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:

1. The reactor is operating at 90 % 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 down and generator output will decrease.

   B. remain stationary and generator output will decrease.

   C. throttle open and generator output will remain constant.

   D. remain stationary and generator output will remain constant.

Proposed Answer: A 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.

Answers B, C, and D are 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.

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:

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 add positive reactivity to the core.

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

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

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

Proposed Answer: D - 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 (B), is incorrect because reactor recirc pump seals are not a concern in this scenario.

Answer (C), 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 (A), 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:

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

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

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

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

Proposed Answer: D - The loss of RPS PP 1B will de-energize the outboard 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:

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 AC bus that provides power to the pump motor.

   D. the loss of the DC 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 (d) is incorrect because DC power is not provided to the pump motor.

Answer (c) 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:

5. A routine shutdown of the unit is being performed when at about 29 % 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 RPS scram logic.

   C. indicates a problem with the main turbine trip 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 %. It informs the operator the turbine trip induced scram is bypassed due to power being less than 30 %.

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 monitor or operate RPS as applies to main turbine generator trip.

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

Answer (b) 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	295006	2.4.45
	Importance Rating	3.3	

Proposed Question 6:

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

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

   A. Break condenser vacuum.

   B. Trip all but two Condensate Pumps.

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

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

Proposed Answer: C - 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 prioritize and interpret the significance of each annunciator and alarm as applied to 295006, SCRAM.

Answer A is incorrect because there is no requirement to break vacuum as an immediate action.

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

Answer B 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	295016	AA2.01
	Importance Rating	4.1	

Proposed Question 7:

7. The Control Room has been abandoned due to presence of a toxic gas. Prior to leaving the control room the reactor operator announced “several” control rods did not insert on the reactor scram.

   In order to obtain a direct indication of reactor power outside the control room, an operator would go to the:

   A. Critical Switchgear Room.

   B. Technical Support Center.

   C. Alternate Shutdown Room.

   D. Non-Critical Switchgear Room.

Proposed Answer: (B) The TSC has PMIS computers capable of monitoring reactor power.

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.

Answers A, C, and D are incorrect because there are no APRM indications of reactor power in these locations.

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:

8. The reactor is operating RTP with the REC “A” Heat Exchanger and “A” REC 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 a rapid power reduction.

   The reason for the power reduction is to:

   A. reduce the effects on drywell pressure.

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

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

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

Proposed Answer: (A) 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 (B), <u>Maintain REC Pump A discharge pressure below 105 psig</u>, 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 (C), <u>minimize REC Pump A cavitation due to the higher REC fluid temperature</u> is not correct because the REC pump will not cavitate at 105 degrees.

Answer (D), <u>Reduce the chance of damaging Recirculation Pumps MG set(s)</u>, 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:

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.

   According to procedure 5.2AIR, Loss of Instrument Air, if instrument air header pressure continues a downward trend, the operator is required to immediately scram the reactor if:

   A. an MSIV drifts closed.

   B. an ADS valve drifts open.

   C. a scram valve drifts open.

   D. the in-service CRDH flow control valve drifts closed.

Proposed Answer: C - The referenced procedure states that if one or more scram valves open, scram the reactor and enter the reactor scram procedure.

Technical References: Procedure 5.2Air, Loss of Instrument Air, Revision 5, Step 4.1.
Procedure 2.4SRV, Stuck Open Relief Valve, Revision 0
Procedure 2.4MSIV, Inadvertent MSIV Closure, Revision 0
Procedure 2.4CRD, CRD Trouble, Revision 5

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) 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 a drifting MSIV does not require an immediate scram.

Answer (b) is incorrect because an ADS valve drifting open does not require an immediate reactor scram unless suppression pool temperature reaches 110 degrees.

Answer (d) is incorrect because closure of the in-service CRDH flow control valve does not require a scram.

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:

10. The reactor is shutdown with RHR A in service for shutdown cooling (SDC). A loss of all off-site power occurs followed by an automatic start of the emergency diesel generators (EDGs). Reactor water level is 55 inches (Shutdown Instrument) and drywell pressure is 0.2 psig.

With regard to restoring SDC, operator action is:

A. required for restarting the RHR A pump only.

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

C. 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.

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

Proposed Answer: B - The loss of offsite power trips the RPS MG Sets thus resulting in a Group isolation on Shutdown Cooling. When the diesels re-power the busses, the group isolation will cause the valves to reposition, thus requiring the valves to be realigned.

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.

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:

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) <u>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</u>. 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), 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, 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), 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, 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:

12. The RPS setpoint for the high drywell pressure scram is:

A. $\leq$ 0.75 psig

B. $\leq$ 1.50 psig

C. $\leq$ 1.84 psig

D. $\leq$ 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 $\leq$ 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:

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.

With no operator action, after the initial transient 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:

14. Loop A of RHR is in the suppression pool cooling mode of operation. A reduction in the amount of heat (BTUs/hour) being removed from the suppression pool would occur if:

A. MCC Q is lost.

B. an SRV is opened.

C. river temperature increases.

D. emergency diesel generator 1 is started.

Proposed Answer: C - If river water temperature increases, then the differential temperature difference between the suppression pool and the RHR service water cooling the RHR heat exchanger decreases. This will result in a reduction in the amount of heat being rejected from the suppression pool. (q=mcdt)

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 a loss of bus Q will result in a loss of power to some of the SPC valves but the valves will not reposition. Thus there will be no effect on

Answer (b) is incorrect because SPC.opening an SRV will increase the amount of heat being put into the suppression pool. This will increase the differential temperature between the pool and the RHR heat exchanger thus causing the amount of heat being rejected to increase.

Answer (d) is incorrect because while starting the EDG will increase the heat load on the Service Water System, the SWS is an open system and the supply to the RHR heat exchanger is independent of the EDG. Thus there will be no effect on SPC.

During the exam a question was asked about whether bus Q as stated in distractor A was the same thing as MCC Q. The distractor was changed to read “MCC Q”

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:

15 Annunciator H-1/C-1, DRYWELL FCU C HI DISCH TEMP, has just annunciated in the control room and the operator suspects the alarm setpoint is incorrect.

The operator can determine the correct setpoint for this alarm by looking at:

A. Procedure 5.2REC, Loss of REC.

B. the alarm card for annunciator H-1/C-1.

C. Procedure 2.4PC, Primary Contaiment Control.

D. Procedure 4.13.3, Primary Containment Temperature.

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), Check temperature indication on recorders and PCIS & Ensure drywell REC lineup is correct & Enter procedure 4.13.3, Primary Containment Temperature, 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), Check temperature indication on recorders & Ensure all REC pumps are running & Enter procedure 6.PC.306, 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), Check that drywell pressure is between 0.25 and 0.45 psig & Ensure valves REC-MO-700 and REC-MO-711 are open & Enter procedures 2.4PC and 5.2REC, 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:

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:

17. A feedwater level control system malfunction has resulted in RPV level trending down. The operator notices the trend at +20 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. wide range indicator on Panel 9-4.

    C. GEMAC level indicator on Panel 9-5.

    D. narrow range indicator on Panel 9-5.

Proposed Answer: D - 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 XX
Comprehension or Analysis

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:

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 % reactor power. The reactor recirc pumps were tripped and reactor water level is +20 inches narrow range and lowering.

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

    A. wide

    B. narrow

    C. shutdown

    D. steam nozzle

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

Technical References: Nuclear Boiler Instrumentation

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

Answer B is incorrect because narrow range is calibrated with the reactor recirc pumps running.

Answer D is incorrect because steam nozzle range is calibrated with the reactor recirc pumps running.

Answer C 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	3.7	

Proposed Question 19:

19. The Control Room Emergency Filtration System (CREFS) is designed to automatically initiate during accident conditions in order to:

    A. maintain a negative pressure in the control room.

    B. maintain control room personnel dose below limits.

    C. enhance the reliability of control room electronics and instrumentation.

    D. provide a backup to the non-safety related Control Room HVAC System.

Proposed Answer: B is correct because the system is designed to maintain control room personnel dose below limits.

Technical References: COR0010801R13-S-OPS, Heating, Ventilation, and Air Conditioning, page 61.

Proposed References to be provided to applicants during examination: None

Learning Objective: LO 10a, 13a

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) is incorrect because the system maintains a positive pressure in the CR.

Answer C) is incorrect because initiating CREVS will not enhance anything.

Answer (D) is incorrect because the CREVS is not intended to be a backup to the CR HVAC.

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:

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:

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 CRDH "B" pump, the CRDH flow controller (CRD-FC-301) is placed in the manual and minimum position in order to:

    A. prevent rod drifts.

    B. prevent lifting any directional control valves.

    C. minimize check valve slam on the tripped pump.

    D. minimize the pressure transient on the HCU 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.

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:

22. A loss of feedwater event resulted in a reactor scram and HPCI was manually initiated by the reactor operator at -30 inches (wide range) RPV level.

    If an automatic trip on high RPV water level were to occur, it would cause the HPCI turbine:

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

    B. stop valve to close and the turbine governor valves to open.

    C. stop valve and exhaust line drain valves (AO-70 and AO-71) to close.

    D. governor valve to close and exhaust line drain valves (AO-70 and AO-71) 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:

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:

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 changed from a condition where average temperature was lowering to a condition where temperature is not changing.

    Which of the following conditions could help explain the problem with suppression pool cooling?

    A. An SRV is partially open.

    B. Refueling activities are in progress.

    C. A reactor scram has not been reset.

    D. The RHR minimum flow valve has opened.

Proposed Answer: D - 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 (c) is incorrect because a reactor scram signal will not influence suppression pool cooling.

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

Answer (a) 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:

25. The plant is in Mode 4 and preparations for a reactor startup are in progress with no LCOs in effect. An operator performing a rod control startup surveillance selects and attempts to withdraw a control rod.

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

A. Rod Worth Minimizer (RWM).

B. Rod Select Sequencer (RSS).

C. Rod Position Indication System (RPIS).

D. Reactor Manual Control System (RMCS).

Proposed Answer: D - 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:

26. The reactor was manually scrammed following a main condenser tube rupture and the reactor operator reported there are "many" 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
- Both Reactor Recirc pumps are in operation
- SRMs and IRMs have been fully inserted
- Indicated reactor period is infinity

At this point the crew should:

A. enter EOP 1A, RPV Control.

B. trip both reactor recirc pumps.

C. stabilize RPV pressure and temperature.

D. commence a 100 degree per hour cooldown rate.

Proposed Answer: C - With 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 D is incorrect because a cooldown could initiate a restart accident.

Answer B 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:

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

- Reactor Building Floor Drain Sump Pump is operating
- Reactor Building Equipment Drain Sump Pump is 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. there is a steam leak from the HPCI System.

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

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

Proposed Answer: (C) A steam leak from the RWCU System. 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:

28. The plant was operating at full power when a LOCA signal initiated RHR and aligned both loops in the LPCI mode. While walking down the boards, the operator notes the RHR Heat Exchanger Bypass Valve (RHR-MO-66A) is closed.

With this valve closed;

A. RPV level recovery will be faster.

B. RPV level recovery will be slower.

C. RHR injection temperature will be higher.

D. RHR injection temperature will be lower.

Proposed Answer: B - with the the heat exchange bypass valve closed, the injection rate will be less and therefore level recovery will take longer.

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 with the bypass valve closed all the injection flow is forced through the heat exchanger resulting in a lower injection rate. This will result in a slower level recovery.

Answers (c) and (d) are incorrect because having the bypass valve open or closed will have no effect on RHR temperatures during a LPCI injection.

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:

29. With RHR A in 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 - Cooldown rate is controlled using either the LPCI injection valve or the heat exchanger outlet valve. The suction valve should not be throttled.

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:

30. The High Pressure Coolant Injection (HPCI) keep-fill system is normally supplied by the _____ and the backup source of water is from the ________.

A. CST Recirc Pump; CRDH System

B. CRDH System; Condensate System

C. Reactor Building Auxiliary Condensate Pump; CST Recirc Pump

D. Condensate System; Reactor Building Auxiliary Condensate Pump

Proposed Answer: D

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.

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:

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:

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

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

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

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

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: Fundamental XX

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 (C), 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 (D), 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 (A), 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:

33. According to Technical Specification 3.1.7, Standby Liquid Control (SLC), the maximum allowed temperature for the sodium pentaborate solution is:

A. 110

B. 130

C. 150

D. 170

Proposed Answer: (C) According to the referenced techspec, the maximum allowable temperature for sodium pentaborate is 150 degrees.

The answer key for this question was changed from C to B post exam. The reason for the change is that the referenced specification is entered at 140 degrees due to SLC pump inoperability making C incorrect and 130 degrees the maximum allowed temperature of those listed.

Technical References: Technical specification 3.1.7, Standby Liquid Control.

Proposed References to be provided to applicants during examination: None

Learning Objective

Question Source: New Question

Question History: Never Used

Question Cognitive Level: Fundamental Knowledge

10 CFR Part 55 Content: 55.41.5

Comments:

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

Answers A, C, and D are all incorrect because the only maximum allowable temperature is 150 degrees.

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:

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

A. L.

B. T.

C. LX.

D. TX.

Proposed Answer: A - The power supply is from bus L.

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:

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. RPS actuation and a scram.

   B. control rod insert block for all rods.

   C. control rod withdraw block for all rods.

   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.

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:

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

A. Distribution Panel DC-A.

B. Distribution Panel CPP-2.

C. Distribution Panel CDP-1-A.

D. RPS Power Distribution Panel A.

Proposed Answer: A - The power supply is DC - 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.

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:

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

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 Based on these indications, there would be a flow comparator trip on channel 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:

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, 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. a decrease in total pump flow.

B. an increase in RCIC pump discharge pressure.

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

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

Proposed Answer: D - 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 (C) 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 (A) is incorrect because RCIC pump flow will increase.

Answer (B) 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:

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

    If ADS actuation were to occur, what affect would the loss of reactor building reliable instrument air have on the operation of the ADS valves?

    A. The ADS valves will not open.

    B. If open, the ADS valves will not close.

    C. There is no effect on the ADS valves.

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

Proposed Answer: C - The valves will continue to operate normally on the nitrogen supply system.

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.

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:

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

A. all ECCS pumps are lost.

B. the valve control switches are taken to CLOSE.

C. RPV level increases to +10 inches (wide range).

D. RPV level increases to -100 inches (wide range).

Proposed Answer: A - 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 (C) is incorrect because the level signal seals in and will not reset without operator action.

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

Answer (B) 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:

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 isolation lights should the operator expect to see reset on Panel 9-5 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:

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 - This is a function of pressure only.

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:

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

A. the SRV tailpipe is pressurized.

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

C. the solenoid to actuate the SRV is energized.

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

Proposed Answer: C - The light indicates the solenoid is energized.

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.

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:

44. Reactor power is 80 % 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 %. The reactor operator just increased power to 85 %.

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 just below the setpoint level.

D. RPV level will initially increase and will then return to just above the 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:

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. drywell pressure reaches 1.84 psig.

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

    C. reactor water level reaches 3.0 inches (narrow range) and decreasing.

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

Proposed Answer: B

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:

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. OCB 1602 opens.

    B. ACB 110 remains closed.

    C. OCB 1606 remains closed.

    D. Breakers 1AS and 1BS open.

Proposed Answer: (C) 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 (B), 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 (A), 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:

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:

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 RPS Bus A Pwr Transfer switch in the:

A. TRANS position and expect a half scram.

B. MG SET position and expect a half scram.

C. TRANS position and expect no change in RPS scram status.

D. MG SET position and expect no change in RPS scram status.

Proposed Answer: A - The switch must be placed in the TRANS 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 XX
Comprehension or Analysis

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

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:

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

    If panel 125 VDC AA3 **or** BB3 is lost, which of the following Alternate Rod Insertion (ARI) system functions are still available?

    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.”

During the exam clarification was asked with regard to which bus was lost, panel 125 VDC AA3 **or** BB3. Clarification was provided that first assume one bus is lost, then assume the other bus is lost. Which of the four potential answers would still be available regardless of which one was lost.

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:

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

The initial response is 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 droop associated with both the governor and regulator will decrease. This will result in real load (KW) to increasing. While the voltage droop will decrease, the actual reactive load (KVAR) change will depend on whether the power factor is leading or lagging.

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 voltage regulator droop decreases when the switch is placed in the isochronous mode.

Answers C and D are incorrect because real load (KW) will increase with a decrease in the droop setting.

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:

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. overheat; decrease

B. overheat; increase

C. trip on low voltage; 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.

Answers C and D are incorrect because the output voltage of the generator are controlled by the grid so it will not trip on undervoltage.

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:

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. air filter damage; bypass the air filters

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

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

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

Proposed Answer: C - 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:

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:

54. A symmetric rod pattern is one in which:

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

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

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 (B), Shallow control rods within a group are not more than one notch different from the other control rods, 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), Intermediate control rods within a group are not more than one notch different from the other control rods, 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 (A), Control rods at position 48 have been moved to position 44 due to high CRD temperatures, 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 not 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:

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. lower Suppression Pool inventory.

B. increase Suppression Pool temperature.

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

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

Proposed Answer: (A) 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 (C), Cause operating limits on the operating RHR heat exchanger to be exceeded, 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), Increase suppression pool temperature, 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),Raise Suppression Pool inventory, 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:

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. close when drywell pressure reaches 0.6 psig.

C. throttle to maintain drywell pressure about 0.25 psig.

D. open at a drywell pressure of 0.1 psig and closes at 0.45 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 (D), 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 $\leq$ 0.45 psig by venting containment through the SGT system.

Answer (B), 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:

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 in containment.

D. mitigate the effects of a potential 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:

58. The plant is operating at 70 % 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. open AO-80A because steam flow through the open steam line drains maintained an equal pressure across AO-80A.

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

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

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

Proposed Answer: (B) Cannot & Steam flow through the open steam line drain may have prevented the drainage of condensate. 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 (C), <u>Cannot & The valve cannot open against a 20-50 psid pressure difference between the idle and flowing steam lines</u>, 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 (B), <u>Can & Steam flow through the open steam line drain maintained an equal pressure across AO-80A</u>, 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), <u>Can & Steam flow through the open steam line drain maintained temperatures in the A steam line within operating limits</u>, 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:

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 (RHR-MO-20), 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. RHR Subsystem B flow ≤ 2107 gpm

C. Suppression pool level = - 1.00 inches

D. Fuel Pool Cooling Pump A discharge ≥ 120 psig

Proposed Answer: (B) <u>RHR Subsystem B flow ≤ 2107 gpm</u>. 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 (D), 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 K6.06	
	Importance Rating	3.3	

Proposed Question 60:

60. Reactor power is 90% and a tube rupture in feedwater heater A5 has resulted in a high heater water level and the closure of the heater's extraction steam isolation valves.

With no operator action, an hour later the control room operators should observe reactor power has:

A. increased and condensate hotwell temperature has increased.

B. increased and condensate hotwell temperature has decreased.

C. decreased and condensate hotwell temperature has decreased.

D. increased and condensate hotwell temperature is the same as it was.

Proposed Answer: (B) With the loss of extraction steam to heater A5, feedwater temperature will decrease. The decrease in temperature will add positive reactivity to the reactor and increase power. The power increase will heat the subcooled feedwater but overall steam flow will decrease with a constant recirc flow. The turbine control valves will throttle down to maintain a constant throttle pressure thereby reducing steam flow through the turbine. With the throttling of the governor valves, exhaust temperature and steam flow will be lower resulting in hotwell temperature decreasing.

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

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 reactor feedwater system will have on the reactor condensate system.

Answer (A), <u>increased and condensate hotwell temperature has increased</u> is incorrect because with the reduction in turbine exhaust steam flow and temperature, hotwell condensate temperature will go down.

Answer (C), <u>decreased and condensate hotwell temperature has decreased</u> is incorrect because with a constant reactor recirc flow colder feedwater temperatures will result in a higher reactor power.

Answer (D), <u>increased and condensate hotwell temperature is the same as it was</u> is incorrect because of the reasons stated in distractor A.

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:

61. A Group 2 PCIS isolation signal will cause the:

A. offgas filter to have a high ΔP.

B. mechanical vacuum pump to trip.

C. AOG steam supply valves to close.

D. standby offgas dilution fan to auto start.

Proposed Answer: (C) The AOG steam supply valves will close on a Group 2 isolation.

Technical References: COR001-16-01, “OPS Off Gas,” Revision 20, page 48.

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 (B), <u>Mechanical Vacuum Pump Trip</u>, is incorrect because the mechanical vacuum pumps will only auto trip in the event of high steam line radiation.

Answer (D), Standby Offgas Dilution Fan Auto-Start, is incorrect because the standby dilution fan will only auto-start if system flow is $\leq$ 1100 cfm.

Answer (A), Offgas Filter High ΔP, is incorrect because with the steam supply valves closed, there will be a low flow condition and this will result in decreasing filter differential.

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:

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

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

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

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

Proposed Answer: (A) 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 (D), 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 (C), 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:

63. While investigating a low supervisory air pressure trouble alarm for the sprinkler system in the Electrical Cable Spreading Room, an operator discovers the supervisory air system pressure for that area is 14 oz and slowly decreasing.

The concern associated with this alarm is the deluge valve:

A. will not operate in the event of a fire.

B. cannot be opened without supervisory air.

C. may inadvertently open, charging the sprinkler header.

D. can only be operated from its local pull handle, delaying the response to a fire.

Proposed Answer: (C) <u>The deluge sprinkler valve may inadvertently open, creating a local hazard</u>. 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, page 24 of 66, step 3.e.

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 (B), sprinkler valve cannot be opened without supervisory air. This answer is incorrect because without supervisory air the valve will actuate but cannot be closed.

Answer (A), The deluge valve sprinkler valve will not operate in the event of a fire is incorrect because the sprinkler valve is actuated by the local temperature and does not use supervisory air.

Answer (D), The deluge valve can only be operated locally, delaying the 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.” Low supervisory air will cause the valve to open.

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:

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 __________. The alarm condition can be reset by ___________.

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

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

C. close fire smoke dampers 1544, 1545, 1546, 1547, 1581 and 1582; manually resetting using the reset magnets

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

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 (D), 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 (B), 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:

65. Given the design of the reactor vessel and internals, RPV water level must at least be up to the ____________ for natural circulation to take place.

A. reference zero

B. active fuel (TAF)

C. top of the jet pump nozzles

D. bottom of the moisture separators

Proposed Answer: (D) RPV level must be at least up to the bottom of the moisture separators for natural circulation to occur. Otherwise there is no return path to the downcomer.

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: Fundamental Knowledge XX

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.

Answers A, B, C are incorrect because each of these levels are below the bottom of the moisture separators.

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:

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

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

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

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

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

Proposed Answer: C - With power at 30 %, 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 (B) is incorrect because the RPV pressure safety limit is 1337 psig.

Answer (D) is incorrect because the RPV level safety limit is the top of active fuel.

Answer (A) 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:

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

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

B. the automatic signal failure swap-over feature is armed.

C. correct auctioneering of the RFPT Controlling Governor Modules.

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

Proposed Answer: (A) <u>Ensure overfill protection in the event of a sensing line break</u>. 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 (D), Ensure overfill protection in the event of an upscale pressure transmitter failure, 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), Ensures correct auctioneering of the RFPT Controlling Governor Modules, 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 (B), the automatic signal failure swap over feature is armed is incorrect because this switch has no function associated with an automatic swap-over.

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:

68. A Technical Specification Limiting Condition of Operation (LCO) entry would be required if:

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

B. average air temperature in the drywell is 125 degrees.

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

D. RCIC Valve MO-30 (RCIC Test Bypass to ECST Valve) is declared inoperable.

Proposed Answer: (C) 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.

Technical References: Technical Specification 3.5.1, “ECCS - Operating,’” Amendment 203

Proposed References to be provided to applicants during examination: None

Learning Objective

Question Source: New Question

Question History: Never Used

Question Cognitive Level: Fundamental

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 (B) is incorrect because a tech spec entry is not required until drywell average temperature reaches 135 degrees.

Answer (D), 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 (A), 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.

During the exam a question was asked if distractor A was referring to REC water temperature or Service Water temperature. Clarification was provided that the distractor was referring to REC water temperature.

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:

69. According to technical specification 2.2, Safety Limit Violations, the operator has up to _______ 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: (C) This answer is correct because the referenced tech spec 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.

Answers A, B, and D are is incorrect because the referenced tech spec gives two hours as the limit.

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:

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. terminate fuel loading.

B. evacuate the refueling floor.

C. actuate the reactor building evacuation alarm.

D. enable the audible SRM count rate in the reactor building.

Proposed Answer: (A) Terminate fuel loading. 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 (D) is incorrect because this is not required per procedure 10.25.

Answer (B), evacuate the refueling floor is incorrect because procedure 10.25 does not require evacuating the refueling floor.

Answer (C), actuate 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:

71. A different type of resin is being considered for the condensate filters. Based on some events at other nuclear plants, the resin needs to be evaluated for reactivity effects should it get into the reactor.

This reactivity evaluation would be done via a:

A. Reactivity Impact Determination.

B. Technical Specification Change Request.

C. Procurement Change in Material Evaluation.

D. submittal to the NRC for review and approval.

Proposed Answer: (A) Reactivity Impact Determination. This answer is correct because procedure 0.52 states in step 3.1, "[The] Department initiating an activity that has the potential to impact reactivity will request a Reactivity Impact Determination from Reactor Engineering."

Technical References: Procedure 0.52, Reactivity Impact Determination, Revision 5

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

Learning Objective

Question Source: New Question

Question History: Never Used

Question Cognitive Level: Fundamental Knowledge

10 CFR Part 55 Content: 55.43.10

Comments:
(KA) Knowledge of the process for determining the internal and external effects on core reactivity.

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:

72. Procedure 9.ALARA.1, Personnel Dosimetry and Occupational Radiation Exposure Program, states NPPD shall monitor the occupational radiation dose of any person that:

A. enters an RCA.

B. enters a high radiation area.

C. is expected to receive greater than 100 mrem during the calender year.

D. is expected to receive an occupational dose in excess of 300 mrem during the current quarter.

Proposed Answer: (B) Enter a high radiation area. This is correct because procedure 9.ALARA.1, step 6.2.8.3 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 (C), is expected to receive greater than 100 mrem during the calender year, 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 receive 300 mrem during the current quarter, is incorrect because procedure 9.ALARA.1, step 6.2.9.3 states, “NPPD shall monitor occupational radiation dose if: An individual will receive in excess of 10% of the limits in 10CFR20. There is no limit on the exposure received in a quarter.

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:

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 total 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; 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:

74. The plant experienced a LOCA approximately 4 hours ago and currently;

- Suppression pool temperature is 200 degrees
- 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) Consistent with plant conditions if suppression pool temperature is increasing AND HPCI system flow is fluctuating. 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:

75. The plant is at 80 % power when operators note condenser vacuum is degrading (trending toward no vacuum) and the Low Pressure Turbine Absolute Recorder (MS-PR-73A) is showing an increasing trend.

Procedure 2.4VAC, Loss of Condenser Vacuum, states the operators are to immediately scram the reactor if vacuum cannot be maintained:

A. ≥ 7" Hg

B. ≥ 12" Hg

C. ≥ 21" Hg.

D. ≥ 23" Hg.

Proposed Answer: (D) 2.4VAC states the operator immediate action is to SCRAM the reactor when vacuum cannot be maintained ≥ 23" Hg.

Technical References: Procedure 2.4VAC, Loss of Condenser Vacuum, Revision 10

Proposed References to be provided to applicants during examination: None

Learning Objective CAF

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 although lowering vacuum is an entry condition to 2.4OG, starting the standby dilution fan does not require operator action because it starts automatically on low dilution flow.

Answer (B), 2.4OG, and perform a normal shutdown of OWC, is incorrect because 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 $\leq$ 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. A special reactivity test that involves lowering feedwater temperature following a turbine trip from 25 % reactor power is to be performed this shift. This evolution is not covered by any existing procedure.

   As part of the review for the conduct of this test, the SRO should find the feedwater temperature abort criteria in:

   A. the COLR (Core Operating Limits Report).

   B. the 50.59 Unreviewed Safety Question review.

   C. Technical Specification 3.1.2, Reactivity Anomalies.

   D. the STP (Special Test Procedure) for this evolution.

Proposed Answer: D - This test would be performed using procedure 2.0.1.1, Special Tests and Evolutions. This procedure requires the abort criteria to be a part of the STP.

Technical References: Procedure 2.0.1.1, Special Tests and Evolutions

Justification:

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) Ability to determine or interpret feedwater temperature as applied to a main turbine generator trip.

Answers A, B, and C are all incorrect because the only source for abort criteria for this test from those listed would be in the STP.

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. Service air pressure is below the green band and the AIR RECEIVER A OR B LOW PRESSURE alarm has annunciated. Service air pressure is now 80 psig and is continuing to lower.

   According to Operation Policy Procedure 2.0.1.2, Operations Procedure Policy, at this point in time the SRO should have ensured the following procedure(s) has (have) been implemented:

   A. EOP 1A, RPV Control ONLY

   B. 5.2AIR, Loss of Instrument Air ONLY

   C. A-4/A-4 Alarm Procedure (AIR RECEIVER A OR B LOW PRESSURE) ONLY

   D. A-4/A-4 Alarm Procedure (AIR RECEIVER A OR B LOW PRESSURE) AND 5.2AIR, Loss of Instrument Air ONLY

Proposed Answer: D - Procedure 5.2 Air and the alarm procedure should be implemented concurrently.

Technical References: Operation Policy Procedure 2.0.1.2, Operations Procedure Policy, Revision 22
CNS Operations Manual Alarm Procedure 2.3.1, General Alarm Procedure

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 XX

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 Graph 1 and Graph 15 from the Emergency Operating Procedures, the:

A. level should be considered unknown and EOP 2B, RPV Flooding, should be entered.

B. Narrow Range indication should be used and level should be recovered using EOP 1A, RPV Control.

C. Steam Nozzle Range indication should be used and level should be recovered using EOP 1A, RPV Control.

D. Shutdown Flooding Range indication-should be used and level should be recovered using EOP 1A, RPV Control.

Proposed Answer: A - Level should be considered unknown because the curves in graphs 1 and 15 show the instruments are in the saturation region and are unreliable. In response EOP 2B, RPV Flooding should be entered.

Technical References: EOP 1A, 2B, Caution 1, Graph 1, Graph 15

Proposed references to be provided to applicants during examination: **Graph 1 and 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 an ATWS, the SRO has determined the average torus water temperature and RPV pressure cannot be maintained within the Heat Capacity Temperature Limit (HCTL). Current conditions are:

- 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
- MSIVs are closed
- Boron injection was initiated 5 minutes ago

The SRO should now transition to:

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 - With the suppression pool temperature approaching the HCTL, an emergency depressurization is required. Therefore EOP 6B must be implemented.

Technical References: EOPs

Proposed References to be provided to applicants during examination: EOP Graph 7

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 vacuum breaker operability is to:

   A. reduce the peak drywell pressure following a LOCA.

   B. prevent damage to the SRV tailpipes upon the opening of an SRV.

   C. prevent an implosion of the drywell following the initiation of drywell sprays.

   D. prevent a low suppression pool level due to a loss of inventory to the drywell following the initiation of drywell sprays.

Proposed Answer: A - The basis for this techspec is to reduce the drywell peak pressure following a LOCA.

Technical References: Technical Specification Bases background for TS 3.6.1.8.

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 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 % power during a plant startup with all systems operable and in their normal configuration. On June 13th at 1600, the surveillance coordinator reported that due to a missed surveillance, the 4 RPV pressure sensors used in the RPS (Reactor Vessel Steam Dome Pressure - High) were not functional tested as required by Technical Specification Surveillance Requirement 3.3.1.1.9 (attached).

If a risk evaluation is not performed, and the Action Statement(s) are not completed, the plant required to be in Mode 3 no later than:

A. June 14th at 1600.

B. June 14th at 1700.

C. June 15th at 0500.

D. June 15th at 1000.

Proposed Answer: C - Missing the surveillance invokes Surveillance Requirement SR 3.0.3 and gives the SRO 24 hours OR the length of the surveillance frequency which ever is greater to get the surveillance done before having to declare the associated LCO not met. To get the allowance to exceed the 24 hours a risk assessment must be performed and that risk managed. This question states that risk has not been evaluated, therefore the surveillance must be performed within 24 hours and if that time is exceeded the LCO shall be declared not met. At which time LCO 3.3.1.1 Condition C would apply because with all four RPV Pressure Instruments declared inoperable, trip capability for that function can not be assured. If RPS trip capability is not restored within 1 hour Condition D would apply and require entry into the condition referenced in Table 3.3.1.1-1 for the channel immediately. The condition referenced from the table is Condition G, Be in Mode 3 within 12 hours. So adding all the times; 24

hours + 1 hour + 12 hours = 37 hours from time of notification of the missed surveillance. June 13th at 1600 + 37 hours is June 15th at 0500.

A. is wrong because this is the amount of time allowed for the completion of the surveillance.

B. is wrong because this is the Surveillance time plus the one hour to restore trip capability.

D. is wrong because this is the time allowed if they performed Condition B instead of Condition C.

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 and Table 3.3.1.1-1**. **The allowable values are to be excluded from 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. A unisolable primary system leak is discharging outside the primary and secondary containment. The radioactive release rate is now approaching the level requiring a General Emergency be declared.

   Based on this the SRO should be preparing to:

   A. flood the RPV.

   B. open the MSIVs.

   C. vent the containment.

   D. emergency depressurize the RPV.

Proposed Answer: D - With the off-site release in progress, procedure 5A should be in use. According to EOP 5A, when the off-site release rate approaches the criteria for a general emergency, an ED is required.

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. 2 and 3 only.

Proposed Answer: D - Twelve mR/hr exceeds the entry condition of 10mR/hr for EOP 5A. This also an entry condition for 5.1RAD and it should be entered as well. If there are any conflicts between the two procedures the EOP is the top tier procedure for this situation and shall take precedence.

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. Plant Manager / is

B. Responsible Manager / is

C. Control Room Supervisor / is not

D. Responsible Maintenance Supervisor / is not

Proposed Answer: B - 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 SRO Only Task per licensee.

Comments:
(KA) Knowledge of the process for managing maintenance activities during shutdown operations as applied to 203000, RHR LPCI Injection Mode.

Answer C is incorrect because the Control Room Supervisor is not the approval authority and a Complex Troubleshooting Plan is required.

Answer A 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. This morning, June 13 at 0600, it was discovered Technical Specification Surveillance Requirement SR 3.5.1.11 was overlooked on the most recent startup schedule and was consequently not performed for any of the ADS valves. The surveillance overdue date (1.25 times the periodicity) was May 1, 2005 and the plant is now at 90% power. Engineering has performed a risk evaluation and put together a plan to manage the risk impact. All other ECCS components addressed by this LCO are operable.

    In response to this sequence of events, the:

    A. plant must be in Mode 3 no later than 1800 today.

    B. plant must be in Mode 3 no later than 0600 tomorrow.

    C. plant must be in Mode 3 no later than 1800 tomorrow.

    D. surveillance can be delayed for up to 18 months from the due date.

Proposed Answer: D - The surveillance must be performed within the next 18 months. According to TS SR 3.0.3, if the risk has been evaluated and a risk management plan is in place the surveillance can be delayed 24 hours or up to the periodicity, whichever is longer. Thus the first 3 answers are incorrect because the LCO does not get entered for 18 months.

Technical References: Technical Specification 3.5.1 and SR 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 will require opening a high voltage disconnect located in the Startup/Emergency Transformer Switchyard. In order to preclude ____________, CNS personnel are required to ____________ prior to opening the disconnects.

A. unauthorized vehicle entry to the switchyard / establish a CNS Safety Watch

B. mis-operating and damaging the disconnects / establish a CNS Safety Watch

C. mis-operating or damaging the disconnects / provide a pre-job-brief that includes pictures

D. unauthorized vehicle entry to the switchyard / post a Security Officer at the entrance in use

Proposed Answer: C - Procedure 0-CNS-52, Control of Switchyard and Transformer Yard Activities, requires a pre-job-brief be held prior to opening the diconnects that includes pictures. This is to preclude misoperating and damaging the disconnects.

Technical References: Procedure 0-CNS-52, Control of Switchyard and Transformer Yard Activities

Proposed References to be provided to applicants during examination:

Learning Objective: Per the licensee point of contact and Procedure 0-CNS-52, control of the switchyard is an SRO only task.

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 (SRO Only Task per licensee)

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 parallels EDG 1 to Bus 1F. Approximately an hour later the main turbine trips and there is a loss of all off-site power.

The Control Room Supervisor should implement procedure:

A. 5.3EMPWR, Emergency Power, because of the loss of bus 1F.

B. 5.3GRID, Degraded Grid Voltage, because of the loss of bus 1F.

C. 5.3EMPWR, Emergency Power, because of the loss of all Division 1 480V MCCs.

D. 5.3GRID, Degraded Grid Voltage, because of the loss of all Division 1 480V MCCs.

Proposed Answer: B - The loss of Bus 1F is an entry condition for procedure 5.3GRID. A loss of 480V MCCs is not an entry condition for either of the referenced procedures and the loss of bus 1F is not an entry condition for 5.3EMPWR.

Technical References: 5.3EMPWR, Emergency Power

5.3GRID, Degraded Grid Voltage

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 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 (one of the operable SRMs) adjacent to the fuel movement has been lowering and is now reading 2 cps.

    In this situation, fuel movements:

    A. can proceed.

    B. must be suspended until the adjacent SRM reads ≥ 3 cps.

    C. must be suspended until a special movable detector is installed.

    D. must be suspended until at least one of the three inoperable SRMs is restored to an operable status.

Proposed Answer: A - The SRM is not required to have ≥ 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 not installed.
- Recirc Pump B is running at minimum flow.
- RWCU is not aligned for alternate heat removal.
- REC inlet temperature is 50 degrees.
- The reactor has been shutdown for 40 days.

By using the attached Procedure NEDC 00-0105, “Fuel Pool Decay Heat Loads/Alternate Decay Heat Removal Time Limitations”, it can be determined that with the current lineup, fuel pool temperature will be maintained between:

A. 135 and 144 degrees.

B. 145 and 154 degrees.

C. 155 and 164 degrees.

D. 165 and 175 degrees.

Proposed Answer: C - 155 and 164 degrees. 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˚.

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]** and **Procedure 2.1.20.2**

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	202001 A2.09	
	Importance Rating		3.4

Proposed Question:

18. The unit is operating at 90% of rated power with Reactor Recirc MG Set A scoop tube in a lockout condition. While getting ready to repair a faulty limit switch on RRMG A, Reactor Feedwater Pump A trips.

    The Control Room Supervisor should now enter procedure:

    A. 2.4RR, Reactor Recirc Abnormal, because of the inability to control reactor recirc flow.

    B. 2.4RXPWR, Reactor Power Anomalies, because of the inability to control reactor recirc flow.

    C. 2.4RXLVL, Reactor Water Level Control Trouble, because of the loss of a feedpump.

    D. 2.4RXPWR, Reactor Power Anomalies, because reactor power did not runback following the RFP pump trip.

Proposed Answer: (A) The inability to control RR flow is an entry condition to the referenced procedure. There is no condition in this scenario that meets the entry conditions for procedure 2.4RXPWR or 2.4RXLVL.

Technical References:
Procedure 2.4RR, "Reactor Recirculation Abnormal," Revision 16
Procedure 2.4RXPWR, Reactor Power Anomalies, Rev 0
2.4RXLVL, Reactor Water Level Control Trouble, Rev 13
Reactor Recirculation / COR002-22-02, Revision 20

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

Learning Objective: COR002-22-02 Licensed Operator Enabling Objectives 10(o) & 13(f)

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 predict the impacts of recirculation scoop tube lockup on the recirculation system 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		3
	Group Number		1
	K / A Number	2.1.14	
	Importance Rating		3.3

Proposed Question:

19. A human performance error resulted in an inadvertent start of one of the emergency diesel generators. After a review of the event, the Shift Manager has determined that an Event Response Team (ERT) is required per procedure 2.0.6, Operational Event Response and Review.

In accordance with this procedure, once this determination is made the Shift Manager is responsible for notifying the:

A. Vice President.

B. Plant Manager.

C. Operations Manager.

D. Chairman of the Station Operational Review Committee (SORC).

Proposed Answer: C - According to this procedure the Shift Manager is required to notify the Operations Manager.

Technical References: Procedure 2.0.6, Operational Event Response and Review, step 3.10.2.

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: 55.43.10 (SRO only task per the licensee)

Comments:

(KA) Knowledge of system status criteria which require notification of plant personnel.

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. A maintenance task is scheduled 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. Fuel movements during a refueling outage are controlled by the:

    A. Fuel Movement Sequence Checklist.

    B. Fuel Accountability and Tracking Form.

    C. Special Nuclear Materials Transfer Form.

    D. Special Nuclear Materials Sequence Checklist.

Proposed Answer: C; This answer is correct because procedure 10.21, Special Nuclear Materials Control and Accountability Instructions requires a SNM Transfer Form be used for fuel movement during a refueling outage.

Technical References: Procedure 10.21, Special Nuclear Materials Control and Accountability Instructions, Revision 24.

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.

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. The plant is operating at full power with a fuel shuffle in progress in the Spent Fuel Pool (SFP). An unanticipated movement of the SFP bridge while lifting a spent fuel assembly out of the fuel rack results in several failed fuel pins and a release of highly radioactive gases. While none of the radioactive gas escaped the Reactor Building, the Radiation Protection supervisor estimates the fuel movement crew may have received intakes 7 to 9 times the annual limit. This is a one hour reportable event.

As a minimum, the on-shift SRO (or designee) must notify the NRC:

A. Operations Center.

B. Operations Center AND the Region IV Duty Officer.

C. Operations Center AND the NRC Resident (or Senior Resident).

D. Resident (or Senior Resident) OR the NRC Region IV Duty Officer.

Proposed Answer: (C) According to procedure 2.0.5, Reports to NRC Operations Center, the SRO is responsible for notifying the NRCOC and Resident within 1 hour..

Technical References: Procedure 2.0.5, Reports to NRC Operations Center

Proposed References to be provided to applicants during examination:

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 (A), is not correct because the NRC resident is also required to be notified per the referenced procedure.

Answer (B) is incorrect because there is no requirement to notify the region.

Answer (D) is incorrect because the NRCOC must be notified.

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 (ED) can authorize emergency radiation exposures.

For a qualified radiation worker that volunteers for the emergency task, the ED’s authorization is:

A. limited to 5 rem when the worker is a declared pregnant worker.

B. limited to 10 rem when the worker is providing first aid to injured personnel with non-life-threatening injuries.

C. not limited when the worker is providing rescue and/or treatment of personnel with life-threatening injuries.

D. limited to 25 rem when the worker is performing corrective actions to protect large populations from extensive radiological exposure.

Proposed Answer: Answer C is correct because it restates the conditions in EPIP 5.7.12, Attachment 1. In this situation there is no specific limit for life-saving activities involving voluntary workers.

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 the ED can authorize up to 25 rem for this situation 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.

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 %
- The SRO has determined torus pressure and water level cannot be maintained below the Pressure Suppression Pressure (PSP) limit.
- 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 to ensure adequate core cooling.

B. shutdown the reactor to reduce the energy input into containment.

C. emergency depressurize the RPV to maintain containment integrity.

D. restore Division 2 ECCS power to return Division 2 ECCS equipment to service.

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. All sectors.

B. The new centerline sector plus one sector to either side.

C. The original three sectors plus two additional sectors to both sides.

D. The original three sectors plus the two sectors in the clockwise direction only.

Proposed Answer: D ; 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 B (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 A (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.