Examination Outline Cross Reference	Level	RO	SRO
	Tier Number	1	
	Group Number	1	
	K / A Number	295001 AK2.04	
	Importance Rating	3.3	

Proposed Question 1:

The reactor is operating at 90 percent Rated Thermal Power (RTP) with all systems in their normal line-up and operable. A malfunction develops in one Recirc MG set that results in one reactor recirc pump slowing down.

In response to this malfunction, the turbine control valves will:

A. throttle open and generator output will remain constant.

B. remain stationary and generator output will decrease.

C. remain stationary and generator output will remain constant.

D. throttle down and generator output will decrease.

Proposed Answer: D is correct because with decreasing reactor recirc flow, reactor power will decrease. Decreasing power will cause steam flow and pressure to decrease. This is sensed by the EHC system and the turbine governor valves will throttle closed to maintain pressure at setpoint. With the governor valves throttling closed, generator output will decrease.

Technical References: COR0022202R2-S-Reactor Recirc
COR0020902R12-S-Digital Electro-Hydraulic Control System

Proposed References to be provided to applicants during examination:

Learning Objective: LO 4t, u and LO 3a, c

Question Source: New Question

Question History: None

Question Cognitive Level: Comprehension or Analysis XX

10 CFR Part 55 Content: 55.41.5

Comments:
(KA) Relationship between the loss of forced core flow circulation and the reactor or turbine pressure regulating system.

Answer (a), , is incorrect because with decreasing reactor recirc flow, reactor power will decrease. Decreasing power will cause steam flow and pressure to decrease. This is sensed by the EHC system and the turbine governor valves will throttle closed (not open) to maintain pressure at setpoint. With the governor valves throttling closed, generator output will decrease.

Answer (b), , is incorrect because is incorrect because with decreasing reactor recirc flow, reactor power will decrease. Decreasing power will cause steam flow and pressure to decrease. This is sensed by the EHC system and the turbine governor valves will throttle closed (not remain stationary) to maintain pressure at setpoint. With the governor valves throttling closed, generator output will decrease.

Answer (b), , is incorrect because is incorrect because with decreasing reactor recirc flow, reactor power will decrease. Decreasing power will cause steam flow and pressure to decrease. This is sensed by the EHC system and the turbine governor valves will throttle closed (not remain staionary) to maintain pressure at setpoint. With the governor valves throttling closed, generator output will decrease.

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number	1	
	Group Number	1	
	K / A Number	295001	AK3.05
	Importance Rating	3.2	

Proposed Question 2:

The reactor is shut down, all rods are fully inserted, and core cooling is being provided by natural circulation. Bottom head to steam dome temperature differential is 150˚F.

A reactor recirc pump should not be started or loop flow raised in this condition because:

A. this will thermal shock the reactor recirc pump seals.

B. low speed reverse rotation will occur in the idle recirculation pump.

C. temperature-induced stress may occur on the bottom head to CRD stub tube welds.

D. this will add positive reactivity to the core.

Proposed Answer: "C" Temperature-induced stress may occur on the bottom head to CRD stub tube welds. Procedure 2.4RR step 5.9 states, "If RR pump is started or loop flow raised during the period of time when a temperature stratification equal to or exceeding 145˚F exists, serious temperature induced stress may occur on the reactor bottom head to CRD stub tube welds and in-core housing welds."

Technical References: Procedure 2.4RR, "Reactor Recirculation Abnormal," Revision 16, Steps 5.8 & 5.9

Technical Specification Bases, B3.4.1, "Recirculation Loops Operating," Revision 0

USAR §5.5, "Events Resulting in a Core Coolant Flow Decrease,“ 9/19/2000

Proposed References to be provided to applicants during examination: None

Learning Objective: Lesson COR002-22-02, "Reactor Recirculation," Revision 20, Enabling Objective 4(n)

Question Source: New Question

Question History: Never Used

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.5

Comments:
(KA) Reasons for reduced loop operating requirements as applies to a loss of forced core flow circulation

Answer (a), is incorrect because reactor recirc pump seals are not a concern in this scenario.

Answer (b), Low speed reverse rotation will occur in the idle recirculation pump, is incorrect because (1) reverse rotation is not expected to occur in the idle pump if procedure 2.4RR, Attachment 1, step 1.1.3 was performed when the pumps tripped, and (2) reverse rotation would not damage the pump per procedure 2.4RR step 5.14

Answer (d), is incorrect because this is not a concern with all control rods fully inserted.

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number	1	
	Group Number	1	
	K / A Number	295003	AK1.05
	Importance Rating	2.6	

Proposed Question 3:

The plant is in Mode 1 with all systems in a normal configuration and operable. An electrical fault occurs resulting in the loss of RPS PP 1B.

With respect to the MSIVs, the:

A. inboard MSIV solenoids will de-energize and the inboard MSIVs will close.

B. outboard MSIV solenoids will de-energize and the outboard MSIVs will close.

C. inboard MSIV solenoids will de-energize and the inboard MSIVs will remain open.

D. outboard MSIV solenoids will de-energize and the outboard MSIVs will remain open.

Proposed Answer: D - The loss of RPS PP 1B will de-energize the inboard MSIV solenoids but the fail safe feature requires a loss of both power supplies before the MSIVs will close. Therefore the MSIVs remain open

Technical References: COR0021402R14-S-OPS-Main Steam, Rev 14

Proposed References to be provided to applicants during examination:

Learning Objective 7j

Question Source: New Question

Question History: None

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.7

Comments:

(KA) Knowledge of the operational implications of failsafe component design applied to partial or complete loss of AC power

Answer (a) is incorrect because the MSIVs remain open.

Answer (b) is incorrect because the outboard solenoids remain energized and the outboard MSIVs remain open.

Answer (d) is incorrect because the outboard MSIV solenoids remain energized.

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number	1	
	Group Number	1	
	K / A Number	295004	AA2.01
	Importance Rating	3.2	

Proposed Question 4:
While performing a walk down of the control room boards, a licensed operator notices there are no status indicating lights for the Core Spray Pump lit (running or not running). The operator also notes all the other ECCS indicating lights are lit as they normally are. After obtaining permission, the operator changes bulbs for the red and green indicating lights with no change in status.

The most probable cause for the loss of the pump indicating lights is:

A. a blown DC fuse in the control circuit for the pump motor.

B. a blown AC fuse in the control circuit for the pump motor.

C. the loss of the DC bus that provides power to the pump motor.

D. the loss of the AC bus that provides power to the pump motor.

Proposed Answer:	A is correct because DC control power circuit provides power for the pump indicating lights and with a blown fuse the lights will not have power.
Technical References:	COR002060R16-S-OPS-Core Spray

Proposed References to be provided to applicants during examination:

Learning Objective

Question Source:	New Question	
Question History:	None	
Question Cognitive Level:	Comprehension or Analysis	XX
10 CFR Part 55 Content:	55.41.7	

Comments:
(KA) Ability to determine or interpret the cause of partial or complete loss of DC power.

Answer (b) is incorrect because control power for the motor is from DC power.

Answer (c) is incorrect because DC power is not provided to the pump motor.

Answer (d) is incorrect because if the AC power supply were lost to the Core Spray pump motor with no other problems, the indicating lights would still be lit since they are from DC power.

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number	1	
	Group Number	1	
	K / A Number	295005	AA1.02
	Importance Rating	3.5	

Proposed Question 5:

The reactor is operating at 95% rated power when the main turbine trips on high vibration.

Which of the following is one of the actions taken by the reactor operator immediately following the scram?

A. Verify the REFUEL MODE SELECT PERMISSIVE indicator is on.

B. Break condenser vacuum.

C. Ensure the Feedwater Control Station (RFC-CS-RFPTA[B]) for the running feedpump is in AUTO.

D. Trip all but two Condensate Pumps.

Proposed Answer: (A) - This answer is correct per step 1.5 of Attachment 1 in procedure 2.1.5, Reactor Scram.

Technical References: Procedure 2.1.5, “Reactor SCRAM,” Revision 45

Proposed References to be provided to applicants during examination: None

Learning Objective

Question Source: New Question

Question History: Never Used

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

10 CFR Part 55 Content: 55.41.7

Comments:
(KA) Ability to monitor or operate RPS as applies to main turbine generator trip.

Answer (b) is incorrect because the Mode switch is placed in REFUEL.

Answer c) is incorrect because the Reactor Feedpump control station is placed in MDEM.

Answer (d) is incorrect because all but one condensate pump is tripped.

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number	1	
	Group Number	1	
	K / A Number	295006	2.4.45
	Importance Rating	3.3	

Proposed Question 6:

A routine shutdown of the unit is being performed when at about 29 percent power the "TSV & TCV CLOSURE TRIP BYP CHAN A/B", annunciator on Panel 9-5-2 goes into an alarm state.

This alarm:

A. is an expected alarm.

B. indicates a problem with the main turbine trip logic.

C. indicates a problem with the RPS scram logic.

D. indicates the Bypass switch for the TSV & TCV Scram logic has been placed in BYPASS.

Proposed Answer: A - This is an expected alarm when power is reduced to less than 30 percent. It informs the operator the turbine trip induced scram is bypassed due to power being less than 30 percent.

Technical References: COR0022102-16-S-OPS Reactor Protection System, Revision 16.

Proposed References to be provided to applicants during examination: None

Learning Objective

Question Source: New Question

Question History: None Last NRC Examination NA

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

10 CFR Part 55 Content: 55.43.5

Comments:
(KA) Ability to prioritize and interpret the significance of each annunciator and alarm as applied to 295006, SCRAM.

Answer (b) is incorrect because this alarm does not reflect any problem with the main turbine trip logic.

Answer (c) is incorrect because this alarm does not reflect any problem with the RPS scram logic.

Answer (d) is incorrect because this alarm does not reflect any part of the trip logic being in BYPASS.

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number	1	
	Group Number	1	
	K / A Number	295016 AA2.01	
	Importance Rating	4.1	

Proposed Question 7:

The Control Room has been abandoned because of the presence of toxic gas. A reactor scram was not performed prior to leaving the Control Room. Assuming that only the minimum number of required on-shift operators are available, and that personnel are present only at those stations specified in procedure 5.1ASD, Alternate Shutdown, direct indication of reactor power can be read at:

A. the APRM cabinet located in the Critical Switchgear Room.

B. the PCIS in the Technical Support Center.

C. the chart recorder on the Alternate Shutdown Room ADS/REC panel.

D. none of the stations outside the control room.

Proposed Answer: (D) None of the stations outside the control room. This answer is correct because there is no local instrumentation for power outside the Control Room (per M. Barton, 12/17/04).

Technical References: Procedure 5.1ASD, “Alternate Shutdown,” Revision 2

Procedure 5.4FIRE-S/D, “Fire Induced Shutdown from Outside the Control Room,” Revision 8

Proposed References to be provided to applicants during examination: None

Learning Objective

Question Source: New Question

Question History: Never Used

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41.10 55.43.5

Comments:
(KA) Ability to determine or interpret reactor power as applies to control room abandonment.

Answer (A), APRM cabinet located in the Critical Switchgear Room, is incorrect because there is no APRM cabinet located in the Critical Switchgear Room.

Answer (B), PCIS in the Technical Support Center, is incorrect because although PCIS indication of reactor power is available in the TSC, procedure 5.1ASD does not direct an operator to the TSC.

Answer (C), Chart recorder on the Alternate Shutdown Room ADS/REC panel, is incorrect because there is no chart recorder located on the ADS/REC panel in the Alternate Shutdown Room.

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number	1	
	Group Number	1	
	K / A Number	295018 AK3.02	
	Importance Rating	3.3	

Proposed Question 8:

The reactor is operating at 100% RTP with the A REC Heat Exchanger and pump in service. After isolating the B REC Heat Exchanger (MO-713, MO-651, and TCV-451B all closed) due to a tube leak, REC temperature has been slowly trending up. When REC temperature reaches 95˚F, the Control Room Supervisor orders power be reduced to 80%.

The reason for the power reduction is to reduce REC heat loads in order to:

A. maintain REC Pump A discharge pressure below 105 psig.

B. minimize REC Pump A cavitation due to the higher REC fluid temperature.

C. reduce the chance of damaging the Reactor Recirculation Pumps MG set(s).

D. reduce the chance of a high drywell pressure SCRAM.

Proposed Answer: (D) Reduce the chance of a high drywell pressure SCRAM. This answer is correct because COR002-19-02 IV(F)(2) states: "During a partial loss of REC, a reactor power reduction will tend to minimize the heat escalation and therefore reduce the chances of a high drywell pressure scram..." In addition, procedure 5.2REC, step 4.7.3.1 states, "Reduce reactor power, as necessary, to maintain REC HX outlet temperature to ≤ 98˚F per Procedure 2.1.10."

Technical References: Procedure 2.2.65.1, "REC Operations," Revision 36

Procedure 5.2REC, "Loss of REC," Revision 5

Procedure 5.2SW, "Service Water Casualties," Revision 11

Lesson COR002-19-02, "Reactor Equipment Cooling", Revision 16

Proposed References to be provided to applicants during examination: None

Learning Objective	COR002-19-02, “Reactor Equipment Cooling”, Revision 16, Enabling Objective 6(c)
Question Source:	New Question
Question History:	Never Used
Question Cognitive Level:	Comprehension or Analysis
10 CFR Part 55 Content:	55.41.5

Comments:
(KA) Knowledge of the reasons for reactor power reduction as applied to partial or total loss of component cooling water.

Answer (A), Maintain REC Pump A discharge pressure below 105 psig, is not correct because REC pump discharge pressure is not related to reactor power and REC temperature. 105 psig is the REC high discharge pressure alarm setpoint.

Answer (B), Maintain REC Heat Exchanger outlet temperature above the dew point, is not correct because the REC heat exchanger outlet temperature is controlled above the dew point during normal operations to prevent condensation on piping from affecting live equipment. This sets a minimum REC temperature while the question concerns the consequences of a maximum REC temperature.

Answer (C), Reduce the chance of damaging Recirculation Pumps due to slipping and overheating of the associated MG set(s), is not correct because (1) while slippage and overheating the MG sets might cause a Recirculation Pump trip, it would not directly damage the associated pump, and (2) pump damage may be caused by an extended reduction in cooling for Recirculation Pump seal water, and bearing and oil coolers [COR002-19-02 IV(F)(6)]. COR002-19-02 IV IV(F)(6) does state: “...RRMG Set Oil system heat exchangers are also supplied by REC. With a loss of REC, the temperature of the oil would rise, causing increased slippage and overheating at the MG sets.”

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number	1	
	Group Number	1	
	K / A Number	295019 2.1.07	
	Importance Rating	3.7	

Proposed Question 9:

The plant is in Mode 1 with all systems in their normal configuration and operable. A weld fails on an instrument air header and consequently, instrument air header pressure is trending down.

If instrument air header pressure continues a downward trend, the first plant response to this event the operator should expect to see is the:

A. the MSIVs drifting closed.

B. the ADS valves drifting open.

C. Service Air System Isolation valve (SA-PCV-609) closes.

D. reactor feedpump turbine governor valves drifting closed.

Proposed Answer: C - The referenced procedure states this valve will automatically close at 77 psig.

Technical References: Procedure 5.2Air, Loss of Instrument Air, Revision 5, Step 4.1.

Proposed References to be provided to applicants during examination: None

Learning Objective

Question Source: New Question

Question History: None Last NRC Examination NA

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

10 CFR Part 55 Content: 55.43.5

Comments:

(KA) Ability to evaluate plant performance and make operational judgements based on operating characteristics, reactor behavior, and instrument interpretation, as applied to 295019, Partial or Total Loss of Instrument Air

Answer (a) is incorrect because MSIVs have an accumulator to maintain air pressure.

Answer (b) is incorrect because the ADS valves have an accumulator to maintain operability.

Answer (d) is incorrect because the RFPT throttle valves will not drift closed on a loss of air.

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number	1	
	Group Number	1	
	K / A Number	295021	2.4.02
	Importance Rating	3.9	

Proposed Question 10:

The reactor is shutdown with RHR A in service for shutdown cooling (SDC). A loss of off-site power occurs followed by an automatic start of the emergency diesel generators (EDGs). Reactor water level is 55 inches and drywell pressure is 0 psig.

With regard to restoring SDC, operator action is:

A. not required because the breaker for RHR A pump will remain closed and the pump will restart when the bus is re-energized by the EDG.

B. not required because the breaker for RHR A pump will trip open and will automatically reclose when the bus is reenergized by the EDG.

C. required for restarting the RHR A pump only.

D. required for restarting the RHR A pump and restoring the RHR valve lineup.

Proposed Answer: C

Technical References: COR0020802R13-S-Diesel Generator
COR0010101R26-S-OPS AC Distribution

Proposed References to be provided to applicants during examination: None

Learning Objective

Question Source: New Question

Question History: New

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

10 CFR Part 55 Content: 55.41.7

Comments:
(KA) Knowledge of system setpoints, interlocks, and automatic actions associated with EOP entry conditions, as applied to 295021, Loss of Shutdown Cooling. [KA Deleted]

(KA) Knowledge of the purpose and function of major system components and controls as applied to 295021, Loss of Shutdown Cooling.

Answer (a) is incorrect because on a loss of power the 4160 volt loads are stripped from the bus.

Answer (b) is incorrect because an automatic reclosure does not occur on a low voltage in the absence of a LOCA signal.

Answer (d) is incorrect because the 480 volt loads do not change position on a loss of power.

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number	1	
	Group Number	1	
	K / A Number	295023	AA1.04
	Importance Rating	3.4	

Proposed Question 11:

The unit is in an outage with fuel movement in progress. Control Room annunciator Q-1/B-7 (ERP Discharge High Rad) just went into an alarm condition. A non-licensed operator reports there are no indications of fuel pool leakage and fuel pool level remains normal.

If this is a valid annunciator indicating a refueling accident is in progress, what other annunciators should also be received:

a) 9-3-1/A-9, Reactor Building High Rad
9-3-1/A-10, Refuel Area High Rad
9-3-1/C-10, Radwaste Building High Rad

b) 9-3-1/A-9, Reactor Building High Rad
9-3-1/A-10, Refuel Area High Rad
9-4-1/E-5, Rx Bldg Vent High Rad

c) 9-3-1/A-10, Refuel Area High Rad
9-4-1/E-4, Rx Building Vent Hi Hi Rad
9-4-2/A-3, Fuel Pool Cooling Trouble

d) 9-3-1/A-9, Reactor Building High Rad
9-3-1/B-9, TIP Room High Rad
Q-1/C-1, Drywell Gas High Activity

Proposed Answer: (B) 9-3-1/A-9, Reactor Building High Rad & 9-3-1/A-10, Refuel Area High Rad & 9-4-1/E-5, Rx Bldg Vent High Rad. This answer is correct because monitor RA-3 alarms at 2.3 mR/h (A-9), monitor RA-2 alarms at 10 mR/h (A-10), and channels ABCD on the reactor building vent alarm at 5 mR/h (E-5).

Effluent monitor release rates and area radiation monitor results are not directly comparable. Annunciator B-7 alarms at 7.83E3 µCi/s which is 9-10 times the normal station release rate; therefore area radiation levels on the refuel floor can be expected to be at least 10 times normal operating levels, resulting in minimum radiation levels between 20 and 50 mR/h, which is greater than the alarm setpoints for monitors RA-1, RA-2, RA-3, Rx Bldg Vent High Rad, and Rx Bldg Vent Hi Hi Rad. FSAR Section 4.6 states that it is assumed that a Group 6 isolation occurs during a fuel handling accident, which requires Rx Building Vent > 10 mR/h with actual refuel floor radiation levels higher than vent radiation levels (the FSAR does not provide expected refuel floor radiation levels during a refueling accident).

Technical References: Lesson COR001-18-01, “Radiation Monitoring,” Revision 16

Procedure 5.1RAD, “Building Radiation Trouble,” Revision 8

Procedure 2.3_9-3-1, “Panel 9-3 Annunciator 9-3-1,” Revision 7

Procedure 2.3_9-4-1, “Panel 9-4 Annunciator 9-4-1,” Revision 12

Procedure 2.3_Q-1, “Panel Q Annunciator Q-1,” Revision 7

FSAR, Section 6.4, “Fuel Handling Accident,”

Proposed References to be provided to applicants during examination: None

Learning Objective Lesson COR001-18-01, Enabling Objective 11(d)

Question Source: New Question

Question History: Never Used

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.7

Comments:
(KA) Ability to monitor or operate radiation monitoring equipment as applied to refueling accidents.

Answer (A), <u>9-3-1/A-9, Reactor Building High Rad & 9-3-1/A-10, Refuel Area High Rad & 9-3-1/C-10, Radwaste Building High Rad</u>, is not correct because radiation conditions in the Radwaste Building could be expected to be at or near normal during a refueling accident (less than the alarm setpoints of monitors 22 through 30), so annuciator 9-3-1/C-10 would not be expected

Answer (C), <u>9-3-1/A-10, Refuel Area High Rad & 9-4-1/E-4, Rx Building Vent Hi Hi Rad & 9-4-2/A-3, Fuel Pool Cooling Trouble</u>, is not correct because a refueling accident which is not associated with draining the fuel pool would not bring in annunciator 9-4-2/A-3

Answer (D), 9-3-1/A-9, Reactor Building High Rad & 9-3-1/B-9, TIP Room High Rad, & Q-1/C-1, Drywell Gas High Activity, is not correct because (1) the TIP room would not be expected to exceed 200 mR/h on monitor RA-6 due to a refueling accident due to distance and shielding, (2) a refueling accident would not cause actual gaseous activity to be present in the drywell, and (3) the radiation shine from the refuel floor through the shielding formed by the concrete in the ceiling and the fuel pool water might not be sufficient to trip the alarm setpoint on monitor RM-4C (1.9E-5 µCi/cc, which is about 10,000 times its normal value)

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number	1	
	Group Number	1	
	K / A Number	295024	EA1.05
	Importance Rating	3.9	

Proposed Question 12:

Procedure 2.4PC, Primary Containment Control, has been entered due to indications of leakage from the RCS sample line. Drywell pressure is 0.50 psig and is rising slowly.

If pressure continues to increase, the operator should expect a reactor scram to occur when drywell pressure reaches _________ .

A. 0.75 psig

B. 1.50 psig

C. 1.84 psig

D. 2.00 psig

Proposed Answer: (C) 1.84 psig. This answer is correct because procedure 4.13.2, Step 2.3 states that pressure switches 12A, 12B, 12C, and 12D, supply the Reactor Protection System to initiate a reactor Scram at ≤ 1.84 psig.

Technical References: COR02-21-02, "Reactor Protection System," Revision 16

Procedure 2.4PC, "Primary Containment Control," Revision 7

Procedure 4.13.2, "Drywell and Suppression Chamber Preassure," Revision 13

Proposed References to be provided to applicants during examination: None

Learning Objective: COR02-21-02 Enabling Objectives 10(k) & 12

Question Source: New Question

Question History: Never Used

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41.7

Comments:
(KA) Ability to monitor or operate RPS as applies to high drywell pressure.

Answer (A), 0.75 psig, is not correct because this value is associated with Technical Specification 3.6.1.4 and not with RPS actuation.

Answer (B), 1.50 psig, is not correct because this value requires a manual reactor Scram per procedure 2.4PC step 4.1, and is not associated with RPS actuation.

Answer (D), 2.00 psig, is not correct because this value is higher than the actual maximum setpoint of 1.84 psig. Per procedure 4.13.2, “Drywell and Suppression Chamber Preassure,” Revision 13, Step 2.2, 2.0 psig is associated with containment spray interlock circuits.

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number	1	
	Group Number	1	
	K / A Number	295025	EK2.05
	Importance Rating	4.1	

Proposed Question 13:

A reactor transient has occurred that resulted in an MSIV closure and a scram due to high RPV pressure. All the control rods fully inserted. Due to the high pressure condition, 3 SRVs initially opened.

Based on this transient, and with no operator action, the SRVs should maintain RPV pressure between:

A. 1015 psig and 875 psig.

B. 1080 psig and 960 psig

C. 1090 psig and 970 psig

D. 1100 psig and 980 psig.

Proposed Answer: A - This set of conditions will initiate the low-low-set feature of the SRVs thus maintaining pressure between 1015 and 875 psig.

Technical References: COR0021602R14-S-OPS Nuclear Pressure Relief

Proposed References to be provided to applicants during examination:

Learning Objective

Question Source: New Question

Question History: None Last NRC Examination NA

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

10 CFR Part 55 Content: 55.41.7

Comments:
(KA) Knowledge of the interrelations between high reactor pressure and safety relief valves.

Answer (b), , is incorrect because with a high pressure scram signal low low set is in effect which reduces the lift setpoint to 875 psig.

Answer (c), , is incorrect because with a high pressure scram signal low low set is in effect which reduces the lift setpoint to 875 psig.

Answer (d), , is incorrect because with a high pressure scram signal low low set is in effect which reduces the lift setpoint to 875 psig.

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number	1	
	Group Number	1	
	K / A Number	295026	EK2.01
	Importance Rating	3.9	

Proposed Question 14:

A plant startup is in progress and a special test involving cycling the SRVs with RCIC in operation has just been completed. The current plant conditions are:

- ✓ Reactor power is now 12 percent
- ✓ Two bypass valves are open
- ✓ Suppression pool temperature is 102 degrees
- ✓ Suppression pool level is 3 inches higher than normal
- ✓ The RHR B heat exchanger and RHR Pump B are currently in service in the Fuel Pool Cooling Assist mode of operation.
- ✓ RHR Pump A is in service to reduce suppression pool level by rejecting water to radwaste.

In this configuration, suppression pool cooling may be placed in service using RHR:

A. Pump A only.

B. Pump D only.

C. Pump A or C only.

D. Pump A or D only.

Proposed Answer: C

Technical References: COR0022302R20-S-OPS Residual Heat Removal

Proposed References to be provided to applicants during examination: None

Learning Objective

Question Source: New Question

Question History: None Last NRC Examination NA

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41.7	

Comments:
(KA) Knowledge of the interrelations between suppression pool high water temperature and suppression pool cooling.

Answer (a) is incorrect because the C RHR Pump may also be used.

Answer (b) is incorrect because with heat exchange B in service for FPC Pump D or B cannot be lined up for suppression pool cooling.

Answer (d) is incorrect because of the same reason stated in answer b.

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number	1	
	Group Number	1	
	K / A Number	295028	2.4.50
	Importance Rating	3.3	

Proposed Question 15:

The unit is operating at 100% power with the following plant conditions:

- ✓ Drywell pressure is 0.25 psig
- ✓ Drywell temperature is 125˚F
- ✓ 4 Drywell Fan Coil Units are in operation.
- ✓ Annunciators H-1/C-1 (Drywell FCU C Hi Discharge Temperature), H-1/A-2 (Drywell Zone 1 High Temperature), and H-1/C-2 (Drywell Zone 3 High Temperature) are received.

The annunciator response procedure directs the operator to check _________ , ensure _______ , and enter _______ :

A. CRT alarm messages and temperature indications on recorders
the drywell REC lineup is correct
procedures 2.4PC and 5.2REC

B. temperature indication on recorders and PCIS
the drywell REC lineup is correct
procedure 4.13.3, Primary Containment Temperature

C. temperature indication on recorders
all REC pumps are running
procedure 6.PC.306, Primary Containment Temperature Element Comparison Check

D. that drywell pressure is between 0.25 and 0.45 psig
valves REC-MO-702 and REC-MO-709 are open
procedures 2.4PC and 5.2REC

Proposed Answer: (A) Check CRT alarm messages and temperature indication on recorders & Ensure drywell REC lineup is correct & Enter procedures 2.4PC and 5.2REC. This answer is correct because lists annunciator procedure A-2, steps 2.1, 2.2, 2.4, and 2.6.

Technical References: Procedure 2.4PC, “Primary Containment Control,” Revision 7

Procedure 4.13.3, “Primary Containment Temperature,” Revision 18

Procedure 2.3 H-1, “Panel H Annunciator H-1,” Revision 2

COR-002–03-02, “Containment,” Revision 18

Proposed References to be provided to applicants during examination: None

Learning Objective: Lesson COR-002–03-02, Enabling Objective 16(a)

Question Source: New Question

Question History: Never Used

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content:

Comments:
(KA) Ability to verify alarm setpoints and operate controls as identified in the alarm response manual as applied to 295028, High Drywell Temperature

Answer (B), <u>Check temperature indication on recorders and PCIS & Ensure drywell REC lineup is correct & Enter procedure 4.13.3, Primary Containment Temperature</u>, is incorrect because (1) the annunciator procedure does not direct the operator to verify temperature on PCIS, and (2) the annunciator procedure directs the operator to enter procedures 2.4PC and 5.2REC and not procedure 4.13.3.

Answer (C), <u>Check temperature indication on recorders & Ensure all REC pumps are running & Enter procedure 6.PC.306</u>, is incorrect because (1) ensuring all REC pumps are running is a step in procedure 5.2REC, and (2) the annunciator procedure directs the operator to enter procedures 2.4PC and 5.2REC and not procedure 6.PC.306.

Answer (D), <u>Check that drywell pressure is between 0.25 and 0.45 psig & Ensure valves REC-MO-702 and REC-MO-709 are open & Enter procedures 2.4PC and 5.2REC</u>, is incorrect because checking drywell pressure and ensuring the REC valve positions are both steps in procedure 2.4PC and are not in the annunciator procedure.

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number	1	
	Group Number	1	
	K / A Number	295030	EA1.02
	Importance Rating	3.4	

Proposed Question 16:

An EOP caution states that suppression pool water level should be at least _______ feet in order to operate RCIC without vortexing occurring at the pump suction.

A. 10

B. 8

C. 6

D. 4

Proposed Answer: C

Technical References: EOP 5.8, Attachment 2, Graph 4, Vortex Limits

Proposed References to be provided to applicants during examination: None

Learning Objective

Question Source: New Question

Question History: None Last NRC Examination NA

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

10 CFR Part 55 Content: 55.41.7

Comments:
(KA) Ability to operate or monitor RCIC as applies to low suppression pool water level.

Answer (a) is incorrect because the correct level is 6 feet.

Answer (b) is incorrect because the correct level is 6 feet.

Answer (d) is incorrect because the correct level is 6 feet.

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number	1	
	Group Number	1	
	K / A Number	295031	2.1.31
	Importance Rating	4.2	

Proposed Question 17:

A feedwater level control system malfunction has resulted in RPV level trending down. The operator notices the trend at +10 inches and takes manual control of feedwater to recover RPV level before the low level scram occurs.

To most accurately monitor RPV level as "seen" by the Reactor Protection System, the operator should be monitoring the:

A. wide range indicator on Panel 9-5.

B. narrow range indicator on Panel 9-5.

C. narrow range indicator on Panel 9-4.

D. wide range indicator on Panel 9-4.

Proposed Answer: B - The RPS instrumentation comes off the narrow range and the indicators are on Panel 9-5.

Technical References: COR0021502R15-S-OPS Nuclear Boiler Instrumentation

Proposed References to be provided to applicants during examination:

Learning Objective

Question Source: New Question

Question History: None Last NRC Examination NA

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

10 CFR Part 55 Content: 55.41 (7)

Comments:
(KA) Ability to locate control room switches and indications and determine that they are correctly reflecting the desired plant lineup as applied to 295031, Reactor Low Water Level.

Answer (a), is incorrect because the scram signal is generated from the narrow range inst.

Answer (c), is incorrect because there are no narrow range indicators on the 9-4 panel.

Answer (d), is incorrect because the scram signal is generated from the narrow range inst.

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number	1	
	Group Number	1	
	K / A Number	295037	EA2.02
	Importance Rating	4.1	

Proposed Question 18:

A steam leak resulted in a reactor scram due to high drywell pressure. About one-half of the control rods failed to insert and reactor power has stabilized at 7 percent reactor power. The reactor recirc pumps were tripped and reactor water level is +20 inches and lowering.

Based on current conditions, the _________ range indicator is the most accurate indicator of RPV level.

A. narrow

B. wide

C. upset

D. shutdown

Proposed Answer: B - The wide range indicator is calibrated for these conditions.

Technical References:

Proposed References to be provided to applicants during examination: None

Learning Objective

Question Source: New Question

Question History: None Last NRC Examination NA

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

10 CFR Part 55 Content: 55.41.10____ 55.43.5

Comments:

(KA) Ability to determine and/or interpret reactor water level as applied to a SCRAM condition present and reactor power above APRM downscale or unknown.

Answer (a), , is incorrect because narrow range is calibrated with the reactor recirc pumps running.

Answer (c), , is incorrect because upset range is calibrated with the reactor recirc pumps running.

Answer (d), , is incorrect because shutdown range is calibrated at 0 psig RPV pressure.

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number	1	
	Group Number	1	
	K / A Number	295038	EK3.03
	Importance Rating	3.0	

Proposed Question 19:

In order to maintain the dose rates for control room personnel below desired limits, the control room ventilation system will automatically realign in the event of a:

A. high high reactor building area radiation monitor signal.

B. PCIS Group 6 isolation signal.

C. high high control room Continuous Air Monitor (CAM) alarm.

D. control building Essential Heating and Ventilation failure alarm.

Proposed Answer: B is the only signal that realigns the system for the reason stated.

Technical References: COR0010801R13-S-OPS, Heating, Ventilation, and Air Conditioning, page 60.

Proposed References to be provided to applicants during examination: None

Learning Objective: None

Question Source: New Question

Question History: Never Used

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

10 CFR Part 55 Content: 41.5 & 45.6

Comments:
(KA) Knowledge of the reasons for control room ventilation isolation during conditions of high offsite release rate.

Answer (a) will not cause an automatic realignment.

Answer c) will not cause an automatic alignment.

Answer (d) will not cause an automatic alignment.

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number	1	
	Group Number	1	
	K / A Number	600000	AK1.02
	Importance Rating	2.9	

Proposed Question 20:

A fire and the followup fire fighting efforts have resulted in the shift manager directing the operating crew to evacuate the control room.

If time permits, prior to leaving the control room the reactor operator should lineup the feed and condensate system so that there is:

A. no feedwater pump, 1 condensate booster pump, and 1 condensate pump running.

B. no feedwater pump, 2 condensate booster pumps, and 2 condensate pumps running.

C. 1 feedwater pump, 1 condensate booster pump, and 1 condensate pump running.

D. 1 feedwater pump, 2 condensate booster pumps, and 2 condensate pumps running.

Proposed Answer: C

Technical References: Procedure 5.4 Fire-SD, Revision 8, step 4.4.1.3.

Proposed References to be provided to applicants during examination: None

Learning Objective

Question Source: New Question

Question History: None Last NRC Examination NA

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

10 CFR Part 55 Content: 55.41 (10)

Comments:
(KA) Knowledge of the operational applications of fire fighting as applies to a plant fire on site.

Answer (a) is incorrect because the referenced procedure directs one feedpump be left in service.

Answer (b) is incorrect because of same as answer a.

Answer (d) is incorrect because the referenced procedure directs 1 condensate booster and one condensate pump be left in service.

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number	1	
	Group Number	2	
	K / A Number	295022	AA1.01
	Importance Rating	3.1	

Proposed Question 21:

The plant is in Mode 1 and the A Control Rod Drive Hydraulic (CRDH) pump just tripped with an associated "CRD PUMP A BREAKER TRIP" alarm.

Prior to starting the B CRDH pump, the CRDH flow control valve (CRD-FC-301) is placed in the manual and minimum position in order to:

A. prevent rod drifts.

B. minimize pump starting current.

C. prevent lifting any directional control valves.

D. minimize the pressure transient on the hydraulic accumulators.

Proposed Answer: A is correct because if the pump is started with the FCV open, the pressure transient can result in rod drifts.

Technical References:

Proposed References to be provided to applicants during examination: None

Learning Objective

Question Source: New Question

Question History: NA

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

10 CFR Part 55 Content: 55.41.7

Comments:
(KA) Ability to determine or interpret reactor water level as applies to high reactor pressure.

Answer (b)

Answer (c)

Answer (d)

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number	1	
	Group Number	2	
	K / A Number	295008	AA1.04
	Importance Rating	3.1	

Proposed Question 22:

A loss of feedwater event resulted in a reactor scram and HPCI was manually initiated by the reactor operator at -30 inches RPV level. If an automatic trip on high RPV water level were to occur, it would cause the HPCI turbine:

A. stop valves to close and the HPCI min flow valve to open.

B. governor valves to close and exhaust line drain valves to open.

C. stop valves and exhaust line drain valves to close.

D. stop valves to close and the turbine governor valves to open.

Proposed Answer: C

Technical References: COR0021102R20-S-OPS High Pressure Coolant Injection System, Revision 20, page 14.

Proposed References to be provided to applicants during examination:

Learning Objective LO 12-c

Question Source: New Question

Question History: None Last NRC Examination NA

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

10 CFR Part 55 Content: 55.41.7

Comments:
(KA) Abilty to monitor or operate HPCI as applies to high reactor water level.

Answer (a) is incorrect because the min flow valve will close.

Answer (b) is incorrect because the drain valves will close.

Answer (d) is incorrect because the turbine governor valves will close.

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number	1	
	Group Number	2	
	K / A Number	295009	AK2.02
	Importance Rating	3.9	

Proposed Question 23:

The plant is operating at RTP with RPV water level at +35" and being controlled by Channel A. The operator receives indication of both high and low reactor level conditions and notices that channel A level instruments show level increasing while the channel B level instruments show level decreasing. Actual vessel level is _____ because of a _______.

a) decreasing
break or leak on level instrument reference line A

b) decreasing
shift to MDEM mode on a level instrument failure

c) increasing
shift to MDEM mode on a level instrument failure

d) increasing
break or leak on level instrument reference line A

Proposed Answer: (A) Decreasing & Break or leak on level instrument reference line A. This answer is correct because procedure 2.4RXLVL, step 5.8 states, "A reference line break or leak on the controlling instrument could cause all the level instruments connected to that line to indicate a higher level than actual reactor water level...The level instruments connected to the actual affected reference leg will indicate a rising level while the instruments connected to the unaffected line will indicate the actual lowering level."

Technical References: Lesson COR002-32-02, "Reactor Vessel Level Control," Revision 14

Procedure 2.4RXLVL, "RPV Water Level Control Trouble," Revision 13

Proposed References to be provided to applicants during examination: None

Learning Objective: Lesson COR002-32-02, Enabling Objective 6

Question Source: New Question

Question History: Never Used

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.7

Comments:
(KA) Knowledge of the interrelations between reactor water level control and low reactor water level.

Answer (B), Stable & Reactor Feed Pumps shift to MDEM mode on a level instrument failure, is incorrect because (1) actual vessel level is decreasing, and (2) per COR002-32-02, V(E), reactor feed pumps shift to MDEM and maintain current speed when there is a complete loss of the selected channel of reactor vessel level indication.

Answer (C), Stable & Reactor Feed Pumps shift to MDVP mode on a level instrument failure, is incorrect because actual vessel level is decreasing. Reactor feed pumps do not automatically switch to MDVP mode.

Answer (D), Increasing & Loss of Bias to the Level/Flow Error Comparator, is incorrect because (1) actual vessel level is decreasing.

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number	1	
	Group Number	2	
	K / A Number	295013 AK2.01	
	Importance Rating	3.6	

Proposed Question 24:

The plant is in Mode 5 and the suppression pool cooling mode of RHR was placed in service about two hours ago to reduce suppression pool temperature. The licensed operator has noted suppression pool temperature has not changed over the same time span.

Which of the following conditions would require followup to resolve the problem with suppression pool cooling?

A. A reactor scram has not been reset.

B. The RHR minimum flow valve is open.

C. Refueling activities are in progress.

D. An SRV is partially open.

Proposed Answer: B - with the min flow valve open, this means there is low RHR system flow and therefore there is little water going through the heat exchanger.

Technical References: COR002-23-02, figure 1

Proposed References to be provided to applicants during examination: None

Learning Objective

Question Source: New Question

Question History: None Last NRC Examination NA

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

10 CFR Part 55 Content: 55.41.7

Comments:
(KA) Knowledge of the interrelations between high suppression pool temperature and suppression pool cooling.

Answer (a) is incorrect because a reactor scram signal will not influence suppression pool cooling.

Answer (c) is incorrect because refueling will not influence suppression pool cooling.

Answer (d) is incorrect because being in Mode 5 means the RPV temp is 200 degrees or less. This means there will be no heating affect from the open SRV.

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number	1	
	Group Number	2	
	K / A Number	295014	2.2.01
	Importance Rating	3.7	

Proposed Question 25:

The plant is in Mode 4 and preparations for a reactor startup are in progress. An operator performing a rod control startup surveillance selects and attempts to withdraw a control rod.

The attempt to withdraw the control rod should be blocked by the:

A. Rod Worth Minimizer (RWM).

B. Rod Position Indication System (RPIS).

C. Reactor Manual Control System (RMCS).

D. Rod Select Sequencer (RSS).

Proposed Answer: C - Rod blocks are enforced by the RMCS.

Technical References: COR0022002R15-S-OPS Reactor Manual Control System, page 36.

Proposed References to be provided to applicants during examination: None.

Learning Objective

Question Source: New Question

Question History: None Last NRC Examination NA

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

10 CFR Part 55 Content: 55.41 (6)

Comments:
(KA) Ability to perform pre-startup procedures for the facility including operating those controls associated with plant equipment that could affect reactivity (295014, Inadvertent Reactivity

Addition).

Answer (a) is incorrect because the RWM enforces rod blocks based on Rod pattern.

Answer (b) is incorrect because RPIS only causes a rod block if it is inoperable.

Answer (d) is incorrect because there is no rod select sequencer on a BWR 4.

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number	1	
	Group Number	2	
	K / A Number	295015 AK1.03	
	Importance Rating	3.8	

Proposed Question 26:

The reactor was manually scrammed following a main condenser tube rupture and the reactor operator reported there are 18 control rods that do not indicate fully inserted. Plant conditions are as follows:

- ✓ RPV water level is +20 inches and lowering
- ✓ RPV pressure is 1000 psig
- ✓ Drywell pressure is 0.2 psig and stable
- ✓ All APRMs indicate downscale
- ✓ Indicated reactor period is infinity
- ✓ Both Reactor Recirc pumps are in operation
- ✓ SRMs and IRMs have been fully inserted

At this point the crew should:

A. enter EOP 1A, RPV Control.

B. commence a 100 degree per hour cooldown rate.

C. trip both reactor recirc pumps.

D. stabilize RPV pressure and temperature.

Proposed Answer: D - With 8 control rods not inserted and an infinite period, RPV temperature should be stabilized until a shutdown margin analysis is completed.

Technical References:

Proposed References to be provided to applicants during examination:

Learning Objective

Question Source: New question

Question History: None Last NRC Examination NA

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

10 CFR Part 55 Content: 41.8 to 41.10

Comments:
(KA) Knowledge of the operational implications of an incomplete SCRAM as it applies to reactivity effects.

Answer (a) is incorrect because the EOP entry conditions are not met.

Answer (b) is incorrect because a cooldown could initiate a restart accident.

Answer (c) is incorrect because it is desired to maintain forced flow through the reactor.

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number	1	
	Group Number	2	
	K / A Number	295032	EK2.07
	Importance Rating	3.6	

Proposed Question 27:

The plant is operating at RTP when the following conditions occur.

- ✓ Reactor Building Floor Drain Sump Pump are operating
- ✓ Reactor Building Equipment Drain Sump Pump are operating
- ✓ Reactor Building Floor Drain Sump High Level Alarm
- ✓ Reactor Building High Radiation alarm on detectors RA-4 (RWCU Precoat) and RA-5 (RWCU Sludge/Decant Pump Area)
- ✓ Area High Temperature Alarm on channels TS-117A and TS-117B (RWCU pump and heat exchanger rooms).

These conditions indicate:

A. the HPCI turbine is in operation.

B. water is being rejected to the Main Condenser through the RWCU System.

C. there is a steam leak from the HPCI System.

D. there is a steam leak from the RWCU System.

Proposed Answer: (D) <u>A steam leak from the RWCU System</u>. This answer is correct because (1) condensate from a steam leak would be collected in the Reactor Building Floor Drain Sump and Reactor Equipment Drain Sump, (2) area radiation monitors RA-4 and RA-5 are located in the RWCU Precoat area and RWCU Sludge/Decant Pump area, and (3) inputs to temperature switches 117A and 117B are located in the RWCU pump and heat exchanger rooms.

Technical References: COR001-11-02, “Leak Detection System,” Revision 11

COR002-03-02, “Containment,” Revision 18

Proposed References to be provided to applicants during examination: None

Learning Objective COR001-11-02, Enabling Objective 2(a)
COR002-03-02, Enabling Objective 14(h) & 19(d)

Question Source: New Question

Question History: Never Used

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.7

Comments:
(KA) Knowledge of the interrelations between high secondary containment area temperatures and leak detection concepts.

Answer (A), Operation of the HPCI turbine, is not correct because (1) while area temperature and radiation alarms may be received when HPCI is in operation due to the flow and leakage of contaminated steam, radiation monitors RA-4 and RA-5, and temperature switches TS-117A and 117B are not associated with HPCI, and (2) operation of the HPCI turbine does not result in water being transferred to the Reactor Building Floor Drain Sump.

Answer (B), Rejection of water to the Main Condenser through the RWCU System, is not correct because (1) water is not rejected through RWCU during normal full power operation, (2) area temperature alarms and radiation alarms are not normally received while rejecting water through the RWCU system, and (3) rejection of water through the RWCU system does not result in flow into the Floor Drain and Equipment Drain sumps.

Answer (C), A steam leak from the HPCI System, is not correct because the radiation monitor(s) and high temperature alarm channels in alarm are not associated with the HPCI system.

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number	2	
	Group Number	1	
	K / A Number	203000	K3.01
	Importance Rating	4.3	

Proposed Question 28:

A LPCI signal has initiated RHR and aligned both loops in the LPCI mode. While walking down the boards verifying all automatic actions have occurred, the operator notes the RHR Heat Exchanger Bypass Valve (RHR-MO-66A) is closed.

The operator should

A. leave the valve in its present configuration as it is supposed to be closed.

B. open the valve to maximize RPV injection rates.

C. open the valve to minimize RPV thermal shock.

D. open the valve to minimize RPV inventory loss through the min flow valve.

Proposed Answer: B - the heat exchange bypass valve should be open to maximize RPV injection flow rates.

Technical References: COR0022302R20-S-OPS Residual Heat Removal, Rev 20, page 15.

Proposed References to be provided to applicants during examination: None

Learning Objective

Question Source: New Question

Question History: None Last NRC Examination NA

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

10 CFR Part 55 Content: 55.41.7

Comments:

(KA) Knowledge of the effect that a loss or malfunction of RHR LPCI mode will have on reactor water level.

Answer (a) is incorrect because it is supposed to be open.

Answer (c) is incorrect because having the bypass valve open or closed will have little influence on RHR temperatures.

Answer (d) is incorrect because the mini flow valve will be closed in the LPCI mode.

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number	2	
	Group Number	1	
	K / A Number	205000 K4.05	
	Importance Rating	3.6	

Proposed Question 29:

With RHR A in service for shutdown cooling, which of the following are procedurally acceptable for controlling the reactor cooldown rate?

1. Throttle RHR flow using the LPCI Injection Valve (MO-27A)
2. Throttle RHR flow using the heat exchanger outlet valve (MO-12A)
3. Throttle RHR flow using the RHR pump suction valve (MO-15A)

A. 1 only

B. 1 and 2 only

C. 1, 2, and 3

D. 2 and 3 only.

Proposed Answer: B

Technical References: Procedure 2.2.69.2, revision 44, step 5.26.

Proposed References to be provided to applicants during examination:

Learning Objective

Question Source: New Question

Question History: None Last NRC Examination NA

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

10 CFR Part 55 Content: 55.41.7

Comments:
(KA) Knowledge of RHR shutdown cooling mode design features and/or interlocks that provide

for reactor cooldown rate.

Answer (a) is incorrect because MO-12A can also be used.

Answer (c) is incorrect because MO-15A cannot be used.

Answer (d) is incorrect because MO-15A cannot be used.

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number	2	
	Group Number	1	
	K / A Number	206000	K1.09
	Importance Rating	4.0	

Proposed Question 30:

The High Pressure Coolant Injection (HPCI) keep-fill system is normally provided by the _____. Should the normal supply fail, then the ________ provides a backup source of keep-fill water.

A. Condensate System; Reactor Building Auxiliary Condensate Pump

B. CST Recirc Pump; CRDH System.

C. Reactor Building Auxiliary Condensate Pump; CST Recirc Pump

D. CRDH System; Condensate System

Proposed Answer: A

Technical References: COR002-11-02, “High Pressure Coolant Injection,” Revision 20

Procedure 2.2.7, “Condensate Storage and Transfer System,” Revision 44

Procedure 2.2.33, “High Pressure Coolant Injection System,” Revision 54

Proposed References to be provided to applicants during examination: None.

Learning Objective

Question Source: New Question

Question History: Never Used

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

10 CFR Part 55 Content: 55.41.2 to 55.41.9

Comments:

(KA) Knowledge of the physical connections and/or the cause/effect relationships between HPCI and the ECCS keep-fill system.

Answer (b) is incorrect because neither of these systems interface with HPCI.

Answer (c) is incorrect because the CST Recirc pump does not provide makeup to HPCI.

Answer (d) is incorrect because the CRDH System does not provide makeup to HPCI.

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number	2	
	Group Number	1	
	K / A Number	209001	A1.08
	Importance Rating	3.3	

Proposed Question 31:

An operator wants to open Low Pressure Core Spray valves MO-11 (Outboard Injection Valve) and MO-12 (Inboard Injection Valve) as part of a surveillance.

The interlocks associated with these valves will permit this lineup if reactor pressure is between:

A. 436 and 500 psig and MO-11 is opened first.

B. 291 and 436 psig and MO-12 is opened first.

C. 436 and 500 psig and MO-12 is opened first.

D. 291 and 436 psig and MO-11 is opened first.

Proposed Answer: D

Technical References: Procedure 2.2.9, "Core Spray System," Revision 57

COR002-06-02, "Core Spray System," Revision 16

Proposed References to be provided to applicants during examination: None.

Learning Objective: COR002-06-02, Enabling Objectives 8 & 12

Question Source: New Question

Question History: None Last NRC Examination NA

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

10 CFR Part 55 Content: 55.41.5

Comments:
(KA) Ability to predict and/or monitor changes in parameters, including system lineup, associated with operating LPCS controls.

Answer (a) is incorrect because pressure must be between 291 and 436 psig.

Answer (b) is incorrect because MO-11 must be opened first.

Answer (c) is incorrect because MO-11 must be opened first.

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number	2	
	Group Number	1	
	K / A Number	211000	K1.03
	Importance Rating	2.5	

Proposed Question 32:

When adding boric acid and borax to the Standby Liquid Control Storage Tanks, the reason valve DW-170 (SA/DW Crosstie Supply to SLC Tank) must be opened to the minimum position necessary for tank agitation is:

A. over-agitation of the tank can cause borax to be ejected from the tank.

B. opening the valve can cause low service air supply header pressure.

C. increased air flow can cause the bubbler level indication system to read higher than actual.

D. increased air flow can reduce the boron concentration in solution.

Proposed Answer: (B) Opening the valve can cause low service air supply header pressure. This answer is correct because the caution at procedure 2.2.74, step 8.5, states: "Opening DW-170 too far can cause a low service air supply header pressure and auto start of standby air compressor."

Technical References: COR00-29-02, "Standby Liquid Control System," Revision 14

Procedure 2.2.74, "Standby Liquid Control System," Revision 35

Proposed References to be provided to applicants during examination: None

Learning Objective COR00-29-02, Enabling Objective 10(a)

Question Source: New Question

Question History: Never Used

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.2 to 55.41.9

Comments:
(KA) Knowledge of the physical connections and/or cause/effect relationships between standby liquid control and plant air systems.

Answer (A), Over-agitation of the tank can cause borax (a hazardous material) to be ejected from the tank, is incorrect because procedure 2.2.74, step 8.5, states: "Opening DW-170 too far can cause a low service air supply header pressure and auto start of standby air compressor."

Answer (C), Increased air flow can cause the bubbler level indication system to read higher than actual, is incorrect because (1) procedure 2.2.74, step 8.5, states: "Opening DW-170 too far can cause a low service air supply header pressure and auto start of standby air compressor," and (2) the bubbler level indication system is supplied from the instrument air system.

Answer (D), Increased air flow can reduce the boron concentration in solution by cooling the solution, is incorrect because (1) sparger air flow has a minimal effect on solution temperature, and (2) procedure 2.2.74, step 8.5, states: "Opening DW-170 too far can cause a low service air supply header pressure and auto start of standby air compressor."

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number	2	
	Group Number	1	
	K / A Number	211000	2.1.12
	Importance Rating	2.9	

Proposed Question 33:

With the reactor in Mode 1, an operator performs a surveillance of the Standby Liquid Control system with the following results:

✔ 4000 gallons of 15 percent sodium pentaborate solution by weight is in the storage tank
✔ The sodium pentaborate solution in the test tank is at 82˚F
✔ The A SLC Pump suction piping is at 80˚F
✔ The B SLC Pump suction piping is at 76˚F

What action statement, if any, is in effect by the Technical Specifications for this situation:

A. No action is necessary. Both SLC subsystems are operable.

B. Restore one SLC subsystem to operable immediately.

C. Restore one SLC subsystem to operable within 8 hours.

D. Restore one SLC subsystem to operable within 7 days.

Proposed Answer: (D) Restore one SLC subsystem to operable within 7 days. This answer is correct because Technical Specification 3.1.7, LCO Condition A, is to restore the subsystem to operable within 7 days when one subsystem is inoperable. The B subsystem is inoperable because the B SLC Pump suction piping is at 76˚F which is in the unacceptable range on Figure 3.1.7-2 for 15 percent sodium pentaborate solution (it would have to be at approximate 78.5˚F to be in the acceptable region).

Technical References: Technical Specification 3.1.7 and Bases.

Proposed References to be provided to applicants during examination: Technical Specification 3.1.7 including figures 3.1.7-1 (Sodium Pentaborate Solution Volume versus Concentration Requirements) and 3.1.7-2 (Sodium Pentaborate Solution Temperature versus Concentration Requirements).

Learning Objective

Question Source: New Question

Question History: Never Used

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43.2 , 55.43.5

Comments:
(KA) Ability to apply technical specifications for 211000, Standby Liquid Control system.

Answer (A), No action is necessary. Both SLC subsystems are operable, is incorrect because the B subsystem is inoperable because the B SLC Pump suction piping is at 76˚F which is in the unacceptable range on Figure 3.1.7-2 for 15 percent sodium pentaborate solution.

Answer (B), Restore one SLC subsystem to operable immediately, is incorrect because there are no immediate actions associated with Technical Specification 3.1.7.

Answer (C), Restore one SLC subsystem to operable within 8 hours., is incorrect because this is the required action for two SLC subsystems being inoperable. Only the B subsystem is inoperable because the B SLC Pump suction piping is at 76˚F which is in the unacceptable range on Figure 3.1.7-2 for 15 percent sodium pentaborate solution.

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number	2	
	Group Number	1	
	K / A Number	212000 K2.01	
	Importance Rating	3.2	

Proposed Question 34:

The power supply for RPS MG set 1A is MCC:

A. L.

B. T.

C. LX.

D. TX.

Proposed Answer: A

Technical References: COR0010101R26-S-OPS AC Distribution

Proposed References to be provided to applicants during examination: None

Learning Objective LO 07e

Question Source: New

Question History: None

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

10 CFR Part 55 Content: 55.41.7

Comments:
(KA) Knowledge of the electrical power supplies to the RPS MG sets.

Answer (b) is incorrect because it is the power supply for RPS B.

Answer (c) is incorrect because it is the alternate power supply for RPS A.

Answer (d) is incorrect because it is the alternate power supply for RPS B.

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number	2	
	Group Number	1	
	K / A Number	215000	K5.03
	Importance Rating	3.0	

Proposed Question 35:

Inserting the Intermediate Range Detectors into the reactor core at a high reactor power level is not desired because this may result in a(n):

A. control rod withdraw block for all rods.

B. control rod insert block for all rods.

C. RPS actuation and a scram.

D. premature burnup of the detector coatings.

Proposed Answer: D

Technical References: COR0021202R11-S-Intermediate Range Monitor, Revision 11

Proposed References to be provided to applicants during examination: None

Learning Objective

Question Source: New Question

Question History: None Last NRC Examination NA

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

10 CFR Part 55 Content: 55.41.5

Comments:
(KA) Knowledge of the operational implications of changing detector positions as applied to the intermediate range monitoring system.

Answer (a) is incorrect because all IRM rod blocks are bypassed with the mode switch in RUN.

Answer (b) is incorrect because all IRM rod blocks are bypassed with the mode switch in RUN.

Answer (c) is incorrect because all IRM scrams are bypassed with the mode switch in RUN.

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number	2	
	Group Number	1	
	K / A Number	215004	K2.01
	Importance Rating	2.6	

Proposed Question 36:

The power supply breaker for SRM Detector A would be found on:

A. Distribution Panel DC-A.

B. Distribution Panel CPP-2.

C. RPS Power Distribution Panel A

D. Motor Control Center RA

Proposed Answer: A

Technical References: COR0023002R11-S-Source Range Monitor, Revision 11

Proposed References to be provided to applicants during examination: None

Learning Objective

Question Source: New Question

Question History: None Last NRC Examination NA

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

10 CFR Part 55 Content: 55.41.7

Comments:
(KA) Knowledge of the electrical power supplies to SRM channels or detectors.

Answer (b), , is incorrect because the power supply is DC-A.

Answer (c), , is incorrect because the power supply is DC-A.

Answer (d), , is incorrect because the power supply is DC-A.

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number	2	
	Group Number	1	
	K / A Number	215005	A3.06
	Importance Rating	3.3	

Proposed Question 37:

While observing the output signals from the APRM Flow Comparator Circuits an operator notes Flow Comparator Channel A indicates 73 percent and Channel B indicates 60 percent.

Based on this the operator should also expect to see a:

A. flow comparator trip on channel A.

B. flow comparator trip on channel B.

C. white APRM UPSC indicating light on Panel 9-5.

D. white APRM INOP indicating light on Panel 9-5.

Proposed Answer: A

Technical References: COR0020102R16-S-OPS Average Power Range Monitor, Revision 16.

Proposed References to be provided to applicants during examination: None

Learning Objective

Question Source: New Question

Question History: None Last NRC Examination NA

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

10 CFR Part 55 Content: 55.41.7

Comments:
(KA) Ability to monitor automatic operation of the maximum disagreement between flow comparator channels as used in the APRM / LPRM system.

Answer (b) is incorrect because the higher of the flow comparator channels will trip if the difference is >10%. This means channel A will trip - not channel B.

Answer (c) is incorrect because there is no UPSC indication in this scenario.

Answer (d) is incorrect because there is no INOP indication in this scenario.

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number	2	
	Group Number	1	
	K / A Number	217000	A1.07
	Importance Rating	3.3	

Proposed Question 38:

Following a shutdown, RCIC was placed in operation for RPV level control taking a suction from the ECST and discharging to the RPV. As the need for RCIC diminished with time, the reactor operator has been reducing RCIC flow by adjusting the RCIC Flow Controller setpoint. The operator just reduced the flow controller from 45 gpm to 30 gpm.

Following this adjustment, the operator would expect to see:

A. ECST level to decrease at a slower rate than before.

B. Supression Pool level to increase at a faster rate than before.

C. a decrease in total pump flow.

D. an increase in RCIC pump discharge pressure.

Proposed Answer: B - pool level will increase at a faster rate due to the opening of minimum flow valve MO-27. This valve will open at 40 GPM effectively increasing total system flow and discharges to the suppression pool.

Technical References: COR0021802R17-S-OPS-RCIC

Proposed References to be provided to applicants during examination: None

Learning Objective

Question Source: New Question

Question History: None Last NRC Examination NA

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

10 CFR Part 55 Content: 55.41.5

Comments:
(KA) Ability to predict and/or monitor changes in parameters associated with suppression pool level as associated with operating RCIC controls.

Answer (a) is incorrect because ECST level will lower faster than before due to the opening of the min flow valve will cause an overall increase in RCIC flow.

Answer (c) is incorrect because RCIC pump flow will increase.

Answer (d) is incorrect because discharge pressure will decrease with increased flow.

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number	2	
	Group Number	1	
	K / A Number	218000 K6.04	
	Importance Rating	3.8	

Proposed Question 39:

The plant is operating in Mode 1 when a leak occurs causing pressure in the Reactor Building Reliable Instrument Air Header to decrease.

What affect will the loss of reactor building reliable instrument air have on the operation of the ADS valves?

A. The ADS valves will not close.

B. The ADS valves will not open.

C. There is no effect on the ADS valves.

D. The ADS valves will open a limited number of times.

Proposed Answer: D - The valves will cycle on the nitrogen contained in the accumulators. Depending on the conditions in the reactor building, the valves can cycle between 1 and 5 times.

Technical References: COR002-16-02, “Nuclear Pressure Relief,” Revision 14

COR001-17-01, “Plant Air,” Revision 17

Procedure 2.2.1, “Nuclear Pressure Relief System,” Revision 34

Proposed References to be provided to applicants during examination: None.

Learning Objective

Question Source: New Question

Question History: Never Used

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.7

Comments:
(KA) Knowledge of the effect that a loss or malfunction of the air supply to ADS valves will have on the ADS system.

Answer (A)

Answer (C)

Answer (D)

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number	2	
	Group Number	1	
	K / A Number	218000	A3.01
	Importance Rating	4.2	

Proposed Question 40:

ADS has actuated and all ADS valves are open. The ADS valves will close if:

A. RPV level increases to -100 inches.

B. RPV level increases to +10 inches.

C. the valve control switches are taken to CLOSE.

D. all ECCS pumps are lost.

Proposed Answer: D - if all ECCS pumps are lost the ADS valves will close.

Technical References: COR0021602R14-S-OPS Nuclear Pressure Relief

Proposed References to be provided to applicants during examination: None

Learning Objective

Question Source: New Question

Question History: None Last NRC Examination NA

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

10 CFR Part 55 Content: 55.41.5

Comments:
(KA) Ability to predict the impacts of a small break LOCA on ADS, and based on those predictions use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations. [Delete]

(KA) Ability to monitor automatic operations of the ADS including ADS valve operation.

Answer (a) is incorrect because the level signal seals in and will not reset without operator action.

Answer (b) is incorrect because the level signal seals in and will not reset without operator action.

Answer (c) is incorrect because the valve control switch will not close the valve with an ADS signal present.

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number	2	
	Group Number	1	
	K / A Number	223002 A1.04	
	Importance Rating	2.6	

Proposed Question 41:

The initiating signals for all PCIS isolations have cleared and the operator turns the two Group Isolation Reset switches on Panel 9-5 to the RESET position to the right.

What PCIS group isolations should the operator expect to see reset when these actions are taken?

A. Group 1 only.

B. Groups 4 and 5 only.

C. Groups 2, 3, 6, and 7 only.

D. Groups 2, 3, 4, and 5 only.

Proposed Answer: C - Groups 2, 3, 6, and 7 will reset given these conditions.

Technical References: COR0020302R18-S-OPS Containment

Proposed References to be provided to applicants during examination: None

Learning Objective

Question Source: New Question

Question History: None Last NRC Examination NA

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

10 CFR Part 55 Content: 55.41.5

Comments:
(KA) Ability to predict and/or monitor changes in parameters associated with operating the Nuclear Steam Supply Shutoff controls including individual system relay status.

Answer (a) is incorrect because Group 1 is reset if the switches are turned to the left.

Answer (b) is incorrect because these groups are not reset with these switches.

Answer (d) is incorrect because Groups 4 and 5 will not reset by turning these switches to the right.

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number	2	
	Group Number	1	
	K / A Number	239002	K1.03
	Importance Rating	3.5	

Proposed Question 42:

The logic associated with the opening and closing function of the Low Low Set safety relief valves is based on RPV:

A. pressure only.

B. pressure and level only.

C. pressure, level, and time.

D. level and time only.

Proposed Answer: A

Technical References: COR0021602R14-S-OPS Nuclear Pressure Relief

Proposed References to be provided to applicants during examination: None

Learning Objective

Question Source: New Question

Question History: None Last NRC Examination NA

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

10 CFR Part 55 Content: 55.41.2 to 55.41.9

Comments:
(KA) Knowledge of the physical connections and/or cause/effect relationship between safety relief valves and the nuclear boiler instrumentation system.

Answer (b) is incorrect because there is no level input to LLS logic.

Answer (c) is incorrect because there is no level or time input to the LLS logic.

Answer (d) is incorrect because there is no level or time input to the LLS logic.

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number	2	
	Group Number	1	
	K / A Number	239002	A3.08
	Importance Rating	3.6	

Proposed Question 43:

The red indicating light located over each SRV on Panel 9-3 actually indicates:

A. the SRV valve stem is in the open position.

B. the solenoid to actuate the SRV is energized.

C. opening air pressure has been admitted to the SRV actuator.

D. the SRV tailpipe pressure is high.

Proposed Answer: B

Technical References: COR0021602R14-S-OPS Nuclear Pressure Relief

Proposed References to be provided to applicants during examination: None

Learning Objective

Question Source: New Question

Question History: None Last NRC Examination NA

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

10 CFR Part 55 Content: 55.41.7

Comments:
(KA) Ability to monitor automatic operation of the safety relief valves and acoustical monitor noise. [Deleted]

(KA) Ability to monitor automatic operation of the safety relief valves including: lights and alarms.

Answer (a) is incorrect because there are no valve stem position indicators.

Answer (c) is incorrect because there are no air pressure sensors in the actuator.

Answer (d) is incorrect because the tailpipe pressure indicator is amber.

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number	2	
	Group Number	1	
	K / A Number	259002	A3.03
	Importance Rating	3.2	

Proposed Question 44:

Reactor power is 80 percent and the RPV Water Level Control System is in automatic three element control. The main steam flow sensor on the A Main Steam Line has developed a problem that has resulted in the sensed flow being stuck at 80 percent. The reactor operator just increased power to 85 percent.

With no other operator action:

A. the reactor will scram on low level.

B. the reactor will scram on high level.

C. RPV level will initially decrease and will then return to setpoint level.

D. RPV level will initially increase and will then return to setpoint level.

Proposed Answer: C - level will initially decrease due to the faulty steam flow sensor leading to steam flow being greater than feed flow. The level will recover because of the integrating level error signal that will increase feedflow until the level error becomes 0.

Technical References: COR0023202R14-S-OPS Reactor Vessel Level Control

Proposed References to be provided to applicants during examination: None

Learning Objective

Question Source: New Question

Question History: None Last NRC Examination NA

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

10 CFR Part 55 Content: 55.41.7

Comments:
(KA) Ability to monitor automatic operation of the reactor water level control system including changes in main steam flow.

Answer (a) is incorrect because the reactor will not scram.

Answer (b) is incorrect because the reactor will not scram.

Answer (d) is incorrect because level will initially go down as feed flow will be less than steam flow because of the faulty steam flow sensor.

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number	2	
	Group Number	1	
	K / A Number	261000	A4.03
	Importance Rating	3.0	

Proposed Question 45:

The Standby Gas Treatment (SGT) A fan control switch is in the STANDBY position. The SGT B fan control switch is in the RUN position and the fan is operating.

With no operator action, the A SGT fan will auto start if:

A. reactor water level reaches 3.0 inches and decreasing.

B. there is a high radiation level in the reactor building exhaust plenum of 49 mR/hr.

C. drywell pressure reaches 1.84 psig.

D. there is a low flow signal in the operating train.

Proposed Answer: D

Technical References: COR0022802R13-S-OPS Standby Gas Treatment

Proposed References to be provided to applicants during examination: None

Learning Objective

Question Source: New Question

Question History: None Last NRC Examination NA

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

10 CFR Part 55 Content: 55.41.7

Comments:
(KA) Ability to manually operate and/or monitor standby gas treatment system fans from the control room.

Answer (a) is incorrect because the LOCA signals do not auto start the fans with the switch in Standby.

Answer (b) is incorrect because the LOCA signals do not auto start the fans with the switch in Standby.

Answer (c) is incorrect because the LOCA signals do not auto start the fans with the switch in Standby.

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number	2	
	Group Number	1	
	K / A Number	262001	K1.03
	Importance Rating	3.4	

Proposed Question 46:

The plant is operating at RTP when the site experiences a lightning strike in the 345 kV switchyard. This results in a loss of the ring bus and a trip of the main generator due to a loss of load.

The station startup transformer will remain energized provided that:

A. ACB 110 remains closed

B. OCB 1606 remains closed

C. OCB 1602 opens

D. Breakers 1AS and 1BS open

Proposed Answer: (B) OCB1606 remains closed. This answer is correct because COR001-01-01 / 1398, IV.F.1, states, "If the ring bus is lost, but OCB-1606 remains closed the Startup Transformer will be powered from the 161kV Auburn line and remain available for station loads."

Technical References: Procedure 22.2.15, "Startup Transformer," Revision 35

COR001-01-01 / 1398, "AC Electrical Distribution," Revision 26

Proposed References to be provided to applicants during examination: None.

Learning Objective: COR001-01-01 / 1398, Enabling Objectives 6(b), 7(d), 8(h) & 13(c)

Question Source: New Question

Question History: Never Used

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41.3 to 55.41.9

Comments:
(KA) Knowledge of the physical connections and/or cause/effect relationships between AC electrical distribution systems and offsite power sources.

Answer (A), ACB110 remains closed, is incorrect because ACB 110 does not provide power to the Auto Transformer or to the Startup Transformer. COR001-01-01 / 1398, states, "The Auto Transformer (Transformer #2) is powered from the 345 kV switchyard and normally supplies power the Startup Transformer (the Auburn 161 KV line provides an alternate supply to the Startup Transformer). Power for the 12.5 kV system is supplied from a 13.8 kV tertiary winding on the Auto Transformer through ACB 110 and a step down transformer."

Answer (C), OCB1602 opens, is incorrect because OCB 1602 supplies power from the 1.61 kV Auburn line to the Auto Transfomer only if it and is closed (OCB 1604 would also have to be closed to supply power to the Startup Transformer).

Answer (D), Breakers 1AS and 1BS open, is incorrect because breakers 1AS and 1BS supply power from the Startup Transformer to the A and B vital busses, and only if they are closed. These breakers have nothing to do with supplying power to the Startup Transformer. COR001-01-01 / 1398, states, "Breakers 1AS and 1BS will close in to supply the buses from the Startup Transformer."

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number	2	
	Group Number	1	
	K / A Number	262002	K4.01
	Importance Rating	3.1	

Proposed Question 47:

The plant is in Mode 1 when annunciator "C-1/A-1, 250 VDC BUS 1A BLOWN FUSE" alarms. The CRT alarm message indicates "(3705) Static Inverter 1A feeder". While investigating the above alarm, annunciators "C-2/C-1, 4160V BUS 1A BKR 1AN LOCKOUT" and "C-1/B-6, 4160V BUS 1F BKR 1FA TRIP" also alarm.

Under the current conditions the operator would expect the No Break Power Panel to be:

A. deenergized.

B. energized by DG-1.

C. energized from the inverter.

D. energized by the Emergency Transformer.

Proposed Answer: (D) Energized by the Emergency Transformer. This answer is correct because Static Inverter 1A loses power from 250 VDC Switchgear 1A due to the blow fuse. However, when 4160V BUS 1A is deenergized undervoltage on 4160V bus 1F signals breaker 1FS to close, powering bus 1F from the Emergency Station Service Transformer; thus, the alternate AC source to the NBPP from MCC-R is available from 480V switchgear 1F and MCC-K which are powered from 4160V BUS 1F. COR002-07-02, II.A.6(g), states, "The NBPP inverter 1A provides 120V AC power to critical loads. NBPP is supplied from 250 VDC division I. If normal 250 VDC power is lost, the critical loads are supplied from MCC-R."

Technical References: Procedure 2.2.20, "Standby AC Power System (Diesle Generator)," Revision 56

COR002-07-02, "DC Electrical Distribution ," Revision 22

Proposed References to be provided to applicants during examination: None.

Learning Objective COR002-07-02, Enabling Objective 6(c), 8(q) & 9(a)

Question Source: Licensee Examination Bank (#263)

Question History: Last NRC Examination UNKNOWN

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.7

Comments:
(KA) Knowledge of the UPS design features and/or interlocks which provide for the transfer from preferred to alternate power.

Answer (A), Deenergized, is incorrect because the NBPP is energized from the alternate AC source. COR002-07-02, II.A.6(g), states, "The NBPP inverter 1A provides 120V AC power to critical loads. NBPP is supplied from 250 VDC division I. If normal 250 VDC power is lost, the critical loads are supplied from MCC-R."

Answer (B), Energized by DG-1, is incorrect because DG-1 will start on the 4160V bus 1F undervoltage, however, breaker EG1 will NOT close automatically unless breaker 1FS fails to close. Procedure 2.2.20, "Standby AC Power System (Diesle Generator)," Revision 56, Attachment 1, step 2.7.1 states. "All of following conditions must be met for Breaker EG1(EG2) to automatically close:...2.7.1.4...Breaker 1FS(1GS) is open."

Answer (C), Energized from the inverter, is incorrect because Static Inverter 1A has lost power from 250 VDC Switchgear 1A due to the blow fuse. COR002-07-02, II.A.6(g), states, "The NBPP inverter 1A provides 120V AC power to critical loads. NBPP is supplied from 250 VDC division I. If normal 250 VDC power is lost, the critical loads are supplied from MCC-R."

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number	2	
	Group Number	1	
	K / A Number	212000 A1.04	
	Importance Rating	2.8	

Proposed Question 48:

A reactor operator has been directed to place RPS A on the alternate power supply.

When this is performed, the operator will place the switch in the:

A. ALT A position and expect no change in RPS scram status.

B. ALT A position and expect a half scram.

C. ALT B position and expect no change in RPS scram status.

D. ALT B position and expect a half scram.

Proposed Answer: B - The switch must be placed in the ALT A position and because it is a break before make switch, there will be a momentary de-energization of RPS A. This will cause a half scram.

Technical References: COR0022102R16-S-OPS-Reactor Protection System

Proposed References to be provided to applicants during examination: None

Learning Objective

Question Source: New Question

Question History: None Last NRC Examination NA

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

10 CFR Part 55 Content: 55.41.5

Comments:
(KA) Ability to predict and monitor changes associated with motor-generator outputs associated with operating the uninterruptible power supply controls. [Delete KA]

(KA) Ability to predict and monitor changes associated with operating the RPS controls including: Bus Voltage

Answer (a) is incorrect because a half scram will occur.

Answer (c) is incorrect because the switch must be placed in ALT A.

Answer (d) is incorrect because the switch must be placed in ALT A.

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number	2	
	Group Number	1	
	K / A Number	263000	K3.03
	Importance Rating	3.4	

Proposed Question 49:

With the operating at 84% power and Ronan unavailable, the"ARI & ATWS RPT LOGIC POWER FAILURE" alarm on Panel 9-5-2/E7 is received.

Which of the following Alternate Rod Insertion (ARI) system functions will be available regardless of whether panels 125 VDC AA3 or BB3 are lost?

A. Manual initiation of control rod insertion only

B. Automatic initiation of control rod insertion only.

C. Automatic initiation of the recirculation pump trips only

D. Manual initiation of control rod insertion and automatic initiation of the recirculation pump trips

Proposed Answer: (C) <u>Only automatic initiation of the recirculation pump trips</u>. This answer is correct because the only feature that is available on loss of either AA3 or BB3 is the ability to automatically initiate a recirculation pump trip. COR002-33-02, V.A, states, “All ARI power is supplied from 125 VDC panels AA-3 and BB-3. A loss of 125 VDC panel AA-3 will prevent actuation of ARI rod insertion but ARI "B" logic will still cause a trip of both RRMG set field breakers. A loss of 125 VDC panel BB-3 will not prevent the actuation of ARI and will not prevent ARI initiation from tripping the RRMG set field breakers.”

Technical References: COR002-33-02, “Alternate Rod Insertion,” Revision 14

Proposed References to be provided to applicants during examination: None.

Learning Objective COR002-33-02, Enabling Objectives 6(a) & 8(a)

Question Source: Licensee Examination Bank (#1243)

Question History: Last NRC Examination UNKNOWN

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.7

Comments:
(KA) Knowledge of the effect that a loss or malfunction the DC electrical system will have on systems with DC components (valves, motors, solenoids, etc.).

Answer (A), Only manual initiation of control rod insertion, is incorrect because these functions are available on a loss of BB3 but are not available on a loss AA3.

Answer (B), Only automatic initiation of control rod insertion, is incorrect because these functions are available on a loss of BB3 but are not available on a loss AA3.

Answer (D), Both manual initiation of control rod insertion and automatic initiation of the recirculation pump trips, is incorrect because manual control rod insertion is not available on a loss of AA3. COR002-33-02, V.A, states, “All ARI power is supplied from 125 VDC panels AA-3 and BB-3. A loss of 125 VDC panel AA-3 will prevent actuation of ARI rod insertion but ARI "B" logic will still cause a trip of both RRMG set field breakers. A loss of 125 VDC panel BB-3 will not prevent the actuation of ARI and will not prevent ARI initiation from tripping the RRMG set field breakers.”

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number	2	
	Group Number	1	
	K / A Number	264000 K5.05	
	Importance Rating	3.4	

Proposed Question 50:

The EDG is operating in parallel with off-site power when an operator inadvertently places the "DROOP PARALLEL" switch in the "ISOCH" position.

In response, the KW being carried by the EDG will _______ and the voltage droop will _______.

A. increase; increase.

B. increase; decrease.

C. decrease; increase.

D. decrease; decrease.

Proposed Answer: B - With the switch in the ISOCH position the speed droop decreases causing an increase in KW. It will also cause the voltage droop to decrease.

Technical References: Procedure 2.2.20, "Standby AC Power System (Diesel Generator)," Revision 62, Step 1.4.3.6.

Proposed References to be provided to applicants during examination: None.

Learning Objective

Question Source: New Question

Question History: Never Used

Question Cognitive Level: Comprehension or Analysis Comprehension

10 CFR Part 55 Content: 55.41.7

Comments:
(KA) Knowledge of the operational implications of paralleling AC power sources as applied to

emergency diesel generators.

Answer (a) is incorrect because the generator will not trip.

Answer (c) is incorrect because the output breaker will not remain closed.

Answer (d) is incorrect because the output breaker will not remain closed.

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number	2	
	Group Number	1	
	K / A Number	264000 A2.04	
	Importance Rating	2.9	

Proposed Question 51:

A reactor operator is performing a post maintenance test that requires running EDG 1 in parallel with off-site power for 1 hour. The operator observes the EDG is loaded to 4000 KW and 900 KVAR (underexcited). The operator also observes the high KVAR load is causing the EDG to be outside the procedural limit on total amps.

Operating the EDG in this condition may cause the generator to __________. The operator should __________ voltage to reduce the KVAR load.

A. slip a pole; decrease

B. overheat; increase

C. trip on over current; decrease

D. trip on low voltage; increase

Proposed Answer: B - the operator should increase voltage to reduce the KVAR load as this could cause overheating of the generator.

Technical References: Procedure 2.2.20, “Standby AC Power System (Diesel Generator),” Revision 56

Procedure 2.2.20.1, “Diesel Generator Operations,” Revision 19

COR002-08-02, “Diesel Generators ,” Revision 13

Proposed References to be provided to applicants during examination: None.

Learning Objective: COR002-08-02, Enabling Objective

Question Source: New Question

Question History: Never Used

Question Cognitive Level: Comprehension or Analysis Analysis

10 CFR Part 55 Content: 55.41.5

Comments:
(KA) Ability to predict the consequences of over-excited or under-excited operation on emergency diesel generators, and based on those predictions, use procedures to correct, control, or mitigate the consequences of the abnormal condition or operation.

Answer (a) is incorrect because if the operator decreases voltage, KVAR loading will increase.

Answer (c) is incorrect because if the operator decreases voltage, KVAR loading will increase.

Answer (d) is incorrect because voltage is controlled by the grid.

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number	2	
	Group Number	1	
	K / A Number	300000 A2.01	
	Importance Rating	2.9	

Proposed Question 52:

Both instrument air dryers are out of service and the air dryer bypass valve has been opened in order to maintain IA header pressure.

This situation can result in _________ and the operators should________ to compensate.

A. premature air compressor failure; minimize air compressor run times

B. air filter damage; bypass the air filters

C. reduced instrument air compressor efficiency; increase air compressor cooling water flow

D. water accumulating in instrument air lines; periodically open the IA drain valves

Proposed Answer: D - water will accumulate in the IAS and end up in the instrumentation.

Technical References: Procedure 2.2.59, “Plant Air,” Revision 39

Procedure 5.3AIR, “Loss of Instrument Air,” Revision 5

COR001-17-01, “Plant Air,” Revision 17

Proposed References to be provided to applicants during examination: None.

Learning Objective COR001-17-01, Enabling Objectives 6(h) & 7(c)

Question Source: New Question

Question History: Never Used

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

10 CFR Part 55 Content: 55.41.5

Comments:
(KA) Ability to predict the consequences of air dryer and filter malfunctions on the instrument air system, and based on those predictions, use procedures to correct, control, or mitigate the consequences of the abnormal condition or operation.

Answer (a) is incorrect because there will be no impact on the air compressors.

Answer (b) is incorrect because higher moisture content will not damage the filters.

Answer (c) is incorrect because there will be no change in compressor efficiency.

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number	2	
	Group Number	1	
	K / A Number	400000	K6.05
	Importance Rating	3.0	

Proposed Question 53:

The plant is in Mode 1 with Reactor Equipment Cooling (REC) pumps A, B, and D running. REC pump C is tagged out for maintenance. Following a trip of the B REC pump, REC pressure initially dropped to 40 psig before stabilizing 2 minutes later at 53 psig.

Which of the following loads can be supplied with REC?

A. "A" Drywell Fan Coil Unit

B. "A" Station Air Compressor

C. "A" Control Rod Drive pump

D. Northwest Quad Fan Coil Unit

Proposed Answer: (D) Northwest Quad Fan Coil Unit. This is correct because with an isolation signal present REC-MO-702MV can be reopened, however, the REC-MO-712 and 713 will auto close on the low pressure and cannot be overriden. This will isolate REC to the non-critical loops/components. The fan coil unit is the only load listed supplied from the critical loop. Procedue 2.2.65, section 2.5 states, “REC-MO-702, DRYWELL SUPPLY ISOLATION, closes when its switch is in AUTO, low pressure of 61.2 psig is sensed by REC-PS-452A, and a 40 second time delay has timed out. The valve can be opened with the low pressure isolation signal present by placing its switch to OPEN.” Section 2.9 states,“REC-MO-712, HX A OUTLET VLV, closes when low pressure of 62.4 psig is sensed by REC-PS-452B1 and a 40 second time delay has timed out.” and section 2.10 states “REC-MO-713, HX B OUTLET VLV, closes when low pressure of 60.2 psig is sensed by REC-PS-452B2 and a 40 second time delay has timed out.”

Technical References: Procedure 2.2.65, “Reactor Equipment Cooling Water System,” Revision 46

COR002-19-02, “Reactor Equipment Cooling,” Revision 16

Proposed References to be provided to applicants during examination: None.

Learning Objective COR002-19-02, Enabling Objectives 3(b), 4(d) & 11(c)

Question Source: Licensee Exam Bank (#504)

Question History: Last NRC Examination UNKNOWN

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.7

Comments:
(KA) Knowledge of the effect that a loss or malfunction of pumps will have on the component cooling water system.

Answer (A), "A" Drywell Fan Coil Unit, is incorrect because The drywell fancoil will remain isolated because REC pressure remains below the isolation setpoint.

Answer (B), "A" Station Air Compressor, is incorrect because REC flow to the air compressor will remain isolated because REC pressure remains below the isolation setpoint.

Answer (C), "A" Control Rod Drive pump, is incorrect because REC flow to the CRD pump will remain isolated because REC pressure remains below the isolation setpoint.

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number	2	
	Group Number	2	
	K / A Number	214000	A4.02
	Importance Rating	3.8	

Proposed Question 54:

A symmetric power distribution results from a control rod pattern in which:

A. shallow control rods within a group are not more than one notch different from the other control rods.

B. intermediate control rods within a group are not more than one notch different from the other control rods.

C. deep control rods are not more than two notches different from the other control rods of the same group.

D. control rods at position 48 have been moved to position 44 due to high CRD temperatures.

Proposed Answer: (C) Deep control rods are not more than two notches different from the other control rods of the same group. This answer is correct because procedure 10.13, Attachment 10, §2.6, states, "The power distribution is not significantly affected by the position of deep control rods (Positions 00 to 22). Therefore, deep control rods that are not more than two notches different from the other control rods in that group are considered to be symmetric."

Technical References: Procedure 4.3, "Reactor Manual Control System and Rod Position Information System," Revision 23

Procedure 10.13, "Control Rod Sequence and Movement Control," Revision 47

COR002-05-02, "Control Rod Drive Mechanism," Revision 9

Proposed References to be provided to applicants during examination: None.

Learning Objective COR002-05-02, Enabling Objectives 5(h) & 12(e)

Question Source: New Question

Question History: Never Used

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41.7

Comments:
(KA) Ability to manually operate or monitor control rod position from the control room.

Answer (A), <u>Shallow control rods within a group are not more than one notch different from the other control rods</u>, is incorrect because procedure 10.13, Attachment 10, §2.6, states, "Intermediate or shallow control rods within a group shall be at the same notch position in order to be considered symmetric."

Answer (B), <u>Intermediate control rods within a group are not more than one notch different from the other control rods</u>, is incorrect because procedure 10.13, Attachment 10, §2.6, states, "Intermediate or shallow control rods within a group shall be at the same notch position in order to be considered symmetric."

Answer (D), <u>Control rods at position 48 have been moved to position 44 due to high CRD temperatures</u>, is incorrect because procedure 10.13, Attachment 10, §2.7, states "If high CRD temperatures are encountered, affected control rods at Position 48 can be moved to Position 46 provided the control rod sequence package is updated to document the actual control rod positions. Control rods moved to notch Position 46 for any reason are <u>not</u> considered asymmetric."

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number	2	
	Group Number	2	
	K / A Number	219000	A1.04
	Importance Rating	3.2	

Proposed Question 55:

Procedure 2.2.69.3, “RHR Suppression Pool Cooling and Containment Spray,” does not permit valves RHR-101 (RHR Crosstie Loop A Side) and RHR-102 (RHR Crosstie Loop B Side) to be open simultaneously.

This is because having both valves open could:

A. cause operating limits on the operating RHR heat exchanger to be exceeded.

B. increase Suppression Pool temperature.

C. lower Suppression Pool inventory.

D. violate the Class 1E mechanical separation requirements contained in 10CFR50.

Proposed Answer: (C) Lower Suppression Pool inventory. This answer is correct because COR002-23-02, II.G.2(c)(1), states, “Procedural guidance is imposed requiring that one of the two valves must be fully closed before the other valve may be opened. This is to prevent cross-connecting the A and B RHR discharge headers, which may lead to one or more of the following undesirable consequences:...reduction of Suppression Pool water inventory.”

Technical References: Procedure 2.2.69, “Residual Heat Removal System,” Revision 68

Procedure 2.2.69.3, “RHR Suppression Pool Cooling and Containment Spray,” Revision 34

COR002-23-02, “Residual Heat Removal System ,” Revision 20

Proposed References to be provided to applicants during examination: None.

Learning Objective COR002-23-02, Enabling Objectives 6(g) & 7(b)

Question Source: New Question

Question History: Never Used

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.5 & 45.5

Comments:
(KA) Ability to predict and/or monitor changes in parameters associated with RHR LPCI Suppression Pool Cooling Mode controls, including suppression pool level.

Answer (A), <u>Cause operating limits on the operating RHR heat exchanger to be exceeded</u>, is incorrect because COR002-23-02, II.G.2(c)(1), states, "Procedural guidance is imposed requiring that one of the two valves must be fully closed before the other valve may be opened. This is to prevent cross-connecting the A and B RHR discharge headers, which may lead to one or more of the following undesirable consequences:...reduction of Suppression Pool water inventory."

Answer (B), <u>Increase suppression pool temperature</u>, is incorrect because COR002-23-02, II.G.2(c)(1), states, "Procedural guidance is imposed requiring that one of the two valves must be fully closed before the other valve may be opened. This is to prevent cross-connecting the A and B RHR discharge headers, which may lead to one or more of the following undesirable consequences:...reduction of Suppression Pool water inventory."

Answer (D),<u>Raise Suppression Pool inventory</u>, is incorrect because COR002-23-02, II.G.2(c)(1), states, "Procedural guidance is imposed requiring that one of the two valves must be fully closed before the other valve may be opened. This is to prevent cross-connecting the A and B RHR discharge headers, which may lead to one or more of the following undesirable consequences:...reduction of Suppression Pool water inventory."

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number	2	
	Group Number	2	
	K / A Number	223001	A3.05
	Importance Rating	4.3	

Proposed Question 56:

The plant is starting up following a refueling outage. Primary containment has been inerted per procedure 2.2.60, “Primary Containment Cooling and Nitrogen Inerting System.” While verifying drywell pressure on recorder PC-R-FR/PR-513, “DW Press,” the operator notices flow through the Nitrogen Supply Pressure Indicating Controller (PC-PIC-513).

If PC-PIC-513 is in automatic and operating normally it will:

A. throttle to maintain nitrogen flow at 250 scfh.

B. open at a drywell pressure of 0.1 psig and close at 0.45 psig.

C. throttle to maintain drywell pressure about 0.25 psig.

D. close when drywell pressure reaches 0.6 psig.

Proposed Answer: (C) PIC-513 will throttle to maintain drywell pressure about 0.25 psig. This answer is correct because COR002-03-02, II.6.d, states, “Pressure Control Valve (PCV-513) on Panel H, compares Drywell pressure to an operator adjusted setpoint, normally 0.25 psig. If Drywell pressure drops below the controller setpoint, PCV-513 opens. This allows Nitrogen from the supply header to be routed to the Drywell and torus as necessary to makeup for any losses in the Primary Containment.”

Technical References: Procedure 2.2.60, “Primary Containment Cooling and Nitrogen Inerting System,” Revision 69C1

Procedure 2.4PC, “Primary Pressure Control,” Revision 7

COR002-03-02, “Containment ,” Revision 18

Proposed References to be provided to applicants during examination: None.

Learning Objective COR002-03-02, Enabling Objective 16(b)

Question Source: New Question

Question History: Never Used

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41.7

Comments:
(KA) Ability to monitor automatic operation of primary containment and auxiliaries, including drywell pressure.

Answer (A), PIC will throttle to maintain nitrogen flow at 250 scfh, is incorrect because per COR002-03-02 the high flow alarm [H-1/A-4 , DW NITROGEN MAKE UP HIGH FLOW] setpoint is 250 scfh.

Answer (B), PIC-513 will open at a drywell pressure of 0.1 psig and close at 0.45 psig, is incorrect because (1) COR002-03-02, II.6.d, states, “Pressure Control Valve (PCV-513) on Panel H, compares Drywell pressure to an operator adjusted setpoint, normally 0.25 psig. If Drywell pressure drops below the controller setpoint, PCV-513 opens.” The drywell low pressure [9-5-2/G-3 , DRYWELL LOW PRESSURE] alarm setpoint is 0.1 psig. Procedure 2.4PC, step 4.2 requires the operator to maintain drywell pressure ≤ 0.45 psig by venting containment through the SGT system.

Answer (D), PIC-513 will throttle closed when drywell pressure reaches 0.6 psig, is incorrect because per the high drywell pressur alarm [9-5-2/F-3 , DRYWELL HIGH PRESSURE] setpoint is 0.6 psig.

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number	2	
	Group Number	2	
	K / A Number	226001	2.4.22
	Importance Rating	3.0	

Proposed Question 57:

The plant has experienced a LOCA and currently:

- ✓ RPV water level is -32 inches on fuel zone instruments (corrected) and stable
- ✓ Drywell pressure is 5 psig and increasing
- ✓ RHR is injecting in the LPCI mode of operation

The basis for directing the operator to take the Containment Cooling 2/3 Core Valve Control Permissive Switch to manual override in order to initiate drywell sprays is to:

A. ensure NPSH for the RHR pumps

B. maintain safety relief valve operability

C. maximize fission product scrubbing

D. mitigate the effects of a hydrogen deflagration in containment.

Proposed Answer: (D) To mitigate the effects of a hydrogen deflagration in containment. This answer is correct because INT008-06-13, F.3, states, “If the Table 7 limits have been met, spraying takes precedence over adequate core cooling to mitigate the effects of a deflagration.”

Technical References: USAR § IV-8.5.3

Procedure 5.8.7, “Primary Containment Flooding / Spray Systems,” Revision 16

Procedure 2.2.69.3, “RHR Suppression Pool Cooling and Containment Spray,” Revision 34

COR002-23-02, “Residual Heat Removal System,” Revision 20

	INT008-06-13, “Flowchart 3A, Primary Containment Control,” Revision 13

Proposed References to be provided to applicants during examination: None.

Learning Objective COR002-23-02, Enabling Objectives 6(l) & 17(c)

Question Source: New Question

Question History: Never Used

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content:

Comments:
(KA) Knowledge of the bases for prioritizing safety functions during abnormal or emergency operations as applies to 226001, RHR LPCI Containment Spray Mode.

Answer (A), To maintain Net Positive Suction Head for the RHR pumps, is incorrect because the purpose is to mitigate the effects of a hydrogen explosion in containment. INT008-06-13, F.3, cautions operators that drywell sprays may reduce containment pressure so that Net Positive Suction Head for the RHR pumps is not maintained.

Answer (B), To maintain operability of the safety relief valves, is incorrect because the purpose is to mitigate the effects of a hydrogen explosion in containment. INT008-06-13, F.3, states that initiation and continued use of drywell sprays using suction sources outside containment (RHRSW crosstie) is only permitted if containment can be restored to the conditions of PCPL-A (SOP Graph 11), and the purpose of maintaining compliance with PCPL-A is to prevent a challenge to SRV operability.

Answer (C), To mitigate off-site releases through fission product scrubbing, is incorrect because the purpose is to mitigate the effects of a hydrogen explosion in containment. INT008-06-13, F.3, states that interlocks preventing operation of drywell sprays at low containment pressures may be defeated in order to allow the use of drywell sprays for fission product scrubbing in cases where the containment has already failed.

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number	2	
	Group Number	2	
	K / A Number	239001	K4.09
	Importance Rating	3.3	

Proposed Question 58:

The plant is operating at 70 percent power when an inboard MSIV (AO-80A) experiences a pneumatic supply failure and closes. The reactor did not scram and the remaining main steam lines are handling the increased steam flow.

After the failure is corrected, operators can:

A. not open AO-80A because steam flow through the open steam line drain may have prevented the drainage of condensate.

B. not open AO-80A because the valve cannot open against a 20-50 psid pressure difference between the idle and flowing steam lines.

C. open AO-80A because steam flow through the open steam line drain maintained an equal pressure across AO-80A.

D. open AO-80A because steam flow through the open steam line drain maintained temperatures in the A steam line within operating limits.

Proposed Answer: (A) <u>Cannot & Steam flow through the open steam line drain may have prevented the drainage of condensate</u>. This answer is correct because procedure 2.4MSIV, step 5.3 states, '...an idle steam line will not drain due to a 20 to 50 psid difference between the idle steam line and flowing steam lines. This may result in steam flow into the idle steam line through the drain header which prevents condensate from draining, forming water slugs..."

Technical References: Procedure 2.4MSIV, "Inadvertent MSIV Closure," Revision 0

COR002-14-02, "Main Steam," Revision 14

Proposed References to be provided to applicants during examination: None.

Learning Objective COR002-14-02, Enabling Objective 4(h)

Question Source: New Question

Question History: Never Used

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.7

Comments:
(KA) Knowledge of main and reheat steam design features and/or interlocks which provide for equalization of main steam isolation valve pressure.

Answer (B), Cannot & The valve cannot open against a 20-50 psid pressure difference between the idle and flowing steam lines, is incorrect because COR002-14-02, F.2(b)(2), states the valve is capable of opening against a pressure difference of 200 psid.

Answer (C), Can & Steam flow through the open steam line drain maintained an equal pressure across AO-80A, is incorrect because (1) procedure 2.4MSIV states that operators are not to open the valve until after an engineering evaluation is performed, and (2) procedure 2.4MSIV, step 5.3 states, '...an idle steam line will not drain due to a 20 to 50 psid difference between the idle steam line and flowing steam lines.

Answer (D), Can & Steam flow through the open steam line drain maintained temperatures in the A steam line within operating limits, is incorrect because (1) procedure 2.4MSIV states that operators are not to open the valve until after an engineering evaluation is performed, and (2) procedure 2.4MSIV, step 5.3 states, '...an idle steam line will not drain due to a 20 to 50 psid difference between the idle steam line and flowing steam lines. This may result in steam flow into the idle steam line through the drain header which prevents condensate from draining, forming water slugs..."

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number	2	
	Group Number	2	
	K / A Number	233000 K1.02	
	Importance Rating	2.9	

Proposed Question 59:

The plant is in Day 7 of a refueling outage with one-half the core off-loaded into the Fuel Pool. Fuel Pool Cooling Heat Exchanger 1B has been isolated due to a leak. Fuel Pool temperature is steady at 130˚F with about nine hours to boil. Fuel pool cooling is being augmented by RHR Subsystem A using the crosstie, per 2.4FPC, “Fuel Pooling Cooling Trouble,” Attachment 3.

Which of the following conditions could have an adverse effect on fuel pool cooling, considering the effects of any automatic or operator actions which result from the described condition:

A. High Effluent Strainer ΔP (≥ 5 psid)

B. Fuel Pool Cooling Pump A discharge ≥ 120 psig

C. Suppression pool level = - 1.00 inches

D. RHR Subsystem B flow ≤ 2107 gpm

Proposed Answer: (D) RHR Subsystem B flow ≤ 2107 gpm. This answer is correct because procedure 2.4FPC states, “Note - if RHR Subsystem B flow ≤ 2107 gpm RHR-MO-16A, Loop A Min Flow BYP VLV remains open.”

Technical References: Procedure 2.4FPC, “Fuel Pool Cooling Trouble,” Revision 5

COR001-06-01, “Fuel Pool Cooling,” Revision 16

Proposed References to be provided to applicants during examination: None.

Learning Objective: COR001-06-01, Enabling Objective 5(b)

Question Source: New Question

Question History: Never Used

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 41.2 to 41.9 & 45.7 to 45.8

Comments:
(KA) Knowledge of the physical connections and/or cause/effect relationship between Fuel-Pool cooling and cleanup, and the RHR system.

Answer (A), High Effluent Strainer ΔP (≥ 5psid),.is incorrect because per COR001-06-01 III.B.I, this condition indicates F/D outlet strainer is ready for cleaning and has no effect on fuel pool temperature.

Answer (B), Fuel Pool Cooling Pump A discharge ≥ 120 gpm , is incorrect because procedure 2.4FPC, step 1.7.2.6 states, "Throttle FPC-29...as necessary to maintain...pump discharge 120 to 150 psig.

Answer (C), Suppression pool level = - 1.00 inches, is incorrect because procedure 2.4FPC states that when RHR subsystem A crosstie is used, maintain suppression pool level ≥ -1.5" and ≤ +1.5"

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number	2	
	Group Number	2	
	K / A Number	256000 K6.10	
	Importance Rating	2.9	

Proposed Question 60:

The plant is starting up and currently:

- ✓ Reactor Vessel pressure at 300 psig
- ✓ Procedure 2.2.28, “Feedwater System Startup and Shutdown” is being performed

What is the effect on Reactor Feed Pump B if one safety relief valve opens to its full-open position under these conditions?

A. There is no effect because the pump is not in operation under these conditions.

B. The turbine inlet pressure will decrease because steam is being supplied from the equalizing header.

C. The feed pump will trip because pump suction pressure will decrease to less than 260 psig.

D. The feed pump turbine will trip because RPV pressure will decrease to less than 260 psig.

Proposed Answer: (A) No effect because the pump is not in operation under these conditions. This is correct because, (1) procedure 2.1.1, step 6.6 states “When feedwater flow is 1.5E6 to 1.9E6 lbs/hour and all main turbine trips have been completed slowly jog open RFP discharge valve per procedure 2.2.28.1,” and (2) procedure 2.1.1, step 5.22 states “If an RFP is not in service, begin placing first RFP in service per procedure 2.2.28.1 when RPV pressure is ~ 350 psig.”

Technical References: Procedure 2.1.1, “Startup Procedure,” Revision 109

Procedure 2.2.28, “Feedwater System Startup and Shutdown,” Revision 70

Procedure 2.2.28.1, “Feedwater System Operation,”’ Revision 38

COR001-15-01, “Nuclear Boiler,” Revision 18

COR002-02-02, “Condensate and Feed,” Revision 15

Proposed References to be provided to applicants during examination: None.

Learning Objective COR001-15-01, Enabling Objective 7(a)

COR002-02-02, Enabling Objectives 3(r) and 8(d)

Question Source: New Question

Question History: Never Used

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.7

Comments:
(KA) Knowledge of the effect that a loss or malfunction of the main steam system will have on the reactor condensate system.

Answer (B), No effect because the pump is being supplied from low-pressure extraction steam, is incorrect because (1) the RFP is initially supplied from high pressure steam from the main steam header, and (2) the RFP is not normally in operation under the conditions described.

Answer (C), Pump trips because pump suction pressure decreases to less than 260 psig, is incorrect because the RFP should be on its turning gear and not in operation under these conditions. COR002-02-02 lists 260 psig as the RFP low suction pressure trip setpoint.

Answer (D), Pump trips because reactor vessel pressure decreases to less than 260 psig, is incorrect because the RFP should be on its turning gear and not in operation under these conditions. COR002-02-02 lists 260 psig as the RFP low suction pressure trip setpoint.

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number	2	
	Group Number	2	
	K / A Number	271000	A3.05
	Importance Rating	2.9	

Proposed Question 61:

The plant is at RTP when the following conditions occur:

- ✔ Low Reactor Water Level at 3 inches
- ✔ Group 2 PCIS Isolation
- ✔ Main Steam Line Hi-Hi Radiation Alarm
- ✔ Offgas Timer Initiated
- ✔ Offgas High Radiation

The operator should also expect the:

A. mechanical vacuum pump to trip.

B. standby offgas dilution fan to auto start.

C. AOG to isolate.

D. offgas filter to have a high ΔP.

Proposed Answer: (C) AOG Auto Isolation. This answer is correct because (1) procedure 2.2.58, step 2.6 states, "A Group 2 isolation signal causes following AOG supply steam valves to close...RHR-920MV, SUPPLY VALVE & RHR-921MV, SUPPLY VALVE.", and (2) COR001-16-01 states that AOG Auto Isolation is received on Low Train supply steam pressure ($\leq$850 psig) or low dilution steam press ($\leq$425 psig). Closing RHR-920MV and RHR-921MV will create a low train supply steam pressure.

Technical References: Procedure 2.2.58, "AOG System," Revision 51

COR001-16-01, "OPS Off Gas," Revision 20

Proposed References to be provided to applicants during examination: None.

Learning Objective COR001-16-01, Enabling Objectives 10(h) and 13

Question Source: New Question

Question History: Never Used

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.7

Comments:
(KA) Ability to monitor automatic operations of the offgas system, including lights and alarms.

Answer (A), <u>Mechanical Vacuum Pump Trip</u>, is incorrect because although a Group 2 signal trips the mechanical vacuum pumps, this alarm occurs with a Breaker Trip on Undervoltage or Overcurrent.

Answer (B), <u>Standby Offgas Dilution Fan Auto-Start</u>, is incorrect because the standby dilution fan auto-starts when system flow is ≤ 1100 cfm.

Answer (D), <u>Offgas Filter High ΔP</u>, is incorrect because procedure 2.2.58, step 2.6 states, "A Group 2 isolation signal causes following AOG supply steam valves to close...RHR-920MV, SUPPLY VALVE & RHR-921MV, SUPPLY VALVE.", and (2) COR001-16-01 states that AOG Auto Isolation is received on Low Train supply steam pressure (≤850 psig) or low dilution steam press (≤425 psig). Closing RHR-920MV and RHR-921MV will create a low train supply steam pressure.

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number	2	
	Group Number	2	
	K / A Number	272000	2.1.2
	Importance Rating	3.0	

Proposed Question 62:

The reason to warn plant personnel to remain clear of the South Scram Discharge Area and Mitigation Monitoring Skid following a reactor SCRAM with elevated radiation levels on RMA-RA-9, CRD Hydraulic Equipment Area (North) is that:

A. transitory “High Radiation” conditions are expected near the North and South SCRAM discharge volumes following a SCRAM due to the collection of primary coolant

B. areas near the RWCU filters are expected to experience short-term elevated radiation levels due to scram-induced iodine spikes

C. local area radiation monitors may not indicate elevated radiation levels due to a scram-induced crud burst

D. leakage past the SCRAM discharge header isolation valves may allow I-131 to escape into the Reactor Building following a SCRAM

Proposed Answer: (C) Area radiation monitors in those areas may not indicate elevated radiation levels due to a scram-induced crud burst. This answer is correct because procedure 5.1RAD, step 5.8, states, “RMA-RA-8, CRD Hydraulic Equip Area (South) and RMA-RA-5, RWCU Sludge and Decant Pump Area, may not provide indication of elevated radiation levels in vicinity of South Scram Discharge Volume (R-902-S) and Mitigation Monitoring Skid (R-932-SW) following a scram induced crud burst due to detector locations. Elevated radiation levels in these locations can be expected concurrent with elevated levels on RMA-RA-9...”

Technical References: Procedure 5.1Rad, “Building Radiation Trouble,” Revision 8

Procedure 4.8, “Area Radiation Monitoring System,” Revision 12

COR001-18-10, “Radiation Monitoring,” Revision 16

Proposed References to be provided to applicants during examination: None.

Learning Objective COR001-18-10, Enabling Objective 10(g)

Question Source: New Question

Question History: Never Used

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.10

Comments:
(KA) Knowledge of operator responsibilities during all modes of operations as they pertain to 272000, Radiation Monitoring System.

Answer (A), Transitory High radiation conditions are expected near the North and South SCRAM discharge volumes following a SCRAM due to the collection of primary coolant, is incorrect because under normal conditions the area surrounding the scram discharge volume(s) do not reach 100 mR/h general area. Per 10CFR20, "high" radiation is ≥100 mR/h at 30 cm from the source.

Answer (B), Areas near the RWCU filters are expected to experience short-term elevated radiation levels due to scram-induced iodine spikes, is incorrect because (1) iodine spikes are generally insufficient to elevate area radiation levels, (2) the RWCU filters do not concentrate iodine following an iodine spike, and (3) area near the South Scram Discharge Volume (R-902-S) and Mitigation Monitoring Skid (R-932-SW) may see elevated radiation levels following a scram induced crud burst.

Answer (D), Leakage past the SCRAM discharge header vent isolation valves may allow I-131 to escape into the Reactor Building following a SCRAM, is incorrect because (1) gas leakage past the SCRAM discharge header vent isolation valves is not generally a radiological problem, and (2) any leakage past the vent isolation valves is directed to the gaseous radwaste system and not to reactor building atmosphere.

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number	2	
	Group Number	2	
	K / A Number	286000 K3.02	
	Importance Rating	3.2	

Proposed Question 63:

A low Supervisory Air pressure trouble alarm is received in the Main Control Room for the sprinkler system in the Electrical Cable Spreading Room. The operator has determined that the supervisory air system pressure for that area is 14 oz and slowly decreasing.

The concern associated with this alarm is the:

A. sprinkler valve cannot be opened without supervisory air.

B. deluge valve can only be operated from its local pull handle, delaying response to a fire.

C. deluge valve cannot be operated by any means, preventing a response to a fire.

D. sprinkler valve may open, creating a local safety hazard.

Proposed Answer: (D) The deluge sprinkler valve may inadvertently actuate, creating a local hazard. This answer is correct because COR001-05-01 II.H(4) states, "At approximately 16 oz of supervisory air, the deluge valve should actuate and charge the closed head system."

Technical References: Procedure 2.2.30, "Fire Protection System," Revision 50

COR001-05-01, "Fire Protection System," Revision 18

Proposed References to be provided to applicants during examination: None.

Learning Objective COR001-05-01, Enabling Objective 7(d)

Question Source: New Question

Question History: Never Used

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.7

Comments:
(KA) Knowledge of the effect that a loss or malfunction of the fire protection system will have on personnel protection.

Answer (A), There is no immediate effect on personnel protection, is incorrect because COR001-05-01 II.H(4) states, “At approximately 16 oz of supervisory air, the deluge valve should actuate and charge the closed head system.” Inadvertent activation of a sprinkler system can be a personnel hazard.

Answer (B), The deluge valve can only be operated from its local pull handle, delaying response to a fire, is incorrect because COR001-05-01 II.H(4) states, “At approximately 16 oz of supervisory air, the deluge valve should actuate and charge the closed head system.” The position of the local pull handle, local heat sensor(s), and the Control Room pushbutton are not relevant to system actuation.

Answer (C), The deluge valve cannot be operated by any means, delaying response to a fire, is incorrect because COR001-05-01 II.H(4) states, “At approximately 16 oz of supervisory air, the deluge valve should actuate and charge the closed head system.” The low supervisory air system pressure does not make the deluge valve inoperative.

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number	2	
	Group Number	2	
	K / A Number	290003	A2.01
	Importance Rating	3.1	

Proposed Question 64:

The reactor is at RTP when a high temperature alarm is received from the temperature element located in the cable spreading room exhaust duct (SD-1001). There is no indication of an actual fire in the Control Room or Cable Spreading Room.

However, the operator should expect the high temperature signal to trip the running supply fan (1-SF-C-1A/B) and __________. This alarm can be mitigated by ___________.

A. start Emergency Supply Fan 1-BF-C-1A; manually resetting SD-1001 with a magnet

B. trip Control Room Exhaust Fan 1-EF-C-1B; manipulating manual transfer switch HV-SW (BF-C-1A/B) in the Auxiliary Relay Room.

C. close fire smoke dampers 1544, 1545, 1546, 1547, 1581 and 1582; manually resetting the FS and HL reset buttons on the Firestat

D. opening Recirc Damper 1021D-2; manually resetting dampers 1545 and 1547 using the duct tool.

Proposed Answer: (C) Closing fire smoke dampers 1544, 1545, 1546, 1547, 1581 and 1582 & Manually resetting the FS and HL reset buttons on the Firestat. This answer is correct because COR001-08-01 step II.W.2 states, "Fire smoke dampers AD-1544, AD-1545, AD-1546, AD-1547, AD-1581 and AD-1582 will also close on high temperature," and, "Once they have been closed they must be manually reset by pressing the local Primary Heat Switch Manual Reset (FS) and the High Limit Temperature Sensor Manual Reset (HL) buttons on the Firestat."

Technical References: COR001-08-01, "Heating, Ventilation and Air," Revision 13

Procedure 2.4HVAC, "Building Ventilation Abnormal," Revision 6

Procedure 2.2.84, "HVAC Main Control Room and Cable Spreading Room," Revision 37

Proposed References to be provided to applicants during examination: None

Learning Objective COR001-08-01, Enabling Objective 10

Question Source: New Question

Question History: Never Used

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.5

Comments:
(KA) Ability to predict the impacts of system initiation or reconfiguration on control room HVAC, and based on those predictions use procedures to correct, control, or mitigate the consequence of those abnormal conditions or operations.

Answer (A), Starting Emergency Supply Fan 1-BF-C-1A & Manually resetting SD-1001 with a magnet, is incorrect because (1) 1-BF-C-1A is not started on a high temperature signal in the Cable Spreading Room exhaust (all fans trip and none are started), and (2) although SD-1001 is reset using a magnet, SD-1001 is a smoke detector which is not associated with a spurious high temperature signal.

Answer (B), Tripping Control Room Exhaust Fan 1-EF-C-1B & Manipulating manual transfer switch HV-SW (BF-C-1A/!B) in the Auxiliary Relay Room, is incorrect because manual transfer switch BF-C-1A/1B transfers fan power and is not associated with a spurious high temperature signal.

Answer (D), Opening Recirc Damper 1021D-2 & Manually resetting dampers 1545 and 1547 using the duct tool, is incorrect because damper 1021D-2 is not opened on a high temperature signal.

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number	2	
	Group Number	2	
	K / A Number	290002 K4.05	
	Importance Rating	3.3	

Proposed Question 65:

The plant is operating at RTP when a design basis earthquake results in a reactor scram and a loss of both Reactor Recirc pumps. RPV water level and pressure are stable at 40 inches and 980 psig respectively. The reactor operator then notices indicated jet pump flows are much less than expected from the Power-to-Flow Map.

These conditions are indicative of a:

A. Recirculation Line Break due to seismic shear

B. Jet Pump plugging due to internal debris

C. Loss of Natural Circulation due to core shroud cracking or displacement

D. Jet Pump Riser separation due to seismic shear

Proposed Answer: (C) <u>Loss of Natural Circulation due to core shroud cracking or displacement</u>. This answer is correct because indicated jet pump flows significantly below the Natural Circulation line on the Power-Flow Map are indicative of bypass flow through the core shroud into the core. In addition, one concern during seismic events is cracking or displacement of sections of the core shroud.

Technical References: COR001-15-01, “Nuclear Boiler,” Revision 18

COR002-22-02, “Reactor Recirculation,” Revision 20

Proposed References to be provided to applicants during examination: None.

Learning Objective COR002-22-02, Enabling Objective 5(h)

Question Source: New Question

Question History: Never Used

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41.7

Comments:
(KA) Knowledge of how the reactor vessel internals design and/or interlocks provide for natural circulation.

Answer (A), Recirculation Line Break due to seismic shear, is incorrect because (1) reactor water level and pressure remain steady at normal values for the postulated condition, and (2) indicated jet pump flows significantly below the Natural Circulation line on the Power-Flow Map are indicative of bypass flow through the core shroud into the core.

Answer (B), Jet Pump plugging due to internal debris, is incorrect because (1) if some jet pumps become plugged flow should increase on the unaffected pumps so that overall core flow remains nearly constant, and (2) indicated jet pump flows significantly below the Natural Circulation line on the Power-Flow Map are indicative of bypass flow through the core shroud into the core.

Answer (D), Jet Pump Riser separation due to seismic shear, is incorrect because (1) separation of the jet pump riser would have no effect with the Recirculation Pumps tripped, and (2) indicated jet pump flows significantly below the Natural Circulation line on the Power-Flow Map are indicative of bypass flow through the core shroud into the core.

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number	3	
	Group Number	1	
	K / A Number	2.1.10	
	Importance Rating	2.7	

Proposed Question 66:

With both reactor recirc loops operating normally and reactor power at 30 percent, it would be a violation of a Technical Specification Safety Limit if:

A. RPV pressure increases to 1325 psig following an MSIV closure.

B. RPV level decreases to 10 inches above the top of active fuel following a station blackout.

C. RPV pressure decreases to 750 psig following the opening of the bypass valves.

D. MCPR decreases to 1.21 following the loss of a recirc pump..

Proposed Answer: C - With power at 30 percent, RPV pressure must be at least 785 psig.

Technical References: Technical Specification 2.0, “Safety Limits (SLs)”

Proposed References to be provided to applicants during examination: None.

Learning Objective ??

Question Source: New Question

Question History: Never Used

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43.5

Comments:
(KA) Knowledge of conditions and limitations in the facility license.

Answer (a) is incorrect because the RPV pressure safety limit is 1337 psig.

Answer (b) is incorrect because the RPV level safety limit is the top of active fuel.
Answer (d) is incorrect because the MCPR safety limit is 1.09 with both recirc pumps operating.

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number	3	
	Group Number	1	
	K / A Number	2.1.28	
	Importance Rating	3.2	

Proposed Question 67:

The purpose of the administrative requirement to place the Reactor Water Level Control Switch on the 9-5 Panel in the B position is to ensure:

A. overfill protection in the event of an upscale pressure transmitter failure.

B. overfill protection in the event of a sensing line break.

C. correct auctioneering of the RFPT Controlling Governor Modules.

D. proper operation of the Startup Master Control Station LCCNETs.

Proposed Answer: (B) Ensure overfill protection in the event of a sensing line break. This answer is correct because COR002-32-02, II.A.4, states: “An administrative procedure requires the selector switch to be positioned to Level B. This ensures the reactor vessel overfill protection is in effect, even if a sensing line break on the controlling element were to occur. If level "A" were selected and a sensing line failure occurred, the 2 out of 3 trip logic would be inoperative and overfill protection not in effect.”

Technical References: COR002-32-02, “Reactor Vessel Level Control,” Revision 14

Proposed References to be provided to applicants during examination: None

Learning Objective COR002-32-02, Enabling Objective 3(j)

Question Source: New Question

Question History: Never Used

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41.7

Comments:
(KA) Knowledge of the purpose and function of major system components and controls.

Answer (A), <u>Ensure overfill protection in the event of an upscale pressure transmitter failure</u>, is incorrect because although the reactor water level control switch does effect which pressure transmitters are displayed on Recorder 97, the pressure input is not used in the reactor water level control system.

Answer (C), <u>Ensures correct auctioneering of the RFPT Controlling Governor Modules</u>, is incorrect because COR002-32-02 II.I.1 states, “The RFPT A(B) controlling governor module autioneering function is accomplished in the LEC. The healthiest of the three Governor Modules is selected for RFPT A(B) speed control.”

Answer (D), <u>Ensure proper operation of the Startup Master Control Station LCCNETs</u>, is incorrect because COR002-32-02 II.L.1, states, ‘The Startup Master OCS communicates with the Startup Valve Control Module in each of the Signal Processor Cabinets. Each Signal Processor Cabinet has two LCCNETs: A and B. The Startup Master Control Station communicates on B LCCNETs only.”

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number	3	
	Group Number	1	
	K / A Number	2.1.33	
	Importance Rating	2.7	

Proposed Question 68:

The plant is in Mode 1 and the HPCI turbine failed surveillance 6.HPCI.103, "HPCI IST and 92 Day Test Mode Surveillance."

With HPCI inoperable, another LCO 3.5.1 entry would be required if:

A. HPCI Valve MO-16 (HPCI Injection Isolation) is declared inoperable.

B. RCIC Valve MO-30 (RCIC to Suppression Pool) is declared inoperable.

C. REC supply water temperature increases to 95˚F.

D. ADS Pneumatic Supply Header pressure decreases to 85 psig.

Proposed Answer: (D) ADS Pneumatic Supply Header pressure at 85 psig. This answer is correct because ADS valves are inoperable when the supply header pressure is less than or equal to 88 psig. Both LCO 3.1.5E and 3.1.5H are entered when both HPCI and ADS are inoperable.

Technical References: Technical Specification 3.5.1, "ECCS - Operating,'" Amendment 203

Proposed References to be provided to applicants during examination: TS 3.5.1 including the surveillances.

Learning Objective

Question Source: New Question

Question History: Never Used

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43.2 , 55.43.3

Comments:
(KA) Ability to recognize indications for system operating parameters which are entry conditions into technical specifications.

Answer (A), LPCI pump B is declared inoperable, is incorrect because LCO 3.5.1A requires one LPCI pump in both subsystems to be inoperable.

Answer (B), RCIC Valve MO-30 is declared inoperable, is incorrect because RCIC Valve MO-30 is in the test return line and does not affect RCIC operabilty.

Answer (C), REC supply water temperature at 95˚F, is incorrect because REC is operable per SR 3.7.3.2 if supply water temperature is ≤ 100˚F. If REC were declared inoperable then low pressure ECCS would also be inoperable and LCO 3.5.1A would be entered.

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number	3	
	Group Number	2	
	K / A Number	2.2.22	
	Importance Rating	3.4	

Proposed Question 69:

According to technical specification 2.2, Safety Limit Violations, the operator has _____ to insert all insertable control rods following a Safety Limit Violation.

A. 30 minutes

B. 1 hour

C. 2 hours

D. 4 hours

Proposed Answer: (B) This answer is correct because the referenced techspec states all insertable control rods must be inserted within 2 hours of a safety limit violation.

Technical References: Technical Specification 2.2 Safety Limit Violations

Proposed References to be provided to applicants during examination: None

Learning Objective

Question Source: New Question

Question History: Never Used

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43.2

Comments:
(KA) Knowledge of limiting conditions for operations and safety limits.

Answer (B) is incorrect because this is for when pressure is greater than 785 psig.

Answer (C) is incorrect because this is for when pressure is greater than 785 psig.

Answer (D) is incorrect because power is limited to 25%.

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number	3	
	Group Number	2	
	K / A Number	2.2.30	
	Importance Rating	3.5	

Proposed Question 70:

When acting as Control Room Refueling Monitor during fuel loading, a reactor operator notices the SRM count rate is 300 counts per second.

In response to this condition, the reactor operator should:

A. bypass the affected SRM detector.

B. withdraw the affected SRM detector until the count rate reduces to 100 cps.

C. terminate fuel loading.

D. actuate the reactor building evacuation alarm.

Proposed Answer: (D) Terminate fuel loading because 300 cps is an abnormally high detector response. This is correct because procedure 10.25, step 8.1.18, states that the Control Room Refueling Monitor is to immediately terminate fuel loading in the event of an unexpected increase, and the note states that SRM count rates do not normally exceed 100 cps during fuel loading.

Technical References: Procedure 10.25, “Refueling - Core Unload, Reload, and Shuffle,” Revision 37

COR001-21-01, “Refueling,” Revision 12

Proposed References to be provided to applicants during examination: None.

Learning Objective: COR001-21-01, Enabling Objective 10

Question Source: New Question

Question History: Never Used

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content:

Comments:
(KA) Knowledge of RO duties in the Control Room during fuel handling (alarms from fuel handling area, communications with fuel storage facility, systems operated from the control room in support of fueling operations, and supporting instrumentation).

Answer (A), Bypass the affected SRM detector, is incorrect because procedure 10.25, step 8.1.18, states that the Control Room Refueling Monitor is to immediately terminate fuel loading in the event of an unexpected increase.

Answer (B), Drive out the affected SRM detector until the count rate reduces to 100 cps, is incorrect because procedure 10.25, step 8.1.18, states that the Control Room Refueling Monitor is to immediately terminate fuel loading in the event of an unexpected increase, and the note states that SRM count rates do not normally exceed 100 cps during fuel loading.

Answer (C), Activate the reactor building evacuation alarm, is incorrect because there is no requirement to evacuate the reactor building.

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number	3	
	Group Number	2	
	K / A Number	2.2.34	
	Importance Rating	2.8	

Proposed Question 71:

The plant is at RTP when the operator notices:

- ✓ Condensate Resin Trap ΔP has increased
- ✓ Main Steam Line radiation levels have increased by a small amount
- ✓ Core Plate Differential pressure has increased by a small amount
- ✓ Conductivity has increased to 0.15 μmho/cm
- ✓ PH has decreased to 6.5

The operator should also expect core thermal power to:

A. decrease because of the increase in voids following the breakdown of resins.

B. decrease because of resin plugging in the control rod channels reducing core flow.

C. remain constant because resins have nearly the same neutron capture cross section as does water.

D. increase because of the decrease in voids caused by the resins in the reactor.

Proposed Answer: (A) <u>Thermal Power decreases because of an increase in voids following the breakdown of resins</u>. This answer is correct because (1) procedure 2.2.5 step 1.1.2 states "An alarm annunciates resin trap high differential pressure which indicated possible leakage from the F/D," and (2) COR002-02-02 states that resin breakdown leads to an increase in voids due to lower water surface tension, and that indicators of resin intrusion are higher conductivity, lower PH, higher core plate DP, and higher Main Steam Line radiation levels.

Technical References: Procedure 2.2.5, "Condensate Filter Demineralizer System," Revision 36

COR002-02-02, "Condensate and Feed," Revision 15

Proposed References to be provided to applicants during examination: None.

Learning Objective COR002-02-02, Enabling Objectives 6(b), 9(a), 9(p) and 11(c)

Question Source: New Question

Question History: Never Used

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43.6

Comments:
(KA) Knowledge of the process for determining the internal and external effects on core reactivity.

Answer (B), Thermal Power decreases because of resin plugging in the Core Plate, is incorrect because (1) the amount of resins which escape the Resin Trap are not sufficient to plug the Core Plate, and (2) COR002-02-02 states that resin breakdown leads to an increase in voids due to lower water surface tension.

Answer (C), Thermal Power remains constant because resins have nearly the same neutron capture cross section as does water, is incorrect because COR002-02-02 states that resin breakdown leads to an increase in voids due to lower water surface tension.

Answer (D), is incorrect because COR002-02-02 states that resin will breakdown leading to an increase in voids due to lower water surface tension.

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number	3	
	Group Number	3	
	K / A Number	2.3.01	
	Importance Rating	2.6	

Proposed Question 72:

NPPD shall monitor the occupational radiation dose of any person that:

A. enters an RCA.

B. is expected to receive greater than 100 mrem duing the calender year.

C. enters a high radiation area.

D. is expected to receive a dose in excess of 500 mrem from external sources.

Proposed Answer: (C) Enter a high radiation area, locked high radiation area, or very high radiation area. This is correct because procedure 9.ALARA.1, step 6.2.9.3 states, “NPPD shall monitor occupational radiation dose if: An individual enters a high, locked high, or very high radiation area.”

Technical References: Procedure 9.ALARA.1, “Personnel Dosimetry and Occupational Radiation Exposure Program,” Revision 15

Proposed References to be provided to applicants during examination: None

Learning Objective *******

Question Source: New Question

Question History: Never Used

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41.12 , 55.43.4

Comments:
(KA) Knowledge of 10 CFR 20 and related facility radiation control requirements.

Answer (A), Enter the radiologically controlled area, is incorrect because procedure 9.ALARA.1, step 6.2.9.1 states that NPPD shall monitor the exposure of any person expected to receive a dose greater than 10% of the limits of 10 CFR 20.1201, which is 500 mrem.

Answer (B), Are expected to receive in a year a dose in excess of 100 mrem from external sources, is incorrect because procedure 9.ALARA.1, step 6.2.9.1 states that NPPD shall monitor the exposure of any person expected to receive a dose greater than 10% of the limits of 10 CFR 20.1201, which is 500 mrem.

Answer (D), Are expected to enter during a year a radiation field in excess of 500 mrem/hour from external sources, is incorrect because procedure 9.ALARA.1, step 6.2.9.3 states, "NPPD shall monitor occupational radiation dose if: An individual enters a high, locked high, or very high radiation area." A high radiation area is one where the radiation field is in excess of 100 mrem/hour and a locked high radiation area is one where the radiation field is in excess of 1000 mrerm/hour.

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number	3	
	Group Number	3	
	K / A Number	2.3.10	
	Importance Rating	2.9	

Proposed Question 73:

Operators are expected to hang 12 DANGER tags in a radiological area tomorrow. The highest expected dose rate is 750 mRem/hr and the job is planned to take two operators 30 minutes per person to hang and verify the tags. The dose for the job is expected to be 600 mRem per person.

Based on this information, this would be classified as a Category ________ job and an ALARA pre-job-briefing ________ be required.

A. 0 (zero); would not

B. 1; would not

C. 1; would

D. 2; would

Proposed Answer: (B) This would be a Category 1 job because the total person dose for the job is between 0.5 and 1.0 Rem. A Category 1 job does not require a pre-job-brief unless the local work area dose rates are greater than 1 Rem/hr. In this situation, the highest expected dose rate is 750 mRem/hr. Therefore, a pre-job-briefing is not required.

Technical References: Procedure 9.ALARA.5, “ALARA Planning and Controls,” Revision 12, Step 6.2

Proposed References to be provided to applicants during examination: None.

Learning Objective: N/A

Question Source: New Question

Question History: Never Used

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41.12

Comments:
(KA) Ability to perform procedures to reduce excessive levels of radiation and guard against personnel exposure.

Answer (a) is incorrect because the person-rem is projected to be more than 0.5 Rem.

Answer (c) is incorrect because a pre-job-briefing is not required.

Answer (d) is incorrect because this is a Category 1 job.

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number	3	
	Group Number	4	
	K / A Number	2.4.46	
	Importance Rating	3.5	

Proposed Question 74:

The plant experienced a LOCA approximately 4 hours ago and currently;

- ✔ The reactor is shutdown with all control rods fully inserted
- ✔ RPV water level is -50 inches and slowly lowering
- ✔ Drywell pressure is 4 psig and slowly rising
- ✔ Suppression Pool level is 8 inches above normal and steady
- ✔ HPCI and RCIC are injecting

Alarm 9-3-2/E-4, HPCI Pump Suction Low Pressure, would be consistent with plant conditions if suppression pool temperature is______ and HPCI system flow is_______.

A. increasing; fluctuating.

B. increasing; steady.

C. decreasing; fluctuating.

D. decreasing; steady.

Proposed Answer: (A) <u>Consistent with plant conditions if suppression pool temperature is increasing AND HPCI system flow is fluctuating</u>. This answer if correct because (1) per EOP 5.8 Attachment 2, Graph 16, NPSH Limits, loss of NPSH can occur if suppression pool temperature is greater than approximately 180˚F, and (2) per COR002-11-02, Revision 20, II.3(c), loss of NPSH can result in pump cavitation as indicated by erratic suction and discharge pressure and erratic system flow.

Technical References: Procedure 2.3-9-3-2, “,” Revision , D-4

COR002-11-02, “High Pressure Coolant Injection,” Revision 20

Proposed References to be provided to applicants during examination: None

Learning Objective	COR002-11-02, Enabling Objective 10(o)
Question Source:	New Question
Question History:	Never Used
Question Cognitive Level:	Comprehension or Analysis
10 CFR Part 55 Content:	55.43.5

Comments:
(KA) Ability to verify that alarms are consistent with plant conditions.

Answer (B), <u>Consistent with plant conditions if suppression pool temperature is increasing AND HPCI system flow is steady</u>, is incorrect because steady system flow is not indicative of a loss of NPSH or pump cavitation.

Answer (C), <u>Consistent with plant conditions if suppression pool temperature is decreasing AND HPCI system flow is fluctuating</u>, is incorrect because decreasing suppression pool temperature is not indicative of a loss of NPSH or pump cavitation

Answer (D), <u>Consistent with plant conditions if suppression pool temperature is decreasing AND HPCI system flow is steady</u>, is incorrect because neither decreasing suppression pool temperature nor steady system flow are indicative of a loss of NPSH or pump cavitation.

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number	3	
	Group Number	4	
	K / A Number	2.4.49	
	Importance Rating	3.3	

Proposed Question 75:

The plant is at 80% of RTP when operators observe low SJAE air flow and steam supply pressure. They also note condenser vacuum is lowering (trending toward no vacuum) and recorder MS-PR-73A (Low Pressure Turbine Absolute Recorder) shows an increasing trend. SJAE radiation levels and offgas dilution ΔP are both normal.

For these conditions the operator should be in procedure __________ and ____________:

A. 2.4OG; start the standby dilution fan.

B. 2.4OG; perform a normal shutdown of OWC.

C. 2.4VAC; close MSIV's if vacuum cannot be maintained $\geq$ 23" Hg.

D. 2.4VAC; SCRAM the reactor if vacuum cannot be maintained $\geq$ 23" Hg.

Proposed Answer: (D) 2.4VAC, and SCRAM the reactor when vacuum is $\leq$ 23" Hg, if the TSV & TCV closure trip is not bypassed. This answer is correct because

Technical References: Procedure 2.4VAC, Loss of Condenser Vacuum, Revision 10

Procedure 2.4OG, Off-Gas Abnormal, Revision 4

Proposed References to be provided to applicants during examination: None

Learning Objective n/a

Question Source: New Question

Question History: Never Used

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.10 , 55.43.2

Comments:
(KA) Ability to perform without reference to procedure those actions which require immediate operations of system components ands controls.

Answer (A), 2.4OG, and start the standby dilution fan, is incorrect because (1) conditions are not consistent with off-gas problems, and (2) a standby dilution fan starts automatically on low dilution flow.

Answer (B), 2.4OG, and perform a normal shutdown of OWC, is incorrect because (1) conditions are not consistent with off-gas problems, and (2) a normal shutdown of OWC occurs following an off-gas isolation signal, which occurs as an automatic action 5 minutes after low dilution flow.

Answer (C), 2.4VAC, and close MSIV's when vacuum is ≤ 23" Hg, is incorrect because closure of MSIV's is an immediate action when vacuum cannot be maintained above 12" Hg.

Comments:
(KA) Knowledge of limiting conditions for operations and safety limits as applied to 295001, Partial or Complete Loss of Forced Core Circulation.

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number		1
	Group Number		1
	K / A Number	295005	AA2.06
	Importance Rating		2.7

Proposed Question:

2. Reactor power is 29 percent during a reactor startup when the reactor operator trips the main turbine due to high vibration.

   The SRO should now anticipate implementing procedures that will:

   A. maintain reactor power $\leq$ 29 percent since power will increase after the main turbine trip.

   B. recover from the reactor scram caused by the turbine trip.

   C. recover vessel level using the feed and condensate system.

   D. scram the reactor.

Proposed Answer: A

Technical References:

Justification: Reactor power will increase following a trip of the main turbine due to a loss of feedwater heating. Power should be reduced until bypass valves are closed unless a restart of the main turbine is anticipated in the near future. The reactor should not scram following the turbine trip and RPV level should remain constant in this case as the feedpumps are steaming off the equalizer header. This makes answer a) the only correct answer.

Proposed References to be provided to applicants during examination: None

Learning Objective

Question Source: New Question

Question History: None Last NRC Examination NA

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

10 CFR Part 55 Content: 55.41.10 , 55.43.5

Comments:
(KA) Ability to determine or interpret feedwater temperature as applied to a main turbine generator trip.

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number		1
	Group Number		1
	K / A Number	295019	2.4.31
	Importance Rating		3.4

Proposed Question:

3. The Instrument Air (IA) Pressure - Low alarm has annunciated and instrument air pressure is now 80 psig. If pressure continues to lower, according to Emergency Procedure 5.2 Air - Loss of Instrument Air, the SRO should anticipate implementing procedures that will:

   1. Close the MSIVs
   2. Trip the reactor if any scram valve opens
   3. Establish RPV level control using RCIC/HPCI

   A. 1 only.

   B. 1 and 2 only

   C. 2 and 3 only

   D. 1, 2 and 3

Proposed Answer: D - Emergency Procedure 5.2 Air requires all three actions listed to be performed.

Technical References: Emergency Procedure 5.2 Air, Revision 5

Proposed References to be provided to applicants during examination: None

Learning Objective

Question Source: New

Question History: None Last NRC Examination NA

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

10 CFR Part 55 Content: 55.43.5

Comments:
(KA) Knowledge of annunciators, alarms, and indications, and use of proper response instructions as applied to 295019, Partial or Total Loss of Instrument Air

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number		1
	Group Number		1
	K / A Number	295028	EA2.03
	Importance Rating		3.9

Proposed Question:

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

- ✓ Drywell temperature is 300 degrees
- ✓ All control rods indicate fully inserted
- ✓ Reactor pressure is 0 psig
- ✓ Narrow Range indicates +7 inches and has been erratic for the last 30 minutes
- ✓ Shutdown Range indicates +7 inches
- ✓ Steam Nozzle Range indicates +7 inches
- ✓ All other level indications have been judged unreliable.

Using the attached Caution 1 and Graphs 1 and 15 from the Emergency Operating Procedures, determine the correct level indication and procedure action for these conditions.

A. Level should be considered unknown and EOP 2B, RPV Flooding, should be entered.

B. The Steam Nozzle Range indication should be used and level should be recovered using EOP 1A, RPV Control.

C. The Shutdown Flooding Range indication-should be used and level should be recovered using EOP 1A, RPV Control.

D. Narrow Range indication should be used and level should be recovered using EOP 1A, RPV Control.

Proposed Answer: A

Technical References: EOP 1A, 2B, Caution 1, Graph 1, Graph 15

Proposed references to be provided to applicants during examination: **Caution 1, Graph 1, Graph 15**

Learning Objective

Question Source: New Question

Question History: None Last NRC Examination NA

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

10 CFR Part 55 Content: 55.43.5

Comments:
(KA) Ability to determine and interpret suppression pool level as applied to high drywell pressure. [Deleted]

(KA) Ability to determine and/or interpret the following as they apply to high drywell temperature: Reactor Water Level.

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number		1
	Group Number		1
	K / A Number	295025	EA2.03
	Importance Rating		4.1

Proposed Question:

5. During a severe reactor transient involving an ATWS, the SRO notes that the high RPV pressure and elevated suppression pool temperatures are now approaching the Heat Capacity Temperature Limit (HCTL). Current conditions are:

- ✓ EOP 7A, RPV - Level (Failure to Scram) is the primary procedure in use
- ✓ RPV Pressure is 800 PSIG and rising
- ✓ RPV Level is -100 inches and lowering
- ✓ All available suppression pool cooling is in service
- ✓ Suppression Pool temperature is 200 degrees and rising
- ✓ Suppression Pool level is 11 feet 9 inches
- ✓ MSIVs are closed

The SRO should now transition from EOP 7A, RPV - Level (Failure to Scram) and implement:

A. EOP 2B, RPV Flooding

B. EOP 2A, Steam Cooling

C. EOP 7B, RPV Flooding (Failure to Scram)

D. EOP 6B, Emergency RPV Depressurization (Failure to Scram)

Proposed Answer: D

Technical References: EOPs

Proposed References to be provided to applicants during examination: None

Learning Objective

Question Source: New Question

Question History: None Last NRC Examination NA

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.43.5

Comments:
(KA) Ability to determine or interpret suppression pool temperature as applied to high reactor pressure.

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number		1
	Group Number		1
	K / A Number	295027	2.1.33
	Importance Rating		4.0

Proposed Question:

6. With the reactor in Mode 1, the technical specification requirement for high drywell temperature is that it be maintained less than or equal to:

   A. 110 degrees.

   B. 135 degrees.

   C. 150 degrees.

   D. 175 degrees.

Proposed Answer: C - The referenced techspec (3.6.1.5) limits drywell temperature to 150 degrees.

Technical References: Technical specification 3.6.1.5, Drywell air temperature.

Proposed References to be provided to applicants during examination: None

Learning Objective

Question Source: New Question

Question History: None Last NRC Examination NA

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

10 CFR Part 55 Content: 55.43.2

Comments:
(KA) Ability to recognize indications for system operating parameters which are entry level conditions for technical specifications as applied to 295027, High Containment Temperature.

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number		1
	Group Number		1
	K / A Number	295030	EA2.04
	Importance Rating		3.7

Proposed Question:

7. Technical Specification 3.6.1.8 requires 9 of 12 suppression chamber-to-drywell vacuum breakers be operable. The basis for this requirement is that the suppression chamber-to-drywell vacuum breakers prevent an excessive negative differential pressure across the wetwell drywell boundary.

   This prevents drawing water up into the SRV downcomers, and thereby:

   A. prevents damage to the SRV tailpipes upon the opening of an SRV.

   B. reduces the peak drywell pressure following a LOCA.

   C. prevents a low suppression pool level due to a loss of inventory to the drywell following the initiation of drywell sprays.

   D. prevents an implosion of the drywell following the initiation of drywell sprays.

Proposed Answer: B - The basis for this techspec is to reduce the drywell peak pressure following a LOCA.

Technical References:

Proposed References to be provided to applicants during examination: None

Learning Objective

Question Source: New Question

Question History: None Last NRC Examination NA

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

10 CFR Part 55 Content: 55.41.10 , 55.43.5

Comments:
(KA) Ability to determine or interpret drywell to suppression pool differential pressure as applied to low suppression pool water level.

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number		1
	Group Number		2
	K / A Number	295007	2.2.12
	Importance Rating		3.4

Proposed Question:

8. The reactor is operating at 70 percent power during a plant startup with all systems operable and in their normal configuration. Today at 1600, a maintenance supervisor reported the 4 RPV pressure sensors used in the RPS (Reactor Vessel Steam Dome Pressure - High) were not calibrated as required by Technical Specification Surveillance Requirement 3.3.1.1.10 (attached).

If no safety evaluation is performed, one Reactor Vessel Steam Dome Pressure - High channel must be placed in a tripped condition no later than:

A. 2200 today.

B. 0400 tomorrow.

C. 1600 tomorrow.

D. 2200 tomorrow.

Proposed Answer: D - Missing a surveillance initially invokes techspec statement 3.0.3 and gives the SRO 24 hours to get the surveillance done before having to declare the associated LCO. In this case, action statement 3.3.1.1.B.1 gives the SRO 6 hours to place one channel in the tripped condition. Therefore, the SRO has 30 hours (24 + 6) from 1600 today to trip the channel. This makes answer D correct.

Technical References: TS 3.3.1.1

Proposed References to be provided to applicants during examination: **Technical Spec. 3.3.1.1 including the surveillance requirements but excluding table 3.3.1.1-1**.

Learning Objective

Question Source: New Question

Question History:	None	Last NRC Examination	NA

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

10 CFR Part 55 Content: 55.43.2

Comments:
(KA) Knowledge of surveillance procedures as applied to 295007, High Reactor Pressure.

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number		1
	Group Number		2
	K / A Number	295017	2.4.6
	Importance Rating		4.0

Proposed Question:

9. According to EOP 5A, Radioactivity Release Control, if a primary system is discharging outside primary and secondary containment, an emergency depressurization is required when the radioactive release rate approaches a level requiring a(n) _____ be declared.

   A. Unusual Event

   B. Alert Emergency

   C. Site Area Emergency

   D. General Emergency

Proposed Answer: D

Technical References: EOP 5A

Proposed References to be provided to applicants during examination: None

Learning Objective

Question Source: New Question

Question History: None Last NRC Examination NA

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

10 CFR Part 55 Content: 55.43.5

Comments:
(KA) Knowledge of symptom-based emergency operating procedures mitigation strategies as applied to 295017, High Offsite Release Rate.

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number		1
	Group Number		2
	K / A Number	295034	EA2.01
	Importance Rating	4.2	

Proposed Question:

10. Following up on a RB VENT HI-HI RAD alarm, the reactor operator reports the Reactor Building Exhaust Plenum radiation level indicates 12 mR/hour.

    Based on this report, the SRO should immediately:

    1. Shutdown the Reactor if operating.
    2. Enter EOP 5A, Secondary Containment Control/Radioactive Release Control.
    3. Enter Emergency Procedure 5.1RAD, Building Radiation Trouble.

    A. 2 only.

    B. 3 only.

    C. 1 and 2 only.

    D. 1 and 3 only.

Proposed Answer: A - Twelve mR/hr exceeds the entry condition of 10mR/hr for EOP 5A. This is the top tier procedure for this situation.

Technical References: EOP 5A, Alarm Procedures 9-4-1/E-5 and E-4, 5.1RAD, 2.0.1.2 (Procedure Use Policy),

Proposed References to be provided to applicants during examination:

Learning Objective

Question Source: New Question

Question History: None Last NRC Examination NA

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

10 CFR Part 55 Content: 55.41.10 , 55.43.5

Comments:
(KA) Ability to determine or interpret ventilation radiation levels as applies to secondary containment high radiation levels.

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number		2
	Group Number		1
	K / A Number	203000	2.2.18
	Importance Rating		3.5

Proposed Question:

11. The plant is in Mode 4 and maintenance has requested they be allowed to troubleshoot a problem with the RHR LPCI Outboard Injection valve MO-27A. Engineering has determined the Level of Rigor for this troubleshooting effort falls into Category B classification.

    With a Category B Level of Rigor, the approval authority for this work is the __________ (with Shift Manager concurrence) and a Complex Troubleshooting Form __is / is not__ required for the work.

    A. Responsible Manager / is

    B. Control Room Supervisor / is not

    C. Plant Manager / is

    D. Responsible Maintenance Supervisor / is not

Proposed Answer: A - According to Procedure 7.0.1.7, the approval authority is the Responsible Manager and a Complex Troubleshooting form is required.

Technical References: Procedure 7.0.1.7, Troubleshooting Plant Equipment, Revision 11,step 2.7.2.

Proposed References to be provided to applicants during examination: None.

Learning Objective

Question Source: New Question

Question History: None Last NRC Examination NA

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

10 CFR Part 55 Content: 55.43.5

Comments:
(KA) Knowledge of the process for managing maintenance activities during shutdown operations as applied to 203000, RHR LPCI Injection Mode.

Answer B is incorrect because the Control Room Supervisor is not the approval authority and a Complex Troubleshooting Plan is required.

Answer C is incorrect because the Plant Manager is not the approval authority.

Answer D is incorrect because the Responsible Maintenance Supervisor is not the approval authority and a Complex Troubleshooting Plan is required.

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number		2
	Group Number		1
	K / A Number	206000	2.1.20
	Importance Rating		4.2

Proposed Question:

12. Yesterday, June 5, with the plant operating at 100% power the following sequence of events took place:

1100,	one ADS valve was declared inoperable due to a problem causing the valve's fuses to blow.
1400,	Maintenance began troubleshooting the problem.
1600,	HPCI failed quarterly Surveillance Requirement 3.5.1.7.
1615,	RCIC was verified operable.
2100,	Maintenance discovered and repaired a ground on the ADS valve and it was declared operable.

You have just taken the day shift with the plant at 100 percent power and have reviewed this scenario. Based on this sequence of events you conclude the plant;

A. should have been placed in Mode 2 last night.

B. must be in Mode 3 no later than June 20 at 0900.

C. must be in Mode 3 no later than June 20 at 0400.

D. must be in Mode 3 no later than June 19 at 2300

Proposed Answer: C - When HPCI was declared inoperable at 1600 with the inoperable ADS valve, it required an entry into TS 3.0.3. The TS 3.0.3 clock was exited at 2100 returning the station to the clock that started at 1600. For HPCI being inoperable, the station must be in mode 3 no later than 14 days and 12 hours from the time of inoperability. This make answer C the only correct answer.

Technical References: Technical Specification 3.5.1 and 3.0.3.

Proposed References to be provided to applicants during examination: **TS 3.5.1 including the surveillance requirements**.

Learning Objective

Question Source: New Question

Question History: None Last NRC Examination NA

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

10 CFR Part 55 Content: 55.41.10 , 55.43.5

Comments:
(KA) Ability to execute procedure steps as applied to 206000, High Pressure Coolant Injection. [Deleted]

(KA) Knowledge of surveillance procedures.

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number		2
	Group Number		1
	K / A Number	215003	2.1.22
	Importance Rating		3.3

Proposed Question:

13. All the Intermediate Range Monitors (IRMs) are fully inserted into the core and are indicating mid-scale on range 6. There are no rod blocks or scram signals present. Reactor coolant temperature and RPV pressure are currently 140 degrees and 0 psig respectively.

According to the Technical Specifications, the reactor is currently in Mode:

A. 1.

B. 2.

C. 3.

D. 4.

Proposed Answer: B - In this scenario, if the IRMs are on range 6 then more than one control rod must be withdrawn. This means the reactor must be in Mode 1 or 2. Because the IRMs are on Range 6, reactor power is less than 1% which means the Mode switch must be in Startup. Otherwise there would be a rod block due to APRMs being downscale. Therefore, the reactor is in Mode 2.

Technical References: Technical Specification Table 1.1-1.

Proposed References to be provided to applicants during examination: None

Learning Objective

Question Source: New Question

Question History: None Last NRC Examination NA

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

10 CFR Part 55 Content: 55.43.2

Comments:
(KA) Ability to determine Mode of Operation as applied to 215000, Intermediate Range Monitoring system.

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number		2
	Group Number		1
	K / A Number	262001	A2.08
	Importance Rating		3.6

Proposed Question:

14. A cracked insulator needing repair has been located on the high voltage side of the startup transformer. Electrical maintenance requires the disconnect located on the high voltage side of the transformer to be opened to repair the insulator. Prior to opening the disconnect, the SRO should ensure procedures are implemented that will:

    A. minimize the load on the transformer.

    B. deenergize the high voltage side of the transformer.

    C. deenergize the low voltage side of the transformer.

    D. deenergized both the high and low voltage sides of the transformer.

Proposed Answer: D - Both sides of the transformer will need to be deenergized.

Technical References:

Proposed References to be provided to applicants during examination:

Learning Objective

Question Source: New Question

Question History: None Last NRC Examination NA

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

10 CFR Part 55 Content: 55.41.5

Comments:
(KA) Ability to predict the impact of opening a disconnect under load and on the basis of the prediction use procedures to correct, control, or mitigate the consequences of the abnormal conditions or operations.

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number		2
	Group Number		1
	K / A Number	264000	A2.05
	Importance Rating		3.6

Proposed Question:

15. The plant is at full power and a reactor operator performing a quarterly surveillance on EDG 1 parallels the EDG to Bus 1F. Approximately an hour later the main turbine trips and there is a loss of off-site power.

    Because the EDG output breaker will __________, procedures should be implemented to ________:

    A. will trip on overcurrent; reset the generator lockout and restart EDG 1.

    B. open and then reclose in 10 seconds later; restore any lost equipment.

    C. remain closed and all Bus 1F 4160 load breakers trip open; restore the lost 4160 loads.

    D. remain closed and all other Bus 1F supply breakers trip open; place the EDG in the isochronous mode of operation.

Proposed Answer: B - the output breaker opens and recloses 10 seconds later.

Technical References: Procedure 2.2.20, "Standby AC Power System (Diesel Generator)," Revision 56

Procedure 2.2.20.1, "Diesel Generator Operations," Revision 19

COR002-08-02, "Diesel Generators ," Revision 13

Proposed References to be provided to applicants during examination: None

Learning Objective

Question Source: New Question

Question History: None Last NRC Examination NA

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis XX

10 CFR Part 55 Content: 55.41.5

Comments:
(KA) Ability to predict the impacts of synchronization of the emergency diesel generator with other electrical supplies, and on the basis of the prediction use procedures to correct, control, or mitigate the consequences of the abnormal conditions or operations.

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number		2
	Group Number		2
	K / A Number	234000	2.2.23
	Importance Rating		3.8

Proposed Question:

16. The plant is in Mode 5 with 2 Operable SRMs. A normal non-spiral off-load is in progress and the last fuel bundle in a core quadrant is about to be removed. The reactor operator informs you that the SRM adjacent to the fuel movement (one of the operable SRMs) is reading 0 cps.

In this situation, fuel movements:

A. can proceed.

B. can proceed with only one operable SRM.

C. must be suspended until the adjacent SRM reads $\geq$ 3 cps.

D. must be suspended until at least one of the inoperable SRMs is restored.

Proposed Answer: A - The SRM is not required to have $\geq$ 3 cps for this situation per TS SR 3.3.1.2.4 and would still be considered operable and meeting the TS requirements. Therefore, fuel movement can proceed.

Technical References: TS 3.3.1.2 & SR 3.3.1.2.4

Proposed References to be provided to applicants during examination: **TS 3.3.1.2 including the surveillance requirements.**

Learning Objective

Question Source: New Question

Question History: None Last NRC Examination NA

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

10 CFR Part 55 Content:

Comments:
(KA) Ability to perform specific system and integrated plant procedures during different modes of plant operations, as applied to 201003, Control Rod and Drive Mechanism. [Deleted]

(KA) Ability to track limiting conditions for operations.

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number		2
	Group Number		2
	K / A Number	233000	2.1.25
	Importance Rating		3.1

Proposed Question:

17. During a refueling outage the following conditions exist

- ✓ Fuel Pool Cooling Pump B is out of service and disassembled for maintenance.
- ✓ Fuel Pool Cooling Pump A is supplying both FPC Heat Exchangers.
- ✓ SFP gates are <u>not</u> installed.
- ✓ Recirc Pump B is running at minimum flow.
- ✓ RWCU is <u>not</u> aligned for alternate heat removal.
- ✓ REC cold leg temperature is 50˚F.
- ✓ The reactor has been shutdown for 40 days.

By using the fuel pool cooling capacity graphs found in procedure _________, it can be determined that with the current fuel pool cooling lineup the fuel pool temperature can be maintained at _____ .

A. NEDC 00-0105, “Fuel Pool Decay Heat Loads/Alternate Decay Heat Removal Time Limitations” / 155˚F

B. NEDC 00-0105, “Fuel Pool Decay Heat Loads/Alternate Decay Heat Removal Time Limitations” / 160˚F

C. Procedure 2.1.20.2, “Cycle Specific Fuel Transfer and Alternate Cooling Guideline” / 155˚F

D. Procedure 2.1.20.2, “Cycle Specific Fuel Transfer and Alternate Cooling Guideline” / 160˚F

Proposed Answer: (A) <u>NEDC 00-0105, “Fuel Pool Decay Heat Loads/Alternate Decay Heat Removal Time Limitations” / 155˚F</u>. This answer is correct because (1) procedure 2.1.20.2, step 3.1 states, “This section provides guidance to determine minimum amount of time that must elapse after reactor shutdown before various FPC configurations combined with other heat removal methods may replace RHR-SDC. These limitations are derived from technical information provided by the latest revision of NEDC 00-0105...” and (2) the point for a shutdown time of 40 days and REC temperature at 50˚ on the graph for Case 2 (1 FPC Pump and 2 FPC HX) is between 155˚ and 160˚, therefore the lower temperature should be selected.

Technical References: Procedure 2.1.20.2, "Cycle Specific Fuel Transfer and Alternate Cooling Guideline," Revision 8

NEDC 00-0105 (latest revision), Fuel Pool Decay Heat Loads/Alternate Decay Heat Removal Time Limitations

Proposed References to be provided to applicants during examination: **NEDC 00-0105 (latest revision), Fuel Pool Decay Heat Loads/Alternate Decay Heat Removal Time Limitations, Attachment C, Case 1 Graph & Case 2 Graph [calculation number not to be shown]**

Learning Objective

Question Source: New Question

Question History: Never Used

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.10 , 55.43.5

Comments:
(KA) Abilty to obtain and interpret station reference materials (graphs, monographs, tables) which contain performance data as applied to 233000, Fuel Pool Cooling and Cleanup.

Answer (B), NEDC 00-0105, "Fuel Pool Decay Heat Loads/Alternate Decay Heat Removal Time Limitations" / 160˚F, is incorrect because the point for a shutdown time of 40 days and REC temperature at 50˚ on the graph for Case 2 (1 FPC Pump and 2 FPC HX) is between 155˚ and 160˚, therefore the lower temperature should be selected.

Answer (C), Procedure 2.1.20.2, "Cycle Specific Fuel Transfer and Alternate Cooling Guideline" / 155˚F, is incorrect because procedure 2.1.20.2, step 3.1 states, "This section provides guidance to determine minimum amount of time that must elapse after reactor shutdown before various FPC configurations combined with other heat removal methods may replace RHR-SDC. These limitations are derived from technical information provided by the latest revision of NEDC 00-0105..."

Answer (D), Procedure 2.1.20.2, "Cycle Specific Fuel Transfer and Alternate Cooling Guideline" / 160˚F, is incorrect because (1) procedure 2.1.20.2, step 3.1 states, "This section provides guidance to determine minimum amount of time that must elapse after reactor shutdown before various FPC configurations combined with other heat removal methods may replace RHR-SDC. These limitations are derived from technical information provided by the latest revision of NEDC 00-0105..." , and (2) the point for a shutdown time of 40 days and REC temperature at 50˚ on the graph for Case 2 (1 FPC Pump and 2 FPC HX) is between 155˚ and 160˚, therefore the lower temperature should be selected.

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number		2
	Group Number		2
	K / A Number	290002	A2.05
	Importance Rating		4.2

Proposed Question:

18. The reactor is operating at 100% rated thermal power when an inadvertent HPCI initiation causes power to increase to 105%. Shortly thereafter radiation alarms are received on the main steam lines and SJAE's. A Periodic Case is generated and it shows the highest MFLCPR value at 1.10.

What impact on the reactor vessel internals would be expected to result from these conditions and what procedure should be entered to mitigate the conditions:

A. 0.1% of fuel rods experience an average fuel rod surface temperature of 1210˚F / Procedure 10.7, "Core Thermal Limits"

B. 0.1% of fuel rods experience an average fuel rod surface temperature of 1800˚F / Procedure 2.4RXPWR, "Reactor Power Anomalies"

C. 0.1% of fuel rods experience fuel damage / Procedure 5.2FUEL, "Fuel Failure"

D. 0.1% of fuel rods experience fuel damage / Procedure 10.7, "Core Thermal Limits"

Proposed Answer: (D) <u>0.1% of fuel rods experienced fuel damage / Procedure 10.7, "Core Thermal Limits"</u>. This answer is correct because per USAR section III-7 if the MCPR is not exceeded at least 99.9% of the fuel rods would be expected to avoid boiling transition. A condition exceeding MCPR is given in the problem, therefore 100% - 99.9% = 0.1% of fuel rods experience boiling transition, and per USAR III-7 fuel damage is assumed to occur when a boiling transition occurs. Procecedure 10.7, step 1, states "This procedure presents the method used to obtain Core Thermal Limits from 3D MONICORE. This procedure also supplies corrective action and documentation to be completed if the Core Thermal Limits are exceeded."

Technical References: Procedure 10.7, "Core Thermal Limits," Revision 24

COR002-10-02, "Fuel," Revision 14

Proposed References to be provided to applicants during examination: None.

Learning Objective

Question Source: New Question

Question History: Never Used

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.5

Comments:
(KA) Ability to predict the impacts of exceeding thermal limits on reactor vessel internals and on the basis of the prediction use procedures to correct, control, or mitigate the consequences of the abnormal conditions or operations.

Answer (A), 0.1% of fuel rods experienced an average fuel rod surface temperature of 1210˚F / Procedure 10.7, "Core Thermal Limits". This answer is incorrect because 1210˚F is a normal average volumetric temperature per USAR section III-7, and safety analyses do not predict an average fuel rod surface temperature of 1210˚F as a result of exceeding MCPR.

Answer (B), 0.1% of fuel rods experienced an average volumetric temperature of 1800˚F / Procedure 2.4RXPWR, "Reactor Power Anomalies". This answer is incorrect because safety analyses do not predict an average volumetric temperature of 1800˚F as a result of exceeding MCPR, and Procedure 10.7, step 1, states "This procedure also supplies corrective action and documentation to be completed if the Core Thermal Limits are exceeded."

Answer (C), 0.1% of fuel rods experienced fuel damage / Procedure 5.2FUEL, "Fuel Failure". This answer is incorrect because Procedure 10.7, step 1, states "This procedure also supplies corrective action and documentation to be completed if the Core Thermal Limits are exceeded."

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number		3
	Group Number		1
	K / A Number	2.1.14	
	Importance Rating		3.3

Proposed Question:

19. The announcement, “All personnel using breathing equipment supplied by plant air shall move to an area with clean atmosphere” is required to be made when which of the given conditions occurs:

    A. SA-MO-81, SA to IA Crosstie, closes

    B. Service Air pressure is 95 psig and lowering

    C. Bypass Valve SA-81MV opens

    D. Loss of CDP-1B

Proposed Answer: (B) Service Air pressure is 95 psig and lowering. This answer is correct because procedure 5.2AIR, step 4.4, directs the announcement when the procedure is entered. One entry condition for 5.2AIR is service air pressure below the green band. The green band for service air pressure is 100-110 psig. Service air pressure of 95 psig is less than 100 psig and is below the green band.

Technical References: Procedure 5.2AIR, “Loss of Instrument Air,” Revision 5

Procedure 0-CNS-02, “Site Work Practices,” Revision

Proposed References to be provided to applicants during examination: None

Learning Objective

Question Source: New Question

Question History: None

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43.5

Comments:
(KA) Knowledge of system status criteria which require notification of plant personnel.

Answer (A), SA-MO-81, <u>SA to IA Crosstie, closes</u>, is incorrect because closing the cross-tie does not [by itself] isolate either instrument air or service air.

Answer (C), <u>Bypass Valve SA-81MV opens</u>, is incorrect because opening a bypass valve around the filters does not isolate either instrument air or service air

Answer (D), <u>Loss of CDP-1B</u>, is incorrect because it doesn't have any effect on the instrument or service air systems. A loss of CDP-1A causes SA-PCV-609, Service Air System Isolation, to close. PCV-609 closing requires the announcement be made.

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number		3
	Group Number		2
	K / A Number	2.2.14	
	Importance Rating		3.0

Proposed Question:

20. Maintenance is scheduled in the plant that will require a temporary alteration. A 50.59 review of the Maintenance Work Order is required if the temporary alteration is expected to be in place for _______.

A. ≥ 30 days (Mode 1 or 2 only)

B. ≥ 30 days (all Modes)

C. ≥ 90 days (Mode 1 or 2 only)

D. ≥ 90 days (all Modes)

Proposed Answer: (C) ≥ 90 days (Mode 1 or 2 only). This answer is correct because procedure 3.4.4, step 2.2.5.1, states "If the temporary alteration is expected to be in place ≥ 90 days at power (MODES 1 or 2) or if the temporary alteration's expected duration changes and it will be in place ≥ 90 days at power (MODES 1 or 2), perform a 50.59 review of the MWO. This review shall be completed, as soon as possible, prior to exceeding 90 days at power (MODES 1 or 2)."

Technical References: Procedue 0-NPG-4.4, "Configuration Management," Revision 4

Procedure 3.4, "Configuration Change Control," Revision 35

Procedure 3.4.4, "Temporary Configuration Change," Revision 2

Proposed References to be provided to applicants during examination: None.

Learning Objective

Question Source:	New Question
Question History:	Never Used
Question Cognitive Level:	Memory or Fundamental Knowledge
10 CFR Part 55 Content:	55.43.3

Comments:
(KA) Knowledge of the process for making configuration changes.

Answer (A), ≥ 30 days (Mode 1 or 2 only), is not correct because procedure 3.4.4, step 2.2.5.1, states “If the temporary alteration is expected to be in place ≥ 90 days at power (MODES 1 or 2) or if the temporary alteration's expected duration changes and it will be in place ≥ 90 days at power (MODES 1 or 2), perform a 50.59 review of the MWO...”

Answer (B), ≥ 30 days (all Modes), is not correct because procedure 3.4.4, step 2.2.5.1, states “If the temporary alteration is expected to be in place ≥ 90 days at power (MODES 1 or 2) or if the temporary alteration's expected duration changes and it will be in place ≥ 90 days at power (MODES 1 or 2), perform a 50.59 review of the MWO...”

Answer (D), ≥ 90 days (all Modes), is not correct because procedure 3.4.4, step 2.2.5.1, states “If the temporary alteration is expected to be in place ≥ 90 days at power (MODES 1 or 2) or if the temporary alteration's expected duration changes and it will be in place ≥ 90 days at power (MODES 1 or 2), perform a 50.59 review of the MWO...”

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number		3
	Group Number		2
	K / A Number	2.2.26	
	Importance Rating		3.7

Proposed Question:

21. The plant is in Mode 5 with irradiated fuel assemblies in the reactor vessel and the fuel grapple expected to be used.

    Which of the following are performed as part of procedure 6.Refuel302, "Daily Refueling Equipment Check:"

    A. Check fuel pool level, cycle the fuel grapple closed and open to verify GRAPPLE CLOSED light indication, and raise the fuel grapple to verify the GRAPPLE NORMAL UP light indication.

    B. Check fuel pool level, lift the test weight using the grapple until the GRAPPLE NORMAL UP light illuminates, and check that pins which secure the grapple to the swivel and hoist cable do not show excessive wear.

    C. Check that pins which secure the grapple to the swivel and hoist cable do not show excessive wear, lower the grapple onto a solid surface and verify the SLACK CABLE light illuminates, and verify the Reactor Mode Switch is locked in REFUEL with the key removed.

    D. Verify the Reactor Mode Switch is locked in REFUEL with the key removed, check that grapple motor, drum, and brake drive line components are securely in place, and verify operability of the Refuel Mode One-Rod-Out Interlock.

Proposed Answer: A; This answer is correct because it restates steps 4.1, 4.2.8.2 and 4.2.8.3, and 4.2.8.8 of procedure 6.Refuel302.

Technical References: Procedure 6.Refuel302, "Daily Refueling Equipment Check," Revision 4

Proposed References to be provided to applicants during examination: None.

Learning Objective

Question Source: New Question

Question History: Never Used

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43.5

Comments:
(KA) Knowledge of refueling administrative requirements.

Answer B (Check fuel pool level, lift the test weight using the grapple until the GRAPPLE NORMAL UP light illuminates, and check that pins which secure the grapple to the swivel and hoist cable do not show excessive wear) is not correct because “lift the test weight using the grapple until the GRAPPLE NORMAL UP light illuminates” is performed once per 7 days using procedure 6.Refuel304, step 4.43.

Answer C (Check that pins which secure the grapple to the swivel and hoist cable do not show excessive wear, lower the grapple onto a solid surface and verify the SLACK CABLE light illuminates, and verify the Reactor Mode Switch is locked in REFUEL with the key removed) is not correct because “verify the Reactor Mode Switch is locked in REFUEL with the key removed” is performed once per shift per procedure 6.Refuel301, step 4.2

Answer D (Verify the Reactor Mode Switch is locked in REFUEL with the key removed, check that grapple motor, drum, and brake drive line components are securely in place, and verify operability of the Refuel Mode One-Rod-Out Interlock) is not correct because (1) “verify the Reactor Mode Switch is locked in REFUEL with the key removed” is performed once per shift per procedure 6.Refuel301, step 4.2, and (2) “verify operability of the Refuel Mode One-Rod-Out Interlock” is performed once per 7 days using procedure 6.Refuel304, steps 4.7.1 through 4.7.17.

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number		3
	Group Number		3
	K / A Number	2.3.3	
	Importance Rating		2.9

Proposed Question:

22. A 1000-gallon tank mounted on a trailer located outside of radwaste is being used for temporary storage of radioactive liquids. The tank has a pressure/overflow relief valve that discharges to liquid radwaste and is surrounded by magenta-and-yellow cord with radiation protection postings . Routine sampling of the tank determines the total radioactivity content is 8.5 Curies.

    This tank (is/is not) an unprotected outdoor liquid storage tank for radioactive liquids as described in TS 5.5.8 and DLCO 3.1.4. The total amount of radioactivity (is/is not) within TS limits ?

    A. is / is

    B. is / is not

    C. is not / is

    D. is not / is not

Proposed Answer: (A) IS an unprotected outdoor liquid storage tank / IS within TS 5.5.8 limits. This answer is correct because TS 5.5.8 states, "This program provides controls for potentially explosive gas mixtures contained in the Augmented Offgas Treatment System, and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks...(b) A surveillance program to ensure the quantity of radioactivity contained in each outside temporary liquid radwaste tank that is not surrounded by a liner, dike, or wall capable of holding the tank's contents and that does not have a tank overflow and surrounding area drain connected to the Liquid Radwaste System is ≤ 10 curies, excluding H-3 and dissolved noble gases."

Technical References: TS 5.5.8

Proposed References to be provided to applicants during examination: TS 5.5.8

Learning Objective

Question Source: New Question

Question History: Never Used

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43.4

Comments:
(KA) Knowledge of SRO responsibilities for auxiliary systems outside the control room (waste disposal and handling systems).

Answer (B), IS an unprotected outdoor liquid storage tank / IS NOT within TS 5.5.8 limits, is not correct because although it is an unprotected tank (is not surrounded by a liner, dike, or wall capable of holding the tank's contents and that does not have a tank overflow and surrounding area drain connected to the Liquid Radwaste System), it is within TS 5.5.8 limits (< 10 Ci).

Answer (C), IS NOT an unprotected outdoor liquid storage tank / IS within TS 5.5.8 limits, is not correct because the tank is an unprotected tank (is not surrounded by a liner, dike, or wall capable of holding the tank's contents and that does not have a tank overflow and surrounding area drain connected to the Liquid Radwaste System), although it is within tS 5.5.8 limits (< 10 Ci).

Answer (D), IS NOT an unprotected outdoor liquid storage tank / IS NOT within TS 5.5.8 limits, is not correct because it is an unprotected tank (is not surrounded by a liner, dike, or wall capable of holding the tank's contents and that does not have a tank overflow and surrounding area drain connected to the Liquid Radwaste System), and is within TS 5.5.8 limits (< 10 Ci).

Examination Outline Cross Reference	Level	RO	SRO
	Tier Number		3
	Group Number		3
	K / A Number	2.3.4	
	Importance Rating		3.1

Proposed Question:

23. In accordance with procedure EPIP 5.7.12, "Emergency Radiation Exposure Control," the Emergency Director can authorize:

A. a declared pregnant worker to receive an emergency radiation exposure of 500 mrem and not to exceed 5 rem.

B. workers providing first aid to injured personnel with non-life-threatening injuries to receive an emergency radiation exposure not to exceed 10 rem.

C. workers performing corrective actions to protect large populations from radiological exposure emergency radiation to receive an exposure greater than 10 rem and not to exceed 25 rem.

D. workers providing rescue and/or treatment of personnel with life-threatening injuries to receive an emergency radiation exposure not to exceed 75 rem.

Proposed Answer: C; This is correct because it restates the conditions of EPIP 5.7.12, steps 3.1.2 and 3.1.2.1.

Technical References: Emergency Plan Implementing Procedure 5.7.12, "Emergency Radiation Exposure Control," Revision 14

Proposed References to be provided to applicants during examination: None.

Learning Objective

Question Source: New Question

Question History: Never Used

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43.4

Comments:

(KA) Knowledge of radiation exposure and contamination control limits , including permissible levels in excess of those authorized.

Answer A (A declared pregnant worker may be authorized emergency radiation exposures between 500 mrem and not to exceed 5 rem) is not correct because 10 CFR 20.1208 restricts a declared pregnant worker to less than or equal to 500 mrem TEDE during the length of their declared pregnancy without reference to emergency conditions, and, EPIP 5.7.12, step 2.6.3, states that declared pregnant workers are excluded from receiving (planned) emergency radiation exposures.

Answer B (Workers providing first aid to injured personnel with non-life-threatening injuries may be authorized emergency radiation exposures not to exceed 10 rem) is not correct because a worker providing first aid to injured personnel or acting in support of life saving activities may be authorized emergency radiation exposure between 10 rem and 25 rem, per EPIP 5.7.12, step 3.1.2.1.

Answer D (Workers providing rescue and/or treatment of personnel with life-threatening injuries may be authorized emergency radiation exposures not to exceed 75 rem) is not correct because workers providing rescue and/or treatment of personnel with life-threatening injuries may be authorized emergency radiation exposures above 25 rem without specific limit, per EPIP 5.7.12, step 3.1.1. Federal Guidance prior to the current (1993) revision of the EPA Manual of Protective Actions (EPA400) had a limit of 75 rem for emergency worker life-saving actions.

Examination Outline Cross Reference	Level		RO	SRO	
	Tier Number		3		
	Group Number		4		
	K / A Number	2.4.22			2.4.22
	Importance Rating		4.0		

Proposed Question:

24. The plant experienced a significant transient and the current plant conditions are as follows:

- ✓ RPV water level is at the top of the active fuel
- ✓ Reactor power is 2 percent
- ✓ The Pressure Suppression Pressure (PSP) limit is about to be exceeded due to increasing suppression pool level.
- ✓ Reactor pressure is 700 psig
- ✓ All Division 1ECCS components are operable
- ✓ All Division 2 ECCS components are inoperable due to a loss of AC power

At this point in time, the highest priority for the SRO is to implement emergency procedures that will:

A. maintain RPV water level.

B. shutdown the reactor.

C. emergency depressurize the RPV.

D. restore division 2 ECCS power

Proposed Answer: C - An emergency depressurization is required prior to exceeding the PSP limit or the suppression pool boundary (and hence the containment) are threatened. This priority over rides all others.

Technical References: EOP 3A, EOP basis documents.

Proposed References to be provided to applicants during examination: None

Learning Objective

Question Source: New Question

Question History:	Never Used	
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	XX
10 CFR Part 55 Content:	55.43.5	

Comments:
(KA) Knowledge of the bases for prioritizing safety functions during abnormal or emergency operations.

Examination Outline Cross Reference

Level	RO	SRO
Tier Number	3	
Group Number	4	
K / A Number	2.4.44	2.4.44
Importance Rating	4.0	

Proposed Question:

25. During a “General Emergency” classification, a three-sector wide protective action recommendation is made to offsite authorities. Subsequent to this protective action recommendation, the wind shifts so that a new centerline direction is steady and located two sectors in the clockwise direction from its previous position. For purposes of making a new protective action recommendation, the downwind affected sectors following the wind shift are:

    A. The new centerline sector plus one sector to either side.

    B. The original three sectors plus the two sectors in the clockwise direction only.

    C. The original three sectors plus two additional sectors to both sides.

    D. All sectors.

Proposed Answer: B ; The answer is correct because after affected sectors are included in a protective action recommendation, they cannot be deleted from subsequent protective action recommendations, and new sectors are added according to actual wind direction, per EPIP 5.7.20, steps 1.1.6 and 1.1.7.

Technical References: “Emergency Plan for Cooper Nuclear Station,” Revision 49, Sections: 4.1.4, “General Emergency, ”5.2, “Emergency Response Organization,” 5.2.3, “EOF Director,” and 6.5, “Protective Actions” and Emergency Plan Implementing Procedure 5.7.20, “Protective Actions,” Revision 17

Proposed References to be provided to applicants during examination: None.

Learning Objective

Question Source: New Question

Question History: Never Used

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43.5

Comments:
(KA) Knowledge of emergency plan protective action recommendations.

Answer A (The new centerline sector plus one sector to either side) is not correct because once recommended, sectors may not be deleted from subsequent protective actions per EPIP 5.7.20, step 1.1.6.

Answer C (The original three sectors plus two additional sectors to both sides) is not correct because only sectors in the direction of the wind shift are added to a protective action per EPIP 5.7.20, step 1.1.6.

Answer D (All sectors) is not correct because only those additional sectors determined from the actual wind direction(s) are added, per EPIP 5.7.20, steps 1.1.6 and 1.1.7.