/CV-31094, Boric Acid Blender To Volume Control Tank, will cycle from OPEN to CLOSE.

Answer
c.
Reference
E-2024, Integrated Logic Diagram, Chemical & Volume Control System
System Description 35, Chemical and Volume Control System (CVC)
ARP 47044-L, RX MAKEUP BORIC ACID FLOW DEVIATION
ARP 47045-L, RX MAKEUP WATER FLOW DEVIATION
OPERXK-100-36, Flow Diagram, Chemical & Volume Control Sys.
NOP-CVC-001, Boron Concentration Control
HIGHER
NEW
K/A Ability to determine and interpret the following as they apply to the Loss of Reactor Coolant Makeup: Failures of flow control valve or controller.
JUSTIFICATION:
a. Wrong. The fail position for CVC-403, Boric Acid To Blender, is open. This is a plausible distractor if the fail position is not understood.
b. Wrong. CVC-403 failing affects only the Boric Acid flow to the blender. It does not affect the reactor makeup water flow to the blender. This is a plausible distractor in that it is similar in structure to the correct answer and provides balance to the distractors.
c. Correct. VCT level at 20% and increasing indicates that an automatic makeup is in progress. CVC-403 fails open on a loss of power. CVC-403 going open, given MOL boric acid concentrations, would result in a deviation in demanded and actual boric acid flow rate during the makeup in excess of +0.2 gpm. The deviation can reasonably be expected to last for > 10 seconds, therefore annunciator 47044-L, RX MAKEUP BORIC ACID FLOW DEVIATION, will alarm.

d. Wrong. CVC-406, Boric Acid Blender To Volume Control Tank, does not open during an automatic makeup. This is a plausible distractor because if CVC-406 is open at the time that annunciator 47044-L is actuated, it receives a closing signal.

Question Number: 007
Exam Date 20090413
Exam Level R
K/A 025A 2.4.45
Question
Given the following:

- The plant is at 255°F, cooling down to Cold Shutdown with RHR Train 'A'.
- RHR Train 'B' is out of service for testing.
- SI Accumulators are ISOLATED.
- Annunciator 47021-G, "RHR TO RCS COOLDOWN FLOW LOW" is LIT.
- Annunciator 47033-43, "TLA-18 RHR SYSTEM MONITOR ABNORMAL" is LIT.
- Annunciator 47031-R, "REACTOR CAVITY SUMP LEVEL HIGH/LOW" is LIT.
- Annunciator 47043-K, "LETDOWN HX OUTLET PRESS HIGH" is LIT.
- RHR flow is oscillating between 950 gpm and 2000 gpm

The crew's priority is implementation of . . .

a. E-0, "Reactor Trip Or Safety Injection".

b. AOP-RC-001, "Reactor Coolant Leak".

c. ARP-47031-R, "REACTOR CAVITY SUMP LEVEL HIGH/LOW".

d. AOP-RHR-001, "Abnormal Residual Heat Removal System Operation".

Answer
d.
Reference
AOP-RHR-001, Abnormal Residual Heat Removal System Operation
AOP-RC-001, Reactor Coolant Leak
E-0, Reactor Trip Or Safety Injection
ARP-47031-R, REACTOR CAVITY SUMP LEVEL HIGH/LOW
ARP-47021-G, RHR TO RCS COOLDOWN FLOW LOW
ARP-47033-43, TLA-18 RHR SYSTEM MONITOR ABNORMAL
ARP-47043-K, LETDOWN HX OUTLET PRESS HIGH
NEW
HIGHER
K/A Loss of RHR System: Ability to prioritize and interpret the significance of each annunciator or alarm.
JUSTIFICATION:
a. Wrong. The EOP network is not used once the SI accumulators are isolated during shutdown conditions. This is a plausible distractor because the reactor cavity sump level alarm on its own may be a false indicator of an RCS leak.
b. Wrong. While in Intermediate Shutdown with the SI accumulators isolated the appropriate procedure to address RCS leaks is AOP-RHR-002, Shutdown Loss Of Coolant Accident. This is a plausible distractor because the reactor cavity sump level alarm on its own may be a false indicator of an RCS leak.
c. Wrong. ARP-47031-R provides guidance on ensuring that the sump pump is operating properly to restore sump level. Addressing it is a low priority when compared to the potential loss of core cooling. The K/A for this question specifically requires prioritizing and interpreting the significance of annunciators during a loss of RHR condition. This is

a plausible distractor because the reactor cavity sump level alarm on its own may be a false indicator of an RCS leak and because it ensures the primary aspects of the K/A can be accomplished.

d. Correct. Annunciators 47021-G and 47033-43 coupled with RHR flow oscillating provides sufficient indications that entry into AOP-RHR-001 is appropriate and of the highest priority due to the only available RHR pump being at risk and subsequent risk to core cooling.

Question Number: 008
Exam Date 20090413
Exam Level R
K/A 026A 2.2.37
Question
Given the following:

- Plant at 100% Power.
- Component Cooling Water Pump 'A' tripped.
- Component Cooling Water Pump 'B' DID NOT auto start.
- The crew manually started Component Cooling Water Pump 'B' 90 seconds after Component Cooling Water Pump 'A' tripped.

Which components were INOPERABLE when Component Cooling system pressure and flow were zero?

a. SI Pump 'A' & ICS Pump 'A'.

b. RHR Pump 'A' & Charging Pump 'A'.

c. RXCP 'A' & Letdown Heat Exchanger.

d. Component Cooling Pump 'A' & Waste Gas Compressor 'A'.

Answer
a.
Reference
47021-H, CC Pumps Discharge Pressure Low.
47021-I, RXCP CC Low.
47022-I, SI pimps CC Flow Low
47023-I, RHR Pumps CC Flow Low
47024-I, ICS Pumps CC Flow Low
Fundamental
New
K/A Loss of Component Cooling water: Ability to determine operability and/or availability of safety related equipment. RO Imp / SRO Imp - 3.6 / 4.6

JUSTIFICATION:
a. Correct: Both SI and ICS pumps are declared inoperable according to ARPs when they lose component cooling flow.
b. Wrong: Charging Pump ‘A’ operability is not affected by component cooling flow.
c. Wrong: Letdown heat exchanger is not a technical specification item.
d. Wrong: Waste gas compressor 'A' is not a technical specification item.

Question Number: 009
Exam Date 20090413
Exam Level R
K/A 027AA1.01
Question
Given the following:

- The unit is operating at 78% power.
- PRZR pressure initially is 2235 psig and stable.
- PRZR Master Pressure Controller output, PC-431K, has failed AS IS.
- Main Feedwater Pump 'B' has TRIPPED.

What is the initial response of the Pressurizer Pressure Control System after Main Feedwater Pump 'B' has tripped?

a. Backup heaters turn OFF due to the pressure increase.

b. Backup heaters turn ON to heat the incoming surge volume.

c. Both PRZR spray valves THROTTLE OPEN to reduce pressure to normal.

d. Both PRZR PORVs OPEN to maintain pressure below the high reactor trip setpoint.

Answer
b.
Reference
Dwgs E-2038, E-2039
System Description SD-05A, Feedwater System, 1.6, page 5
OP-KW-ARP-47061- D, FEEDWATER PUMP B TRIP, 4.3
KPS USAR, 4.1.5.4 & 4.1.5.5
BANK
HIGHER
K/A Ability to operate and/or monitor the following as they apply to Pressurizer Pressure Control Malfunctions: PRZR heaters, sprays and PORVs.
JUSTIFICATION:
a. Wrong. Output failed as is, no response
b. Correct. +10% deviation will cause heaters to energize regardless of RCS pressure (master controller output)
c. Wrong. Output failed as is, no response
d. Wrong. Runback would occur and RCS Temp/press will increase, but not high enough to actuate both PRZR PORVs (even w/o spray). PR-2A is actuated by the master controller.

Question Number: 010
Exam Date 20090413
Exam Level R
K/A 038EK1.03
Question
Given the following:

- The plant has experienced a Steam Generator Tube Rupture in Steam Generator 'A' and a Loss of Off-Site Power
- The crew has cooled down the RCS to the target CET temperature of 444°F using Steam Generator 'B ' PORV and is depressurizing the RCS using PR-2B, PRZR PORV.
- A RED path on Integrity is indicated because of a low RCS Loop 'A' Cold Leg temperature.

What is the cause for the RED path on Integrity?

a. Re-direction of safety injection flow.

b. Voiding of the upper head region.

c. Backfill from the ruptured steam generator to the RCS.

d. Establishing natural circulation in RCS loop 'A'.

Answer
a.
Reference
E-3, Steam Generator Tube Rupture, Step 11 CAUTION 2
E-3 background for CAUTION 2 step 11
HIGHER
NEW
K/A 038EK1.03 Knowledge of the operational implications of the following concepts as they apply to the SGTR: Natural Circulation.
Justification:
a. Correct - With no RXCPs running, during cooldown with the intact steam generator or RCS depressurization with a Pressurizer PORV, a red or orange path condition could develop because of the re-direction of SI flow in the ruptured loop.
b. Wrong - Voiding of the upper head region is a concern during depressurization with no RXCPs, but the voiding is not the reason for the red path on integrity.
c. Wrong - The temperature of the ruptured SG 'A' water is greater than the 274 degrees required for a red path on integrity.
d. Wrong - Natural circulation in the RCS loop with the ruptured steam generator is never established.

Question Number: 011
Exam Date 20090413
Exam Level R
K/A E12EK2.2
Question
Given the following:

- Operators are performing ECA 2.1, "Uncontrolled Depressurization of Both Steam Generators," due to a steam leak on BOTH S/G main steam lines inside containment.

Parameter	Value	Trend
Cooldown rate	125°F/hour	Stable
RCS Cold Leg 'A' temperatures	360°F	Decreasing
RCS Cold Leg 'B' temperatures	320°F	Decreasing
S/G 'A' Narrow Range level	OFF Scale Low	---------
S/G 'A' Wide Range level	45%	Stable
S/G 'A' pressure	130 psig	Slowly Decreasing
S/G 'A' AFW flow	85 gpm	Stable
S/G 'B' Narrow Range level	10%	Slowly Increasing
S/G 'B' Wide Range level	67%	Slowly Increasing
S/G 'B' pressure	105 psig	Slowly Decreasing
S/G 'B' AFW flow	125 gpm	Stable
Containment pressure	8 psig	Slowly Decreasing

To mitigate the effects of the steam leaks and control the subsequent transient the operator should . . .

a. minimize additional cooldown by decreasing AFW flow to 25 gpm to each Steam Generator.

b. minimize steam flow to containment by isolating AFW flow to Steam Generator 'A' and maintaining at 25 gpm AFW flow to Steam Generator 'B'.

c. prevent Steam Generator 'A' tube dryout by decreasing Steam Generator 'A' AFW flow to 25 gpm and increasing Steam Generator 'B' AFW flow to at least 185 gpm.

d. ensure an adequate heat sink by maintaining total AFW flow between 210 gpm and 250 gpm until at least one Steam Generator narrow range level is greater than 13%.

Answer
a.
REFERENCE
ECA-2.1, Step 4
HIGHER
NEW

K/A Knowledge of the interrelationships between the Uncontrolled depressurization of all Steam Generators and the following: Facility's heat removal systems, including primary coolant, emergency coolant, decay heat removal systems, and relations between the proper operation of these systems to the operation of the facility.

JUSTIFICATION:

a. Correct reduce AFW flow to both Steam Generators to 25 gpm to minimize the cooldown to help mitigate the effects of the steam leaks and control plant.

b. Wrong because AFW flow should not be isolated to Steam Generator 'A' and reducing AFW flow at this point is to minimize the effects of the cooldown.

c. Wrong because 25 gpm is what is required to maintain Steam Generator tubes wet and both S/G's are faulted therefore, 25 gpm each. Normally 210 gpm AFW flow or 5% [13%] narrow range level in one SG is required for adequate heat sink. For uncontrolled depressurization of both steam generators this requirement does not apply.

d. Wrong because maintaining between 210 gpm and 250 gpm is for reducing a cooldown in E-0 when neither Steam Generator Narrow range level is greater than 13% (adverse containment). At this step in ECA-2.1 the operator is directed to maintain at least 25 gpm feed flow to both steam generators if narrow range level in both Steam Generators is not greater than 13%.

Question Number: 012
Exam Date 20090413
Exam Level R
K/A 054AK1.02
Question
Given the following:

- Operators are performing FR-H.1, "Response to Loss of Secondary Heat Sink", due to a loss of Main Feedwater event.
- Feed and bleed has been initiated.
- Main Feedwater is now available.

What is the reason for establishing feed flow in a slow and controlled manner?

a. Limit the RCS temperature decrease.

b. Decrease the time to termination of feed and bleed.

c. Prevent stratification of S/G as cold water is introduced.

d. Minimize potential water hammer in the Main Feed Water System.

Answer
a.
Reference
FR-H.1, CAUTION Step 36
Background FR-H.1, CAUTION Step 36
FUNDAMENTAL
NEW
K/A Knowledge of the operational implications of the following concepts as they apply to Loss of Main Feedwater (MFW): Effects of feedwater introduction on dry S/G.
JUSTIFICATION:
a. Correct. When feedwater flow is restored to a steam generator, high flow rates may cause RCS temperature to rapidly decrease. This is especially applicable when main feedwater is used to restore steam generator inventory. Although rapid cooldown of the RCS can produce temperature-induced stress with the potential for a loss of vessel integrity under high pressure conditions, it is likely that the vessel would have already been subjected to a rapid cooldown prior to reestablishing feedwater flow, due to the injection of relatively cold SI water during bleed and feed. After reestablishing feedwater flow, minimizing the rate of RCS cooldown is more beneficial for operator controllability of the plant (by minimizing coolant shrinkage and pressure transients) than integrity of the vessel.
b. Wrong. Limiting feedwater flow would increase the time to termination of feed and bleed however this is a plausible distractor in that it provides balance to the distractor choices presented. Also, the stem of this question does not state whether the intent of the step is to establish a maximum or a minimum value of 25 gpm feed flow therefore, if this is misunderstood as establishing a minimum of 25 gpm, this could be seen as a valid selection.
c. Wrong. The reason for establishing feedwater flow at 25 gpm is to limit the temperature decrease. This is a plausible distractor because stratification is a concern normally in systems with large volumes of water.
d. Wrong. At this time establishing a heat sink with the steam generators is the priority and

although water hammer may be introduced, the background document does not refer to this as being a concern.

Question Number: 013
Exam Date 20090413
Exam Level R
K/A 055EA1.04
Question
Given the following:

- Station Blackout occurred 2 hours ago.
- Crew is performing the actions of ECA-0.0, "Loss of All AC Power", Step 36 (CAS) MONITOR DC BUS VOLTAGES.

- DC BUS voltages:

  BRA-102 voltage is 120 VDC
  BRB-102 voltage is 119 VDC
  BRC-102 voltage is 120 VDC
  BRD-102 voltage is 119 VDC
  BRE-102 voltage is 205 VDC

Based on the above voltages, what is the appropriate action?

a. Shed loads from BRB-102 and BRD-102.

b. Monitor all individual cell voltages to ensure they are > 1.02 volts.

c. Cross-connect BRA-102 with BRB-102 and BRC-102 with BRD-102.

d. Report when generator seal oil pressure within 5 psig of generator gas pressure.

Answer
d.
Reference
ECA-0.0, Step 36
NEW
HIGHER
K/A Ability to operate and/or monitor the following as they apply to a Station Blackout: Reduction of loads on the battery.
JUSTIFICATION:
a. Wrong because direction to load shed is when bus voltage reaches 117 VDC
b. Wrong taking individual cell voltages is not directed in the procedure, but it will inform the operators of the condition of the batteries.
c. Wrong because the procedure does not direct cross connecting DC Buses and is not permitted when equipment powered from these buses are required to be operable.
d. Correct because when BRE-102 VDC is not greater than 220 VDC than generator seal oil pressure is monitored to ensure hydrogen does not escape the generator

Question Number: 014
Exam Date 20090413
Exam Level R
K/A 056AK3.02
Question
Given the following:

- SI Pump 'A' is OOS for maintenance.
- The plant was operating at 100% power when a Loss of Offsite Power occurred.
- The crew is performing the Immediate Actions of E-0, "Reactor Trip or Safety Injection".
- Bus 5 and 6 are NOT energized.
- Annunciator 47091-G, "BUS 5 LOCKOUT", is LIT.
- Annunciator 47091-J, "BUS 6 LOCKOUT", is NOT LIT.
- Emergency Diesel Generator 'A' is running unloaded.

Per the Immediate Actions of E-0, "Reactor Trip or Safety Injection", the Balance of Plant Operator should first . . .

a. manually start Emergency Diesel Generator ‘B’ to provide a power source for a train of Safeguards equipment.

b. place Emergency Diesel Generator 'B' output breaker 43 switch to manual in preparation for manual start of Emergency Diesel Generator 'B'.

c. place Emergency Diesel Generator 'A' in PULLOUT to prevent damage to the Emergency Diesel Generator due to running without Service Water.

d. close SW pump 1A2 breaker prior to closing Emergency Diesel Generator 'A' output breaker to provide Service Water to Emergency Diesel Generator 'A'.

Answer
a.
Reference
E-0, Step 3 (Immediate Operator Action)
E-0 Background, Step 3
NEW
HIGHER
K/A Knowledge of the reasons as they apply to the loss of Offsite Power: Actions contained in EOP for loss of offsite power. RO Imp / SRO Imp - 4.4 / 4.7
Justification:
a. Correct: No Train 'B' Safeguard equipment is OOS, thus Bus 6 should be restored to have one complete train of Safeguard equipment available.
b. Wrong: The operator should manually start the Emergency Diesel Generator 'B' first and see if the EDG will automatically load on the bus. By placing the 43 switch to manual first, could delay restoration of power and safeguards equipment.
c. Wrong: Emergency Diesel Generator 'A' output breaker should be closed if the Emergency Diesel Generator is running and the bus is not faulted.
d. Wrong: Closing SW pump 1A2 breaker prior to energizing bus by holding the breaker closed for 5 seconds is required when re-energizing Bus 5 in ECA-0.0 but not in E-0

Question Number: 015
Exam Date 20090413
Exam Level R
K/A 057 2.1.19
Question
Using the attached D/T PARAMETER and TAVE trend snapshots from the PPCS, identify the event that results in the displayed parameters

a. Loss of power to 480V MCC 1-62C.

b. Loss of power to 125V DC Bus BRB-104.

c. Loss of power to 120V AC Bus BRB-113.

d. Loss of power to 120V AC cabinet BRB-105.

Answer
c.
Reference
Plant Process Computer System
E-233 Rev AR
E-2042 Rev J (B-1)
E-2044 Rev. U (H-6)
HIGHER
NEW
Provided Reference: D/T parameter and Tave Trend Snapshots.
K/A: Loss of Instrument Bus: Ability to use plant computers to evaluate system or component status.
JUSTIFICATION:
a. Wrong. MCC 62C is the normal source of power to Instrument Bus 2 through Inverter BRB-111. The DC Bus automatically supplies power through BRB-111 without interruption if MCC 62C is lost. The MCC does supply Battery Charger BRB-108 and is the normal supply to 120V AC cabinet BRB-105, the alternate supply to Instrument Bus 2.
b. Wrong. Loss of the DC Bus would result in a reactor trip affecting all 4 channels of instrumentation. The DC Bus does provide the standby source of power to Instrument Bus 2 through Inverter BRB-111.
c. Correct. Loss of BRB-113, Instrument Bus 2 will result in associated RPS instrumentation failing low. Using the provided PPCS screens, it can be determined that T0411A, RCLA CH2 OTEMP DT SETPT, and T0401A, RCLA CHAN 2 TAVE, values have failed to the low value while then other channels indicate current RCS values. The trend values for all 4 channels show the effects of the loss of Instrument Bus 2.
d. Wrong. 120V AC cabinet BRB-105 is the alternate supply to Bus 2. It also supplies power to fuse panel RR175 which does not affect the RPS instrumentation.

Question Number: 016
Exam Date 20090413
Exam Level R
K/A 058AK1.01
Question
Battery Charger, BRA-108, is provided with an automatic current limiting device to . . .

a. limit charging current to a value that doesn't cause excessive gassing from Safeguards Battery, BRA-101.

b. automatically place Safeguards Battery, BRA-101, on a normal float charge when Equalizing charge is complete.

c. protect Safeguards Battery, BRA-101, from high charging current should a rectifier fail in Battery Charger, BRA-108.

d. prevent tripping the Battery Charger, BRA-108, output circuit breaker when Safeguards Battery, BRA-101, is completely discharged.

Answer
d.
Reference
System Description 38, DC & Emergency AC Electrical Distribution System 3.3.1 and 3.3.4, pages 12-14.
NEW
HIGHER
K/A Knowledge of the operational implications of the following concepts as they apply to Loss of DC Power: Battery charger equipment and instrumentation.
JUSTIFICATION:
a. Wrong. One of the purposes of the automatic current limiting device is to "limit the charging current during overloads to a value that does not cause damage to the charger." This is a plausible distractor because excessive charging current causes excessive gassing; however, that is not the purpose of the automatic current limiting device.
b. Wrong. A 0 - 24 hour timer is provided to automatically terminate the manually started equalizing charge after a set time.
c. Wrong. One of the purposes of the automatic current limiting device is to "limit the charging current during overloads to a value that does not cause damage to the charger" not the battery.
d. Correct. The battery chargers are provided with an automatic current limiting device (device 90) that 1). Limits the charging current during overloads to a value that does not cause damage to the charger; 2). Also prevents tripping of the input or output circuit breakers when the charger is connected to a completely discharged battery; and 3). Factory set at 115% of full load rating (172.5 amps). In this example, the connected battery (BRA-101) is completely discharged and the output breaker of BRA-108 is specifically identified.

Question Number: 017
Exam Date 20090413
Exam Level R
K/A E04EK3.4
Question
Given the following:

- A plant cooldown for a refueling outage is in progress.
- RCS pressure is 400 psig.
- RCS temperature is 375°F.
- Two Charging Pumps are running, both in MANUAL, with Letdown and Charging balanced (Letdown flow is 40 gpm).
- The RHR system has just been placed in service for decay heat removal.
- Auxiliary Building radiation levels are now rising.
- Pressurizer level is lowering.
- Annunciator 47032-Q, "RHR PUMP PIT A/B LEVEL HIGH", is LIT.
- Annunciator 47032-R, "RHR PUMP PIT SUMP LEVEL HIGH", is LIT.
- SER 1599, RHR pump A pit level high, is in ALARM.
- SER 1600, RHR pump B pit level high, is in ALARM.

Assuming all plant systems function as designed, . . .

a. an automatic Safety Injection (SI) will occur.

b. the SI Accumulators will inject into the RCS.

c. Containment Sump recirculation will be unavailable.

d. Containment Isolation (CI) signal will isolate the leak.

Answer
c.
Reference
ECA-1.2, LOCA Outside Containment, Step 5
Background ECA-1.2, LOCA Outside Containment, Step 5
BANK
HIGHER
K/A Knowledge of the reasons for the following responses as they apply to the LOCA Outside Containment: RO or SRO function within the control room team as appropriate to the assigned position, in such a way that procedures are adhered to and the limitations in the facilities license and amendments are not violated.
JUSTIFICATION:
a. Wrong. Safety Injection system is not aligned for automatic operation given the as described plant conditions.
b. Wrong. Given the as described plant conditions, the Safety Injection Accumulators should be aligned for RCS pressure < 1000 psig and are therefore not aligned for injection.
c. Correct. The as described plant conditions point to a LOCA outside of Containment. Given a LOCA outside Containment, Reactor Coolant is lost from the RCS but does not accumulate in the Containment Sump making Containment Sump recirculation unavailable because of the lack of coolant inventory in the Containment Sump.
d. Wrong. Based on the plant conditions provided, the source of RCS leakage is from RHR

into the RHR pump pit. Given these conditions, CI will not isolate the source of RCS leakage.

Question Number: 018
Exam Date 20090413
Exam Level R
K/A E05EA2.2
Question
Given the following:

- A loss of normal feedwater has occurred.
- Conditions for a reactor trip have occurred, but the Reactor has NOT tripped.
- Control Rods are being MANUALLY inserted.
- Attempts to trip the Reactor have failed.

What is the reason the operator trips the Main Turbine, even though the Reactor is NOT tripped?

a. To cause the Reactor to trip.

b. To generate an AMSAC signal.

c. To maintain SG water inventory.

d. To prevent reverse powering the Main Generator.

Answer: C
Reference
FR-S.1 Background Document, Step 2.
USAR 14.1.12, page 14.1-45
BANK
FUNDAMENTAL
K/A Ability to determine and interpret the following as they apply to the loss of secondary Heat Sink: Adherence to appropriate procedures and operation within the limitations in the facility's licensee and amendments.
Justification:
a. Wrong: At this point in the procedure the Reactor has already failed to trip by multiple RPS signals and opening of the Bus 33 and 43 breakers have failed.
b. Wrong: The turbine tripping does not generate an AMSAC signal. AMSAC Signal generated from 3 out of the 4 non controlling narrow SG levels <13% for 25 seconds. The AMSAC signal trips the main turbine, Starts all AFW pumps, and disables the output of the rod drive MG sets.
c. Correct: Tripping of the main turbine maintains adequate SG inventory for decay heat removal to prevent fuel damage.
d. Wrong: Reverse powering the main generator is not the reason for tripping the Main Turbine.

Question Number: 019
Exam Date 20090413
Exam Level R
K/A 003AA1.06
Question
Given the following:

- Reactor is at 8% power, BOL.
- The Main Turbine is NOT LATCHED.
- RCS Tavg is 546°F and slowly lowering.
- Pressurizer pressure is 2235 psig and steady.
- Control Bank D position is 124 steps.
- RCS Boron is 2248 ppm.
- Control Bank D Rod G-11 drops.

The crew has entered AOP-CRD-001, "Control Drive System Malfunction".

To maintain Tavg on program the operators will . . .

a. step rods out.

b. dilute the RCS.

c. raise steam demand.

d. allow power defect to provide positive feedback.

Answer
b.
REFERENCE
Ref AOP-CRD-001 Step 5
NEW
HIGHER
K/A Ability to operate and/or monitor the following as they apply to the dropped control rod: RCS pressure and temperature.
JUSTIFICATION:
a. Wrong because procedure directs Tavg control by boration/dilution/turbine load. Rods only get moved if operable rods not at desired rod height as determined by delta flux and rod insertion insertion limits not Tavg.
b. Correct because procedure directs Tavg control by boration/dilution/turbine load.
c. Wrong because raising steam demand will lower Tavg
d. Wrong because power defect will not add enough feedback for the dropped rod to recover temperature

Question Number: 020
Exam Date 20090413
Exam Level R
K/A 036AK2.02
Question
Given the following:

- An irradiated fuel assembly has been dropped in the core.
- Annunciator 47011-B, "RADIATION INDICATION HIGH", is LIT.
- Radiation monitor R-11 Cnmt Vent (particulate) is in HIGH alarm.
- Radiation monitor R-12 Cnmt Vent (gas) is in HIGH alarm.

Which of the following describes the automatic response to this event?

a. Containment Ventilation Isolation (CVI) of Train 'A' equipment will occur.

b. Containment Ventilation Isolation (CVI) of Train 'B' equipment will occur.

c. Control Room ventilation shifts to the Post-Accident Recirculation mode.

d. Zone SV Boundary Dampers Train 'A' and 'B' Close and R-11 and R-12 sample return shifts to the Containment.

Answer
b.
Reference
AOP-RM-001, ATTACHMENT A, A2 & A3.
System Description 45, Radiation Monitoring System (RM), Section 3.11
E-1636, Integrated Logic Diagram, Diesel Generator Electric
E-2001, Integrated Logic Diagram, Diesel Generator Electric
E-1608, Integrated Logic Diagram, Reactor Building Vent System
E-2068, Integrated Logic Diagram, Reactor Building Vent System
MODIFIED
HIGHER
K/A Knowledge of the interrelations between the Fuel Handling Incidents and the following: Radiation monitoring equipment (portable and installed).
JUSTIFICATION:
a. Wrong. R-11 and R-12 actuate Containment Ventilation Isolation of Train B equipment. This is a plausible distractor because the opposite train is used.
b. Correct. R-11 and R-12 actuate Containment Ventilation Isolation of Train B equipment.
c. Wrong. Control Room ventilation shifts to Post-Accident Recirculation mode based on a signal from R-23 and NOT R-11 or R-12. This is a plausible distractor in that it describes an operator response due to an elevated radiation reading. AOP-FH-002, Dropped/Damaged Fuel Assembly, will direct manual starting if it hasn't automatically started.
d. Wrong. Zone SV Boundary Dampers close and R-11 and R-12 sample return shifts to Containment based on a signal from R-13 and R-14 and NOT R-11 or R-12. This is a plausible distractor in that it describes a plant response due to an elevated radiation reading and includes R-11 and R-12 responses.

Question Number: 021
Exam Date 20090413
Exam Level R
K/A 037AA1.11
Question
Given the following:

- A primary to secondary leak has developed in SG 'A'.
- The crew is reducing power at 3% per min per AOP-RC-004, "Steam Generator Tube Leak", and AOP-GEN-002, "Rapid Power Reduction".
- Reactor Power is 50% and lowering.
- Tavg is 559°F.
- PRZR Level is 22% and lowering.
- Charging Pump 'A' is in AUTO with 100% output and Charging Pump 'B' is in MANUAL with demand at 20%.
- Makeup Control is in AUTO.
- Letdown flow is 40 gpm.
- VCT level is 30% and lowering.
- Annunciator 47043-E, "PRESSURIZER LEVEL DEVIATION, has just LIT.

The operators should first . . .

a. isolate letdown.

b. maximize charging.

c. manually trip the reactor.

d. adjust loading rate to 1% per min.

Answer
b.
Reference
ARP-47043-E, Rev. 0
AOP-RC-004 steps 2 and 17, Rev. 0
MODIFIED
HIGHER
K/A Ability to operate and/or monitor the following as they apply to the Steam Generator Tube Leak: PZR level indicator.
JUSTIFICATION:
a. Wrong. Isolation of letdown is performed after charging is maximized and it is determined that PRZR level is still decreasing.
b. Correct. The annunciator 47043-E will alarm at +/- 10% from program. AOP-RC-004 (CAS) step 2 RNO, when PRZR level is determined NOT to be stable the first action is to maximize charging to maintain level.
c. Wrong. Conditions do NOT warrant a reactor trip. Indications for this action are PRZR level less than 3%, VCT level less than 5%, PRZR level decreasing with charging maximized and letdown isolated, or SG tube leak greater than available charging system capacity. Two charging pumps at 20% with letdown in service are within the capacity of the makeup system.
d. Wrong. Adjusting loading rate is not directed until 45% power and is not tied to PRZR level or makeup capacity.

Question Number: 022
Exam Date 20090413
Exam Level R
K/A 051AA2.02
Question
With a degrading condenser vacuum at 100% power, what is the sequence of events? (Assume NO operator action)

a. As Generator MWe decreases and MVARS become negative, G-1 will open at 5% reverse power.

b. Reactor Thermal Power will increase as plant efficiency degrades, this causes an OPDT Runback.

c. The Reactor Trips, G-1 opens in 30 seconds, and the SG PORV's will open to control Tavg at 547°F.

d. At a condenser vacuum of 10" HgA the Main Turbine will Trip, causing a Reactor Trip and Permissive P-7 to become lit.

Answer
d.
REFERENCE
OP-KW-ARP-47051-W, 3.4
System Description 9, Air Removal, 1.6, page 7
System Description 54, Main Turbine and Auxiliaries, 3.11.4, page 38
System Description 47, 3.12.2, page 19
E-2058, Integrated Logic Diagram - Turbine System
XK-100-156, Logic Diagrams Turbine Trip Runbacks & Other Signals
MODIFIED
HIGHER
K/A Ability to determine and interpret the following as they apply to the loss of Condenser Vacuum: Conditions requiring reactor and/or turbine trip.
a. Wrong: G-1 does not open on reverse power, opens 30 seconds after the turbine trip.
b. Wrong: With turbine on VPL Limit, turbine load will only decrease. This does not increase the reactor thermal power and OPDT does not change.
c. Wrong: The atmospheric steam dumps open to control Tave
d. Correct. The turbine trips at a vacuum equal to 10" HgA. The Atmospheric Steam Dumps are set to automatically open based on Tave following the trip.

Question Number: 023
Exam Date 20090413
Exam Level R
K/A 061AK3.02
Question
Given the following:

- Annunciator 47013-B, "RAD MONITOR FAILURE", is LIT.
- R-23, Control Room Vent Monitor, has failed low.
- R-23 Key Switch is in KEYPAD.
- TLA-15, "RMS ABOVE NORMAL", is LIT.
- R-5, Fuel Handling Area Monitor, reads above normal at 6 mr/hr and increasing slowly.
- HP has verified R-5 readings are valid
- Annunciator 47012-B, "HIGH RADIATION INDICATION ALERT", is NOT LIT
- Annunciator 47011-B, "RADIATION INDICATION HIGH", is NOT LIT

In response to the conditions above, the operating crew manually starts Control Room Post Accident Recirc to . . .

a. ensure that the Control Room environment is protected.

b. prevent discharging contamination from the Control Room to the Aux Building.

c. establish operability of the system prior to radiation levels reaching EAL actions levels.

d. place R-24, CRPA Recric Rad Monitor, in service to monitor the control atmosphere.

Answer
a.
Reference
ARP 47013-B, RAD MONITOR FAILURE
AOP-RM-001, Abnormal Radiation Monitoring, Steps 2.1; 7 (R-5), 24 (R-23)
NEW
HIGHER
K/A Knowledge of the reasons for the following responses as they apply to the Area Radiation Monitoring (ARM) System Alarms: Guidance contained in alarm response for ARM system.
JUSTIFICATION:
a. Correct. Control Post Accident Recirculation is designed to protect the control environment.
b. Wrong. Normal Control Air Conditioning discharges to the Turbine Building.
c. Wrong. Not establishing operability of the system by starting.
d. Wrong. There is not a rad monitor that gets put is service when CRPAR is started.

Question Number: 024
Exam Date 20090413
Exam Level R
K/A 068AK3.07
Question
Given the following:

- A fire has forced the control room to be evacuated.
- Control Operator 'A' is establishing Auxiliary Feedwater at the DSP.

AFW-10A, AFW Train 'A' Crossover Valve, is verified closed to . . .

a. prevent feeding Steam Generator 'B'.

b. prevent feeding Steam Generator 'A'.

c. control TDAFW pump feed flow to Steam Generator 'A'.

d. prevent a smart short from opening AFW-10A inadvertently.

Answer
a.
Reference
OP-KW-AOP-FP-002, Fire in Alternate Fire Zone, APPENDIX A, Step A14
System Description 87, 10CFR50 Appendix R, Section 3.7.1
NEW
FUNDAMENTAL
K/A Knowledge of the reasons for the following responses as they apply to the Control Room Evacuation: Maintenance of S/G level, using AFW flow control valves.
JUSTIFICATION:

a. Correct. With a fire in the Alternate Zone the secondary side inventory control shutdown function is established at the DSP using Steam Generator "A" and Train “A” equipment. To that end, AFW-10A being a Train "A" component is closed to prevent feeding Steam Generator "B" and thereby ensuring a continued supply of AFW to Steam Generator "A".
b. Wrong. With a fire in the Alternate Zone the secondary side inventory control shutdown function is established at the DSP using Steam Generator "A" and Train "A" equipment NOT Steam Generator "B." This is a plausible distractor in that only one Steam Generator is relied upon to establish the secondary side inventory control shutdown function.
c. Wrong. With a fire in the Alternate Zone the secondary side inventory control shutdown function is established at the DSP using Steam Generator "A" and Train "A" equipment. MDAFW Pump A is used and feed flow established through AFW-2A. This is a plausible distractor in that closing AFW-10A would control TDAFW pump feed flow to Steam Generator A however, this is not the reason for closing AFW-10A.
d. Wrong. With a fire in the Alternate Zone the secondary side inventory control shutdown function is established at the DSP using Steam Generator "A" and Train "A" equipment. This is because the Train "A" equipment is protected against fire damage. Therefore AFW-10A would be protected against experiencing a smart short. This is a plausible distractor in that a smart short is a concern during a fire and one addressed by Appendix R.

Question Number: 025
Exam Date 20090413
Exam Level R
K/A E06 2.4.31
Question
Given the following:

- Loss of Off-Site Power
- A Small Break LOCA is in progress.
- Both trains of ECCS failed to actuate.
- Core Exit Thermocouples are 560°F and stable.
- Reactor Vessel Level Indication is 0%.
- Subcooling Margin Monitor is 0°F and stable.
- Annunciator 47044-F, "ICCMS PANEL TROUBLE", is LIT.
- Annunciator 47033-12, "TLA-2 RCS SUBCOOLING HIGH/LOW", is LIT.

The plant is progressing towards inadequate core cooling because . . .

a. RCS coolant has reached superheated conditions.

b. RCS subcooling is decreasing towards saturation conditions.

c. coolant inventory has decreased to below the top of the core.

d. coolant inventory is between the bottom of the hot leg and the top of the core.

Answer
d.
REFERENCE
System Description 50, Incore Instrumentation & Inadequate Core Cooling Monitor & Loose Parts Monitoring (IE), Sections 3.5.2 and 3.5.3.
NEW
HIGHER
K/A Degraded Core Cooling: Knowledge of annunciator alarms, indications, or response procedures. RO Imp / SRO Imp - 4.2 / 4.1
JUSTIFICATION:
a. Wrong. The RCS has reached saturation but has not yet become superheated as indicated by SMM being 0°F and steady. If superheated conditions were reached the SMM would indicate a negative value. This is a plausible distractor and would be indicative of the third and final stage in approaching ICC.
b. Wrong. The RCS has already reached saturation conditions as indicated by SMM = 0°F. This is a plausible distractor and would be indicative of the first stage in approaching ICC.
c. Wrong. The core has not yet begun to uncover as indicated by SMM = 0°F and steady. Once core uncovery begins the CETs will begin to indicate superheated conditions and the SMM will become negative in value. This is a plausible distractor and would be indicative of the third and final stage in approaching ICC.
d. Correct. Reactor vessel level is below the bottom of the hot leg as indicated by RVLIS = 0% and above the top of the core as indicated by the SMM being steady.

Question Number: 026
Exam Date 20090413
Exam Level R
K/A 076AA2.02
Question
The following plant conditions exist:

- 100% power with all systems in normal lineup.
- Annunciator 47033-35, "TLA-15 RMS ABOVE NORMAL", is LIT.
- Annunciator 47012-B, "RADIATION INDICATION ALERT", is LIT.
- Annunciator 47011-B, "RADIATION INDICATION HIGH", is LIT.
- The radiation level indication on R-9, RCS Letdown Monitor, is 15 R/hr which has been confirmed by portable monitoring equipment.

Procedurally . . .

a. letdown is isolated to minimize the radiation hazard.

b. excess letdown is placed in service to minimize the radiation hazard.

c. letdown flow is maximized through the mixed bed demineralizers to reduce RCS activity.

d. letdown flow is maximized through the cation bed demineralizer to reduce RCS activity.

Answer
c.
Reference
AOP-RM-001, Abnormal Radiation Monitoring System, Step 10.b
AOP-RC-003, High Reactor Coolant Activity, Step 2.1 and Step 2 RNO
MODIFIED
FUNDAMENTAL
K/A Ability to determine and interpret the following as they apply to the High Reactor Coolant Activity: Corrective actions required for high fission product activity in RCS.
JUSTIFICATION:
a. Wrong. This is a plausible distractor in that isolating letdown would minimize the radiation hazard; however, recovery actions as directed in AOP-RC-003 are primarily geared towards reducing the RCS activity level more so than minimizing the radiation hazard.
b. Wrong. This is a plausible distractor in that placing excess letdown in service would reduce the radiation hazard because of the piping configuration and reduced flow rate however, recovery actions as directed in AOP-RC-003 are primarily geared towards reducing the RCS activity level more so than minimizing the radiation hazard.
c. Correct. With R-9 operating properly but having abnormally high radiation levels the RNO of AOP-RM-001 directs performing AOP-RC-003 while continuing in AOP-RM-001. AOP-RC-003, Step 2 has the operator check that maximum letdown flow is established. If not, the RNO directs the operator to operate the CVC mixed bed demineralizers at maximum letdown flow rate to reduce RCS activity.
d. Wrong. This is a plausible distractor because apart from the wrong demineralizer (e.g. cation vs. mixed bed), the actions are correct. The cation demineralizer is used for pH control, not RCS activity reduction.

Question Number: 027
Exam Date 20090413
Exam Level R
K/A W/E08EK1.1
Question
Given the following:

- Steam Line Break has occurred in containment with a Loss of Off-Site Power.
- Bus 5 and 6 are powered from the Emergency Diesel Generators.
- The crew is performing the actions of FR-P.1, "Response to Imminent Pressurized Thermal Shock".
- Both SI pumps have been stopped and placed in AUTO.
- Both RHR pumps have been stopped and placed in AUTO.
- Charging pump 'A' is running with charging flow established at 25 gpm.
- Letdown is unavailable.

Which of the following is the preferred method to depressurize the plant?

a. Use auxiliary spray to prevent rupturing the PRT rupture disk.

b. Turn off all pressurizer heaters to prevent thermal stratification.

c. Use normal spray to prevent dilution of PRZR boron concentration.

d. Open a PRZR PORV to minimize thermal shock to the PRZR spray nozzle.

Answer
d.
REFERENCE
FR-P.1, Step 21
Background Document for FR-P.1 step 21
NEW
FUNDAMENTAL
K/A: Knowledge of the operational implications of the following concepts as they apply to the Pressurized Thermal Shock: Components, capacity, and function of emergency systems.
Justification:
a. Wrong: With letdown isolated, auxiliary spray is not preferred because it could thermal shock the pressurizer spray nozzle.
b. Wrong: Turning off PRZR heaters will have minimal effect on reducing RCS pressure.
c. Wrong: With loss of offsite power, the RXCPs are not running and normal spray is not available. Also dilution of the RCS is not the primary concern at this time.
d. Correct: With letdown isolated and RXCPs not available, the PRZR PORV is the preferred method of depressurization to prevent thermal shock to the pressurizer spray nozzle.

Question Number: 028
Exam Date 20090413
Exam Level R
K/A 003K3.02
Question
Given the following:

- The plant is at 100% power.
- Control Bank D is at 225 steps.
- Reactor Coolant System pressure is 2235 psig and stable.
- Tavg is 572°F and stable.
- Reactor Coolant Pump 'A' trips.

With NO operator action, 2 minutes after Reactor Coolant Pump 'A' trips . . .

a. AFW flow to S/G 'A' is > AFW flow to S/G 'B'.

b. AFW Pump 'A' tripped on low discharge pressure.

c. SD-3B, S/G 'B' PORV, will be cycling to maintain RCS Temperature.

d. FW-12A, S/G 'A' Feedwater Isolation, will be closed to limit feed water flow.

Answer
a.
REFERENCE
Accident and Transient Analysis, Chapter VI (page VI-25)
Dwg E-1625
HIGHER
MODIFIED
K/A Knowledge of the effect that a loss or malfunction of the RCPs will have on the following: S/G.
JUSTIFICATION:
a. Correct: When the reactor coolant pump is stopped the flow in the idle loop will reverse. The A cold leg temperature will be the same as the B loop however, the A hot leg temperature will decrease to a value below the cold leg temperature as the water flowing through the steam generator in the reverse direction is cooled. The drop in the A loop hot leg temperature decreases pressure in S/G A therefore, AFW flow will go to S/G A.
b. Wrong: Steam Dumps will restore Tavg to 547ºF. Corresponding SG pressure will remain high enough that AFW low discharge pressure trip will not be challenged.
c. Wrong: Given plant conditions are that all systems operate as designed so that the Condenser Steam Dumps restore Tavg to no load value which is below the setpoint for the SG PORV's.
d. Wrong. There is no SI signal or high S/G water level that will cause FW-12A to shut.

Question Number: 029
Exam Date 20090413
Exam Level R
K/A 006K6.14
Question
Given the following:

- Plant Startup and Heatup in progress.
- The RCS is solid and vented.
- RHR system is in service with RHR/CVC crossconnect aligned: RHR-210 OPEN, RHR-211 OPEN, RHR/CVC Spectacle Flange INSTALLED.
- RCS Cold Leg Temperatures are 175°F.
- RCS Hot Leg Temperatures are 175°F.
- RCS pressure is 350 psig.
- SW-1306A, CC HX A Temperature CV, is in MANUAL and Component Cooling Heat Exchanger Outlet Temperature, T0621A, is 100°F

RXCP 'B' is the first RXCP to be started. Which action is required prior to starting RXCP 'B'?

a. Form a bubble in the Pressurizer.

b. Measure Steam Generator temperatures locally.

c. Verify PS-1A, Przr Spray Control Valve A, in MANUAL and CLOSED.

d. Reduce Component Cooling Heat Exchanger outlet temperature to <95°F.

Answer
b.
REFERENCE
OP-KW-NOP-RCS-001, Reactor Coolant Pump Operation, Section 4.5, 4.6 & 5.4
NEW
FUNDAMENTAL
K/A Knowledge of the effect of a loss or malfunction on the following will have on the RCPS: Starting Requirements.
Justification:
a. Wrong: A bubble must be formed in the pressurizer if all of the following conditions exist: RCS temperature > 180°F, both RXCP have been stopped for greater than 5 minutes, and a differential temperature exists in the RCS due to RXCP seal injection flow. The parameters given in the questions indicate zero differential temperature. P&L 4.6 of NOP-RCS-001.
b. Correct. Must determine steam generator temperatures prior to starting RXCPs while solid and cold to ensure that the resulting pressure transient does not exceed the capacity of the LTOP. P&L 4.5 and Step 5.4.4 of NOP-RCS-001.
c. Wrong: The requirement for PS-1A to be in MANUAL and CLOSED applies only if there is a bubble in the pressurizer. Step 5.4.3 of NOP-RCS-001.
d. Wrong: The maximum component cooling water heat exchanger outlet temperature is 105°F. The minimum temperature is 95°F. P&L 4.9 of NOP-RCS-001.

Question Number: 030
Exam Date 20090413
Exam Level R
K/A 004K5.14
Question
Given the following:

- Cooldown to Cold Shutdown in progress.
- RCS temperature at 310°F.
- RCS pressure at 345 psig.
- One RXCP in operation with spray control in AUTO.
- PRZR level on program with control in AUTO.
- PRZR PORV, PR-2A, is being cycled to vent hydrogen from the Pressurizer and is currently open.
- PRT level and pressure were normal before cycling PR-2A.

Close PRZR PORV, PR-2A, before . . .

a. RCS subcooling decreases to 15ºF.

b. PRT pressure increases to 20 psig.

c. PRT temperature increases to 110°F.

d. RXCP #1 seal DP decreases to 200 psid.

Answer
d.
REFERENCE
GOP-202, Shutdown From RHR To Cold Shutdown, 4.6 and Step 5.2.4
AOP-RC-005, Abnormal RXCP Operation, Foldout Page
NOP-RCS-002, Pressurizer Relief Tank Operation, 4.5 and 4.8
NEW
HIGHER
K/A Knowledge of the operational implications of the following concepts as they apply to the CVCS: Reduction process of gas concentration in RCS: vent-accumulated non-condensable gases from PZR bubble space, depressurized during cooldown or by alternatively heating and cooling (spray) within allowed pressure band (drive more gas out of solution).
JUSTIFICATION:
a. Wrong. GOP-202 P&L 4.6, states that during cooldown, when < 350°F and a bubble exists in the pressurizer, RCS subcooling must be maintained 30 – 300°F to minimize fatigue stress to PZR and PZR surge line. This is a plausible distractor because RCS pressure will be decreasing during this activity however closing PR-2A at 15°F would be later than necessary. In addition, 15°F subcooling is a limitation used within the EOP network.
b. Wrong. IAW NOP-RCS-002, P&L 4.5, PRT Rupture Disc blows at 100 psig. PRT pressure shall NOT be allowed to exceed 75 psig during normal operation. This is a plausible distractor because GOP 202 Step 5.2.4.f sets the limit for PRT pressure while venting H2 at 25 psig. Closing PR-2A at 20 psig would be earlier than required, reducing the effectiveness of the venting operation.
c. Wrong. IAW NOP-RCS-002, P&L 4.8, PRT water temperature shall be maintained less

than or equal to 120°F. This is a plausible distractor because GOP 202 Step 5.2.4.f sets the limit for PRT temperature while venting H2 at 120°F. Closing PR-2A at 110°F would be earlier than required, reducing the effectiveness of the venting operation.

d. Correct. IAW GOP-202, Step 5.2.4.f, the limiting RXCP #1 seal DP is 200 psid. This is also the limiting RXCP #1 seal DP identified in the FOP for AOP-RC-005, Abnormal RXCP Operation, and is the minimum DP to ensure proper RXCP #1 seal operation.

Question Number: 031
Exam Date 20090413
Exam Level R
K/A 05K5.01
Question
Which design feature is associated with minimizing the probability of brittle fracture of the reactor pressure vessel?

a. Low Temperature Overpressure Protection relief valve is installed on the common suction of the RHR pumps.

b. The inside surface of the reactor vessel is clad with a stainless steel material that is not subject to brittle fracture.

c. Pressurizer safety valves designed to limit reactor vessel pressure transients to 110% of design are installed.

d. Reactor vessel flow is directed downward on both sides of the thermal shield to provide cooling of the thermal shield.

Answer
a.
REFERENCE
System Description 34, Residual Heat Removal System, 3.1.6, pages 10-11
SOER 82-7 (OEA 82-205)
MODIFIED
FUNDAMENTAL
K/A Knowledge of the operational implications of the following concepts as they apply to the RHRS: Nil ductility transition temperature (brittle fracture).
JUSTIFICATION:
a. Correct. The LTOP valve RHR-33-1, is designed to prevent a pressurized thermal shock (PTS) event from occurring while the RCS is being cooled by RHR.
b. Wrong. The inside of the reactor vessel is clad with a stainless steel material to provide a corrosion resistant layer. It does not reduce the probability of brittle fracture of the carbon steel vessel. The cladding layer is not considered as structural and thus, is not included in the vessel stress and fracture analysis.
c. Wrong. The installed safety valves prevent pressure transients from exceeding design pressure limits. These would be operating transients where temperature is well above RTNDT and brittle fracture is not a concern.
d. Wrong. Reactor vessel flow is directed down both sides of the thermal shield to remove the heat generated by neutron attenuation. This has no effect on the number of fast neutrons reaching the vessel wall and thus, does not impact the shift of RTNDT and brittle fracture probability.

Question Number: 032
Exam Date 20090413
Exam Level R
K/A 05A4.03
Question
With the RHR system in the normal cooldown mode and total RHR flow rate being maintained using RHR-101, RHR Flow Control Bypass Valve, how is RCS temperature controlled using the RHR system?

a. Throttle valves at the inlet of the RHR Heat Exchangers are positioned by the Control Room operator to maintain RCS temperature.

b. Throttle valves at the outlet of the RHR Heat Exchangers are positioned by the Control Room operator in response to RCS temperature trends.

c. A temperature control valve at the outlet of the RHR Heat Exchanger controls RHR flow through the RHR Heat Exchanger to maintain the preset RHR Heat Exchanger outlet temperature automatically.

d. A temperature control valve in the Component Cooling Water (CCW) system controls CCW flow through the RHR Heat Exchanger to maintain the preset RHR Heat Exchanger outlet temperature automatically.

Answer
b.
REFERENCE
System Description 34 – Residual Heat Removal, 3.5
OPERXK-100-18, Flow Diagram, Residual Heat Removal System
MODIFIED
FUNDAMENTAL
K/A Ability to manually operate and/or monitor in the control room: RHR temperature, PZR heaters and flow, and nitrogen. RO Imp / SRO Imp - 2.8 / 2.7
JUSTIFICATION:
a. Wrong. RCS temperature is controlled by manually adjusting RHR-8A/B which are located on the outlet of the respective RHR heat exchanger not at the inlet. This is a plausible distractor in that it moves the throttle valve to the wrong end of the heat exchanger.
b. Correct. RCS temperature is controlled by manually adjusting RHR-8A/B which are located on the outlet of the respective RHR heat exchanger.
c. Wrong. RCS temperature is controlled by adjusting RHR-8A/B manually not automatically. This is a plausible distractor because automatically controlled heat exchanger outlet valves are used in other applications to control process fluid temperature.
d. Wrong. RCS temperature is controlled by adjusting RHR-8A/B not by controlling CCW flow. This is a plausible distractor in that CCW flow is controlled in other applications to control process fluid temperature.

Question Number: 033
Exam Date 20090413
Exam Level R
K/A 006K2.02
Question
Given the following:

- A cooldown to COLD SHUTDOWN is in progress.
- RCS Tavg is 450°F.
- RCS pressure is 990 psig.
- Both SI Accumulator Discharge Isolation power supply breakers are ON.
- SI-20A/MV-32091 Accumulator 'A' Isolation valve is CLOSED with its switch in AUTO.
- SI-20B/MV-32096 Accumulator 'B' Isolation valve is OPEN with its switch in AUTO.
- Bus 5 LOCKOUT occurs.
- Large break LOCA occurs on Loop 'B' Cold Leg (Double Ended Shear).
- RCS pressure decreases to 400 psig in 15 seconds.

What is the response of the SI Accumulators?

a. Both SI Accumulators will inject to the reactor vessel.

b. Neither SI Accumulator will inject to the reactor vessel.

c. Only SI Accumulator 'A' will inject to the reactor vessel.

d. Only SI Accumulator 'B' will inject to the reactor vessel.

Answer
b.
REFERENCE
KNPP USAR, Sections 6.2.2.2.1, 6.2.3.7, 14.3.2.1 and 14.3.3.1
E-2034, Integrated Logic Diagram – Safety Injection System, Rev. R
Operator Aid # 02-22 MCC-52B and MCC-62B
BANK
HIGHER
K/A Knowledge of the bus power supplies to the following: Valve operators for accumulators.
JUSTIFICATION
a. Wrong: SI Accumulator 'A' will not inject, the valve will not reposition on the SI signal because of a loss of power to bus 5 and SI Accumulator 'B' will dump onto the floor.
b. Correct: SI Accumulator 'A' will not inject, the valve will not reposition on the SI signal because of a loss of power to bus 5 and SI Accumulator 'B' will dump onto the floor.
c. Wrong: SI Accumulator 'A' will not inject, the valve will not reposition on the SI signal because of a loss of power to bus 5.
d. Wrong: SI Accumulator 'B' will dump onto the containment floor.

Question Number: 034
Exam Date 20090413
Exam Level R
K/A 007A1.01
Question
Given the following:

- Reactor is at 100% power.
- Annunciator 47043-B, "PRESSURIZER RELIEF TANK ABNORMAL", is LIT.
- PRT level indication is 62% and slowly lowering.
- PRT temperature is 119°F.

If a PRZR PORV fails OPEN before the operators restore PRT level, PRT pressure will . . .

a. lower.

b. increase slightly.

c. remain constant.

d. increase to over-pressurization.

Answer
d.
REFERENCE
System Description 36, RCS, 3.6.8, pages 32-34
BANK
HIGHER
K/A Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the PRTS controls including: Maintaining Quench tank water level within limits. RO Imp / SRO Imp - 2.9 / 3.1
JUSTIFICATION
a. Wrong. The lower level would reduce the mass available to quench the steam from the PORV.
b. Wrong. The added mass and temperature would compress and heat the gas bubble, which would significantly increase pressure.
c. Wrong. The added heat and increased mass in the lower PRT quench volume would significantly increase pressure.
d. Correct. The design of the PRT is that with normal levels and temperatures, a lifting relief will not overpressure the PRT. At a reduced level the over pressurization would occur.

Question Number: 035
Exam Date 20090413
Exam Level R
K/A 007G2.1.20
Question
The crew is performing a pumpdown of the PRT to CVC HUT per NOP-RCS-002 section 5.2, "Pressurizer Relief Tank Operation".

When the Operator opens PR-40, Przr Relief Tank Drain Isolation, . . .

a. RCDT pump 'A' starts and then stops when RCDT level is 12%.

b. RC-503-1, RCDT to Rx Clnt Drain pumps, closes and RCDT pump 'A' starts.

c. RCDT pump 'B' starts when RCDT level is 50% and stops when RCDT level is 12%.

d. RC-507, Rx Clnt Drain Pump Disch Header Isolation, opens and RCDT pump 'A' will start.

Answer
b.
REFERENCE
OP-KW-NOP-RCS-002, Pressurizer Relief Tank Operation, section 5.2
OP-KW-NOP-LWP-001, Reactor Coolant Drain Tank, section 5.2
Dwg. E-2040
Dwg. E-2046
MODIFIED
FUNDAMENTAL
K/A Ability to execute procedure steps. RO Imp / SRO Imp - 4.6 / 4.6
Justification:
a. Wrong: RCDT pump 'A' will remain running as long as PR-40 is open.
b. Correct: RC-503-1 closes and RCDT pump 'A' starts when PR-40 is opened.
c. Wrong: RCDT pump 'B' starts when RCDT level is 80% and RC-507 & RC-508 are open. The pump will automatically stop when RCDT is 12%.
d. Wrong: RC-507 is a manual valve. When RC-507 and RC-508 are manually opened and RCDT drain tank level is 50% then RCDT pump 'A' starts.

Question Number: 036
Exam Date 20090413
Exam Level R
K/A 08A1.03
Question
Given the following:

- RXCP 'A' thermal barrier has ruptured
- 47022-H, "CC SURGE TANK LEVEL HIGH/LOW", is LIT
- Component Cooling (CC) Surge Tank level is 85% and increasing

The CC Surge Tank is protected from overpressurization by the . . .

a. CC Surge Tank relief valve which is set to open at 150 psig.

b. CC Surge Tank Vent line which is open to the Waste Holdup Tank.

c. CC-630A, RXCP 'A' Thermal Barrier Return relief valve, which is set to open at 2485 psig.

d. CC-104, CC Surge Tank Vent valve, which automatically opens to limit surge tank pressure to 100 psig.

Answer
b.
REFERENCE
OPERXK-100-19, Flow Diagram Component Cooling System
OPERXK-100-20, Flow Diagram Component Cooling System
System Description 31, Component Cooling Water System (CC)
NEW
MEMORY
K/A Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the CCWS controls including: CCW pressure.
JUSTIFICATION:
a. Wrong. There is no relief valve installed on the CC surge tank. This is a plausible distractor in that relief valves are often used to provide overpressure protection.
b. Correct. The CC surge tank vent is sized to relieve maximum flow rate resulting from a ruptured thermal barrier-cooling coil in a RXCP.
c. Wrong. The RXCP thermal barrier relief valves protect the high pressure portion of the CC system from overpressurization. The CC surge tank is not part of the high pressure portion of the CC system. This is a plausible distractor in that a failed RXCP thermal barrier heat exchanger forms the basis for the overpressure protection afforded the CC surge tank via the vent line.
d. Wrong. The CC surge tank vent valve is normally closed and there is no procedural guidance to open the valve. This is a plausible distractor in that the valve actually does exist and at one time was used to isolate the CC surge tank in the event of inleakage.

Question Number: 037
Exam Date 20090413
Exam Level R
K/A 010K5.01
Question
The plant is shutdown with core decay heat being removed by the steam generators (S/Gs).

S/G pressures are being maintained at 415 psig using the Condenser Steam Dumps.

Using steam tables, determine what listed pressure below is the MINIMUM PRZR pressure required to establish a 110°F subcooling margin in the RCS loop cold legs? (Assume a negligible temperature difference between the RCS and the S/Gs.)

a. 1100 psig

b. 1120 psig

c. 1140 psig

d. 1160 psig

Answer
c.
REFERENCE
Steam Tables
BANK
HIGHER
K/A Knowledge of the operational implications of the following concepts as they apply to the PRR PCS: Determination of condition of fluid in the PZR using steam tables.
Justification
a. A: Wrong: Does not provide 110°F of subcooling.
b. Wrong: Does not provide 110°F of subcooling.
c. Correct: The minimum pressure listed for 110°F subcooling
d. Wrong: Not the minimum pressure listed for 110°F subcooling.

Question Number: 038
Exam Date 20090413
Exam Level R
K/A 012A3.02
Question
The plant was operating at 18% power when an event occurred.

The operator then noted the following:

- Tavg is 548°F and decreasing rapidly.
- Main Turbine is LATCHED.
- FW-7A/B, Main Feedwater Control Valves are Throttled OPEN.
- Steam Dumps are NOT Armed.
- Annunciator 47012-K, "RXCP B BREAKER OPEN", is LIT.
- Annunciator 47012-M, "RCS LOOP B FLOW LOW", is LIT.
- Annunciator 47032-D, "SINGLE LOOP LOW FLOW REACTOR TRIP", is LIT.
- Reactor Trip Breakers green lights are LIT.

Which of the following permissive circuits has failed as indicated by this event?

a. P-2 circuit

b. P-4 circuit

c. P-8 circuit

d. P-13 circuit

Answer
b.
REFERENCE
System Description 47, Reactor Protection and Reactor Coolant Temperature Instrument, 3.12.2 and 3.12.3
XK100-144
BANK
HIGHER
K/A Ability to monitor automatic operation of the RPS, including: Bistables.
JUSTIFICATION:
a. Wrong - P-2 circuit is related to Auto Rod control. No impact with reactor tripped.
b. Correct - P-4 circuit causes turbine trip and main feedwater isolation.
c. Wrong - P-8 circuit is single RCS loop loss of flow.
d. Wrong - P-13 circuit is related to turbine impulse pressure and feeds P-7. No relevance to this issue.

Question Number: 039
Exam Date 20090413
Exam Level R
K/A 013K6.01
Question
Given the following:

- Containment pressure instrument PT-945, Containment Pressure (Channel I), has failed downscale.
- The actions of AOP-MISC-001 Attachment I, "Response to Instrument Failure", have been completed for removing the channel from service.
- The following status lights are LIT:
  44908-0901, Containment 4 psig
  44908-1101, Containment 23 psig
- The following annunciators are LIT:
  47023-D, "CNTMT PRESS SI ALERT"
  47022-F, "CNTMT PRESS ICS ALERT"
- After AOP-MISC-001 Attachment I has been completed for the failure of PT-945, PT-949 (Channel III) fails upscale.

What is the response of the plant?

a. Only Safety Injection will occur.

b. Only Main Steam Isolation will occur.

c. Safety Injection and Main Steam Isolation will occur.

d. Safety Injection and Containment Spray Actuation will occur.

Answer
a.
REFERENCE
OP-KW-AOP-MISC-001, ATTACHMENT I, I.5 and I.9
XK100-150
BANK
HIGHER
K/A Knowledge of the effect of a loss or malfunction of the following will have on the ESFAS: Sensors and detectors.
JUSTIFICATION:
a. Correct. Containment pressure has 6 instruments/detectors. P-945, P-947, and P-949 are the pressure channels used to generate a Safety Injection Actuation. Per OP-KW-AOP-MISC-001, the bistables are tripped so the logic goes from 2 out of 3 to 1 out of 2. With bistables tripped for P-945, the failure high of P-949 would complete the logic.
b. Wrong. MSIV closure uses P-946, P-948, and P-950.
c. Wrong. MSIV closure uses P-946, P-948, and P-950.
d. Wrong. Containment Spray logic requires 1 out of 2, 3 times to actuate (i.e. 3 signals) and these failures would only generate 2 of the required 3 signals.

Question Number: 040
Exam Date 20090413
Exam Level R
K/A 022K2.01
Question
Given the following:

- A Small Break LOCA resulted in a MANUAL SI
- A loss of off-site power occurred at the time of trip/SI
- 480 V Bus 1-51 de-energized on overcurrent and locked out
- Emergency Diesel Generators 'A' and 'B' have energized their respective vital busses

Which identifies the status of CFCU's?

a. No CFCU's will be operating.

b. All CFCU's will be operating.

c. Only CFCU 'A' and 'B' will be operating.

d. Only CFCU 'C' and 'D' will be operating.

Answer
d.
REFERENCE
Drawing E-240
480 V Bus 51: CFCU A and B
480 V Bus 61: CFCU C and D
BANK
FUNDAMENTAL
K/A Knowledge of bus power supplies for the following: Containment Cooling Fans.
JUSTIFICATION:
a. Wrong. SI signal would start the two energized CFCU's (C and D).
b. Wrong. A & B are de-energized.
c. Wrong. Both are de-energized.
d. Correct. These are powered from Bus 61.

Question Number: 041
Exam Date 20090413
Exam Level R
K/A 022G2.4.2
Question
Given the following:

- A Safety Injection Signal is actuated.

How is the operation of SW-901B-1/CV-31705, Header B Shroud Clg Coil A/B Bypass, affected by this condition?

a. It gets a close signal to provide Containment Isolation for the Shroud Cooling Coil 'A/B'.

b. It gets an open signal to ensure adequate cooling flow through Containment Fan Coil Unit 'B'.

c. It gets a close signal to prevent excessive cooling flow conditions for Containment Fan Coil Unit 'B'.

d. It gets an open signal to maintain a minimum flow of 75 gpm through Shroud Cooling Coil 'A/B'.

Answer
b.
REFERENCE
E-3174, Rev. F, (C-2)
OPERM-547, Rev. R, (A-3)
System Description Number 18, 3.7, pages 18 & 19
BANK
MEMORY
K/A Knowledge of the system setpoints, interlocks and automatic actions associated with EOP entry conditions. RO Imp / SRO Imp - 4.5 / 4.6
JUSTIFICATION:
a. Wrong: The valve goes open.
b. Correct: The valve is normally throttled automatically to maintain 75 gpm flow through the CRDM Shroud Cooling Coils. When the SI signal is generated, the valves go to the full open position to provide unrestricted flow through the CFCUs. The CRDM Shroud Cooling coils supply and return isolation valves go closed to stop flow through the CRDM coils.
c. Wrong: The valve goes open.
d. Wrong: Normal flow is 75 gpm. Valves go flow open to provide full unrestricted flow to CFCUs

Question Number: 042
Exam Date 20090413
Exam Level R
K/A 026A2.08
Question
Given the following:

- LOCA.
- Unable to establish Containment Sump Recirculation.
- Both trains of Containment Spray automatically actuated.
- Both trains of Containment Spray are taking a suction from the RWST.
- Containment Fan Coil Units 'A', 'B', and 'C' are operating in the emergency mode.
- RBV-150D, Containment Fan Coil Unit D Emergency Discharge Damper, has failed closed.

According to the attached ECA-1.1, "Loss of Emergency Coolant Recirculation", Step 8, which of the following combinations of RWST level and Containment Pressure would allow the EARLIEST stopping of the LAST Containment Spray pump?

a. 40% and 40 psig

b. 50% and 30 psig

c. 40% and 20 psig

d. 30% and 10 psig

Answer
c.
REFERENCE
ECA-1.1, Loss Of Emergency Coolant Recirculation, Step 8
NEW
FUNDAMENTAL
Provided Reference: ECA-1.1, STEP 8.
K/A Ability to (a) predict the impacts of the following malfunctions or operations on the CSS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Safe securing of containment spray (when it can be done).
JUSTIFICATION:
a. Wrong. Given three operating CFCUs, 40% RWST level, and 40 psig Containment pressure requires 1 ICS pump running per ECA-1.1, Step 8. This is a plausible distractor because it is a valid plant condition addressed by the procedure step.
b. Wrong. Given three operating CFCUs, 50% RWST level, and 30 psig Containment pressure requires 1 ICS pump running per ECA-1.1, Step 8. This is a plausible distractor because it is a valid plant condition addressed by the procedure step.
c. Correct. Given three operating CFCUs, 40% RWST level, and 20 psig Containment pressure requires 0 ICS pump running per ECA-1.1, Step 8.
d. Wrong. Given three operating CFCUs, 30% RWST level, and 10 psig Containment pressure requires 0 ICS pump running per ECA-1.1, Step 8 however, the procedure would have allowed stopping the last ICS pump earlier, when Containment pressure was less than 46 psig. This is a plausible distractor because it is a valid plant condition addressed by the procedure step.

Question Number: 043
Exam Date 20090413
Exam Level R
K/A 039K1.07
Question
Which of the following conditions does NOT result in an INOPERABLE Turbine Driven AFW pump with reactor power at 20%?

a. AFW Recirc Flow aligned to TB standpipe.

b. MS-103, T/D AFW Pump Trip And Throttle Valve, is CLOSED.

c. SW-500A, Service Water Supply to TD AFW Pump, is NOT fully OPEN.

d. AFW-10A or AFW-10B, AFW Train Crossover Valve, is NOT fully OPEN.

Answer
a.
REFERENCE
KPS Technical Specification, 3.4.b
KPS Technical Specification Bases, 3.4.b
OPERM-202 Sh 2
OPERM-203
BANK
FUNDAMENTAL
K/A Knowledge of the physical connections and/or cause-effect relationships between the MRSS and the following systems: AFW.
JUSTIFICATION:
a. Correct. AFW Recirc flow has no effect on operability of the TDAFW pump.
b. Wrong. This is the (overspeed) isolation valve just prior to the TD AFW Pump turbine governor. T.S. 3.4.b.1.B states the TD AFW Pump is capable of delivering flow to both steam generators." With this valve closed there is no source of motive power for the turbine.
c. Wrong. This is a manual valve supplying TD AFW suction from SW Train A Header. T.S. 3.4.b.1.B states the TD AFW Pump is "capable of taking suction from the Service Water System."
d. Wrong. T.S. 3.4.b.7.C states that when Rx power < 15%, AFW-10A and AFW-10B may be in the closed position without declaring the corresponding AFW train inoperable.

Question Number: 044
Exam Date 20090413
Exam Level R
K/A 059K1.02
Question
How will AFW pump 'A' respond to a Bus 1 and Bus 2 LOCKOUT?

The AFW pump 'A' INITIALLY starts because…

a. of a Steam Generator 'A' Lo-Lo Level.

b. both Main Feedwater Pump breakers trip.

c. of an AMSAC Steam Generator Lo-Lo Level.

d. of the undervoltage condition on Buses 1 and 2.

Answer
b.
REFERENCE
E-1602-1, Integrated Logic Diagram - Auxiliary Feedwater
E-1624, Integrated Logic Diagram – Feedwater System
NEW
FUNDAMENTAL
K/A Knowledge of the physical connections and/or cause-effect relationships between the MFW system and the following systems: AFW.
Justification
a. Wrong: Though a lo-lo level in SG 'A' will start 'A' AFW pump the feed pump breakers opening will initially start both MD AFW pumps.
b. Correct: The Bus 1 and 2 lockout will cause both MFW pump breakers to trip which will cause both MD AFW pumps to start.
c. Wrong: The AMSAC signal will cause the AFW pumps to start but 'A' AFW pump will already be running.
d. Wrong: The undervoltage condition on bus 1 and 2 will cause the TD AFW pump to start.

Question Number: 045
Exam Date 20090413
Exam Level R
K/A 061K4.03
Question

The plant is operating at 100% power, with all systems in a normal configuration. Auxiliary Feedwater Pump 'B' is started from the Control Room by placing the control switch to the start position.

Without further operator actions, what is the final condition of the Steam Generator Blowdown valves?

	BT-2A	BT-3A	BT-2B	BT-3B
a.	Open	Open	Closed	Closed
b.	Open	Closed	Closed	Open
c.	Closed	Open	Open	Closed
d.	Closed	Closed	Open	Open

Answer
b.
REFERENCE
E-1602-1, Integrated Logic Diagram - Auxiliary Feedwater
E-1629, Integrated Logic Diagram - Main Steam & Steam Dump System
BANK
FUNDAMENTAL
K/A Knowledge of AFW system design feature(s) and/or interlock(s) which provide for the following: Automatic blowdown/sample isolation. RO Imp / SRO Imp - 2.7 / 2.9
JUSTIFICATION:
The start of B AFW Pump gives a close signal to the SG Blowdown Isolation valves associated with SG A Outside Containment Isolation Valve, BT-3A, and SG B Inside Containment Isolation Valve, BT-2B.
a. Wrong. This would isolate both valves associated with SG B.
b. Correct.
c. Wrong. These are the valves that close upon the start of AFW Pump A.
d. Wrong. This would isolate both valves associated with SG A.

Question Number: 046
Exam Date 20090413
Exam Level R
K/A 062K4.05
Question
Per AOP-EHV-005, "Loss of 4160V Bus 5", the following conditions are required to shift from Diesel Generator 'A' to the TAT:

- Verify Bkr 1-501 43 switch in MAN.
- Position Bkr 1-501 SYNC Switch to ON
- Locally set Governor Speed Droop to 30
- Locally position Parallel-Unit switch to PARALLEL
- Verify Incoming and Running voltages MATCHED
- Verify Synchroscope rotating slowly in FAST direction
- At 11:57 o'clock CLOSE Bkr 1-501

Which condition would prevent Bkr 1-501, TAT to Bus 5, from CLOSING?

a. Bkr 1-501 43 Switch is left in AUTO.

b. The synchroscope is rotating slowly in the SLOW direction.

c. Running voltage is not within 2-3 volts of incoming voltage.

d. The attempt to close Bkr 1-501 does not occur until 12:01 o'clock.

Answer
a.
REFERENCE
AOP-EHV-005, Loss Of 4160V Bus 5, ATTACHMENT A, page 1 and 2
E-914, Interlock Logic Diagrams, Bus 1-5 Source Breakers
NEW
FUNDAMENTAL
K/A Knowledge of A.C. Distribution System design feature(s) and/or interlock(s) which provide for the following: Paralleling of ac sources (synchroscope).
JUSTIFICATION:
a. Correct. If the 43 Switch is left in AUTO it is not possible to close the breaker in the Control Room.
b. Wrong. The DG synchronizing relay is not active when closing a transformer onto the bus, so there are no interlock features that require the synchroscope to be rotating in the FAST direction to close the breaker. Normally the expectation is to rotate in the FAST direction which ensures that the source being paralleled to the bus will pick up load when it is attached vs. becoming a load. This is a plausible distractor in that the direction of rotation is specified by procedure when paralleling sources.
c. Wrong. There are no interlock features that require voltages to be matched when paralleling AC sources however, if they are not, current flow will exist when the breaker is closed and the current flow, if excessive could result in protective relay actuation. This is a plausible distractor both because matched voltage is specified in the procedure and because elsewhere "2 - 3" volts is a specific value for having voltages matched.
d. Wrong. The synchro check relay allows breaker closure as long as frequency is within 20º (plus or minus 3.3 minutes of 12, i.e. 11:56:40 - 12:03:20) of TDC. In the question, the conditions given (i.e. 1201 o'clock) would permit closure. This is a plausible distractor

in that the synchro check relay is an active interlock which must be met when closing a breaker.

Question Number: 047
Exam Date 20090413
Exam Level R
K/A 063A3.01
Question
Given the following:

- Annunciator 47102-D, "INSTRUMENT BUS INVERTER TROUBLE", is LIT.
- SER printout identifies BRA-112 as the affected inverter.

If BRA-105 is supplying the inverter's load . . .

a. the inverter supplying load light would be ON.

b. the alternate source supplying load light would be ON.

c. annunciator 47101-A, "BRA-102 DC Voltage Low", would be ON.

d. the red circuit status light on BRA-104 for BRA-112 supply breaker would be OFF.

Answer
b.
REFERENCE
E-233
OP-KW-ARP-47102-D
System Description 38, DC and Emergency AC Electrical Distribution System, 3.4.3, pages 16-17 and 3.11, page 21.
NEW
FUNDAMENTAL
K/A Ability to monitor automatic operation of the DC Electrical System, Including: Meters, annunciators, dials, recorders, and indicating lights.
JUSTIFICATION
a. Wrong: The normal supply for BRA-112 is MCC-52C.
b. Correct: BRA-105 is the alternate source for BRA-112 and the alternate source supplying the load would be lit if it was supplying the inverter loads.
c. Wrong: Annunciator 47101-A does not activate when BRA-105 is supplying BRA-112. BRA-112 is still energized, but on alternate power.
d. Wrong: Indications for breaker status on BRA-104 does not indicate BRA-105 is supplying BRA-112. Red status light is alarm module for that breaker, not an indication of power.

Question Number: 048
Exam Date 20090413
Exam Level R
K/A 064A2.19
Question
48. The following plant conditions exist:

- Diesel Generator 'B' was started manually.
- Diesel Generator 'B' is currently running at steady state conditions.
- The KW loading is 2400 KW.
- The KVAR loading is 1600 KVAR.
- Diesel Generator 'B' is to be operated based on Attachment 'A' Figure 1 of NOP-DGM-001B, "Diesel Generator 'B' Remote Operation".

Using Attachment 'A' Figure 1 of NOP-DGM-001B, "Diesel Generator 'B' Remote Operation, determine what action should be taken by the operator to maintain safe operation of Diesel Generator 'B'.

a. Position the Speed Control Switch to LOWER in order to decrease the power factor.

b. Position the Voltage Control Switch to LOWER in order to increase the power factor.

c. Position the Diesel Generator Mode Selector Switch to AUTO to allow the governor to automatically decrease the power factor.

d. Position the Man Voltage Control Selector Switch to MAN and position the Voltage Control Switch to RAISE in order to increase the power factor.

Answer
b.
REFERENCE
OP-KW-NOP-DGM-001B, Step 5.3.11, 5.4.1 and Attachment A, Figure 1
BANK
HIGHER
Provided Reference: OP-KW-NOP-DGM-001B, Attachment A: Figure 1.
K/A Ability to (a) predict the impacts of the following malfunctions or operations on the ED/G System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of the malfunctions or operations: Consequences of High Vars on the ED/G integrity.
Justification:
a. Wrong: The operator needs to increase the power factor. This action will lower load (KW).
b. Correct: Positioning the voltage control switch will lower KVARS and raise the power factor.
c. Wrong: Placing the Diesel Generator Mode Selector Switch to Auto does not place a circuit in place to correct the power factor by controlling KVARS.
d. Wrong: This switch is a local switch and should not be operated to adjust voltage during normal operations. Also, raising the voltage will decrease the power factor

Question Number: 049
Exam Date 20090413
Exam Level R
K/A 064A4.05
Question
Given the following:

- Diesel Generator 'A' was noted making an unusual noise during its startup.
- To troubleshoot the problem requires the Diesel Generator be operated between 700 and 800 rpm for a short time.
- The Operator places the Man Voltage Control Selector in OFF position prior to the troubleshooting.

This action was taken to . . .

a. disable the overspeed trip.

b. position the fuel racks to the no-fuel position.

c. minimize the potential for motoring the generator.

d. prevent generator excitation resulting in regulator damage.

Answer
d.
REFERENCE
OP-KW-NOP-DGM-001A, Precaution & Limitation 4.7
E-2022, (A-4)
System Description 42, 3.3.1, page 8
BANK
MEMORY
K/A Ability to manually operate and/or monitor in the control room: Transfer of ED/G control between manual and automatic.
JUSTIFICATION:
a. Wrong: The Man Voltage Control Switch does not disable the overspeed trip
b. Wrong: The Man Voltage Control in off disables the exciter voltage.
c. Wrong: The EDG will not motor is the output breaker is not shut.
d. Correct: Placing the Man Voltage Control Selector to OFF will disable the exciter field and prevent damage to regulator caused by operating the EDG for extended periods of time at low speed with power to the exciter.

Question Number: 050
Exam Date 20090413
Exam Level R
K/A 073A4.01
Question
Given the following:

- Performing a Gas Decay Tank discharge
- WG-36, Waste Gas To Plant Vent Control Valve, automatically closes.

Assuming the automatic closure signal is no longer present, and it is acceptable for the release to be re-initiated, which of the following describes the operation needed to re-open WG-36, Waste Gas To Plant Vent Control Valve?

a. WG-36 control knob must be taken to close to raise control air pressure, and then may be re-opened.

b. WG-36 control knob must be taken to close to reduce control air pressure, and then may be re-opened.

c. WG-36 can be re-opened by placing its control switch at the Waste Disposal Panel in the OPEN position.

d. The automatic closure signal must be manually reset and WG-36 will then automatically re-open.

Answer
b.
REFERENCE
N-GWP-32B – Gaseous Waste Processing & Discharge System Note before step 4.2.8.e.7
E-2048
System Description 32B, 3.5, page 13
BANK
FUNDAMENTAL
K/A Ability to manually operate and/or monitor the control room: Effluent release.
JUSTIFICATION:
a. Wrong: Close the controller to lower air pressure.
b. Correct: Must be reset by running controller closed (Lowering Air pressure < 2.5 psig controller output).
c. Wrong: Must reset the valve with the control knob.
d. Wrong: The valve does not automatically open when the signal is reset.

Question Number: 051
Exam Date 20090413
Exam Level R
K/A 076K1.01
Question
What TWO systems in the Auxiliary Building are supplied with emergency makeup water via the Service Water system?

a. Safety Injection and Spent Fuel Pool.

b. Component Cooling and Spent Fuel Pool.

c. Residual Heat Removal and Safety Injection.

d. Residual Heat Removal and Component Cooling.

Answer
b.
REFERENCE
System Description 2, Service Water, Sections 2.0, 3.5, 3.6.6 and 3.6.8
OPERM 202-2, Flow Diagram Service Water System
BANK
FUNDAMENTAL
K/A Knowledge of the physical connections and/or cause-effect relationships between the SWS and the following systems: CCW system. RO Imp / SRO Imp - 3.4 / 3.3
JUSTIFICATION
Of the various answer choices available, only RHR and Safety Injection are not provided makeup from Service Water. Of the remaining choices, only the second choice is correct.
a. WRONG: Safety Injection is not provided makeup from Service Water. It does provide cooling for the SI Pump stuffing box jacket and lube oil coolers.
b. CORRECT: Answer is correct.
c. WRONG: RHR and SI are not provided makeup from Service Water.
d. WRONG - RHR is not provided makeup from Service Water. It does provide cooling flow to the RHR Pump Pit Fan Coil Units.

Question Number: 052
Exam Date 20090413
Exam Level R
K/A 076 2.4.50
Question
Given the following:

- Operating at 100% power.
- Service Water Pump Preferred Selector Switch is in the '1A' position, with the first three pumps in the sequence running.
- SW-3A and SW-3B, Service Water Header 'A' & 'B' Isolations, are both OPEN.
- Turbine Building Header Selector Switch is in position '1A'.
- Annunciator 47051-P, "SW HEADER PRESSURE LOW", is LIT.
- Annunciator 47052-Q, "TURBINE BLDG SW ISOLATION ALERT", is LIT.
- SER 93, SW header 'A' less than 82 psig is in ALARM.
- SER 841, Service Water Header 'A' pressure Low (SWI), is in ALARM.

With NO operator action, which of the following is consistent with the given alarms?

a. All Service water pumps are running.

b. No change in Service Water System alignment.

c. SW-3A, Service Water Header 'A' Isolation, CLOSED

d. SW-4A, SW Header 'A' to Turbine Bldg Hdr, CLOSED

Answer
b.
REFERENCE
OP-KW-ARP-47051-P, SW Header Pressure Low.
OP-KW-ARP-47052-Q, Turbine Bldg SW Isolation Alert.
OP-KW-ARP-47051-Q, Turbine Bldg Service Water Isolation.
NEW
HIGHER
K/A Service Water: Ability to verify system alarm setpoints and operate controls identified in the alarm response manual.
Justification:
a. Wrong: The third SW pump, 1B1, starts at 82 psig. The fourth SW pump 1B2 will not start until SW header **B** pressure reaches 78 psig; SER point 122, SW header pressure B less than 78 psig. Annunciator 47051-Q would not be alarming since SW-4A and SW-3A would not be closed. Annunciator 47051-Q alarms when both SW-4A and SW-3A are closed.
b. Correct: The alarms indicate a problem but not automatic setpoints for the system have been reached.
c. Wrong: SW-3A would be open. SW-3A closes when SW Header A pressure = 72 psig.
d. Wrong: SW-4A would not be closed. SW-4A closes automatically on the following, Turbine header selector switch is positioned to 1B or ISOL, or SW Header A pressure < 82.5 psig (SER 841) and Safety Injection Sequence Step 9.

Question Number: 053
Exam Date 20090413
Exam Level R
K/A 078K1.02
Question
Given the following:

- Air Compressor 'A' switch in AUTO
- Air Compressor 'B' switch in AUTO
- Air Compressor 'C' switch in OFF
- The station air compressor Preferred Selector Switch is aligned to Compressor 'G'.

What automatically occurs WHEN the Station and Instrument Air System air header pressure reaches 95 psig?

a. Air Compressor 'F' starts.

b. Air Compressor 'A' starts and loads.

c. SA-200, SA Header 'A' Supply Valve, starts to close.

d. SA-60, SA Crossover Pressure Control Valve, closes.

Answer
c.
REFERENCE
Dwg E-1603
Dwg E-2098
Dwg OPERM-213-1
BANK
HIGHER
K/A Knowledge of the physical connections and/or cause-effect relationships between the IAS and the following systems: Service air.
Justification:
a. Wrong: Compressor F would have started at 112 psig.
b. Wrong: Air Compressors 'A' starts at 105 psig and loads at 96 psig.
c. Correct: SA-200 starts to close at header pressure of 95 psig.
d. Wrong: SA-60 regulates air from the SA to IA headers and cycles to maintain IA header pressure at 110 psig.

Question Number: 054
Exam Date 20090413
Exam Level R
K/A 078A4.01
Question
Given the following:

- The plant has experienced a Small Break LOCA with a Loss of Off-Site Power.
- The crew has transitioned to E-1, "Loss of Reactor or Secondary Coolant".
- Air Compressors 'A', 'B' and 'C' were in AUTO before the plant trip.
- Instrument Air Header Pressure is 85 psig.

What action in the CONTROL ROOM is required to allow for starting of additional air compressors?

a. Reset both trains of Safety Injection.

b. Direct a NAO to locally reset Air Compressor 'A'.

c. Position Air Compressor 'C' control switch to ON.

d. Position Air Compressor 'B' control switch to OFF.

Answer
a.
REFERENCE
Dwg E-1603
NEW
HIGHER
K/A Ability to manually operate and/or monitor in the control room: Pressure gauges.
Justification:
a. Correct: Safety Injection must be reset to allow for starting of Air Compressor 'A'. Air compressors 'B' and 'C' are already running. Both would be running after the SI sequence is complete and IA header pressure less than 105 psig.
b. Wrong: Air Compressor 'A' does not have a local reset (Air Compressor 'C' has local controls)
c. Wrong: Air Compressor 'C' would already be running and there is no control switch in the control room for Air Compressor 'C' ('A' and 'B' have control switches in the control room)
d. Wrong: Air Compressor 'B' would already be running. Air compressor has to be reset by taking its control switch in the control room to off after SI has been reset.

Question Number: 055
Exam Date 20090413
Exam Level R
K/A 103K3.03
Question
Refueling Operations are in progress and changes in core geometry are taking place.

Which of the following would require refueling of the reactor to cease?

a. Residual Heat Removal Pump 'B' has tripped and will not restart.

b. Refueling cavity water level was found to be 24 ft above the vessel flange.

c. The containment personnel air lock inner and outer doors were damaged during the movement of equipment and will not close

d. A manual containment isolation valve was found stuck closed, and the upstream automatic containment isolation valve is inoperable.

Answer
c.
REFERENCE
Technical Specifications 3.8 refueling operations
NEW
HIGHER
K/A Knowledge of the effect that a loss or malfunction of the containment system will have on the following: Loss of containment integrity under refueling operations.
Justification:
a. WRONG: Per Technical Specification 3.8.a.4, at least one Residual Heat removal pump shall be operable. This is a plausible distractor because Residual Heat Removal Pumps are covered by Technical Specification 3.8 and changes to RH status may result in the suspension of refueling operations.
b. WRONG: Per Technical Specification 3.8.a.10, the minimum water level above the vessel flange shall be maintained at 23 ft. This is a plausible distractor because refueling level is addressed in technical specification 3.8 and lowered below the specified value would result in suspension of refueling operations.
c. CORRECT: Per Technical Specification 3.8.a.1.a, at least one door in each personnel air lock shall be capable of being closed in 30 minutes or less during Refueling Operations.
d. WRONG: Per Technical Specification 3.8.a.1.b, each line that penetrates containment and which provides a direct air path from containment atmosphere to the outside atmosphere shall have a closed isolation valve or an operable automatic isolation valve. This is a plausible distractor because containment isolation valves are covered by Technical Specification 3.8 and a loss of containment isolation capability would result in a suspension of refueling operations. The manual isolation valve is in the closed position and containment closure requirement is met.

Question Number: 056
Exam Date 20090413
Exam Level R
K/A 002A2.03
Question
Given the following:

- Current time is 1400.
- ES-0.2, "Natural Circulation Cooldown", is being implemented.
- RXCPs are unavailable.
- All CRDM fans are off and CANNOT be started.
- Letdown is NOT isolated.
- SI IS blocked.
- RCS cooldown is in progress.
- RCS cold leg temperature just lowered to 339°F.
- RCS Pressure is 1450 psig.

Using Attachment B of ES-0.2, "Natural Circulation Cooldown", determine the crew's actions and basis.

a. Hold temperature and pressure stable until 2300 to prevent damage to the CRDM coils due to overheating.

b. Cooldown the RCS to < 200°F at 25°F per hour while maintaining RCS pressure >1400 psig to prevent void formation in the U-tubes.

c. Maintain RCS pressure >1400 psig until 2300 while cooling down the RCS to 275°F to minimize void formations in the Reactor Vessel head.

d. Depressurize the RCS while holding RCS cold leg temperature stable until 2300 to ensure the upper head fluid temperature is equal to the cold leg fluid temperature.

Answer
c.
REFERENCE
ES-0.2, Natural Circulation Cooldown, Step 18, ATTACHMENT B
ES-0.2 Background Document, Step 18
BANK
HIGHER
Provided Reference: ES-0.2, ATTACHMENT B
K/A Ability to (a) predict the impact of the following malfunctions or operation on the RCS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequence of those malfunctions or operations: Loss of forced circulation.
Justification:
a. Wrong. During natural circulation cooldown with no CRDM fans, the concern is preventing void formation the reactor vessel head NOT CRDM coils overheating.
b. Wrong. Attachment B of ES-0.2 does not allow cooldown to 200°F at this time. During natural circulation cooldown with no CRDM fans, the concern is preventing void formation the reactor vessel head NOT void formation in the S/G U-tubes.
c. Correct. Maintaining RCS pressure greater than 1400 psig for 9 hours and cooldown to 275°F is consistent with Attachment B of ES-0.2 for minimizing void formation in the

reactor vessel head.

d. Wrong. Depressurizing the RCS increases the chances of void formation in the reactor vessel head.

Question Number: 057
Exam Date 20090413
Exam Level R
K/A 014K3.02
Question
Given the following:

- Plant is operating at 100%.
- Bus 62 experiences a LOCKOUT.

Which of the following is INOPERABLE?

a. Step Counters.

b. All control rods.

c. Axial Flux Difference.

d. Rod Position Deviation Monitor.

Answer
d.
REFERENCE
AOP-CRD-001, Control Rod Drive System Malfunction, ATTACHMENT B, Step B1.
Technical Specification 3.10.f, 3.10.i.
NEW
HIGHER
K/A Knowledge of the effect that a loss or malfunction of the RPIS will have on the following: Plant Computer.
Justification:
a. Wrong. Step counters are not affected by loss of bus 62
b. Wrong. Control rods are not inoperable according to technical specifications.
c. Wrong. Axial Flux Difference is not affected by the loss of bus 62
d. Correct. IRPI indication is lost with a loss of bus 62. The IRPI input is also lost to PPCS, which uses the input to calculate rod deviation.

Question Number: 058
Exam Date 20090413
Exam Level R
K/A 015A4.02
Question
Given the Following:

- Plant Startup is in progress.
- Reactor Power is 15% and increasing.
- Source and Intermediate Range Channel 1 nuclear instrument FAILS: Fission Chamber NE-35, Detector 28044.

Which of the following would NOT be accurate because of the failure of NE-35?

a. NI-41 delta flux.

b. Calculated Tave-Tref.

c. NI-36D SUR.

d. N-31B count rate.

Answer
d.
REFERENCE
Dwg E-2051-1
Dwg E-2051-2
MODIFIED
HIGHER
K/A Ability to manually operate and/or monitor in the control room: NIS indicators.
Justification:
a. Wrong. NI-41 is a power range NI and source and intermediate channels do not have an input into delta flux
b. Wrong. Intermediate and Source range indications are not an input into calculated Tave-Tref.
c. Wrong. NI-36D is a channel 2 indicator.
d. Correct. NI-31B is a channel 1 indicator from NE-35

Question Number: 059
Exam Date 20090413
Exam Level R
K/A 028K5.03
Question
Which of the following is NOT a direct source of hydrogen gas in Containment after a Loss Of Coolant Accident?

a. Radiolysis of water.

b. Metal-water reaction.

c. SI Accumulator gas injection.

d. Chemical corrosion of materials by spray liquids.

Answer
c.
REFERENCE
USAR section 5.8.2.1
BANK
FUNDAMENTAL
K/A Knowledge of the operational implications as they apply to the HRPS: Sources of hydrogen in containment.
Justification:
a. Wrong. Radiolysis is a source of hydrogen
b. Wrong. Metal-water reaction is a source of hydrogen
c. Correct. Nitrogen gas is used to pressurize the SI accumulators and not a source of hydrogen
d. Wrong. Chemical corrosion of material by spray liquids is a source of hydrogen.

Question Number: 060
Exam Date 20090413
Exam Level R
K/A 029A2.03
Question
Given the following:

- The Reactor is at 10% power.
- A Containment 2" vent is planned to begin using Hydrogen Recombiner System Train 'B'. N-RBV-18B, "Reactor Bldg Vent System Cold Operation and Making Releases", Section 4.1.3.

Why is the Hydrogen Recombiner System Train ‘A’ NOT used to vent containment via LOCA-2A, Post-LOCA Hydrogen Cntmt Vent Isol 'A', and LOCA-100A, Post-LOCA Hydrogen to Recombiner 'A'?

a. Hydrogen Recombiner System Train 'A' vents at a higher flow rate than Train B and requires a release permit.

b. Hydrogen Recombiner System Train 'A' CANNOT be operated from the Post LOCA H_2 Control Station panels.

c. Hydrogen Recombiner System Train 'A' isolation valves do NOT receive a Containment Vent Isolation Train 'A' signal.

d. Opening of LOCA-2A and LOCA-100A, with the RCS cold leg temperatures > 200°F requires NRC prior to opening.

Answer
c.
REFERENCE
Dwg E-2068
Dwg OPERM-403
N-RBV-18B, Reactor Bldg Vent System Cold Operation and Making Releases, 4.1.3.
MODIFIED
HIGHER
K/A Ability to (a) predict the impacts of the following malfunctions or operations on the Containment Purge System and (b) based on those predications use procedures to correct, control, or manipulate the consequences of those malfunctions or operations: Startup operations and the associated required valve lineups.
Justification:
a. Wrong: Both Trains have same capability.
b. Wrong: "A" Train can be operated from the Post LOCA H_2 control station panels.
c. Correct: "A" train does not receive any containment vent isolation signals.
d. Wrong: The NRC notification is required for the 36" vent when RCS > 200ºF.

Question Number: 061
Exam Date 20090413
Exam Level R
K/A 034G2.1.32
Question
The design of the New Fuel Elevator prevents inadvertently raising an irradiated fuel assembly.

To accomplish this design feature the New Fuel Elevator will move in the up direction when the . . .

a. lower pushbutton is disabled by a key interlock.

b. load sensing station senses less than 1000 lbs.

c. lower limit switch resets the raise pushbutton control.

d. elevator interlock bypass key is used to bypass the raise limit switch.

Answer
b.
REFERENCE
Dwg E-2052
NEW
FUNDAMENTAL
K/A Fuel handling Equipment: Ability to explain and apply system limits and precautions.
Justification:
a. Wrong. There is no key interlock for the lower pushbutton.
b. Correct. The load sensing station will not allow the new fuel elevator to move up unless it senses less than 1000 lbs or the elevator key bypass switch is in bypass.
c. Wrong. The lower limit resets the lower pushbutton control. The raise limit switch resets the up push button control.
d. Wrong. The elevator interlock bypass is for the load sensing station not the raise limit switch.

Question Number: 062
Exam Date 20090413
Exam Level R
K/A 041K6.03
Question
Given the following:

- Holding at 75% turbine power.
- PPCS Point P0487G, P486 Deviation, is in ALARM.
- The Balance of Plant Operator reports that P-486, Turbine Impulse Pressure, is off scale high.

In response to P-486, Turbine Impulse Pressure, failing high . . .

a. control bank 'D' rods will step out.

b. SG 'B' steam flow indication will increase.

c. condenser steam dumps will not function in Tave mode.

d. blocking of reactor trips associated with P-10 is prevented.

Answer
c.
REFERENCE
Dwg E-1626
Dwg E-2051-2
Dwg XK-100-151
Dwg XK-100-156
OP-KW-AOP-MISC-001, ATTACHMENT N
NEW
HIGHER
K/A Knowledge of the effects of a loss or malfunction of the following will have on the SDS: Controller and positioners including ICS, S/G, CRDS.
Justification:
a. Wrong: Tave-Tref will not change with this failure. P-485, Turbine Impulse pressure is the input to Tref not P-486. (Dwg XK-100-151)
b. Wrong: P-486 is not an input into steam flow indication.
c. Correct: Condenser and atmospheric steam dumps will not arm because P-486 provides the loss of load signal to arm the condenser steam dumps (E-1626).
d. Wrong: Permissive P-7 is affected via P-13 (Turbine Impulse Pressure). Not permissive P-10 which is NIs. (E-2051-2)

Question Number: 063
Exam Date 20090413
Exam Level R
K/A 055A3.03
Question
Given the following:

- The weekly Air Inleakage Test is being performed using the local digital air flow calibrator per NOP-AR-001, "Air Removal System", section 5.8
- An air flow reading has just been obtained, when a high alarm occurs on R-15, Air Ejector Exhaust Monitor.

AR-6, Air Ejector Discharge Vent Valve, will . . .

a. NOT re-position, since it is already aligned to the suction of the Aux. Building ventilation exhaust fans.

b. automatically re-align to direct the air ejector flow to the suction of the Aux. Building ventilation exhaust fans.

c. NOT re-position, since it is already aligned to the suction of the Spent Fuel Pool ventilation exhaust fans.

d. automatically re-align to direct the air ejector flow to the suction of the Spent Fuel Pool ventilation exhaust fans.

Answer
b.
REFERENCE
E-1607
OPERM-212
OPERM-601
NOP-AR-001, Air Removal System, Step 5.8.2
BANK
FUNDAMENTAL
K/A Ability to monitor automatic operation of the CARS, including: Automatic diversion of CARS exhaust.
Justification:
a. Wrong. AR-6 gets re-positioned during the weekly air inleakage test.
b. Correct. AR-6 was re-aligned during the procedure and will re-position on the high alarm associated with R-15.
c. Wrong. AR-6 does not align to SFP ventilation exhaust fans.
d. Wrong. AR-6 does not align to SFP ventilation exhaust fans.

Question Number: 064
Exam Date 20090413
Exam Level R
K/A 056K1.03
Question
Given the following:

- Rx Power 99.4%
- Condensate Pump 'B' trips on overcurrent.

With NO operator action . . .

a. C-701, Condensate Recirculation Control Valve, fails open when Condensate Pump 'B' trips.

b. FW-101A, Feedwater Pump 'A' recirculation valve, modulates to maintain recirculation flow.

c. C-13, Condensate Bypass LP FW Heaters, will open to attempt to maintain feed water pump suction pressure.

d. FW-12A, SG 'A' Feedwater Isolation Valve, will close because of the reactor trip signal generated from the low level on both Steam Generators.

Answer
c.
REFERENCE
Dwg E-1615
OP-KW-ARP-47063-P, Feedwater Htr Bypass Alert.
Dwg E-1625
OP-KW-ARP-47063-A, SG A Feedwater Isolation.
NEW
HIGHER
K/A Knowledge of the physical connection and/or cause-effect relationships between the Condensate System and the following systems: MFW.
Justification:
a. Wrong - C-701 will continue to modulate to maintain > 2200 gpm condensate flow. It would close on interlock if 'A' condensate pump was stopped. The valve is a FO valve.
b. Wrong - If open, FW-101A would close when 'A' Feed pump breaker opens.
c. Correct - Feed water pump suction pressure will decrease to less than 220 psig when 'B' condensate pump trips causing C-13 to open.
d. Wrong - FW-12A does not go closed on a reactor trip signal. Closes on FW isolation signal - High SG Level or SI. The reactor will trip on low steam generator water level.

Question Number: 065
Exam Date 20090413
Exam Level R
K/A 086K4.01
Question
With NO operator action, what is the fire system response to a loss of power to MCC-52D?

a. The Fire System Header pressure will rise to 140 psig because Fire Pumps '1A' and '1B' will automatically start due to loss of voltage.

b. The Fire System header pressure will stabilize at 143 psig when Fire Pump '1B' automatically starts on loss of power to Fire Pump '1A'.

c. The Fire System Header pressure will cycle between 102 to 110 psig after Fire Pump '1B' automatically starts on low header pressure.

d. The Fire System header pressure will lower to 110 psig at which time Fire Pump '1A' will automatically start and raise header pressure.

Answer
d.
REFERENCE
E-1619, Integrated Logic Diagram – Fire Protection System
Operator Aid # 02-22, MCC-52D
BANK
FUNDAMENTAL
K/A Knowledge of the Fire Protection System design feature(s) and/or Interlocks which provide for the following: Adequate supply of water for FPS.
Justification:
The loss of power to MCC-52D affects the Fire Protection System Jockey Pump. Fire Header pressure will then begin to drop.
a. Wrong. There is no loss of voltage signal that starts the fire pumps.
b. Wrong. Fire Pump '1A' does not lose power.
c. Wrong. Fire pump 'B' would start at header pressure of 102 psig but it would have to be locally stopped, it does not cycle between 102 psig and 110 psig to maintain fire header pressure between 102 psig and 110 psig.
d. Correct. Fire Pump '1A' will start automatically when header pressure reaches 110 psig, at which time header pressure will rise.

Question Number: 066
Exam Date 20090413
Exam Level R
K/A 2.1.17
Question
Given the following:

- The plant is at 100%.
- Annunciator 47061-B, "SG A SF > FF", is LIT
- S/G 'A' Level is rising slowly
- S/G 'A' Pressure Instrument PI-468, Red Channel, indicates Off-Scale High.
- S/G 'A' Stm Flow Instrument FI-464, Red Channel, indicates Off-Scale High.

The control board operator should IMMEDIATELY . . .

a. attempt to manually close SD-3A, S/G 'A' PORV.

b. shift the steam dump controller, HC-484, to manual.

c. place S/G 'A' Steam Flow Selector Switch to FI-465, White Channel.

d. position Main Steam Dump Control Mode Selector to RESET then STM PRESS.

Answer
a.
REFERENCE
OP-KW-AOP-GEN-001, Operator Immediate Actions, ATTACHMENT D
Dwg. E-1627
Dwg. XK-100-556
NEW
HIGHER
K/A Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior and instrument interpretation.
Justification:
a. Correct: The failure of the steam pressure instrument will cause the 'A' SG PORV to open and per the operator immediate actions of OP-KW-AOP-GEN-001 the operator should attempt to shut SD-3A from the control room
b. Wrong: Steam Dump controller is affected by a failure of turbine impulse pressure instruments, not SG pressure instruments.
c. Wrong: Placing the Steam Flow Selector Switch to FI-465 is not an immediate action for the failure.
d. Wrong: Steam Dump Mode is not affected by Steam Generator pressure.

Question Number: 067
Exam Date 20090413
Exam Level R
K/A 2.1.37
Question
What is the continuous/steady state startup rate (SUR) allowed by the Dominion Reactivity Management procedure, OP-AP-300, and the Kewaunee Reactor Startup procedure GOP-104, "Startup From Hot Shutdown To Hot Standby", during an approach to criticality?

a. 0.30 dpm

b. 0.50 dpm

c. 0.75 dpm

d. 1.00 dpm

Answer
d.
REFERENCE
OP-KW-GOP-104, Section 4.21
OP-AP-300, Attachment 2.3
NEW
FUNDAMENTAL
K/A Knowledge of procedures, guidelines, or limitations associated with reactivity management.
Justification:
a. Wrong: 0.30 dpm is the max SUR allowed to approach the POAH per OP-KW-GOP-105
b. Wrong: 0.50 dpm is the max SUR allowed from critical to 1E-3 for data per OP-KW-GOP-104
c. Wrong: 1.0 dpm is the limit.
d. Correct: 1.0 dpm is the limit.

Question Number: 068
Exam Date 20090413
Exam Level R
K/A 2.2.15
Question

Given the following:

- Plant is at 100% power
- Makeup Water System Checklist has been verified

Given N-MUP-27A-CL, "Makeup Water System Checklist", and OPERM-209-2, Flow Diagram of Make-up and Demineralized Water Systems determine which of the following components is NOT supplied from the Demineralized Water Header.

a. TSC Computer Room HVAC.

b. Waste Gas Compressor Seals.

c. Membrane Contactor Vacuum Pump.

d. Secondary Sampling System Rad Detector.

Answer
b.
REFERENCE
N-MUP-27A-CL, Makeup Water System Checklist
OPERM-209-2, Flow Diagram Make-up and Demineralized Water Systems
XK-100-132, Flow Diagram Waste Disposal System
NEW
HIGHER
Provided Reference: N-MUP-27A-CL; OPERM-209-2
K/A Ability to determine the expected plant configuration using design and configuration control documentation, such as drawings, line-ups, tag-outs, etc.
Justification:
a. WRONG. The Reactor Make-up Water Storage Tank is normally supplied from Demin Water
b. CORRECT. The Waste Gas Compressor Seals are normally supplied from the Reactor Make-up Water Header.
c. WRONG. Per OPERM-209-2, the Membrane Contactor Vacuum Pump is supplied by the Demin Water Header.
d. WRONG. The Secondary Sampling System Rad Detector is normally supplied by the Demin Water Header

Question Number: 069
Exam Date 20090413
Exam Level R
K/A 2.2.36
Question
Transformer Bank T-10 is going to be removed from service per MOP-SUB-001, Removal and Restoration of Transformer Bank T-10, during power operations at 50% NI power.

To remove Transformer Bank T-10 from service the following equipment alignment is required for PRA concerns:

- Busses 1, 3, 4 & 5 on the RAT
- Busses 2 and 6 on the MAT
- MCC 5262 lined up to Bus 62
- BRA-106 lined up to normal
- BRB-106 lined up to normal
- Fast transfers for 1-5 and 1-6 disabled except to Emergency Diesel Generators
- Fast transfers of bus 1-1 to RAT is disabled

The following loads are aligned such that they are NOT running and will not start coincident with SI actuation.

-	Charging Pump 'C' control switch	PULLOUT
-	Boric Acid Transfer Pump 1A control switch	PULLOUT
-	Boric Acid Immersion Heaters 1A1 and 1B1 breakers	OFF
-	Spent Fuel Pool Pump 1A breaker	OFF
-	Dome Vent Fan 1A control switch and breaker	OFF
-	Turning Gear Oil Pump breaker	OFF
-	Instrument Air Dryer 1A control switch	OFF

The following substation alignment is required.

-	TA-199 OCB	OPEN
-	TA-199 OCB Bus and Transformer Side Disconnects	OPEN
-	1066E OCB	OPEN
-	1066E OCB Transformer Side Disconnect	OPEN
-	1066W OCB	OPEN
-	1066W OCB Transformer Side Disconnect	OPEN

Which equipment alignment requires entry into a Limiting Condition for Operation?

a. Bus 1-5 aligned to the RAT.

b. Spent Fuel Pool Pump 1A breaker OFF.

c. Boric Acid Transfer Pump 1A control switch in PULLOUT.

d. TA-199 OCB Bus and Transformer Side Disconnects OPEN.

Answer
d.
REFERENCE

OP-KW-MOP-SUB-001, Removal and Restoration of Transformer Bank T-10.5.1.6.b, 5.1.6.e, 5.2.6 and 5.4
N-SUB-59, Substation Equipment Switching Procedure, 2.2, 2.8 and Section 4.6
Technical Specification, 3.7.a.2, 3.7.b.1 and 3.2.a
NEW
HIGHER
K/A Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of the limiting conditions for operations.
JUSTIFICATION:

a. Wrong: Bus 1-5 aligned to RAT is not referenced in Technical Specifications. Bus 1-5 and Bus 1-6 is referenced in Technical Specification 3.7.
b. Wrong: Spent Fuel Pool Pump 1A does not have a T.S. associated with it. T.S. 3.8 refers to the spent fuel sweep system.
c. Wrong: T.S. 3.2 there has to be at least one flow path for boric acid to the core. The basis refers to using the CVC system, RWST and SI pumps. Having Boric Acid Transfer pump 1A control switch to pullout does not eliminate all flow paths for boric acid to the core.
d. Correct: TA-199 OCB Bus and Transformer Side Disconnects OPEN removes T-10 from service and makes the TAT inoperable then in a 7day LCO per T.S. 3.7.b.1

Question Number: 070
Exam Date 20090413
Exam Level R
K/A 2.2.41
Question
Given the following:

- VCT level is lowering slowly.
- DDT level is rising slowly.
- TI-127, Regen Heat Exchanger Letdown temperature, rose slightly
- PI-135, Letdown Heat Exchanger Outlet pressure, is unchanged
- LD-10, Letdown Cont Pressure, has throttled closed by 5%

Using the attached OPERXK-100-35 and OPERXK-100-36, Flow Diagrams for the Chemical & Volume Control System, and OPERM-350, Flow Diagram, Reactor Plant Misc. Vents, Drains & Sump Pump Piping, the leak in the CVC System is most likely originating from . . .

a. LD-10, Letdown Cont Pressure, stem leakoff.

b. CVC-7, Chg Line Flow Cont Vlv, stem leakoff.

c. LD-13, Letdown Line Relief Valve, seat leakage.

d. CVC-264, Seal Water Return Filter Drain, seat leakage.

Answer
a.
REFERENCE
OPERXK-100-35, Flow Diagram, Chemical & Volume Control Sys.
OPERXK-100-36, Flow Diagram, Chemical & Volume Control Sys.
OPERM-350, Flow Diagram, Reactor Plant Misc. Vents, Drains & Sump Pump Piping
AOP-RC-001, Step 12
NEW
HIGHER
Provided Reference:

OPERXK-100-35, Flow Diagram, Chemical & Volume Control System

OPERXK-100-36, Flow Diagram, Chemical & Volume Control System

OPERM-350, Flow Diagram, Reactor Plant Misc. Vents, Drains & Sump Pump Piping
K/A - Ability to obtain and interpret station electrical and mechanical drawings.
JUSTIFICATION:

a. Correct. The symptoms presented represent a leak in the letdown line of the CVC System. Based on these indications the leak must be downstream of the LD HX. Of the possible choices, only two (LD-10 and LD-13) are in the letdown line. Based on DDT level change, the leak is going into the DDT. Only LD-10 stem leakoff is directed to the DDT making it the only valid choice.
b. Wrong. The symptoms presented represent a leak in the letdown line of the CVC System. CVC-7 is in the charging line. CVC-7 stem leakoff is directed to the DDT.
c. Wrong. The symptoms presented represent a leak in the letdown line of the CVC System. Although LD-13 is in the letdown line, seat leakage is directed to the RXCP Seal Return line which then goes to the VCT. This is a plausible distractor in that LD-13 symptoms would be reflected in the letdown parameters but not the DDT level change.

d. Wrong. The symptoms presented represent a leak in the letdown line of the CVC System. CVC-264 is in the return line from the RXCP seals. The CVC-264 seat leakage is directed to the Sludge Interceptor Tank.

Question Number: 071
Exam Date 20090413
Exam Level R
K/A 2.3.11
Question
The Control Room operators are responding to a Steam Generator Tube Rupture.

In order to cool down the RCS and establish required subcooling margin, the operators dump steam to the Condenser using the intact S/G.

This method of RCS cooldown is preferred over dumping steam through the PORV of the INTACT S/G because it minimizes . . .

a. radiological releases.

b. RCS subcooling requirements.

c. shrink experienced by the RCS.

d. thermal shock to the reactor vessel.

Answer
a.
REFERENCE
E-3, Step 11
Bkg Doc E-3 Step 11
BANK
HIGHER
K/A Radiation Control: Ability to control radiation releases. RO Imp / SRO Imp - 3.8 / 4.3
Justification:
a. Correct: This will conserve FW and contain Rad releases to the condenser.
b. Wrong: The C/D will control RCS subcooling.
c. Wrong: Inventory issue with leak and cooldown has no direct affect from the Steam Dump.
d. Wrong: Not a concern, based on reducing RCS pressure to match S/G to minimize leak.

Question Number: 072
Exam Date 20090413
Exam Level R
K/A 2.3.14
Question
Why is it preferred that Post Accident Leakage Control System be activated per AOP-MDS-002, "Post Accident Leakage Control System", prior to establishing containment sump recirculation during a Large Break LOCA?

a. To prevent boron dilution of Containment Sump 'B' during containment sump recirculation.

b. The Post Accident Leakage Control System is required to obtain containment sump samples during containment sump recirculation.

c. High radiation levels could prohibit diverting the Deaerator Drain Tank Vent to containment during containment sump recirculation.

d. A large differential pressure across CVC-215B, Seal Water Filter Bypass valve, will develop during containment sump recirculation and prevent it from opening.

Answer
c.
REFERENCE
AOP-MDS-002, Post Accident Leakage Control, NOTE Step 2
NEW
HIGHER
K/A Knowledge of the radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities. RO Imp / SRO Imp - 3.4 / 3.8
Justification:
a. Wrong: Dilution of the sump is not a concern at this time.
b. Wrong: Though the sample lines are the source of the radiation, the system does not need to be activated to sample from the High Rad Sample Room
c. Correct: During a LB LOCA radiation from back leakage of sample line may prohibit access to WG-309, DDT vent to containment.
d. Wrong: Containment Sump Recric does not cause a D/P across CVC-215B, though the valve is operated in AOP-MDS-002. CVC-215B is opened to lower Aux Building rad levels by keeping particles out of the filters.

Question Number: 073
Exam Date 20090413
Exam Level R
K/A 2.4.16
Question
Given the following:

- A loss of BOTH offsite and onsite power has occurred, resulting in a Reactor Trip.
- Immediately the STA reports the status of Critical Safety Functions as follows:

  SUBCRITICALITY - Green
  CORE COOLING - Orange
  HEAT SINK - Red
  INTEGRITY - Green
  CONTAINMENT - Green
  INVENTORY - Yellow

In response to the above conditions the crew should implement . . .

a. ECA-0.0, "Loss of All AC Power".

b. FR-I.3, "Response to Voids in Reactor Vessel".

c. FR-C.1, "Response to Inadequate Core Cooling".

d. FR-H.1, "Response to Loss of Secondary Heat Sink".

Answer
a.
REFERENCE
ECA-0.0, Loss of All AC, Note prior to step 3
Back ground ECA-0.0 for note prior to step 3.
UG-0, Users guide for Emergency and abnormal procedures, section 6.2.3
BANK
HIGHER
K/A Knowledge of the EOP implementation hierarchy and coordination with other support procedures or guidelines such as, operating procedures, abnormal operating procedures, and severe accident management guidelines.
JUSTIFICATION
a. Correct: Per ECA-0.0 , ECA-0.0 takes precedent over the other procedures.
b. Wrong: Per ECA-0.0, ECA-0.0 takes precedent over the other procedures.
c. Wrong: Per ECA-0.0, ECA-0.0 takes precedent over the other procedures.
d. Wrong: Per ECA-0.0, ECA-0.0 takes precedent over the other procedures.

Question Number: 074
Exam Date 20090413
Exam Level R
K/A 2.4.21
Question
Given the following:

- Reactor Trip and Safety Injection actuated 20 minutes ago.
- The Reactor Trip and Safety Injection were caused by a Steam Line Break on Main Steam Header 'A' inside Containment.
- Transition from E-0, "Reactor Trip or Safety Injection," is in progress.

Parameter	Value	Trend
S/G 'A' Narrow range level	Off Scale Low	
S/G 'B' Narrow range level	14%	Slowing Rising
ALL RCS cold leg temperatures	265ºF	Slowly Lowering
SI Flow	400 gpm	Slowing Rising
Total AFW Flow	100 gpm	Slowing Rising
Containment Pressure	27 psig	Slowly Lowering

What CSF would be expected for these plant conditions?

a. A RED path on Integrity.

b. A RED path on Heat Sink.

c. A YELLOW path on Integrity.

d. An ORANGE path on Containment.

Answer
a.
REFERENCE
FR-0, Critical safety Function Status Trees.
MODIFIED
HIGHER
K/A - Knowledge of the parameters and logic used to assess the status of critical safety functions, such as reactivity control, core cooling and heat removal, reactor coolant system integrity, containment conditions, radioactivity release, etc.
Justification:
a. Correct: All RCS Cold leg temperatures are less than 274°F
b. Wrong: One SG level is greater than 13% for adverse containment.
c. Wrong: All RCS Cold leg temperatures are less than 274°F
d. Wrong: There is a least one ICS fan running. ICS actuated, CFCU's operate in emergency mode

Question Number: 075
Exam Date 20090413
Exam Level R
K/A 2.4.35
Question
Given the following:

- At 11:00 today the site experienced a Loss of All AC power
- The crew is performing actions of ECA-0.0, "Loss of All AC Power".

What is an ECA-0.0, "Loss of All AC Power", time critical operator action and the reason for its performance?

a. Energize Bus 52 within 60 minutes to prevent depletion of station emergency batteries.

b. Opening of Relay Room panel room doors within 60 minutes to ensure that temperatures do not exceed 120°F.

c. Close CVC-212, RXCP Seal Water Return Isolation Valve, within 10 minutes to prevent RCS inventory loss in excess of one Charging Pump.

d. Close MU-2A, Condensate Normal Makeup Inlet Valve, within 10 minutes to ensure adequate CST level for 4 hours of decay heat removal.

Answer
d.
REFERENCE
GNP-05.16.06, Time Critical Operator Actions – 5. Respond to a station blackout
NEW
FUNDAMENTAL
K/A Knowledge of local auxiliary operator tasks during and emergency and the resultant operational effects.
JUSTIFICATION:
a. Wrong. Energize bus 52 is for restoring charging.
b. Wrong. Relay room doors need to be opened within 30 minutes
c. Wrong. Close CVC-212, though in ECA-0.0 is not a time critical operator action
d. CORRECT. MU-2A must be closed within 10 minutes for CST inventory

Question Number: 076
Exam Date 20090413
Exam Level S
K/A 017AA2.02
Question
Given the following:

At 0700
- Reactor Power is 1.2%, plant startup is in progress.
- Annunciator 47052-C, "CNTMT EMERG DISCH DMPRS ABNORMAL", is LIT
- RBV-150A, Containment Fan Coil Unit 'A' Emergency Discharge Damper, has failed OPEN.
- Actions taken per ARP-47052-C, "CNTMT EMERG DISCH DMPRS ABNORMAL", did NOT CLOSE RBV-150A, Containment Fan Coil Unit 'A' Emergency Discharge Damper.

At 0714
- Reactor Power is 1.4%.
- TLA-20, "4160V Stator Temperature Hot", is LIT.
- Computer Point T0437A, Reactor Coolant Pump 'B' Stator, is in ALARM at 112°C and rising at 1°C per minute.

At 0715
- Reactor Power is 1.5%.
- Annunciator 47014-K, "RXCP Vibration Abnormal", is LIT
- Computer Point V8198A, RXCP 'B' Motor X Vibration, is in ALARM at 3.2 mils and rising slowly
- Computer Point V8199A, RXCP 'B' Motor Y Vibration, is in ALARM at 3.6 mils and rising slowly.

What action should the Unit Supervisor take in response the above conditions?

a. Declare RXCP 'B' INOPERABLE per T.S. 3.1.a and perform a normal plant shutdown to Hot Shutdown.

b. Enter AOP-RC-005, "Abnormal RXCP Operation" and direct tripping the reactor if RXCP 'B' Motor vibration $\geq$ 5.0 mils.

c. Direct Tripping of the Reactor, Enter E-0, "Reactor Trip or Safety Injection", Stop RXCP 'B' and cooldown the plant to less than 200°F in the next 36 hours.

d. At 0700 declare 'A' train of CFCUs INOPERABLE per T.S. 3.3.c, suspend any mode changes and enter a 72 hour LCO for one train of CFCUs INOPERABLE.

Answer
b.
REFERENCE
ARP 47052-C, CNTMT EMRG DISCH DMPRS Abnormal.
ARP 47014-K, RXCP Vibration Abnormal.
AOP-RC-005, Abnormal RXCP Operation, Step 1.
Technical Specification 3.1.a and Bases
Technical Specification 3.3.c and Bases

ARP-47033-45, 4160V Stator Temperature High, Steps 1-8.
NEW
HIGHER
K/A Ability to determine and interpret the following as they apply to the Reactor Coolant Pump Malfunctions (Loss of RC Flow): Abnormalities in RCP air vent flow paths and/or oil cooling system.
JUSTIFICATION:
a. Wrong. Technical Specification 3.1.a only states that RXCPs have to be in operation when in the operating mode. The RXCPs are still operating and have not reached any trip criteria.
b. Correct. AOP-RC-005 is the correct procedure to enter for the indications given and provides direction for tripping the reactor and stopping the affected RXCP if any of the criteria is reached on the foldout page. 5.1 mils is greater than the trip criteria of 5 mils.
c. Wrong. No trip criteria are reached at this time. And the requirement to cooldown the plant to 200°F in 36 hours does not apply.
d. Wrong. RBV-150A has failed to its ECCS position. The CFCU train remains operable.

Question Number: 077
Exam Date 20090413
Exam Level S
K/A 025AA2.05
Question
Given the following:

- The plant has been shutdown for two weeks and 4 hours.
- RCS temperature is 80°F.
- All Charging Pumps are unavailable due to maintenance.
- RHR is aligned in a normal cooldown lineup with the following valves open:
  RHR-1A and RHR-2A, RCS Loop A Supply To RHR Pumps
  RHR-1B and RHR-2B, RCS Loop B Supply To RHR Pumps
  RHR-10A and RHR-10B, Cross Connect
  RHR-100A and RHR-100B, Heat Exchanger Bypass Line
- RHR pump 'A' was running but was shutdown due to indications of cavitation.
- RCS refueling level is 9.95%.
- RHR pump 'B' is AVAILABLE.
- The Shift Manager has decided to refill the RCS using RHR pump 'B' per AOP-RHR-003, "Loss of RHR While Operating At Reduced Inventory", step 11.a RNO 11.a.2.

Using the attached AOP-RHR-003, "Loss of RHR While Operating At Reduced Inventory", Attachments C, D and F, determine when saturation temperature will be reached in the RCS and the maximum allowable RHR flowrate when establishing makeup to the RCS using RHR pump 'B'.

a. 19 minutes; 2900 gpm

b. 21 minutes; 1500 gpm

c. 34 minutes; 2000 gpm

d. 54 minutes; 1350 gpm

Answer
c.
REFERENCE
AOP-RHR-003, Loss Of RHR While Operating At Reduced Inventory Conditions
N-RHR-34, Residual Heat Removal System Operation. P&L 2.8
ES-1.3, Step 14.b
NEW
HIGHER
Provided Reference: From AOP-RHR-003:
- **Attachment C, Approximate Heatup Rates**
- **Attachment D, Time To Reach Saturation Vs. Time Shutdown**
- **Attachment F, RCS Level Vs. RHR Flow**

K/A Ability to determine and interpret the following as they apply to the Loss of Residual Heat Removal System: Limitations on LPI flow and temperature rates of change.
JUSTIFICATION:
a. Wrong. 19 minutes is the value obtained if the "Initial Temp 140°F" line of Attachment D was inadvertently used to look up the time to saturation. Using AOP-RHR-003,

ATTACHMENT F, “RCS LEVEL VS RHR FLOW,” the RHR intake flow per loop at 9.95% Refueling Level corresponds to a 1450 gpm RHR intake per loop suction flow value. Since both sets of loop suction valves are open, the limiting flow for RHR would be 2900 gpm, if both RHR Pumps were running.

b. Wrong. 21 minutes is the value obtained if the 80°F starting temperature was divided by the expected heatup rate of 3.9°F. 1500 gpm is an established RHR flow value to be obtained when establishing Containment Sump recirc using RHR (ES-1.3, Step 14.b).

c. Correct. The plant had been shutdown for 340 hours (14 days x 24 hrs). Looking up 340 hours on Attachment C gives a corresponding heat up rate of 3.9°F/hr. Subtracting the initial RCS temperature (80°F) from 212°F results in 132°F from current RCS temperature to saturation temperature, Dividing 132°F by 3.9°F/hr results in 33.84 minutes which approximates to 34 minutes. A Precaution and Limitation (2.8) from N-RHR-34 states, “Individual RHR flow rate shall NOT exceed 2000 gpm.” Using AOP-RHR-003, ATTACHMENT F, “RCS LEVEL VS RHR FLOW,” the RHR intake flow per loop at 9.95% Refueling Level corresponds to a 1450 gpm RHR intake per loop suction flow value. Since both sets of loop suction valves are open, the limiting flow for RHR would be 2900 gpm. However, with RHR Pump B only running, the flow is limited to 2000 gpm.

d. Wrong. 54 minutes is the value obtained if the heat up rate was accidentally divided into 212°F without first subtracting the 80°F starting temperature. 1350 gpm is the desired pump maximum flow when checking if RHR Pumps should be stopped (AOP-RHR-003, Step 1.b).

Question Number: 078
Exam Date 20090413
Exam Level S
K/A 029E 2.4.8
Question
Given the following:

- A spurious Reactor Trip and SI have occurred.
- The crew is performing the actions of E-0, "Reactor Trip or Safety Injection".
- There are NO indications of a LOCA, Ruptured Steam Generator, or Faulted Steam Generator.
- Annunciator 47051-N, "CW Pumps Flood Level Trip" is LIT.

According to UG-0, "User's Guide For Emergency and Abnormal Procedures", what is the EARLIEST time that the Unit Supervisor is allowed to perform procedures in parallel, and direct the actions of ARP-47051-N, "CW Pumps Flood Level Trip"?

a. After the transition from E-0, "Reactor Trip or Safety Injection" to ES-1.1 "SI Termination."

b. After the actions to terminate Safety Injection are completed per ES-1.1, "SI Termination."

c. During the immediate actions of E-0, "Reactor Trip or Safety Injection," as long as the actions of the ARP do NOT interfere with the performance of E-0.

d. After the immediate actions of E-0, "Reactor Trip or Safety Injection," as long as the actions of the ARP do NOT interfere with the performance of E-0.

Answer
d.
REFERENCE
UG-0, User's Guide For Emergency and Abnormal Procedures, step 6.2.4 and section 6.4
NEW
HIGHER
K/A Knowledge of how abnormal operating procedure are used in conjunction with EOPs.
JUSTIFICATION:
a. WRONG. UG-0, states that ARPs can be implemented during an IPEOP only after immediate actions are performed.
b. WRONG. UG-0, states that an ARP can be implemented during an IPEOP only after immediate actions are performed.
c. WRONG. UG-0, states that an AOP can be implemented during an IPEOP only after immediate actions are performed.
d. CORRECT. UG-0, states that an AOP can be implemented during an IPEOP only after immediate actions are performed, as occurs in this case.

Question Number: 079
Exam Date 20090413
Exam Level S
K/A 056A 2.2.38
Question
Given the following:

- Plant is operating at 100%.
- ATC has called the control room with information that the grid is degraded and 138KV Bus Post Trip Voltage is 137KV. All contingencies to maintain KPS 138KV Bus Post Trip voltage greater than 140KV have been implemented and have NOT been successful.

The SRO should declare . . .

a. Bus 6 INOPERABLE and transfer Bus 6 to 'B' EDG.

b. transmission lines F-84 and Y-51 INOPERABLE and reduce power to < 50%.

c. the RAT INOPERABLE and continue 100% power operation for up to 7 days.

d. off-site power INOPERABLE and commence a standard shutdown sequence.

Answer
d.
REFERENCE
OP-KW-AOP-EG-001, Abnormal Grid Operations, Step 2
Technical Specification 3.7.a.8, 3.7.b.4 and 3.0.c
NEW
HIGHER
K/A Knowledge of conditions and limitations in the facility license.
JUSTIFICATION:
a. Wrong. Transfer of Bus 6 to EDG 'B' is not required.
b. Wrong. F-84 and Y-51 are inoperable, but the need to commence a standard shutdown sequence is not required. Reducing power to less than 50% is required for 3 of 4 Off-site lines out of service/inoperable.
c. Wrong. Need to commence standard shutdown sequence. 7 day LCO refers to only RAT inoperable.
d. Correct. Off-Site power is inoperable when post trip voltage falls below 140KV on 138KV line. Cannot ensure ESF equipment will function as designed at that voltage.

Question Number: 080
Exam Date 20090413
Exam Level S
K/A 057A 2.4.11
Question
Given the following:

- The plant is at 100% power.
- All systems are in Automatic.
- Loss of Yellow Instrument Bus 4 has occurred.

In addition to AOP-EDC-001, "Loss of Instrument Bus", it will be necessary to implement procedure . . .

a. E-0, "Reactor Trip or Safety Injection," because the reactor will have tripped.

b. N-RM-45, "Radiation Monitoring System," to shift Containment sampling to R-21.

c. NOP-CVC-001, "Boron Concentration Control," to manually makeup to the Charging Pump suction to maintain VCT level.

d. NOP-CVC-002, "Charging and Volume Control," to place Excess Letdown in service.

Answer
c.
REFERENCE
AOP-EDC-001, Loss of Instrument Bus
NEW
HIGHER
K/A Knowledge of abnormal condition procedures for a Loss of Vital AC Instrument Bus.
JUSTIFICATION:

a. Wrong. The NOTE before the first step of AOP-EDC-001 directs stabilizing the plant using the EOPs if the reactor trips as a result of the instrument bus loss. The first step for a loss of the Red Instrument Bus 1 in Attachment A or the Blue Instrument Bus 3 in Attachment C, is to check the reactor tripped but NOT for the Yellow Instrument Bus 4.
b. Wrong. AOP-EDC-001 directs N-RM-45 implementation to shift Containment sampling to R-21 for a loss of the White Instrument Bus 2 due to loss of Train B RMS but NOT for the loss of the Yellow Instrument Bus 4.
c. Correct. AOP-EDC-001 directs NOP-CVC-001 implementation to manually makeup to the Charging Pump suction to maintain VCT level 20 - 50% for a loss of the Yellow Instrument Bus 4.
d. Wrong. AOP-EDC-001 directs NOP-CVC-002 implementation to place Excess Letdown in service for a loss of White Instrument Bus 2 and Blue Instrument Bus 3. In addition, for a loss of the Yellow Instrument Bus 4, AOP-EDC-001 directs implementation of NOP-CVC-002 to place "normal" Letdown in service.

Question Number: 081
Exam Date 20090413
Exam Level S
K/A E04EA2.1
Question
A plant cooldown at 25°F/hour is in progress using RHR when the following timeline of events occur.

At time = 0, annunciator 47032-R, "RHR PUMP PIT SUMP LEVEL HIGH", is LIT and the following plant conditions are noted:

- RCS pressure 380 psig
- RCS hot leg temperature 325°F
- Pressurizer level 27%, lowering
- Letdown In Service with one 40 gpm orifice OPEN
- Charging Pumps 'A' & 'B' are both in manual.

At time = 1 minute, R13 alarms at the ALERT level and its indication is rising.
At time = 4 minutes, charging flow is raised to maximum, pressurizer level is 18% and letdown is isolated.
At time = 5 minutes, pressurizer level reaches 17%.
At time = 7 minutes, charging flow remains at maximum and pressurizer level is 14% and slowly rising.

What is the correct course of action for this situation?

a. Implement actions of AOP-RC-001, "Reactor Coolant Leak."

b. Enter E-0, "Reactor Trip Or Safety Injection," and manually initiate SI.

c. Manually initiate SI and implement ECA-1.2, "LOCA Outside Containment."

d. Implement actions of AOP-RHR-002, "Shutdown Loss Of Coolant Accident."

Answer
d.
REFERENCE
AOP-RC-001, Reactor Coolant Leak
AOP-RHR-002, Shutdown Loss Of Coolant Accident
E-0, Reactor Trip Or Safety Injection
ECA-1.2, LOCA Outside Containment
BANK
HIGHER
K/A - Ability to determine and interpret the following as they apply to the LOCA Outside Containment: Facility conditions and selection of appropriate procedures during abnormal and emergency operations. RO Imp / SRO Imp – 3.4 / 4.3
JUSTIFICATION:
a. Wrong. The first note of AOP-RC-001 states that the procedure does not apply when on RHR.
b. Wrong. Given the plant conditions, SI is blocked and accumulators isolated. Therefore, initiating SI is not required and E-0 does not apply.
c. Wrong. Because the plant is on RHR, the EOP network of procedures does not apply.

d. Correct. This is a loss of coolant outside Containment while the plant is on RHR.

Question Number: 082
Exam Date 20090413
Exam Level S
K/A 024A 2.4.47
Question
Given the following:

- The Reactor is Tripped.
- Safety Injection has NOT occurred.
- The first FOUR steps of E-0, Reactor Trip Or Safety Injection, have been completed.
- Intermediate Range SUR reads -1/3 dpm.
- Tave is 547°F.
- Two control rods are NOT fully inserted.
- Bus 5 LOCKED OUT when G-1 opened, expected return > 6 hours.

Based on the above conditions, what method will the Unit Supervisor direct in order to add the required negative reactivity?

a. Establish a normal boration through CVC-406, BA Blender to VCT, per NOP-CVC-001, "Boron Concentration Control."

b. Start an emergency boration through CVC-440, Emergency Boration to charging pumps, per E-0, "Reactor Trip or Safety Injection."

c. Commence a boration through CVC-403, Boric Acid to Blender, and CVC-408, BA Blender to Charging Pumps, per AOP-CVC-001, "Emergency Boration."

d. Align the RWST to suction of the charging pumps by opening CVC-301, RWST Supply to Charging Pumps, per FR-S.2, "Response to Loss of Core Shutdown."

Answer
c.
REFERENCE
ES-0.1, Reactor Trip Response, Step
AOP-CVC-001, Emergency Boration
FR-0, Critical Safety Function Status Trees, ATTACHMENT A F-0.1 SUBCRITICALITY
OA # 02-22 MCC-52E
NEW
HIGHER
K/A Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material.
Justification:
a. Wrong. With a stuck rod during a reactor trip, ES-0.1 directs boration per AOP-CVC-001 if all rods cannot be verified on the bottom. NOP-CVC-001 is not used for establishing this boration.
b. Wrong. CVC-440 does not have any power with a bus 5 lockout and is not used for this boration. CVC-440 is powered from MCC-52E.
c. Correct. Procedural flow path is to CVC-403 and CVC-408 use if CVC-440 will not open from the control room as directed by AOP-CVC-001. ES-0.1 directs using AOP-CVC-001 to emergency borate if all rods cannot be verified fully inserted.
d. Wrong. FR-S.2 is not entered for the given conditions with IR SUR > -0.2 dpm. CVC-

301 is used only if flow can NOT establish from the Boric Acid Storage Tanks.

Question Number: 083
Exam Date 20090413
Exam Level S
K/A 060A 2.4.41
Question
Given an EAL matrix containing the Offsite Rad Conditions and the following conditions:

- An unplanned release of gaseous radioactivity to the environment is in progress.
- The release is expected to continue.
- R-13 Aux. Bldg. Vent Exhaust reading 1.0E+07 cpm and stable.
- R-14 Aux. Bldg. Vent Exhaust reading 1.0E+07 cpm and stable.
- 01-07 Aux. Bldg. SPING Mid Range reading 1.30E+05 cpm.
- Dose assessment using actual meteorology indicates 150 mRem TEDE and 430 mRem thyroid CDE at the site boundary.

The Shift Manager should declare a . . .

a. General Emergency based on RG1.1.

b. General Emergency based on RG1.2.

c. Site Area Emergency based on RS1.1.

d. Site Area Emergency based on RS1.2.

Answer
d.
REFERENCE
EPIP-AD-02, Emergency Class Determination, Step 5.1
Kewaunee Power Station Emergency Action Level Matrix
Emergency Action Level Technical Bases Document, RG1 Basis
NEW
HIGHER
Provided Reference: EAL matrix containing the Offsite Rad Conditions
K/A Knowledge of the emergency action level thresholds and classifications for an Accidental Gaseous Radwaste Release.
JUSTIFICATION:
a. Wrong. RG1.1 provides a list of monitors and thresholds that if exceeded for > 15 minutes should result in declaration of a GE. Although the 01-07 Aux. Bldg. SPING Mid Range reading of 1.30E+05 cpm exceeds the limit, the classification determined by using the Dose Assessment overrides. The preference for using RS1.2 over using RG1.1 is specified in the NOTE. Therefore declaration as GE would be inappropriate because of the data collected indicating RS1.2 limit met for Dose Assessment but RG1.2 is not met.
b. Wrong. RG1.2 provides limits used when provided with Dose Assessment using actual meteorology. In this case the limits indicated in RG1.2 are not exceeded by the data provided.
c. Wrong. RS1.1 provides a list of monitors and thresholds that if exceeded for > 15 minutes should result in declaration of an SAE. Although the 01-07 Aux. Bldg. SPING Mid Range reading of 1.30E+05 cpm exceeds the limit provided for a SAE, it also exceeds the limit for a GE. Therefore declaration as SAE would be inappropriate because a higher classification is called for.

d. Correct. The Dose Assessment value of 100 mRem TEDE is exceeded. In accordance with the NOTE, if dose assessment results are available at the time of declaration, the classification should be based on RG1.2 or RS1.2 instead of RG1.1.

Question Number: 084
Exam Date 20090413
Exam Level S
K/A 067A 2.4.8
Question
Given the following:

- AOP-FP-001, "Abnormal Operating Procedure - Fire," is in progress.
- The fire is in the Cable Spreading area.
- All Nuclear Instrumentation indication has been lost.
- PR-2A, Pressurizer PORV, has failed open and will NOT close from the Control Room.
- Contingency actions to close PR-2A, Pressurizer PORV, have NOT been successful.
- PT-468, S/G 'A' header pressure, has failed low.
- SM directs Control Room evacuation.
- US directs the RO to trip the reactor.
- RO trips the reactor and checks reactor trip and bypass breakers ALL open.

The US should direct implementation of . . .

a. AOP-FP-002, "Fire In Alternate Fire Zone."

b. AOP-FP-003, "Fire In Dedicated Fire Zone."

c. E-0, "Reactor Trip Or Safety Injection," Immediate Actions.

d. FR-S.1, "Response To Nuclear Power Generation/ATWS."

Answer
a.
REFERENCE
AOP-FP-001, Abnormal Operating Procedure – Fire, Steps 30-31
AOP-FP-002, Fire In Alternate Fire Zone, 2.1 and NOTE at step 1
NEW
HIGHER
K/A Knowledge of how abnormal operating procedures are used in conjunction with EOPs.
JUSTIFICATION:
a. Correct. The Cable Spreading area is an Alternate Fire Zone. Failure of actions to regain plant control and/or monitoring capability are possible entry conditions for AOP-FP-002 from AOP-FP-001 at the Shift Manager's discretion. A NOTE at the start of AOP-FP-002 states that entry into E-0 while performing AOP-FP-002 is not required. With the loss of all NI indications the crew would be unable to verify neutron flux decreasing requiring a transition to the RNO for VERIFY Reactor Trip.
b. Wrong. The Cable Spread area isn’t a Dedicated Fire Zone. Plausible distractor in that AOP-FP-003 also provides direction for fire with a loss of indication or controls.
c. Wrong. This is a plausible distractor because under normal circumstances, a reactor trip requires entry into E-0. In this case however, the NOTE at the start of AOP-FP-002 states that entry into E-0 while performing AOP-FP-002 is not required.
d. Wrong. Plausible distractor because if confirmation of reactor trip is not possible from the Control Room, under normal circumstances, a transition to FR-S.1 would be made.

Question Number: 085
Exam Date 20090413
Exam Level S
K/A E01EA2.2
Question
The operator may go to ES-0.0, "Rediagnosis," if . . .

a. Safety Injection is required and a transition has not been made from E-0, "Reactor Trip Or Safety Injection."

b. Safety Injection is not required and a transition from E-0, "Reactor Trip Or Safety Injection," has been completed.

c. Safety Injection is in service and a transition from E-0, "Reactor Trip Or Safety Injection," has been completed.

d. Safety Injection is in service and a transition has not been made from E-0, "Reactor Trip Or Safety Injection."

Answer
c.
REFERENCE
UG-0, User's Guide For Emergency And Abnormal Procedures, Step 6.6.3.c
ES-0.0, Rediagnosis
NEW
FUNDAMENTAL
K/A Ability to determine and interpret the following as they apply to the Reactor Trip or Safety Injection/Rediagnosis: Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments.
JUSTIFICATION:
a. Wrong. To go to ES-0.0, a transition must be made from E-0. This answer does not meet that requirement. This condition contains part of the requirement and represents the plant conditions that would require staying within E-0 until it is determined whether or not SI is required and then transitioning as required into the next appropriate EOP within the framework of the EOP network of procedures.
b. Wrong. To go to ES-0.0, SI must either be in service or be required. This answer does not meet that requirement. It contains part of the requirement and represents the plant conditions that would exist for a transition to ES-0.1.
c. Correct. Per UG-0, Step 6.3.3.c, if all of the following conditions exist, the operator may go to ES-0.0, REDIAGNOSIS, Step 1: 1. SI is in service or required; and, 2. A transition is made from E-0, REACTOR TRIP OR SAFETY INJECTION. Only this answer satisfies both of those requirements.
d. Wrong. To go to ES-0.0, a transition must be made from E-0. This answer does not meet that requirement. It contains part of the requirement and represents the plant conditions that would require staying within the framework of the EOP network of procedures.

Question Number: 086
Exam Date 20090413
Exam Level S
K/A 010 2.2.22
Question
Which statement describes the Technical Specification Safety Limit for Reactor Coolant System Pressure and its bases?

The Reactor Coolant System pressure shall not exceed . . .

a. 3107 psig at COLD SHUTDOWN during hydrostatic testing to prevent exceeding 110% of design pressure, the maximum transient pressure allowable under ASME Code.

b. 2735 psig with fuel assemblies installed in the reactor vessel with settings providing protection to prevent exceeding this value for all transients except for a rod ejection accident.

c. 2485 psig during at power operation to ensure DNBR during steady-state operation, normal operational transients, Condition I and Condition II transients is maintained greater than or equal to the 95/95 DNBR criterion.

d. 2335 psig while the RCS is greater than or equal to 200ºF in order to prevent exceeding design containment pressure resulting from the postulated Design Basis Accident.

Answer
b.
REFERENCE
Technical Specification 2.2.a and Bases
MODIFIED
FUNDAMENTAL
K/A Knowledge of limiting conditions for operations and safety limits for Pressurizer Pressure Control.
JUSTIFICATION:
a. Wrong. 3107 psig was the limit for the initial hydrostatic test pressure to ensure the integrity of the Reactor Coolant System. The test was performed at normal operating temperature (HOT SHUTDOWN). Currently the RCS Integrity Test is conducted at pressures between 2285 - 2310 psig. The 110% design pressure value for the RCS is 2735 psig. The maximum transient pressure allowable in the reactor pressure vessel under the ASME Code, Section III, is 110% of design pressure.
b. Correct. The specification states, "The Reactor Coolant System pressure shall not exceed 2735 psig with fuel assemblies installed in the reactor vessel." The bases describes that the settings for the PORVs, reactor high pressure trip and Safety Valves have been established to prevent exceeding this limit for all transients, except for a rod ejection accident, in which limit is 3105 psig.
c. Wrong. 2485 psig is the PRZR Safety Setpoint and is the current RCS design pressure. The bases comes from TS Bases 2.1 for reactor core safety limits, specifically the section related to maintaining the integrity of the fuel cladding and prevent fission product release (DNB). While the reactor core safety limits has a RCS pressure component, it is not related to RCS integrity.
d. Wrong. 2335 is setpoint for the PRZR PORV opening. The bases come from TS Bases

3.6.d associate with Containment pressure limits.

Question Number: 087
Exam Date 20090413
Exam Level S
K/A 026 2.2.25
Question
Sodium hydroxide (NaOH) is added to the Internal Containment Spray System to. . .

a. enhance the ability of the spray to scavenge iodine fission products from the containment atmosphere.

b. ensure the sump pH remains acidic during the containment sump recirculation phase of injection.

c. maximize the volatility of iodine in the containment sump.

d. facilitate converting soluble iodine into insoluble iodine.

Answer
a.
REFERENCE
Technical Specification 3.3 Bases
MODIFIED
FUNDAMENTAL
K/A Knowledge of bases in technical specifications for limiting conditions for operations and safety limits for Containment Spray. RO Imp / SRO Imp – 3.2 / 4.6
JUSTIFICATION:
a. Correct. Per TS 3.3 Bases, page TS B3.3-3, "Sodium Hydroxide (NaOH) is added to the spray solution for pH adjustment by means of the spray additive system. The resulting alkaline pH of the spray enhances the ability of the spray to scavenge iodine fission products from the containment atmosphere."
b. Wrong. The NaOH added in the spray also ensures an alkaline pH for the solution recirculated in the containment sump.
c. Wrong. Per TS 3.3 Bases, page TS B3.3-3, "The alkaline pH of the containment sump water inhibits the volatility of iodine..." The key is that the volatility of iodine is inhibited (i.e. minimized) NOT maximized.
d. Wrong. Actually facilitates the conversion of insoluble iodine into soluble iodine.

Question Number: 088
Exam Date 20090413
Exam Level S
K/A 039A2.04
Question
Given the following:

- A spurious reactor trip has occurred from 100% power.
- Red Channel Tave, TI-405, has failed off-scale high during the trip.
- All other Tave channels are indicating 543°F and decreasing slowly.
- Steam Dump Control Mode Selector Switch is in Tavg mode.

What procedure will FIRST be used to address the status of Steam Dumps based on the above failure?

a. E-0, "Reactor Trip Or Safety Injection," will be used to take the Steam Dump Interlock Selector Switches to BYPASS INTLK.

b. ES-0.1, "Reactor Trip Response," will be used to reposition the Steam Dump Control Mode Selector Switch to RESET and then to STM PRESS.

c. NOP-MS-001, "Main Steam and Steam Dump System," will be used to position the Steam Dump Control Mode Selector Switch to RESET.

d. FR-P.1, "Response To Imminent Pressurized Thermal Shock Condition," will be used to reposition the Steam Dump Interlock Selector Switches to OFF/RESET.

Answer
b.
REFERENCE
E-0, Reactor Trip Or Safety Injection, Steps 1-4 (RNO)
ES-0.1, Reactor Trip Response, Step 15.b & 15.c
NEW
HIGHER
K/A Ability to (a) predict the impacts of the following malfunctions or operations on the MRSS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Malfunctioning steam dump.
Justification:
a. Wrong. E-0 will transition at step 4, no manipulations of Steam Dumps will take place until ES 0.1. Steam Dumps are manipulated in E-0 and the student will have to determine based on the information what the status of SI is (not required or actuated).
b. Correct. Based on the failure of T-405, Steam Dumps will stay open until the other Tave channels reach 540°F. No safety injection will occur. The first procedure that will be entered is E-0 and will transition to ES-0.1 at step 4. Step 2 of ES-0.1 is to verify RCS temperature control. The RNO will direct "Position the Main Steam Dump Control Mode Selector Switch to RESET and then STM Press."
c. Wrong. This is not the FIRST procedure that will address the Steam Dump position. Steam Dumps are manipulated in NOP-MS-001 during normal plant operation.
d. Wrong. Based on the above failure the steam dumps will close at 540°F and FR-P.1 will not be entered for excessive cooldown. If a cooldown did continue FR-P.1 entry may be required and steps in FR-P.1 direct controlling cooldown.

Question Number: 089
Exam Date 20090413
Exam Level S
K/A 059A2.03
Question
Given the following:

- Operating at 100% power, 1772 MWt.
- 'A' AFW Pump starts on a spurious start signal.

What is an operational limitation associated with the spurious start of 'A' AFW Pump?

a. Initiate action to restore ARTO and IMMEDIATELY reduce power to $\leq$ 1710 MWt.

b. Reduce actual reactor thermal power to less than 1772 MWt since PPCS "Thermal Output Monitoring" is inaccurate.

c. Reduce Power to $\leq$ 1649 MWt within 4 hours to ensure adequate decay heat removal during a loss of feedwater event.

d. Declare 'A' AFW Pump INOPERABLE and either reduce power to $\leq$ 1673 MWt or restore 'A' AFW Pump OPERABILITY within 2 hours.

Answer
b.
REFERENCE
AOP-CP-001, Abnormal Plant Process Computer System, ATTACHMENT A, page 11
Technical Specification 3.4
NEW
HIGHER
K/A Ability to (a) predict the impacts of the following malfunctions or operations on the MFW system and (b) based on those predictions, use procedures to correct, control or mitigate the consequences of those malfunctions or operations: Overfeeding event.
Justification:
a. Wrong. Power reduction to less than 1710 MWt is required only after 24 hours. AOP-CP-001 Attachment A, A2
b. Correct. Per AOP-CP-001 Attachment A, A2, Do not increase reactor thermal power above its value at the time PPCS or Thermal Output Monitoring program is out of service. RTO is not accurate because it does not account for the feedwater being delivered by 'A' AFW Pump.
c. Wrong. Reduction of power to less than 1649 MWt is not required. The 4 hour time limit is for 2 AFW pumps inoperable.
d. Wrong. Per Technical Specifications 3.4.b.4 enter a 72 hour LCO for one AFW pump OOS. Per Technical Specification 3.4.b.3 if two of three trains of AFW are inoperable then within 2 hours reduce reactor power to less than 1673 MWt.

Question Number: 090
Exam Date 20090413
Exam Level S
K/A 062 2.2.38
Question
Per TS 3.7 Auxiliary Electrical Systems, there are three pairs of physically independent transmission lines to the Kewaunee Substation:

What describes the MINIMUM requirements that must be met for transmission lines, as given in Technical Specification 3.7.a.8?

a. TS 3.7.a.8 requires that three of the physically independent transmission line pairs serving the substation are OPERABLE.

b. TS 3.7.a.8 requires that at least one of the physically independent transmission line pairs serving the substation is OPERABLE.

c. TS 3.7.a.8 requires that at least two of the physically independent transmission line pairs serving the substation are OPERABLE.

d. TS 3.7.a.8 requires that at least one feed of each physically independent transmission line pair serving the substation are OPERABLE.

Answer
b.
REFERENCE
Technical Specification 3.7 Auxiliary Electrical Systems, 3.7.a.8
BANK
FUNDAMENTAL
K/A Knowledge of conditions and limitations in the facility license.
Justification
a. Wrong. Per T.S. 3.7.a.8, the minimum requirement is one pair of physically independent transmission lines serving the substation.
b. Correct. Per T.S. 3.7, at least one pair of physically independent transmission lines serving the substation is OPERABLE. The three pairs of physically independent transmission lines are R-304 and Q-303, F-84 and Y-51, and R-304 and Y-51.
c. Wrong. Per T.S. 3.7.a.8, the minimum requirement is one pair of physically independent transmission lines serving the substation.
d. Wrong. Per T.S. 3.7.a.8, the minimum requirement is one pair of physically independent transmission lines serving the substation.

Question Number: 091
Exam Date 20090413
Exam Level S
K/A 002 2.1.7
Question
Which describes a situation requiring entry into Technical Specification 3.10.n, DNBR Parameters, for Reactor Coolant System Pressure and correctly describes the basis and required action of T.S. 3.10.n?

a. RCS pressure 1904 psig, provides protection against a power excursion at full RATED THERMAL POWER.
Required action is to restore RCS pressure within 5 minutes and be in HOT SHUTDOWN within 4 hours.

b. RCS pressure 2176 psig, ensures that calculated offsite doses are held to within the limits specified in 10 CFR 50.67.
Required action is to place the reactor in INTERMEDIATE SHUTDOWN with an average reactor coolant temp < 500°F within 6 hours.

c. RCS pressure 2211 psig, maintains the integrity of the fuel cladding.
Required action is to restore RCS pressure in two hours or less to within limits or reduce power to < 5% of thermal rated power within an additional six hours.

d. RCS pressure 2485 psig, maintains the integrity of the Reactor Coolant System.
Required action is to restore compliance and place the reactor in HOT SHUTDOWN within 1 hour.

Answer
c.
REFERENCE
T.S. 3.10.k, 3.10.l, 3.10.m and 3.10.n.
COLR 2.11.2
NEW
HIGHER
K/A Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.
Justification:
a. Wrong. The value for RCS pressure is less than 2217 however the basis and required action are wrong. The value is consistent with an RCS operational parameter, SI injection setpoint. The required action is similar to the action required for exceeding a Safety Setpoint.
b. Wrong. The value for RCS pressure is less than 2217 however the basis and required action are wrong. The value is consistent with DNBR entry and is the action and basis from the TS 3.1.c Maximum Coolant Activity.
c. Correct. See references above for T.S. 3.10.l, TS 3.10.n and COLR 2.11 DNB Basis - Maintain the integrity of the fuel cladding.
d. Wrong. This scenario is similar to T.S. 2.2 Safety limit - Reactor Coolant System Pressure and its associated basis is to maintain the integrity of the reactor coolant system.

Question Number: 092
Exam Date 20090413
Exam Level S
K/A 017A2.01
Question
Given the following:

- The plant is operating at 100% power.
- Annunciator 47033-12, "TLA-2 RCS Subcooling High/Low," is LIT.
- Annunciator 47033-24, "TLA-9 Core Exit T/C Tilt," is LIT.
- Annunciator 47044-F, "ICCMS Panel Trouble," is LIT.

The following indications are reported by the RO:

- Core Exit Thermocouple Monitor Train 'A': 673°F, erratic and rising slowly
- Core Exit Thermocouple Monitor Train 'B': 626°F, stable
- ICC SYS ACK train 'A' bright with display flashing
- ICC SYS ACK train 'B' dim

The SRO should direct entry into which procedure to address the current plant condition?

a. E-0, "Reactor Trip or Safety Injection."

b. N-II-50, "Inadequate Core Cooling Monitoring System."

c. AOP-CP-001, "Abnormal Plant Process Computer System."

d. AOP-II-001, "Abnormal Inadequate Core Cooling Monitoring System."

Answer
d.
REFERENCE
ARP-47033-12, TLA-2 RCS Subcooling High/Low
ARP-47033-24, TLA-9 Core Exit T/C Tilt, Recommended Action 3
ARP-47044-F, ICCMS Panel Trouble, Step 12 (RNO)
AOP-II-001, Abnormal Inadequate Core Cooling Monitoring (ICCM) System
AOP-CP-001, Abnormal Plant Process Computer System, Step 23
NEW
HIGHER
K/A Ability to (a) predict the impacts of the following malfunctions or operations on the ITM System and (b) based on those predictions , use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: thermocouple open and short circuits.
Justification:
The alarms and indications are caused by a Train A Core Exit Thermocouple (CETC) failing high. This causes a loss of Subcooling on Train A; and erratic rising Train A CETC temperature. The SRO will need to identify that a Train A CETC is failing high based on the information given and recall the correct procedure to address the issue.
a. WRONG. Based on the given information, the alarm for RCS subcooling is due to a failure and does not require entry into E-0 for Reactor Trip. An actual loss of subcooling would require the operator to trip the reactor and go to E-0. See ARP 47033-12 step 4 RNO b.

b. WRONG. N-II-50 describes the normal operation of ICCMS. This procedure does not address ICCMS failures or malfunctions.

c. WRONG. AOP-CP-001 is referenced in the ARP and provides guidance for responding to total or partial failure of PPCS and/or SPDS. This will not address the failing CET. Issues with SMM can be caused PPCS inputs.

d. CORRECT. Per Alarm Response for 47044-F and 47033-24, the correct procedure to address the CET malfunction is AOP-II-001. This procedure provides guidance for responding to ICCM System faults.

Question Number: 093
Exam Date 20090413
Exam Level S
K/A 034K5.03
Question
Per N-FH-53F, "Reactor Cavity Draining With Fuel and Upper Internals Removed," what is an operational consideration while RHR is running at a reduced inventory condition?

a. Do not drain below 10.2% Reactor Vessel Level, to prevent lowering below the center line of the Hot Leg.

b. At 12.5% Reactor Vessel Level, an operator is required to continuously monitor Reactor Vessel Level via the tygon to prevent lowering below the bottom of the RVLIS instrument range.

c. Draining below 13.0% Reactor Vessel Level will require reducing RHR flow to less than 1000 gpm to prevent cavitation.

d. With Reactor Vessel Level below 19.0%, continuous monitoring is required of RHR parameters for signs of vortexing such as rising suction pressure.

Answer
a.
REFERENCE
N-RHR-34C, RHR Operation At A Reduced Inventory Condition, 2.3, CAUTION Step 4.11, ATTACHMENT A
N-FH-53F, Reactor Cavity Draining With Fuel and Upper Internals Removed, 2.2
NEW
FUNDAMENTAL
K/A Knowledge of the operational implications of the following concepts as they apply to the Fuel Handling System: Residual Heat Removal; decay.
Justification:
a. Correct: Per N-FH-53F, precaution 2.2, "Do NOT drain the Reactor Vessel below 10.2% (center line of the Hot Leg) using RHR pumps.
b. Wrong: The bottom of the Reactor Vessel instrument range is 7.95%. The requirement to have an operator stationed to monitor the tygon is false. If indication range would be lost, the only available indication would be locally at the tygon. The fact that it would be procedurally required is also plausible.
c. Wrong: Per N-RHR-34C, "When refueling level is less than 13%, RHR system flow rate shall not exceed 3000 gpm." As level is reduced it would be plausible that flow would also be required to be reduced. Cavitation is a major concern for RHR, however per N-RHR-34C cavitation is not expected to occur until 9% Reactor Vessel Level.
d. Wrong: Reduced inventory is not reached until 17% Reactor Vessel Level. Also wrong based on the signs of vortexing which would be LOWERING suction pressure. Per N-RHR-34C, NOTE at step 4.1, "Operations at reduced inventory requires increased monitoring of the RHR system. This shall include frequent monitoring for abnormal motor and pump noise which may be indicative of air entrainment. These local checks shall verify adequate suction pressure. This is very similar to the note contained in the procedure except for the level and the change in the trend of RHR suction pressure.

Question Number: 094
Exam Date 20090413
Exam Level S
K/A 2.1.3
Question
According to GNP-02.07.01, "Refueling Operations - Logkeeping, Watchstanding and Shift Turnover," which of the following is NOT required as part of the Shift Turnover for the oncoming Refueling SRO?

a. A review of the detailed fuel movement log.

b. A review of the containment Open Boundary Tracking Log.

c. A face to face turnover in containment or other designated location.

d. The transfer of Refueling SRO responsibilities documented in the detailed fuel movement log.

Answer
b.
REFERENCE
GNP-02.07.01 Refueling Operations - Logkeeping, Watchstanding, and Shift Turnover. Section 6.2 Shift Relief and Turnover
NEW
FUNDAMENTAL
K/A Knowledge of shift or short term relief turnover practices.
Justification:
a. Wrong: Per procedure GNP-02.07.01, a review of the detailed fuel movement log is required.
b. Correct: This is not required by GNP-02.07.01. The Shift Manager is responsible for initiating the Open Boundary Tracking Log, which is maintained from the Control Room.
c. Wrong: Per procedure GNP-02.07.01, the face to face turnover is required.
d. Wrong: Per procedure GNP-02.07.01, the documentation of the transfer of responsibilities is required.

Question Number: 095
Exam Date 20090413
Exam Level S
K/A 2.1.36
Question
What accident provides the basis for the minimum boron concentration and shutdown margin limits for the RCS during REFUELING OPERATIONS when fuel is in the reactor?

a. Loss of Inventory during Refueling

b. Boron Dilution during a Refuel Accident

c. Dropped Fuel Assembly during Refueling

d. Loss of Decay Heat Removal during a Refuel Accident

Answer
b.
REFERENCE
Technical Specification 3.8 Refueling Operations and associated Basis, 3.8.a.5.
Core Operating Limits Report (COLR) Cycle 29, 2.12.
NEW
FUNDAMENTAL
K/A Knowledge of procedures and limitations involved in core alterations.
Justification:
a. WRONG. The minimum water level requirement stated in 3.8.a.10.
b. CORRECT. Per BASIS - Refueling Operations (TS 3.8) page TS B3.8.1, "A minimum shutdown margin of greater than or equal to 5% dk/k must be maintained in the core. The boron concentration as specified in the COLR is sufficient to ensure an adequate margin of safety. The specification for REFUELING OPERATIONS shutdown margin is based on a dilution during refueling accident.
c. WRONG. A probable accident during refueling is a dropped or damaged fuel assembly.
d. WRONG. A common event with the potential to cause damage during refueling is a loss of decay heat removal.

Question Number: 096
Exam Date 20090413
Exam Level S
K/A 2.2.23
Question
Given the current equipment out of service information:

- Packing for SW-301A, Service Water from D/G A Heat Exchanger, was replaced.
- During the post-maintenance review of the work package it was identified that the wrong packing was used.
- An Operability Determination was performed and the valve was found to be operable but degraded.

How is the current operability condition officially tracked until full operability is restored to the component?

a. Tech Spec Tracking Log.

b. Unit Supervisor Turnover.

c. Control Room Narrative Log.

d. Corrective Action Program.

Answer
d.
REFERENCE
OP-KW-ORT-MISC-007, Operations Turnovers, Logs and Briefings, ATTACHMENTS D and E
OP-AA-102, Operability Determination, 2.2 & 3.2.11
BANK
FUNDAMENTAL
K/A Ability to track Technical Specification limiting conditions for operations.
Justification:
a. Wrong. The Tech Spec Tracking log is the tracking mechanism for inoperable equipment. This log does track equipment issues but only those which result in inoperability.
b. Wrong. The OBDs are covered by the Shift Manager (SM) during turnover brief and is listed on the SM Turnover Sheet and Pre-Shift Briefing Sheet.
c. Wrong. The control room narrative log would be used to document the issue however it does not track the equipment degradation until completion. The log would be used to document the issue at the time of discovery.
d. Correct. All OBDs are entered into the Corrective Action Program and active OBDs are covered by the Shift Manager during crew turnover.

Question Number: 097
Exam Date 20090413
Exam Level S
K/A 2.2.37
Question
Given the following:

- Plant is at 100% power.
- CFCU 'A' is INOPERABLE for maintenance.
- ICS Pump 'A' is INOPERABLE for maintenance.
- Plant Electricians request Bus 1-61 be removed from service for maintenance.

Upon SRO review, the Electricians' request will be denied. What is the reason for the denial?

a. De-energizing Bus 1-61 will result in both Containment Spray trains being INOPERABLE.

b. Vital 480V buses shall be energized from their respective station transformers when the reactor is critical.

c. Both Containment Spray trains may only be out of service provided that all CFCUs are operable.

d. Per PRA configuration control guidelines, only one SSC affecting PRA risk may be removed for maintenance at a time.

Answer
a.
REFERENCE
E-240 SH-001, Circuit Diagram – Circuit Diagram 4160V and 480V Power Sources
MODIFIED
HIGHER
K/A: Ability to determine operability and/or availability of safety related equipment.
Justification:
a. Correct. Per Technical Specification 3.3.c.1.A.3.ii, one containment spray train may be out of service for 72 hours provided the opposite containment spray train remains OPERABLE. Bus 1-61 is the power supply for ICS Pump B, thus removing Bus 1-61 from service would result in both ICS Pumps being out of service.
b. Wrong. Tech Spec 3.7 allows the removal from service of buses for short duration at power. This selection is a statement directly from T.S. 3.7. However this condition is allowed by Tech Specs.
c. Wrong. Tech Spec 3.3.c, Containment Cooling Systems, allows for the same containment fancoil unit and containment spray train to be out of service together. This is a plausible distractor because of the allowance to remove from service various combinations of ICS Pumps and CFCU's.
d. Wrong. The total number of SSC's removed for maintenance is based on overall CDF and LERF risk color for the combination of total equipment assumed to be unavailable, not a predefined number of pieces of equipment. PRA risk is evaluated prior to removing equipment from service.

Question Number: 098
Exam Date 20090413
Exam Level S
K/A 2.3.13
Question
In accordance with Technical Specification 6.13, "High Radiation Area," which statement describes the Shift Manager's responsibilities for the required controls of the identified radiation areas?

a. Areas >1000 mrem per hour shall be provided with locked doors and the keys shall be maintained under the administrative control of the Shift Manager and/or Health Physics Supervision.

b. A Health Physics qualified individual reporting directly to the Shift Manager is responsible for providing positive control over the activities within Locked High Radiation areas.

c. Entrance to radiation areas >1000 mrem per hour shall be controlled by requiring issuance of a radiation work permit approved by the Shift Manager.

d. Any individual or group of individuals requiring entry into a Locked High Radiation area shall receive Shift Manager permission prior to entry.

Answer
a.
REFERENCE
T.S. 6.13 High Radiation Area
NEW
FUNDAMENTAL
K/A Knowledge of radiological safety procedures pertaining to licensed operator duties, such as response to radiation monitoring alarms, containment entry requirements, fuel handling responsibilities, access to locked high radiation areas, aligning filters, etc. RO Imp / SRO Imp – 3.4 / 3.8
Justification
TS 6.13.b: "in one hour a dose >1000 mrem shall be provided with locked doors to prevent unauthorized entry, and the keys shall be maintained under the administrative control of the shift manager on duty and/or health physics supervision."
a. Correct. T.S. 6.13.b requires areas >1000 mrem/hr be provided with locked doors and the keys controlled by the Shift Manager and/or Health Physics Supervision.
b. Wrong. A Health Physics qualified individual is only required IF a radiation monitoring device is not available to provide cumulative dose or dose rate and they are not required to report to the shift manager. This is part of TS 6.13.a.3
c. Wrong. Radiation Work Permits are required to be issued per T.S. 6.13 however they are not required to be approved by the Shift Manager. One of the controls required is an RWP.
d. Wrong. Shift Manager permission is not required for entry into Locked High Rad areas. Authorization prior to entry is required; however, not by the Shift Manager.

Question Number: 099
Exam Date 20090413
Exam Level S
K/A 2.4.26
Question
Given the following:

- Shift turnover is completed with Night Shift, on-coming.
- Reactor Power is 70%.
- The operating crew is at minimum Technical Specification staffing.
- The nuclear auxiliary operator assigned to the Fire Brigade has been injured and contaminated.
- The injured nuclear auxiliary operator requires transport to the hospital.

Which replacement will meet the time and personnel requirements to maintain BOTH the Fire Brigade and the On-duty Shift Complement?

a. Within 1 hour, a licensed operator and a contractor radiation technologist.

b. Within 2 hours, a licensed operator and a chemistry technologist.

c. Within 2 hours, a nuclear auxiliary operator and a radiation technologist.

d. Within 4 hours, a nuclear auxiliary operator and a security guard.

Answer
c.
REFERENCE
T.S. 6.2.b.1 and 6.2.b.3
Kewaunee Power Station Fire Protection Program Plan, 7.8.4 and 7.9.
NEW
FUNDAMENTAL
K/A Knowledge of facility protection requirements, including fire brigade and portable fire fighting equipment usage.
Justification:
Each on-duty shift complement shall consist of at least:
One shift manager (SRO)
Two licensed reactor operators
Two nuclear auxiliary operators
One radiation technologist
While above COLD SHUTDOWN, the on-duty shift complement shall consist of the personnel above and an additional SRO.

Minimum compliment for the Fire Brigade is 5 persons, normally:
- One Radiation Technician
- Two Security Guards
- Two Operators (NAOs)

a. The contractor radiation technologist is incorrect because this individual would not be trained in fire fighting, and the licensed operator would also be incorrect for fire fighting. This does identify the correct time limit for re-staffing.
b. Wrong. The chemistry technologist does not meet the Rad Tech requirement. The time

line is correct and the licensed operator can support the shift complement; however, the fire brigade requirements would not be met.

c. Correct. The time required is within 2 hours and the correct requirements for personnel are a Nuclear Auxiliary Operator and a Rad Tech.

d. Wrong. The time frame is incorrect. The security guard does not meet the required shift complement which requires a Rad Tech. This replacement could satisfy the fire brigade manning requirement.

Question Number: 100
Exam Date 20090413
Exam Level S
K/A 2.4.30
Question
Given the following:

- ORT-DGM-002, "Technical Support Center Diesel Generator Monthly Availability Test," is in progress.
- During the diesel run, the Equipment Operator reports that the TSC Diesel is on fire.
- The Fire Brigade has been dispatched to the scene.
- After 10 minutes, the Fire Brigade Leader reports that the fire is still active.

Two hours have now elapsed from the initial report of fire.

By this time, which of the following outside agencies is NOT required to have been notified of this event?

a. NRC.

b. State Warning Center.

c. Manitowoc County Sheriff.

d. Nuclear Electric Insurers Limited (NEIL).

Answer
d.
REFERENCE
FPP-08-13 Fire Report, 5.2
EPIP-AD-07 Emergency Notifications, 3.1
GNP-11.08.04, Reportability Determinations, 6.2.1 and Table 1, item 1
BANK
HIGHER
K/A Knowledge of events related to system operation/status that must be reported to internal organizations or external agencies, such as the state, the NRC, or the transmission system operator.
Justification:
a. Wrong. Per GNP 11.08.04 Section 6.2.1 & Table 1 Item 1, NRC phone notifications shall be initiated prior to exceeding the notification time limit of 1 hour.
b. Wrong. Per EPIP-AD-07 Emergency Notification to the State Warning Center via the NARS form is required within 15 minutes of Classification.
c. Wrong. Per EPIP-AD-07 Emergency Notification to the Manitowoc County Sheriff via the NARS form is required within 15 minutes of Classification.
d. Correct. Notification is required to NEIL per the following: FPP-08-13 section 5.2: The Fire Marshal shall notify NEIL, as soon as possible after the incident, when a fire involves activation or malfunction of a fixed fire extinguishing or detection system. Based on the question, two hours have elapsed since the fire and this does not exceed a timed notification requirement for NEIL.