Examination Outline Cross-Reference	**Level**	RO
295001 Partial or Complete Loss of Forced Core Flow Circulation / 1 & 4 Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION: (CFR: 41.5) **AK3.01 Reactor water level response**	**Tier#**	1
	Group#	1
	K/A #	295001 AK3.01
	Rating	3.4
	Revision	0
Revision Statement:		

Question 1

The plant is operating at rated power when Reactor Recirculation Pump B trips.

Which one of the following completes the statement below regarding the INITIAL Narrow Range RPV Water Level response?

Narrow Range RPV Water level ____(1)____ due to the ____(2)____ of steam voids.

A. (1) rises
(2) increase

B. (1) rises
(2) reduction

C. (1) lowers
(2) increase

D. (1) lowers
(2) reduction

Answer: A. (1) rises (2) increase
Explanation: Indicated RPV water level measures level in the downcomer area vs. actual level in the fuel area. Following a RR pump trip, indicated RPV water level rises due to (1) less water leaving the downcomer area to the RR pump and (2) the reduction of flow to the fuel causing void fraction to rise.
Distracters:

Answer B is plausible to the unprepared applicant who remembers indicated level rises but does not understand why. The applicant may remember the principle of level instrumentation dP cell operation and incorrectly attribute the rise in level to an increase in head in the variable leg due to reduction of voids and replacement by liquid. This is wrong because there are no voids in the downcomer region where narrow range variable taps are located. Voids increase in the core region due to the reduction of core flow. That displaces liquid to the downcomer region, which results in increased downcomer level.

Answer C is plausible to the unprepared applicant who does not remember the level response but may simply remember that a reduction in recirc drive flow causes power to reduce by void increase. This applicant may believe narrow range level instrumentation reflects water level in the core region or that voids increase in the downcomer region. Either of those false assumptions would lead him to conclude that indicated level would lower if void fraction rises. This is wrong because there are no voids in the downcomer region, and that is where narrow range variable taps are located. Voids increase in the core region due to the reduction of core flow. That displaces liquid to the downcomer region, which results in increased downcomer level.

Answer D is plausible to the unprepared applicant who remembers neither the indicated level response nor the effect of flow reduction on void fraction, but associates reduced effects with reduced recirc flow. The applicant may believe narrow range level reflects water level in the core and conclude water level lowers. The applicant may remember that reactor power decreases but may be confused about cause and effects of concepts such as void fraction, power, and boiling boundary and incorrectly conclude that voids reduce because boiling reduced because power reduced. This is wrong because there are no voids in the downcomer region, and that is where narrow range variable taps are located. Voids increase in the core region due to the reduction of core flow. That displaces liquid to the downcomer region, which results in increased downcomer level.

Technical References: USAR Section XIV-5.5

References to be provided to applicants during exam: None

Learning Objective: COR002-22-02 Obj. LO-06h

Question Source:	Bank #	Audit Q#1 for NRC 12/2015
(note changes; attach parent)	Modified Bank #	
	New	
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b)(2),(5)	
Level of Difficulty:	2	

Examination Outline Cross-Reference	Level	RO
295003 Partial or Complete Loss of AC / 6 Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER: (CFR: 41.7) **AA1.02 Emergency generators**	**Tier#**	1
	Group#	1
	K/A #	295003 AA1.02
	Rating	4.2
	Revision	0
Revision Statement:		

Question 2

A Loss of Offsite Power and LOCA occurred 1 hour ago.

DG1 indicates:

Regarding operation and monitoring of DG1 under these conditions …

(1) Which one of the following describes DG1 operation reflected above relative to applicable load limits?

AND

(2) Which one of the following procedures contains the Equipment Load Guideline, which lists individual load values for plant equipment?

A. (1) Allowed for up to two hours in a 24 hour period
(2) 2.2.20 [Standby AC Power System (Diesel Generator)]

B. (1) Allowed for up to two hours in a 24 hour period
(2) 5.3EMPWR [Emergency Power During Modes 1, 2, or 3]

C. (1) NOT allowed, load must be reduced immediately
   (2) 2.2.20 [Standby AC Power System (Diesel Generator)]

D. (1) NOT allowed, load must be reduced immediately
   (2) 5.3EMPWR [Emergency Power During Modes 1, 2, or 3]

Answer: B. (1) Allowed for up to two hours in a 24 hour period.
(2) 5.3EMPWR [Emergency Power During Modes 1, 2, or 3]

Explanation:
Emergency DGs 1 and 2 automatically start and re-energize their respective buses during a loss of offsite power. Procedure 2.2.20.1 [Diesel Generator Operations] section 4 addresses automatic DG initiation and local monitoring of DG parameters but refers the operator to 5.3EMPWR [Emergency Power During Modes 1, 2, or 3] to monitor DG load. 5.3EMPWR Att. 3 step 1.2.2 lists the maximum load limit for DG1/2 as 4000 kW, but it also states the DG may be overloaded up to 4400kW and 763 amps for a maximum of 2 hours in a 24 hour period. 5.3EMPWR Att 7 information sheet states the load on 4160 VAC bus 1F, the bus supplied by DG1, may actually exceed the DG1 steady state load rating of 4000 KW for DBA LOCA conditions. It further states this loading will drop below 4000 KW within the first 2 hours post-accident.

DG1 loading depicted in the stem is 4200 kW and 730 amps. This is higher than the maximum load limit but below the overload limit. Therefore, operation at these conditions is allowed for up to 2 hours in a 24 hour period.

Various loads that were load-shedded do not automatically sequence back on. Manual start of loads may be desired and is allowed, provided DG load limits are not exceeded. A list of individual load values for various AC powered equipment is provided as 5.3EMPWR Attachment 6 [Equipment Load Guideline] for the operator to reference to ensure DG load limits will not be exceeded.

Distracters:
Answer A part 1 is correct. Part 2 is plausible because it is the SOP common to both Div 1 and Div 2 DGs. SOPs contain instructions applicable to some aspects of DG operation following automatic initiation, particularly local monitoring, but 5.3EMPWR prescribes monitoring and controls for DG loading. This answer is wrong because individual equipment load values are found in 5.3EMPWR Att. 6, not in any of the DG SOPs.

Answer C part 1 is plausible because the maximum load rating of DG1 is 4000kW and because operation above the green bands is depicted on the photos of meters in the stem. It is wrong because 5.3EMPWR Att. 3 step 1.2.2 also states the DG may be overloaded up to 4400kW and 763 amps for a maximum of 2 hours in a 24 hour period. 5.3EMPWR also cites a DG loading study documented in USAR Table VIII-5-

1, which shows a maximum DG1 loading above 4000kW is expected for a short period, less than 2 hours, under some conditions. Part 2 is plausible and wrong for the same reasons as stated for distractor A.

Answer D part 1 is plausible and wrong for the same reasons as stated for distractor A. Part 2 is correct.

Technical References: Procedures 5.3EMPWR [Emergency Power During Modes 1, 2, or 3](Rev 65), 2.2.20.1 [Diesel Generator Operations](Rev 69), 2.2.20 [Standby AC Power System (Diesel Generator)](Rev 94)

References to be provided to applicants during exam: none

Learning Objective: INT032-01-31 EO-W, COR002-08-02 LO-10c, INT032-01-05 LO-03,05

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b)(8),(10)	
Level of Difficulty:	2	
SRO Only Justification:	N/A	
PSA applicability:		

Examination Outline Cross-Reference	Level	RO
295004 Partial or Total Loss of DC Pwr / 6 Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER: (CFR: 41.10) **AA2.01 Cause of partial or complete loss of D.C. power**	**Tier#**	1
	Group#	1
	K/A #	295003 AA2.01
	Rating	3.2
	Revision	0
Revision Statement:		

Question 3

A LOCA has occurred.

The Control Room red AND green indicating lights for the following switches are extinguished:

- RHR Pump B
- Core Spray Pump A
- DIESEL GEN 1 BKR 1FE

Which one of the following conditions would cause these indications?

A. Panel AA3 has a blown fuse.

B. Panel BB3 has a blown fuse.

C. Diesel Generator 1 has tripped.

D. 4160 VAC Bus 1F is de-energized.

Answer: A. Panel AA3 has a blown fuse.
Explanation: The breakers associated with the listed components are all on 4160 VAC Bus 1F. 125 VDC Panel AA3 supplies the breaker control circuits for Bus 1F. The Control Room control switch indicating lights are powered from the breaker control circuits.

Distracters: Answer B is plausible because RHR Pump B, which is in RHR Loop B, is listed. Other components in RHR Loop B are Division 2 powered It is wrong because RHR Pump B is powered from Division 1 Bus 1F, so its breaker control circuit is powered by 125 VDC panel AA3. Answer C is plausible because all of the listed breakers are on 4160VAC Bus 1F. It is wrong because the breaker position indicating lights are powered from their 125 VDC control circuit, in this case supplied by Panel AA3. Answer D is plausible because a LOCA has occurred, which would auto start DG1, the supply breaker from DG1 to bus 1F is listed as an affected component, and DG1 is the emergency power source for the other two listed components. It is wrong because breaker 1FE position indicating lights are powered from their 125 VDC control circuit, in this case supplied by Panel AA3.
Technical References: 2.2A_125DC.DIV1 [125 VDC Power Checklist (DIV 1)](Rev 7), Procedure 2.2A_125DC.DIV2 [125 VDC Power Checklist (DIV 2)](Rev 7), Procedure 2.2A_4160.DIV1 [4160 VAC Auxiliary Power Checklist (DIV 1)](Rev 1)
References to be provided to applicants during exam: none
Learning Objective: COR002-07-02 Obj. LO-06b, 08b

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b)(7)	
Level of Difficulty:	2	
SRO Only Justification:		
N/A		
PSA Applicability		
Top 10 Risk Significant System – Emergency DC Power Top 10 Risk Sensitive Components – 125 VDC Panel AA3		

Examination Outline Cross-Reference	Level	RO
295005 Main Turbine Generator Trip / 3 **2.1.23** Ability to perform specific system and integrated plant procedures during all modes of plant operation. (CFR: 41.10)	**Tier#**	1
	Group#	1
	K/A #	295005 G2.1.23
	Rating	4.3
	Revision	0
Revision Statement:		

Question 4

The plant is at 23% power.

This alarm is received:

TG THRUST BEARING PRESS PRE-TRIP	PANEL/WINDOW: B-1/B-5

If the Thrust Bearing continues to degrade,...

(1) Which one of the following is the thrust bearing oil pressure at which a MANUAL trip of the Main Turbine is required IAW Procedure 2.4TURB [Main Turbine Abnormal]?

AND

(2) What TURBINE TRIP ACTION is required to be verified IAW Procedure 2.2.77 [Turbine Generator]?

A. (1) 35 psig
(2) 345KV Bus PCB-3310 and 3312 open

B. (1) 35 psig
(2) Main Turbine Bypass Valves throttle open

C. (1) 70 psig
(2) 345KV Bus PCB-3310 and 3312 open

D. (1) 70 psig
(2) Main Turbine Bypass Valves throttle open

Answer: C. (1) 70 psig (2) 345KV Bus PCB-3310 and 3312 open
Explanation:

The thrust bearing trip device will give a warning should the thrust bearing shoes wear down to a predetermined amount. Further wear will shut down the turbine before serious damage can occur to other turbine parts. The device consists of two nozzles screwed into the control block. Each nozzle discharges high pressure oil against one face of the collar machined on coupling spacer. The oil supply is limited by orifices. Clearances between the thrust collar and nozzles are normally constant and hence the pressure is constant. However, if thrust bearing collar should move closer to either the nozzle face due to wear, the oil flow from that nozzle will be restricted and pressure to that nozzle, through its orifice plug, will rise. When pressure rises to 35 psig, the pressure switch will sound the pre-trip alarm. Should pressure rise to setpoint of 77 psig, the turbine trip will actuate via the Trip TriCon Unit. The scram action at 70 psig thrust bearing oil pressure is based on adequate margin to the automatic trip and is consistent with System Engineering guidance. Per Operations policy, it is preferable to initiate manual action rather than allowing the automatic action to occur. Therefore, thrust bearing wear causes thrust bearing trip oil pressure to rise, and a manual turbine trip will occur at 70 psig, before the automatic trip at 77 psig. Turbine trip verification actions are contained in Procedure 2.2.77 Attachment 4, and include verifying 345KV Bus breakers PCB-3310 and 3312 open.

Distracters:
Answer A part 1 is plausible because 35 psig is the setpoint for the thrust bearing pre-trip alarm. The unprepared applicant who does not understand how the turbine thrust bearing wear detector operates and does not remember the turbine trip criteria (≥70 psig) might select this answer. The unprepared applicant may believe abnormal thrust bearing wear would produce a lowering oil pressure and incorrectly deduce 35 psig would be encountered, not 70 psig. This answer is wrong since increased thrust bearing wear produces a rising backpressure to the trip device. Manual turbine trip criteria requires tripping the Main Turbine when thrust bearing oil trip pressure rises to ≥70 psig is set per engineering guidance to provide margin to the auto trip at 77 psig. Part 2 is correct.

Answer B part 1 is plausible and wrong for the same reasons as given for distractor A. Part 2 is plausible because Bypass Valves open upon a Turbine trip to maintain steam pressure at the DEH pressure setpoint by diverting steam flow to the condenser. It is also plausible because turbine steam admission valves' response is required to be verified per Procedure 2.2.77 Attachment 4. It is wrong because verifying BPV operation is NOT a required step in Procedure 2.2.77 Attachment 4.

Answer D part 1 is correct.. Part 2 is plausible and wrong for the same reasons stated for distractor B.

Technical References: Alarm Card B-1/B-5, Alarm Card B-1/A-5 (Rev 36), Procedure 2.4TURB [Main Turbine Abnormal](Rev 31), Procedure 2.2.77 [Turbine Generator](Rev 114)

References to be provided to applicants during exam: none		
Learning Objective: INT032-01-27 EO-O,P		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b)(10),(4)	
Level of Difficulty:	3	
SRO Only Justification:	N/A	
PSA applicability:		
N/A		

Examination Outline Cross-Reference	Level	RO
295006 SCRAM / 1 Knowledge of the operational implications of the following concepts as they apply to SCRAM: (CFR: 41.8 to 41.10) **AK1.03 Reactivity control**	**Tier#**	1
	Group#	1
	K/A #	295006 AK1.03
	Rating	3.7
	Revision	0
Revision Statement:		

Question 5

What completes the statement below regarding the reactivity control design feature that ensures ample time for a scram sequence to complete?

The Reactor Protection System (RPS) scram reset MINIMUM time delay is ____(1)___ seconds and will delay reset of __(2)___.

A. (1) 10
(2) automatic scrams, only

B. (1) 10
(2) manual and automatic scrams

C. (1) 37.5
(2) automatic scrams, only

D. (1) 37.5
(2) manual and automatic scrams

Answer: B. (1) 10 (2) manual and automatic scrams
Explanation: RPS time delay relays K22A(B) enforce a 10 second (minimum) time delay for resetting RPS A(B) following a scram. This is designed to allow ample time for all control rods to fully insert from fully withdrawn position. Relay coils K21A(B), in parallel with backup scram solenoids, energize upon the respective RPS trip system tripping (logic de-energizing). K21A(B) energizing opens a contact in the scram reset logic that is in parallel with a normally open K22A(B) contact. K21A(B) energizing also closes a contact to energize K22A(B), which is 10 sec time delay to pick up relay. Once K22A(B) picks up, the reset circuit for the respective RPS trip system is enabled, which allows the scram to be reset using the scram reset switch on panel 9-5. The 10 second time delay K22 relays affect both manual and automatic scram

reset circuits. Time delay to close K22 contacts must close to enable re-energizing auto scram K14 relays and manual scram K15 relays.
Distracters: Answer A part 1 is correct. Part 2 is plausible because the unprepared applicant may reason that since the reset delay is an automatic function, it applies only to automatic scrams. The unprepared applicant might not know the purpose of the time delay is to allow time for rod insertion, or he may believe the time delay would not be needed for a manual scram, since that is a deliberate act. This is wrong because the RPS circuitry is arranged such that any scram signal, whether automatic or manual, energizes the TD relay in the respective RPS trip system, which enforces the 10 second time delay for reset. Answer C part 1 is plausible because it reflects the ARI reset time delay. The unprepared applicant may confuse ARI and RPS reset time delays, since they share a common purpose, to allow ample time for control rod insertion. This answer is wrong because RPS and ARI systems are distinct, with separate circuitry and controls. The RPS reset time delay relays are set for 10 seconds. Part 2 is plausible and wrong for the same reasons as given for distractor A. Answer D part 1 is plausible and wrong for the same reasons as given for distractor C. Part 2 is correct.
Technical References: GE drawings 791E256 Sheets 11, 12, 14
References to be provided to applicants during exam: none
Learning Objective: COR002-21-02 Obj. LO-04f

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b)(6),(7)	
Level of Difficulty:	3	
SRO Only Justification:	N/A	
PSA applicability:		
Top 10 Risk Significant System – Reactor Protection System		

Examination Outline Cross-Reference	Level	RO
295016 Control Room Abandonment / 7 Knowledge of the interrelations between CONTROL ROOM ABANDONMENT and the following: (CFR: 41.7) **AK2.01 Remote shutdown panel: Plant-Specific**	**Tier#**	1
	Group#	1
	K/A #	295016 AK2.01
	Rating	4.4
	Revision	0
Revision Statement:		

Question 6

The control room has been abandoned due to toxic gas.

What component is controlled by the Alternate Shutdown Panel (ASD) Operator using control switches in the ASD room?

A. SRV 71C

B. Core Spray Pump B

C. RCIC-MO-131 [STM SUPP TO TURB VLV]

D. RHR-MO-13D [PUMP D TORUS SUCT VLV]

Answer: D. RHR-MO-13D [PUMP D TORUS SUCT VLV]

Explanation:
Three Alternate Shutdown Panels are located in the ASD room, HPCI, RHR, and REC/ADS panels. The designated ASD operator is assigned to operate systems by manipulating control switches on the ASD panels and by coordinating operation of components controlled using the local auxiliary shutdown panels (LASPs) throughout the plant. Of the answers provided, only RHR-MO-13D has a control switch located on an ASD panel in the ASD room.

Distracters:
Answer A is plausible because three SRVs can be controlled from the REC/ASD panel. It is wrong because SRV 71C cannot be controlled from the ASD panel, only SRVs 71E, F, and G.

Answer B is plausible because controls for other pumps, HPCI and REC Pumps C and D, are on the ASD panel and because voltage indication located above CS Pump B breaker is used to determine bus availability in procedure 5.1ASD [Alternate

Shutdown]. It is wrong because controls for CS Pump B do not exist on the ASD panels. Answer C is plausible because controls for other MOVs are on ASD panels and because RCIC-MO-131 is operated per procedure 5.1ASD [Alternate Shutdown]. It is wrong because controls for RCIC-MO-131 do not exist on the ASD panels. Instead, it is operated locally by the Reactor Building Operator at the RCIC 125 VDC starter rack in the reactor building.		
Technical References: Procedure 5.1ASD [Alternate Shutdown](Rev 17), TS bases Table B 3.3.3.2-1		
References to be provided to applicants during exam: none		
Learning Objective: COR002-34-02 Obj. LO-02h		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b)(7)	
Level of Difficulty:	2	
SRO Only Justification:	N/A	
PSA applicability:		
Top 10 Risk Significant System - RHR		

Examination Outline Cross-Reference	Level	RO
295018 Partial or Total Loss of CCW / 8 Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER: (CFR: 41.5) **AK3.07 Cross-connecting with backup systems**	**Tier#**	1
	Group#	1
	K/A #	295018 AK3.07
	Rating	3.1
	Revision	0
Revision Statement:		

Question 7

Which one of the following completes the statement below regarding initiating Service Water backup to Reactor Equipment Cooling (REC) system IAW Procedure 5.2REC [Loss of REC] when critical loop cooling is required and REC pumps are unable to supply cooling?

Service Water backup to REC must be established within a maximum of __(1)__ from loss of REC in order to provide cooling to ______(2)______.

A. (1) 30 minutes
(2) Drywell FCUs

B. (1) 30 minutes
(2) ECCS Quad FCUs

C. (1) 1 hour
(2) Drywell FCUs

D. (1) 1 hour
(2) ECCS Quad FCUs

Answer: D. (1) 1 hour (2) ECCS Quad FCUs
Explanation: Procedure 5.2REC [Loss of REC] states SW backup cooling to the critical loops is required to be restored within 1 hour to ensure postulated temperature limits for ECCS Pump areas are maintained. The critical loops supply cooling to ECCS Quad FCUs, including RHR. They do not supply DW FCUs. Procedure 5.2REC [Loss of REC] provides instructions for aligning SW backup to REC, which involves isolating the non-critical header from the SW supplied critical header.
Distracters: Answer A part 1 is plausible because other operator time critical actions associated with cooling are required to be performed within 30 minutes, like opening control room panel and switchgear room doors during a station blackout and aligning suppression

pool cooling during an ATWS. It is wrong because procedure 5.2REC states the time limit for establishing SW backup cooling to REC is 1 hour, and the stem asks for the maximum time. Part 2 is plausible because REC supplies Drywell FCUs. This answer is wrong because SW backup to REC does not supply the DW FCUs, since they are on the non-critical REC supply header. The non-critical header is manually isolated from the critical header and the SW crosstie valves are opened from VBD-M.

Answer B part 1 is plausible and wrong for the same reasons as stated for distractor A. Part 2 is correct.

Answer C part 1 is correct. Part 2 is plausible and wrong for the same reasons as stated for distractor A.

Technical References: Procedure 5.2REC [Loss of REC](Rev 17), USAR section X-6.5.3, procedure 2.0.1.3 [Time Critical Operator Action Control and Maintenance](Rev 6)

References to be provided to applicants during exam: none

Learning Objective: INT032-01-26 EO-L, INT032-01-05 LO-3,5

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b)(4),(8),(10)	
Level of Difficulty:	2	
SRO Only Justification:	N/A	

PSA applicability:

Top 10 Risk Significant System – Service Water System

Examination Outline Cross-Reference	Level	RO
295019 Partial or Total Loss of Inst. Air / 8 Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR: (CFR: 41.7) **AA1.03 Instrument air compressor power supplies**	**Tier#**	1
	Group#	1
	K/A #	295019 AA1.03
	Rating	3.0
	Revision	0
Revision Statement:		

Question 8

Station Air Compressors (SACs) are aligned as follows:

	Control Switch (Panel A)	Status
SAC A	AUTO	Running
SAC B	OFF	Stopped
SAC C	OFF	Stopped

A Loss of Offsite Power occurs.

Which one of the following describes how a SAC can be started under these conditions?

A. Place SAC C control switch to AUTO.

B. Place SAC B control switch to AUTO.

C. Place SAC A control switch to OFF and then to AUTO.

D. Locally depress the silver TRIP button on SAC C breaker, then start it locally.

Answer: B. Place SAC B control switch to AUTO.
Explanation: Loss of offsite power results in undervoltage trips of SAC 480 VAC supply breakers on buses F, G, and B for SACs A, B, and C, respectively. DG1 and DG2 will re-energize buses F and G. Since SAC B was not running and its control switch was in OFF before the loss of power and its control switch is in AUTO, it can be restarted by placing its control switch to ON, without having to reset its breaker locally.

Distracters:
Answer A is plausible because if a SAC was not running with its panel A control switch in OFF before a loss of power, it can be restarted after power is restored by placing its control switch to AUTO. This answer is wrong because power is not restored to 480 VAC bus B, which supplies SAC C, during a loss of offsite power. DG1 and DG2 only restore power to 4160 VAC buses F and G and their respective 480VAC buses. Therefore, SAC C will not start since it is still de-energized.

Answer C is plausible because some loads have anti-pumping devices that are reset by placing their control switches to Stop/Off in order to remove a start signal. It is wrong in this case, because an undervoltage trip of a SAC that was operating before the trip must be reset by depressing the TRIP button on the front of its breaker, locally.

Answer D is plausible because an undervoltage trip of a SAC that was operating before the trip is reset by depressing the TRIP button on the front of its breaker, and then it can be started locally. This answer is wrong because power is not restored to 480 VAC bus B, which supplies SAC C, during a loss of offsite power. DG1 and DG2 only restore power to 4160 VAC buses F and G and their subordinate 480VAC buses. Therefore, SAC C will not start since it is still de-energized.

Technical References: Procedure 5.3EMPWR [Emergency Power During Modes 1, 2, OR 3](Rev 65), Procedure 2.2.59 [Plant Air System](Rev 74)

References to be provided to applicants during exam: none

Learning Objective: COR001-17-01 Obj. LO-06e

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b)(4)	
Level of Difficulty:	3	
SRO Only Justification:	N/A	

PSA applicability:

Examination Outline Cross-Reference	Level	RO
295021 Loss of Shutdown Cooling / 4 Knowledge of the reasons for the following responses as they apply to LOSS OF SHUTDOWN COOLING: (CFR: 41.5) **AK3.05 Establishing alternate heat removal flow paths**	**Tier#**	1
	Group#	1
	K/A #	295021 AK3.05
	Rating	3.6
	Revision	0
Revision Statement:		

Question 9

The plant is in Mode 3.

Reactor Recirc pump B is operating.

RHR pump A is operating in Shutdown Cooling Mode.

IAW Procedure 2.4SDC [Loss of Shutdown Cooling]…

Which of the following events will require placing RWCU in Alternate Heat Removal mode per Procedure 2.2.66 [Reactor Water Cleanup]?

A. Lockout of 4160 VAC bus 1F

B. Trip of Reactor Recirc pump B

C. RHR A Heat Exchanger tube leak

D. Sealed-in PCIS Drywell High Pressure trip signal

Answer: D. Sealed-in PCIS Drywell High Pressure trip signal
Explanation: Drywell Pressure High results in a Group 2 isolation to SDC suction valves MO-17&18, and return valves MO-25A&B, common to and preventing operation of both SDC subsystems. Procedure 2.4SDC only directs operation of RWCU in Alternate Heat Removal IAW Procedure 2.2.66 for a complete loss of SDC per 2.4SDC Att. 2. This answer is the only one that would result in a loss of both SDC subsystems. The reason for establishing RWCU alternate heat removal is inferred from diagnosing a complete loss of SDC.

Distracters:
Answer A is plausible because it would result in loss of the operating SDC pump and pumps in both SDC subsystems, pump A and pump B. It is wrong because RHR pump C in subsystem A and pump D in subsystem B could be aligned for SDC IAW 2.4SDC section 4.7 or 4.8. Therefore, entry into 2.4SDC Att. 2 would not be directed.

Answer B is plausible to the unprepared applicant who confuses core circulation requirements. With loss of the operating SDC system, TS 3.4.7 requires ensuring core circulation by another method, typically via one recirculation pump in operation. Conversely, the unprepared applicant may infer RWCU operation is required for core circulation if Recirc pumps are lost. It is not unusual for an applicant to confuse SDC requirements with respect to heat removal and circulation. This answer is wrong because only a total loss of SDC requires operation of RWCU for decay heat removal, not necessarily core circulation.

Answer C is plausible because a tube leak in the operating SDC loop is required to be isolated IAW 2.4SDC Att. 1. The heat exchanger is directed to be bypassed, thus removing the heat sink for RHR and defeating its heat removal capacity. It is wrong because SDC subsystem B would still be available and would be started IAW 2.4 section 4.8. Therefore, entry into 2.4SDC Att. 2 would not be directed.

Technical References: 2.4 SDC [Loss of Shutdown Cooling](Rev 15)

References to be provided to applicants during exam: none

Learning Objective: COR002-23-02 obj LO-08o

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b)(7),(9),(10)	
Level of Difficulty:	3	
SRO Only Justification:	N/A	

PSA applicability:
Top 10 Risk Significant System – Residual Heat Removal System

Examination Outline Cross-Reference	Level	RO
295023 Refueling Acc / 8 Ability to determine and/or interpret the following as they apply to REFUELING ACCIDENTS: (CFR: 41.10) **AA2.02 Fuel Pool Level**	**Tier#**	1
	Group#	1
	K/A #	295023 AA2.02
	Rating	3.4
	Revision	0
Revision Statement:		

Question 10

Irradiated fuel is being moved in the Spent Fuel Pool.

Two Fuel Pool Cooling pumps are operating.

A Refueling Accident causes Spent Fuel Pool (SFP) level to lower.

Which one of the following completes the statements below regarding EOP and TS limits being exceeded as SFP level lowers?

EOP-5A entry is FIRST required when ____(1)____.

LCO TS 3.7.6 [Spent Fuel Storage Pool Water level] is required to be entered as soon as SFP level **above fuel seated in the SFP storage racks** goes below _(2)___.

A. (1) alarm FUEL POOL LOW LEVEL on Panel 25-15 is received
(2) 21 feet, 6 inches

B. (1) alarm FUEL POOL LOW LEVEL on Panel 25-15 is received
(2) 37 feet, 6.5 inches

C. (1) Fuel Pool Cooling pumps trip due to low Skimmer Surge Tank level
(2) 21 feet, 6 inches

D. (1) Fuel Pool Cooling pumps trip due to low Skimmer Surge Tank level
(2) 37 feet, 6.5 inches

Answer: A. (1) alarm FUEL POOL LOW LEVEL on Panel 25-15 is received (2) 21 feet, 6 inches
Explanation:

EPG Rev 3 guidance for Spent Fuel Pool level and temperature are a very recent addition to CNS EOPs. EOP-5A entry is required when SFP level lowers 4 inches below normal. Normal level is 37 ft, 6-1/2 in. above the bottom. FUEL POOL LOW LEVEL alarm on Panel 25-15 setpoint is 37 ft, 2-1/2 inches, which corresponds to the EOP-5A entry condition of below 37 ft, 3 inches. TS 3.7.6 requires ≥ 21 feet, 6 inches of water above the top of irradiated fuel assemblies seated in the SFP storage racks.
Distracters: Answer B part 1 is correct. Part 2 is plausible because it reflects normal SFP level, measured from the bottom of the SFP. It is wrong because 21'6" is the limit of LCO 3.7.6, which is referenced to the top of the fuel racks in the SFP. Answer C part 1 is plausible because it represents loss of forced circulation, which is an entry condition for procedure 2.4FPC [Fuel Pool Cooling Trouble]. The unprepared applicant may not understand system operation and confuse which condition would occur first as level lowers or may confuse EOP and AOP entry conditions and select this answer. It is wrong because SFP adjustable weir plates are set at a SFP level of approximately 37 ft 6 inches (i.e. normal level), which is above the low level alarm. Overflow to the Skimmer Surge Tanks would cease when SFP level reached that point, and tank level would lower until all FPC pumps tripped. Level would be above the low level alarm setpoint, so the EOP-5A entry condition would not yet be met. Part 2 is plausible and wrong for the same reasons as given for distractor A. Answer D part 1 is plausible and wrong for the same reasons as given for distractor C. Part 2 is correct.
Technical References: EOP-5A [Secondary Containment Control] (Rev 16) and AMP-TBD00 [EOP/PSTG Technical Basis], Procedure 2.4FPC [Fuel Pool Cooling Trouble](Rev 34), Procedure 2.2.32 [Fuel Pool Cooling and Demineralizer System](Rev 96), TS 3.7.6 [Spent Fuel Cooling Pool Water Level], Alarm Card 25-15-1/A-3 (Rev 3)
References to be provided to applicants during exam: none
Learning Objective: INT008-06-17 EO-1

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b)(4),(10)	
Level of Difficulty:	3	

SRO Only Justification:	N/A
PSA applicability:	
N/A	

Examination Outline Cross-Reference	Level	RO
295024 High Drywell Pressure / 5 **2.2.44** Ability to interpret control room indications to verify status and operation of a system, and understand how operator actions and directives affect plant and system conditions. (CFR: 41.5)	**Tier#**	1
	Group#	1
	K/A #	295024 G2.2.44
	Rating	4.2
	Revision	0
Revision Statement:		

Question 11

The plant is at 60% power during startup.

The following alarm is received:

DRYWELL HIGH PRESSURE	PANEL/WINDOW: 9-5-2/F-3

Drywell pressure is rising 0.1 psig per minute.

Drywell temperature is 140°F, rising 0.5°F per minute.

(1) What action is required NOW with respect to Drywell Fan Coil Units (DW FCU)?

AND

(2) If these trends continue, when is a manual scram required IAW plant procedures?

A. (1) Ensure all available DW FCU control switches in RUN
   (2) ONLY after Drywell pressure exceeds 1.5 psig

B. (1) Ensure all available DW FCU control switches in RUN
   (2) When the crew determines Drywell pressure will reach 1.5 psig

C. (1) Place all available DW FCU control switches in OVERRIDE
   (2) ONLY after Drywell pressure exceeds 1.5 psig

D. (1) Place all available DW FCU control switches in OVERRIDE
   (2) When the crew determines Drywell pressure will reach 1.5 psig

Answer: B. (1) Ensure all available DW FCU control switches in RUN (2) When the crew determines Drywell pressure will reach 1.5 psig
Explanation:

For the conditions stated, no EOP entry condition is met. The setpoint for annunciator 9-5-2/F-3 is 0.6 psig. EOP-3A entry conditions related to the parameters stated are 1.84 psig DW pressure and 150°F DW temperature. DW FCUs automatically trip on DW Pressure High, 1.84 psig. DW FCU control switches are only allowed to be placed in OVERRIDE, defeating the LOCA signal interlocks, per EOP-3A step DW/T-3. 140°F is below the entry condition of 150°F. Since EOP-3A entry is not yet required, defeating this interlock is not allowed now. Procedure 2.4PC [Primary Containment Control] step 4.6 applies for current conditions and requires ensuring all available drywell FCU control switches are in RUN.
Alarm Card 9-5-2/F-3 step 1.3 and 2.4PC step 4.2 state if DW pressure cannot be maintained below 1.5 psig, then scram and enter procedure 2.1.5 [Reactor Scram]. Consistent with EOP terminology, “cannot be maintained” does not prohibit anticipatory action—depending upon plant conditions, the action may be taken as soon as it is determined that the limit will ultimately be exceeded. If the crew determines the trend in DW pressure will eventually reach 1.5 psig, the reactor should be placed in a safer, shutdown state at that time. While true that “cannot be maintained” does not require anticipatory action, this answer remains true and the distractor is false, since the distractor is phrased as the ONLY option.

Distracters:
Answer A part 1 is correct. Part 2 is plausible because the unprepared applicant may believe “cannot be maintained” means actually not maintained, due to the common usage of the verb. He may also remember only faster moving simulator scenarios where DW pressure has already exceeded 1.5 psig by the time DW pressure was reported. This answer is wrong because Alarm Card 9-5-2/F-3 step 1.3 and 2.4PC step 4.2 state if DW pressure cannot be maintained below 1.5 psig, then scram and enter procedure 2.1.5 [Reactor Scram]. Consistent with EOP terminology, “cannot be maintained” does not prohibit anticipatory action—depending upon plant conditions, the action may be taken as soon as it is determined that the limit will ultimately be exceeded. If the crew determines the trend in DW pressure will eventually reach 1.5 psig, the reactor should be placed in a safer, shutdown state at that time.

Answer C part 1 is plausible because DW FCU automatically stop due to high DW pressure and because EOP-3A directs that interlock to be overridden at elevated DW temperatures. It is wrong because the given alarm setpoint is 0.6 psig, below 1.84 psig at which DW FCUs trip. Also, EOP-3A entry conditions have not yet been met; therefore, defeating the LOCA interlock to DW FCUs is not permitted. Part 2 is plausible and wrong for the same reasons as stated for distractor A.

Answer D part 1 is plausible and wrong for the same reasons as stated for distractor C. Part 2 is correct.

Technical References: EOP-3A [Primary Containment Control](Rev 17) and AMP-TBD00 [EOP/PSTG Technical Basis], Procedure 2.4PC [Primary Containment Control](Rev 17), Alarm Card 9-5-2/F-3 (Rev 46)

References to be provided to applicants during exam: none		
Learning Objective: INT008-06-13 EO-1,4b		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	12/2015 NRC Q#11
	New	
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b)(4),(10)	
Level of Difficulty:	3	
SRO Only Justification:	N/A	
PSA applicability:		
N/A		

Examination Outline Cross-Reference:

Level	RO	~~SRO~~
Tier #	1	
Group #	1	
K/A #	295024.G2.4.31	
Importance Rating	4.2	

295024 High Drywell Pressure

2.4.31 Knowledge of annunciator alarms, indications, or response procedures.

Question: 11

While raising power from 75% to 90% power, the following conditions exist:

- Drywell temperature has risen from 135°F to 145°F in the past 10 minutes.
- Drywell pressure is 1.5 psig and rising 0.5 psig/min.

What is/are the next appropriate operator action(s)?

a. Per Alarm 9-5-2/F-3, HIGH DRYWELL PRESSURE, scram and enter Procedure 2.1.5, REACTOR SCRAM.

b. Per Alarm 9-5-2/F-3, HIGH DRYWELL PRESSURE, ensure all available drywell FCU control switches are in OVERRIDE.

c. Per Procedure 2.4MC-RF, CONDENSATE AND FEEDWATER ABNORMAL, trip and isolate both reactor feedwater pumps.

d. Per Procedure 2.1.10, STATION POWER CHANGES, perform rapid power reduction and reduce core flow to $40x10^6$ lbs/hr.

Answer:

a. Per Alarm 9-5-2/F-3, HIGH DRYWELL PRESSURE, scram and enter Procedure 2.1.5, REACTOR SCRAM.

Examination Outline Cross-Reference	Level	RO
295025 High Reactor Pressure / 3 Knowledge of the operational implications of the following concepts as they apply to HIGH REACTOR PRESSURE: (CFR: 41.8 to 41.10) **EK1.03 Safety/relief valve tailpipe temperature / pressure relationships**	**Tier#**	1
	Group#	1
	K/A #	295025 EK1.03
	Rating	3.6
	Revision	0
Revision Statement:		

Question 12

A DEH failure at rated power resulted in RPV pressure peaking at 1110 psig.

NOW, Reactor pressure is 500 psig, slowly falling.

SRV-71F tailpipe temperature is 340°F, stable.

SRV-71F indications on Panel 9-5 are:

What is the status of SRV-71F?

A. Failed open

B. Failed closed

C. Operating properly AND open

D. Operating properly AND closed

Answer: A. Failed open		
Explanation: The stem reflects a high RPV pressure condition that would result in automatic opening of 8 SRVs. The photo depicts SRV-71F control switch in AUTO, solenoid energized (red light ON), and pressure switch not picked up (amber light OFF). (The pressure switch may not be picked up at 500 psig reactor pressure due to the reduced static head, the pressure drop across the SRV, and the venturi effect of the steam flow in the SRV discharge piping.) However, Tailpipe temperature >300°F indicates the SRV is open. (Procedure 2.2.1 note at step 4.2.4 states Tailpipe temperatures for closed safety and relief valves is ~ 140°F to 158°F.) With the handswitch in AUTO, the lowest reset pressure for SRV-71F, in low-low set mode, is 875 psig, so it should be closed at RPV pressure 500 psig. Therefore, SRV-71F has failed open.		
Distracters: Answer B is plausible because the amber pressure switch light is not on. It is wrong because since the solenoid indicates energized and Tailpipe temperature indicates >300°F, the SRV is open. Answer C is plausible because indications are that the SRV open circuit is energized. It is wrong because with the control switch in AUTO, the SRV should have closed with RPV pressure below 875 psig; therefore, it is not operating properly. Answer D is plausible because the amber light is not on. It is wrong because the pressure switch may not be picked up at 500 psig reactor pressure due to the reduced static head, the pressure drop across the SRV, and the venturi effect of the steam flow in the SRV discharge piping, and since the solenoid indicates energized and Tailpipe temperature indicates >300°F, the SRV is open.		
Technical References: Procedure 2.2.1 [Nuclear Pressure Relief System](Rev 38), Alarm Card 9-3-1/C-1 (Rev 37), Operations Instruction #8 [Guideline for Successful Transient Mitigation](Rev 14), GE Drawing 791E253 sh 03		
References to be provided to applicants during exam: none		
Learning Objective: COR002-16-02 LO-12c, 06d, 06e		
Question Source:	Bank #	new bank #175
(note changes; attach parent)	Modified Bank #	

	New	
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b)(5),(7)	
Level of Difficulty:	3	
SRO Only Justification:	N/A	
PSA applicability:		
Top 10 Risk Significant System – ADS and Pressure Relief		

Examination Outline Cross-Reference	Level	RO
295026 Suppression Pool High Water Temp. / 5 Knowledge of the interrelations between SUPPRESSION POOL HIGH WATER TEMPERATURE and the following: (CFR: 41.7) **EK2.06 Suppression pool level**	**Tier#**	1
	Group#	1
	K/A #	295026 EK2.06
	Rating	3.5
	Revision	0
Revision Statement:		

Question 13

Reference Provided

The reactor has failed to scram.

Reactor pressure is 400 psig, stable.

Suppression Pool temperature is 220°F, stable.

Which Suppression Pool water level exceeds the Heat Capacity Temperature Limit (HCTL) under these conditions?

A. 11 feet

B. 12.5 feet

C. 14 feet

D. 15.5 feet

Answer: D. 15.5 feet
Explanation: High Torus water temperature is addressed by EOP-3A. Emergency Depressurization is required if the HCTL (Graph 7) of the suppression pool is exceeded. The unsafe zone of the HCTL is to the right of the curve corresponding to actual RPV pressure. At a RPV pressure of 400 psig and Suppression Pool Temperature of 220°F, of the levels listed in the answers, the only SP level value that is in the unsafe zone of the HCTL is 15.5 feet.

Distracters:
All distractors have plausibility because the shape of the HCTL curve for 400 psig. RPV pressure becomes more limiting at lower than normal SP levels, just as it becomes more limiting at higher than normal SP levels.

Answer A is plausible because Emergency Depressurization is required by EOP-3A solely based on extremely low SP level. The low SP level that requires ED is 9.6 feet and is the lower boundary of the safe zone on the HCTL graph. It is also plausible because the unprepared applicant may believe unsafe operation is depicted to the left of the HCTL curve for 400 psig. It is also plausible because an error in graphing the given conditions, for example plotting by the 600 psig RPV pressure line, could result in the applicant believing this answer represents operation in the unsafe zone of the HCTL cure. This answer is wrong because, for the conditions given, 11 feet SP level leaves approximately 2°F margin to HCTL; therefore, HCTL is not exceeded.

Answer B is It is also plausible because the unprepared applicant may believe unsafe operation is depicted to the left of the HCTL curve for 400 psig. It is also plausible because an error in graphing the given conditions, for example plotting by the 600 psig RPV pressure line, could result in the applicant believing this answer represents operation in the unsafe zone of the HCTL cure. This answer is wrong because, for the conditions given, 11 feet SP level leaves approximately 3°F margin to HCTL; therefore, HCTL is not exceeded.

Answer C is It is also plausible because the unprepared applicant may believe unsafe operation is depicted to the left of the HCTL curve for 400 psig. It is also plausible because an error in graphing the given conditions, for example plotting by the 600 psig RPV pressure line, could result in the applicant believing this answer represents operation in the unsafe zone of the HCTL cure. This answer is wrong because, for the conditions given, 11 feet SP level leaves approximately 2°F margin to HCTL; therefore, HCTL is not exceeded.

Technical References: EOP-3A [Primary Containment Control](Rev 17) and AMP-TBD00 [EOP/PSTG Technical Basis]s, EOPSAG Graph 7 (HCTL)(Rev 16),

References to be provided to applicants during exam: EOPSAG Graph 7 (HCTL)

Learning Objective: INT008-06-18 EO-3

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X

Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X

10CFR Part 55 Content:	55.41(b)(9),(10)

Level of Difficulty:	3
SRO Only Justification:	N/A
PSA applicability:	
N/A	

Examination Outline Cross-Reference	Level	RO
295028 High Drywell Temperature / 5 Ability to operate and/or monitor the following as they apply to HIGH DRYWELL TEMPERATURE: (CFR: 41.7) **EA1.01 Drywell spray: Mark-I&II**	**Tier#**	1
	Group#	1
	K/A #	295028 EA1.01
	Rating	3.8
	Revision	0
Revision Statement:		

Question 14

Drywell spray has been placed in service during a LOCA due to high Drywell temperature.

- Drywell temperature and pressure are lowering.

When is Drywell spray required to be stopped IAW EOP 3A (Primary Containment Control)?

Drywell spray is REQUIRED to be stopped before…

A. drywell pressure lowers to zero psig.

B. suppression chamber to drywell vacuum breakers open.

C. reactor building to suppression chamber vacuum breakers open.

D. drywell temperature and pressure lower to the UNSAFE (Red) region of the Drywell Spray Initiation Limit (DWSIL) curve.

Answer: A. drywell pressure lowers to zero psig.
Explanation: EOP-3A override DS-1 directs if drywell sprays have been started, then before drywell pressure drops to 0 psig ensure drywell sprays are stopped to ensure that primary containment pressure is not reduced below atmospheric. Drywell pressure can be reduced below suppression chamber pressure causing the suppression chamber to DW vacuum breakers to open (non-condensables return to DW). DW pressure would have to be significantly negative to cause the suppression chamber pressure to lower sufficiently to cause the reactor building to suppression chamber vacuum breakers to open due to the reactor building being maintained at a negative pressure.

Distracters:
Answer B is incorrect because Drywell spray is not required to be stopped before the suppression chamber to drywell vacuum breakers open. This choice is plausible due the common misconception that operation of the suppression chamber to drywell vacuum breakers is abnormal and not desired. Applicants that have the misconception of suppression chamber to drywell vacuum breaker operation would select this answer.

Answer C answer is incorrect because Drywell spray is not required to be stopped before the reactor building to suppression chamber vacuum breakers open. This choice is plausible due the operation of the reactor building to suppression chamber vacuum breakers introducing air into the torus being abnormal and not desired. Applicants that have the misconception of the reactor building to suppression chamber vacuum breaker operation and securing drywell sprays would select this answer.

Answer D answer is incorrect because Drywell spray is not required to be stopped before the drywell temperature and pressure lower to the RED region of the Drywell Spray Initiation Limit (DWSIL) curve. This choice is a common misconception because drywell spray cannot be initiated in the unsafe region of the curve but if spray is already in progress it may continue. Applicants that have the misconception of always being in the SAFE region of DWSIL would select this answer

Technical References: EOP-3A [Primary Containment Control](Rev 17), AMP-TBD00 [EOP/PSTG Technical Basis]

References to be provided to applicants during exam: none

Learning Objective: INT00806130010300

Question Source:	Bank #	4/2015 NRC Q#14
(note changes; attach parent)	Modified Bank #	
	New	

Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	

10CFR Part 55 Content:	55.41(b)(7),(9)
Level of Difficulty:	3
SRO Only Justification:	N/A

PSA applicability:

N/A

Examination Outline Cross-Reference	Level	RO
295030 Low Suppression Pool Wtr Lvl / 5 Ability to determine and/or interpret the following as they apply to LOW SUPPRESSION POOL WATER LEVEL: (CFR: 41.10) **EA2.04 Drywell/ suppression chamber differential pressure: Mark-I&II**	**Tier#**	1
	Group#	1
	K/A #	295030 EA2.04
	Rating	3.5
	Revision	0
Revision Statement:		

Question 15

A LOCA occurred at rated power.

Drywell pressure is 9.0 psig, rising.

Torus pressure is 7.6 psig, rising.

Torus water level is 12 feet, lowering

What will cause Drywell-to-Torus differential pressure to fall to 0.0 psid AND remain there?

A. Operation of Torus Spray

B. Operation of Drywell Spray

C. Torus water level lowering to 9 feet

D. Automatic operation of Torus-to-Drywell Vacuum Breakers

Answer: C. Torus water level lowering to 9 feet
Explanation: The drywell-to-torus downcomer openings are at 9.58 feet Torus level. Torus water level at 9 feet results in direct exposure of the drywell atmosphere to the suppression chamber airspace thus compromising the pressure suppression function of the primary containment. Drywell and Torus pressures would equalize; therefore, Drywell-to-Torus dP would fall to 0.0 psid and remain there.
Distracters: Answer A is plausible because operation of Torus Spray affects DW/Torus dP. It is wrong because Torus Spray would cause Torus pressure to fall first, followed by DW

pressure, and would result in an increase in dP. Torus Spray would automatically isolate if DW pressure fell below 2 psig, so there would continue to be a driving head for some DW/Torus dP. DW/Torus dP would not fall to 0.0 psid and stabilize.

Answer B is plausible because operation of DW Spray affects DW/Torus dP. It is wrong because DW Spray would cause DW pressure to fall first, followed by Torus pressure. DW pressure would drop below Torus pressure, causing DW/Torus dp to go negative, below 0.0 psig. Torus-to-DW Vacuum Breakers would then open when DW/Torus dp lowered to -0.5 psid. DW Spray would automatically isolate when DW pressure fell below 2 psig, so there would continue to be a driving head for some DW/Torus dP. DW/Torus dP would eventually go positive again after vacuum breakers closed. DW/Torus dP would not fall to 0.0 psid and stabilize there.

Answer D is plausible because operation of Torus/DW Vacuum Breakers affects DW/Torus dP. It is wrong because Torus-to-DW Vacuum Breakers automatically open when DW/Torus dp lowers to -0.5 psid, which is below 0.0 psid. DW/Torus dP would not fall to 0.0 psid and stabilize there.

Technical References: AMP-TBD00 [EOP/PSTG Technical Basis] for EOP-3A [Primary Containment Control](Rev 17) step SP/L-2.2, Lesson Plan COR002-23-02 [RHR System](Rev 33), Lesson Plan COR002-03-02 [Containment Systems](Rev 30)

References to be provided to applicants during exam: none

Learning Objective: INT008-06-13 EO-4a

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b)(9)	
Level of Difficulty:	3	
SRO Only Justification:	N/A	

PSA applicability:
Top 10 Risk Significant System – Primary Containment (Over-Pressure)

Examination Outline Cross-Reference	Level	RO
295031 Reactor Low Water Level / 2 **2.4.1** Knowledge of EOP entry conditions and immediate action steps. (CFR: 41.10)	**Tier#**	1
	Group#	1
	K/A #	295031 G2.4.1
	Rating	4.6
	Revision	0
Revision Statement:		

Question 16

Regarding EOP entry conditions and Mitigating Task Scram Actions,…

(1) What is the EOP-1A entry condition for RPV water level? **Level below….**

AND

(2) Why is the Reactor Mode Switch placed to REFUEL IAW Procedure 2.1.5 [Reactor Scram]?

A. (1) +3 inches
(2) Insert a redundant scram signal

B. (1) +3 inches
(2) Facilitate checking all control rods full in

C. (1) +12 inches
(2) Insert a redundant scram signal

D. (1) +12 inches
(2) Facilitate checking all control rods full in

Answer: B. (1) +3 inches (2) Facilitate checking all control rods full in
Explanation: The EOP-1A entry condition for RPV water level is water level below +3 inches, consistent with the RPS setpoint of TS 3.3.1.1. Entry into Procedure 2.1.5 [Reactor Scram] is directed by EOP-1A to accomplish the power control leg. Procedure 2.1.5 Attachment 1 [Mitigating Task Scram Actions] step 1.2 directs the operator to place the Reactor Mode Switch to REFUEL. According to Procedure 2.1.5 Att. 6 step 1.3, this is done before placing the RMS to SHUTDOWN IAW Attachment 2 [Reactor Power Control] in order to check rod positions. If all control rods are fully inserted, the

Refuel Mode Select Permissive Light on panel 9-5 will illuminate when the RMS is placed to REFUEL and the Rod Select Power Switch is cycled OFF and back ON. Procedure 2.1.5 Note 3 at step 3.1 states this is an acceptable method of determining all control rods are fully inserted.		
Distracters: Answer A part 1 is correct. Part 2 is plausible because the RMS is finally placed to SHUTDOWN, which inserts a redundant scram signal. It is wrong because the purpose of stopping in REFUEL position is to facilitate verifying all control rods fully inserted, and REFUEL position does not cause a scram. Answer C part 1 is plausible because procedure 2.4RXLVL [RPV Water Level Control Trouble] immediate action 3.1.1 requires a scram to be inserted if RPV level cannot be maintained above +12" on narrow range instruments. With the reactor at higher power levels, a manual scram would result in void collapse and level shrink below the 3 inch entry condition. The unprepared applicant may associate entering EOP-1A during a level reduction transient and confuse the manual scram criteria with the EOP-1A entry condition. It is wrong because the EOP-1A entry condition is consistent with the RPS trip setpoint, +3 inches. If the plant was at higher power, EOP-1A entry would be required following a manual scram inserted at 12 inches, but only because level would shrink below 3 inches. Part 2 is plausible and wrong for the same reasons as stated for distractor A. Answer D part 1 is plausible and wrong for the same reasons as stated for distractor C. Part 2 is correct.		
Technical References: EOP-1A [RPV Control](Rev 20) and AMP-TBD00 [EOP/PSTG Technical Basis], Procedure 2.1.5 [Reactor Scram](Rev 73), Procedure 2.4RXLVL [RPV Water Level Control Trouble](Rev 26)		
References to be provided to applicants during exam: none		
Learning Objective: INT008-06-05 EO-1		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b)(6),(10)	
Level of Difficulty:	2	
SRO Only Justification:	N/A	

PSA Applicability
N/A

Examination Outline Cross-Reference	Level	RO
295037 SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown / 1 Knowledge of the operational implications of the following concepts as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN: (CFR: 41.8 to 41.10) **EK1.02 Reactor water level effects on reactor power**	**Tier#**	1
	Group#	1
	K/A #	295037 EK1.02
	Rating	4.1
	Revision	0
Revision Statement:		

Question 17

Failure to Scram has occurred from 100% power.

Current conditions are:

- Reactor power 2%
- Control rod density 40%
- Reactor water level -90 inches Wide Range
- Amount of SLC Tank injected 28%

How will power respond if level slowly rises above +3 inches?

A. Continually lower, stabilizing below 2%

B. Initially drop then slowly rise, stabilizing below 2%

C. Initially drop then slowly rise, stabilizing above 2%

D. Initially rise then continually lower, stabilizing below 2%

Answer: D. Initially rise then continually lower, stabilizing below 2%
Explanation: The conditions given represent RPV level having been lowered to below the feedwater spargers and hot shutdown boron weight (26%) injected by SLC. EOP-7A step FS/L-8 directs beginning level restoration to the normal band when HSD boron weight has been injected and emergency depressurization is not expected or in

progress. Per PSTG for EOP-7A step FS/L-24, power is expected to initially rise due to resumption of natural circulation as level is raised, then power is expected to lower as boron is mixed within the core. The negative reactivity from boron will outweigh the positive reactivity from natural circulation; therefore, power will stabilize below where it was initially, 2%.

Distracters:
Answer A is plausible to the unprepared applicant who only considers the effects of SLC mixing in the core. It is wrong because power will initially rise due to natural circulation being established as level is raised.

Answer B is plausible to the unprepared applicant who confuses the order and effects of natural circulation and SLC mixing. It is wrong because natural circulation is established before SLC concentration in the core rises, so power initially rises and then falls.

Answer C is plausible to the unprepared applicant who misjudges the effects of natural circulation versus the effects of HSD boron weight distributed within the core. The applicant may select this answer based on what he might have seen most often during simulator training, level/power effects before HSD boron weight injection. It is wrong because the negative reactivity from boron will outweigh the positive reactivity from natural circulation; therefore, power will stabilize below where it was initially, 2%.

Technical References: EOP-7A [RPV Level, Failure-to-Scram](Rev 19), AMP-TBD00 [EOP/PSTG Technical Basis] for EOP-7A step FS/L-24

References to be provided to applicants during exam: none

Learning Objective: INT008-06-10 EO-6,9

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X

Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X

10CFR Part 55 Content:	55.41(b)(2),(5)
Level of Difficulty:	3
SRO Only Justification:	N/A

PSA applicability:
N/A

Examination Outline Cross-Reference	Level	RO
295038 High Off-site Release Rate / 9	Tier#	1
Knowledge of the interrelations between HIGH OFF-	Group#	1
SITE RELEASE RATE and the following:	K/A #	295038 EK2.02
(CFR: 41.7)	Rating	3.6
EK2.02 Offgas system	Revision	0
Revision Statement:		

Question 18

The plant is at rated power.

The following alarms are received due to a valid high radiation release rate:

OFFGAS TIMER INITIATED	PANEL/WINDOW: 9-4-1/C-4
OFFGAS HIGH RAD	PANEL/WINDOW: 9-4-1/C-5

(1) What is the maximum allowed time delay associated with annunciator 9-4-1/C-4?

AND

(2) According to 2.4OG [Off-Gas Abnormal], which one of the following actions should be taken if immediate off-gas isolation is desired?

A. (1) 5 minutes
   (2) Place Off Gas Timer switch to CLOSE on Panel 9-2.

B. (1) 5 minutes
   (2) Place Off Gas System Isolation OG-AO-254 switch to CLOSE on VBD-K.

C. (1) 15 minutes
   (2) Place Off Gas Timer switch to CLOSE on Panel 9-2.

D. (1) 15 minutes
   (2) Place Off Gas System Isolation OG-AO-254 switch to CLOSE on VBD-K.

Answer: C. (1) 15 minutes
(2) Place Off Gas Timer switch to CLOSE on Panel 9-2.

Explanation:
Per Alarm Card 9-4-1/C-4, the offgas timer maximum delay is 15 minutes per the ODAM. It initiates when Offgas Rad Monitors A and B reach their high-high setpoints. IF the high-high radiation condition does not clear within 15 minutes, an offgas isolation occurs. This closes the following valves that discharge to the Elevated Release Point:

- OG-AO-254, OFF/GAS SYSTEM ISOLATION
- AOG-AO-902, AOG RETURN.

Procedure 2.4OG Att. 1 step 1.2 states IF off-gas isolation is immediately desired, THEN at Panel 9-02, place OFFGAS TIMER switch to CLOSE.

Distracters:
Answer A part 1 is plausible because low dilution flow, <1100 cfm, also causes an offgas isolation after a 5 minute time delay. The unprepared applicant who confuses the low dilution flow time delay with the high-high radiation isolation time delay would select this answer. It is wrong because the time delay for the given condition, high-high radiation, is 15 minutes. Part 2 is correct.

Answer B part 1 is plausible and wrong for the same reasons as given for distractor A. Part 2 is plausible because OG-AO-254 is named the Off Gas System Isolation valve and because it automatically closes on an offgas isolation signal, and it would effect isolation of the offgas discharge path to the ERP if Augmented Off Gas was not in service, such as at low power. It is wrong because at rated power, AOG is in service. AO-254 is closed and OG-AO-901 and OG-AO-902 are open. AO-902 must close to isolate offgas discharge to the ERP, and it does auto isolate on an offgas isolation system. If the timer has not elapsed and immediate isolation is desired, 2.4OG requires placing the Off Gas Timer switch from AUTO to CLOSE on panel 9-2, which in effect completes the timer function and closes the associated offgas isolation valves, including AO-902 and AO-254, if open.

Answer D part 1 is correct. Part 2 is plausible and wrong for the same reasons as given for distractor B.

Technical References: Alarm Cards 9-4-1/C-4 and 9-4-1/C-5 (Rev 56), procedure 2.4OG [Off-Gas Abnormal](Rev 24)

References to be provided to applicants during exam: none

Learning Objective: COR001-18-01 Obj LO-10c, 12b, 05b, 05c

Question Source:	Bank #	

(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b)(11)	
Level of Difficulty:	3	
SRO Only Justification:	N/A	
PSA applicability:		
N/A		

Examination Outline Cross-Reference	Level	RO
600000 Plant Fire On Site / 8 Knowledge of the reasons for the following responses as they apply to PLANT FIRE ON SITE: **AK3.04 Actions contained in the abnormal procedure for plant fire on site**	**Tier#**	1
	Group#	1
	K/A #	600000 AK3.04
	Rating	2.8
	Revision	0
Revision Statement:		

Question 19

Why are Service Water pumps required to be immediately started IAW Procedure 5.4FIRE-S/D [Fire Induced Shutdown from Outside Control Room]?

To provide cooling to …

A. HPCI room

B. Fuel Pool Cooling

C. Shutdown Cooling

D. Diesel Generator(s)

Answer: D. Diesel Generator(s)

Explanation:
Procedure 5.4FIRE-S/D is entered from procedure 5.1INCIDENT or 5.4POST-FIRE-CONTROL due to a fire in particular areas of the plant that might require control room evacuation. The first action listed in procedure 5.4FIRE-S/D is subsequent action step 4.2, which states if DG(s) running without SW flow, THEN immediately Start Service Water (SW) pumps from Control Room. If that is unsuccessful, SW pumps are started from the Critical Switchgear room. If a DG runs for ≥5 minutes without SW, then it must be shutdown using Emergency Stop push button.

This concern is also reflected in the first operator action of procedure 5.4POST-FIRE-CONTROL [Control Building Post-Fire Operational Information].

Distracters:
Answer A is plausible because HPCI operation from the Alternate Shutdown panel (ASD) is directed if control room evacuation is required due to effects of a fire in the plant. HPCI is important because it is the primary system for RPV level and pressure control during shutdown from outside the control room, and much emphasis is placed

on preserving HPCI operation. SW is the heat sink for REC, which is the normal cooling water supply for the HPCI room cooler. Eventually in 5.4FIRE-S/D Att. 1, SW is cross-tied to REC to directly supply the HPCI room cooler. This answer is wrong because HPCI room cooling is not the reason SW pumps may have to be manually started as soon as 5.4FIRE-S/D is entered. HPCI room temperature is not as time limiting as DG cooling. IF DG operation is required, SW must be supplied within 5 minutes to avoid DG damage.

Answer B is plausible because SW serves as the heat sink for Fuel Pool Cooling, and loss of cooling could eventually result in uncovery of spent fuel and fuel damage. The effects of loss of Fuel Pool Cooling has received much emphasis related to the Fukushima Daiichi disaster. This answer is wrong because Fuel Pool cooling is not the reason SW pumps may have to be manually started as soon as 5.4FIRE-S/D is entered. Spent Fuel Pool temperature is not as time limiting as DG cooling. IF DG operation is required, SW must be supplied within 5 minutes to avoid DG damage.

Answer C is plausible because of the emphasis placed on loss of shutdown cooling. SW serves as the heat sink for Shutdown Cooling, and loss of cooling could eventually result in uncovery of fuel and fuel damage. This answer is wrong because Shutdown cooling is not the reason SW pumps may have to be manually started as soon as 5.4FIRE-S/D is entered. Reactor coolant temperature and inventory is not as time limiting as DG cooling. IF DG operation is required, SW must be supplied within 5 minutes to avoid DG damage.

Technical References: Procedure 5.4FIRE-S/D [Fire Induced Shutdown from Outside Control Room](Rev 68), Procedure 5.4POST-FIRE-CONTROL [Control Building Post-Fire Operational Information](Rev 3)

References to be provided to applicants during exam: none

Learning Objective: INT032-01-34 EO-I

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b)(10),(8)	
Level of Difficulty:	2	
SRO Only Justification:	N/A	

PSA applicability:

Top 10 Risk Significant Systems – Service Water, Diesel Generators

Examination Outline Cross-Reference	Level	RO
700000 Generator Voltage and Electric Grid Disturbances / 6 Ability to operate and/or monitor the following as they apply to GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES: (CFR: 41.5 and 41.10) **AA1.05 Engineered safety features**	**Tier#**	1
	Group#	1
	K/A #	700000 AA1.05
	Rating	3.9
	Revision	0
Revision Statement:		

Question 20

DG1 was started 1 minute ago for a monthly surveillance.

- DIESEL GEN 1 BKR EG1 is open.

Then, a Loss of Offsite Power (LOOP) results in the following alarm:

4160V BUS 1F UNDERVOLTAGE	PANEL/WINDOW: C-1/A-6

What is the SHORTEST time after alarm C-1/A-6 is received that 4160V Bus 1F is re-energized?

A. 5 seconds

B. 10 seconds

C. 15 seconds

D. 20 seconds

Answer: B. 10 seconds
Explanation: DG1 and DG2 and ESF 4160 VAC Buses 1F/1G are ESF systems. Offsite power supplies 4160V buses 1A and 1B via the SSST, and it supplies the ESST, which is capable of supplying 4160V ESF Buses 1F/1G. When offsite power is lost, undervoltage is sensed on buses 1A and 1B, and their feeder breakers to buses 1F

and 1G trip. The ESST is also de-energized on a LOOP, so it does not automatically supply 4160V Bus 1F. DG1 is already running and at rated speed and voltage due to the surveillance. DG1 output breaker EG1 auto closes when bus 1F has been de-energized for 10 seconds, sensed by UV relay 27X3/1F. Therefore, bus 1F will be re-energized within 10 seconds.

Distracters:
Answer A is plausible because it reflects the maximum time delay for loss of 4160V Bus 1F voltage relay 27/1F1 to drop out. It is wrong because relay 27/1F1 dropping out causes alarm C-1/A-6 and starts 10 second TDD relay 27X3-1F, which provides the close permissive to DG1 output breaker EG1, which re-energizes Bus 1F.

Answer C is plausible because it reflects the approximate time for DG1 to reach rated speed and voltage from a standby start. DG1 is designed to reach rated speed and voltage from standby within 14 seconds. It is wrong because DG1 is initially at rated speed and voltage, since it has been running for 1 minute; therefore, its output breaker will close 10 seconds after the alarm is received.

Answer D is plausible because it reflects the combined times of distractors A and C. The unprepared applicant may chose this answer if they do not recognize DG1 is already at rated speed and voltage and if they mistake the alarm to occur at the moment offsite power degrades. It is wrong because the alarm indicates the time delay for relay 27/1F1 has expired, DG1 is already at rated speed and voltage, and breaker EG1 will close when the 10 second time delay for relay 27X3/1F elapses.

Technical References: Procedure 2.2.18 [4160V Auxiliary Power Distribution System](Rev 208), Lesson Plan COR001-01-01 [AC Distribution System] (Rev 47), B&R drawing 3017 sheet 01 and 3024 sheet 08, USAR section VII (ESF systems), TS SR 3.8.1.7, USAR Section VII-1 (showing ESF designations)

References to be provided to applicants during exam: none

Learning Objective: COR001-01-01 LO-08h

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b)(7)	
Level of Difficulty:	3	
SRO Only Justification:		

N/A
PSA Applicability
Top 10 Risk Significant Systems – Emergency AC Power

Examination Outline Cross-Reference	**Level**	RO
295002 Loss of Main Condenser Vac / 3 Knowledge of the operational implications of the following concepts as they apply to LOSS OF MAIN CONDENSER VACUUM : (CFR: 41.8 to 41.10) **AK1.03 Loss of heat sink**	**Tier#**	1
	Group#	2
	K/A #	295002 AK1.03
	Rating	3.6
	Revision	0
Revision Statement:		

Question 21

The plant is at 50% power.

Condenser vacuum is completely lost due to a failed LP Turbine rupture disc.

A Group 1 isolation occurs due to loss of vacuum.

Which component will be manually started by the operator to mitigate the effects of this condition?

A. Service Water Pump

B. Circulating Water Pump

C. Mechanical Vacuum Pump

D. RHR Service Water Booster Pump

Answer: D. RHR Service Water Booster Pump
Explanation: A complete loss of condenser vacuum results in a Group 1 isolation. Procedure 2.4VAC requires closing MSIVs if vacuum cannot be maintained ≥ 12 inches Hg, since MSIVs automatically close at 8 inches Hg. This results in loss of the main condenser as a heat sink, loss of the steam supply for RFPs, and, ultimately, loss of the hotwell as a source of makeup for Condensate. Pressure control is required to be transferred to SRVs, HPCI, and/or RCIC. Operation of SRVs, HPCI, or RCIC results in energy addition to the Suppression Pool, requiring use of RHR Suppression Pool Cooling. An RHR SW Booster Pump is required to be started IAW procedure 2.2.69.3 [RHR Suppression Pool Cooling and Containment Spray] per procedure 2.2.70 [RHR Service Water System] to support SPC operation. 2.4VAC Attachment 6 [Pre-Staging

Aid] lists placing SPC in service as a contingency action to be performed if MSIV closure is anticipated due to loss of vacuum.

Distracters:
Answer A is plausible for the unprepared applicant who confuses the reasons for the loss of vacuum or proper mitigating actions (i.e. loss of circulating water flow versus air inleakage.) 2.4VAC Att. 2 lists actions for loss of vacuum due to flow blockage in the intake bays. Actions include entering procedure 5.2SW [Service Water Casualties], which includes actions to start additional SW pumps. It is also plausible to the unprepared applicant who does not know the actions associated with placing RHR SPC in service IAW the procedure or confuses SW pumps with RHR SW Booster pumps. This answer is wrong because a complete loss of condenser vacuum due to air in-leakage would not affect SW pump operation but would result in MSIV closure and operation of RHR SPC, which requires manually starting a RHR SW Booster pump.

Answer B is plausible for the unprepared applicant who confuses the reason for the loss of vacuum (i.e. loss of circulating water flow versus air in-leakage) or who believes additional circulating water flow would mitigate the effects of air in-leakage. 2.4VAC Att. 2 lists actions for loss of vacuum due to flow blockage in the intake bays. Actions include shifting Circulating Water Pumps, and the fourth CW pump may be idle at rated power, if river temperature is <60°F. This answer is wrong because 2.4VAC does not list starting an additional CW pump as an action to mitigate air in-leakage, since air in-leakage blankets the condenser tubes and increased flow would be ineffectual.

Answer C is plausible because a MVP malfunction is listed as a possible cause of loss of vacuum in 2.4VAC, MVPs are designed to remove non-condensables from the condenser to achieve condenser vacuum, and MVPs would be idle under the given conditions. This answer is wrong because the problem is gross air in-leakage, not a problem with air removal via SJAEs. MVPs do not achieve a vacuum as high as SJAEs, and MVP operation is prohibited above ~5% power. There are no instructions for starting a MVP in 2.4VAC given these conditions.

Technical References: 2.4VAC [Loss of Condenser Vacuum] Att 6 (Rev 25), Procedure 2.2.69.3 [RHR Suppression Pool Cooling and Containment Spray](Rev 46), 2.2.70 [RHR Service Water System](Rev 76)

References to be provided to applicants during exam: none

Learning Objective: INT032-01-32 EO-G

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X

Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b)(7),(10)	
Level of Difficulty:	2	
SRO Only Justification:	N/A	
PSA applicability:		
Top 10 Risk significant System – Service Water System, RHR Suppression Pool Cooling		

Examination Outline Cross-Reference	**Level**	RO
295008 High Reactor Water Level / 2	**Tier#**	1
Knowledge of the interrelations between HIGH	**Group#**	2
REACTOR WATER LEVEL and the following:	**K/A #**	295008 AK2.03
(CFR: 41.7 / 45.8)	**Rating**	3.6
AK2.03 Reactor water level control	Revision	0
Revision Statement:		

Question 22

The plant is at 70% power during power ascension.

RFP A turbine accelerates due to a problem in its hydraulic oil system.

The following alarm is received:

REACTOR HIGH WATER LEVEL	PANEL/WINDOW: 9-5-2/F-1

The operator trips RFP A.

A manual scram is inserted when reactor water level reaches 47 inches.

What is the Reactor Water Level Control system (RVCLS) Master Level Controller setpoint 10 minutes after the scram?

A. (+) 5 inches

B. (+) 25 inches

C. (+) 35 inches

D. (-) 15 inches

Answer: B. (+) 25 inches
Explanation: The SETPOINT SETDOWN function of the RVLCS mitigates potential high reactor water level conditions following a scram. Void collapse due to a scram from high

power results in a rapid level reduction below 0”. Without Setpoint Setdown, RFPs would feed at a high rate to restore level to normal. The relatively cold water injected would then swell, causing level to rise above the RFP high water level trip setpoint. RVLCS anticipates this swell effect and reduces the amount of FW injected during shrink from a scram to prevent level from reaching the Level 8 trip setpoint, 54”. The SETPOINT SETDOWN switch when positioned to the ENABLED position provides for reactor level setpoint down on a Reactor Scram. Setpoint setdown is enabled during startup before placing the Turbine Generator in service (~15% power). Upon a reactor scram, backup scram logic sends a scram signal to RVLCS to initiate setpoint setdown. Upon a Reactor Scram, if LT59D is valid, level setpoint is set to -15" and then commences ramping to 25" at 5" per minute. If LT59D is invalid, level setpoint is set to +5" and commences ramping to 25" at 5" per minute. In this case, redundant narrow range level channel LT-59D is valid; therefore, the level setpoint will shift to -15 inches upon the scram. Then the setpoint will ramp up at 5 inches/minute until 25” is reached. Since -15” to 25” is 40” total, it will take eight minutes for the signal to ramp from -15” to 25”. Therefore, 10 minutes after the scram the level setpoint is +25”.

Distracters:
Answer A is plausible because if LT59D is invalid, the level setpoint is initially set to +5". The unprepared applicant may only remember this value is involved in setpoint setdown logic and select this answer. It is wrong because LT-59D is not identified as invalid, so the applicant should consider it valid. With LT-59D valid, the level setpoint initially goes to -15” and ramps up 5”/minute. After 10 minutes, the setpoint is +25”.

Answer C is plausible because it reflects the normal level setpoint for power operation and because it reflects the result that would be obtained if the ramp did not stop at 25”. It is wrong because the setpoint stops ramping when it reaches 25”.

Answer D is plausible because with LT59D is valid, the level setpoint is initially set to (-15"). The unprepared applicant may only remember this value is involved in setpoint setdown logic and select this answer. It is wrong because, the level setpoint will ramp to 25” within the specified 10 minutes.

Technical References: Procedure 4.4.1 [Reactor Vessel Level Control System](Rev 8), Alarm Card 9-5-2/F-1 (Rev 46)

References to be provided to applicants during exam: none

Learning Objective: COR002-32-02 Obj. LO-06c, 05f

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X

Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b)(7)	
Level of Difficulty:	3	
SRO Only Justification:	N/A	
PSA applicability:		
N/A		

Examination Outline Cross-Reference	Level	RO
295015 Incomplete SCRAM / 1 Knowledge of the reasons for the following responses as they apply to INCOMPLETE SCRAM : (CFR: 41.5) **AK3.01 Bypassing rod insertion blocks**	**Tier#**	1
	Group#	2
	K/A #	295015 AK3.01
	Rating	3.4
	Revision	0
Revision Statement:		

Question 23

Why are control rod blocks manually bypassed IAW Procedure 5.8.3 [Alternate Rod Insertion Methods] during a Failure-to-Scram situation?

To defeat presence of…

A. Control rod select blocks.

B. Rod Worth Minimizer rod blocks.

C. One-Rod Out Interlock rod block.

D. Reactor Mode Switch SHUTDOWN position rod block.

Answer: B. Rod Worth Minimizer rod blocks

Explanation:
The RWM enforces control rod sequence constraints if below the LPSP with both withdrawal blocks and insertion blocks. The RWM is bypassed IAW procedure 5.8.3 Att. 1 step ARI-20 to allow manual rod insertion when control rods are not in pattern. Manual control rod insertion is necessary during failure to scram conditions as power is reduced below the LPSP until achieving all rods in. Only the RWM enforces insert blocks.

Distracters:
Distractors are all plausible because each represents a type of control rod block that could be present during various phases of a failure to scram event. The unprepared applicant may choose a distractor due to the common usage of the term “rod block” and confusion due to the complexity of the numerous rod block signals.

Answer A is plausible because a Select Block de-energizes the rod select relays and the SELECT light for the selected control rod on the Full Core Display, and prevents further control rod selection until the condition has cleared/reset. This answer is wrong because Select Blocks only occur due to either a withdraw Timer malfunction or due to an RPIS Inop condition. Neither of these conditions would occur solely due to occurrence of a failure to scram.

Answer C is plausible because it reflects a rod block that is present while the Reactor Mode Switch is in the Refuel position, which occurs during performance of procedure 2.1.5 [Reactor Scram] Att. 1 [Mitigating Task Scram Actions]. It is wrong because the One Rod Out interlock is a control rod withdrawal block and is not defeated by bypassing RWM.

Answer D is plausible because a rod block is generated by the Reactor Mode Switch any time it is in Shutdown position. It is wrong because it is a control rod withdrawal block and is not defeated by bypassing RWM.

Technical References: Procedure 5.8.3 [Alternate Rod Insertion Methods](Rev 17), Procedure 4.2 [Rod Worth Minimizer](Rev 29), Lesson Plan COR002-20-02 [RMCS] (Rev 22), Procedure 4.3 [Reactor Manual Control System and Rod Position Information System](Rev 28)

References to be provided to applicants during exam: none

Learning Objective: COR002-20-02 Obj. LO-04c

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b)(6)	
Level of Difficulty:	2	
SRO Only Justification:	N/A	

PSA applicability:

N/A

Examination Outline Cross-Reference	**Level**	RO
295020 Inadvertent Cont. Isolation / 5 & 7 Ability to operate and/or monitor the following as they apply to INADVERTENT CONTAINMENT ISOLATION : (CFR: 41.7) **AA1.03 Containment ventilation system: Plant-Specific**	**Tier#**	1
	Group#	2
	K/A #	295020 AA1.03
	Rating	2.9
	Revision	0
Revision Statement:		

Question 24

Containment Ventilation is aligned to purge the drywell using Reactor Building Exhaust IAW Procedure 2.2.60 [Primary Containment Ventilation and Nitrogen Inerting System].

Consider the position of the following valves on VBD-H for that alignment:

- PC-MO-306, VALVE MO 231 BYPASS VLV
- PC-MO-231, DW EXH INBD ISOL VLV
- PC-AO-246, DW EXH OUTBD ISOL VLV

Which of the above listed valves **CHANGE POSITION** if RPS MG Set B trips?

A. PC-MO-231, only

B. PC-AO-246, only

C. PC-MO-231 AND PC-AO-246, only

D. PC-MO-306 AND PC-MO-231 AND PC-AO-246

Answer: C. PC-MO-231 AND PC-AO-246, only
Explanation: When containment ventilation is aligned for initial drywell purge to prepare for plant startup, PC-MO-231 and PC-AO-246 are open as an exhaust path. PC-MO-306 remains closed. PC-MO-306 , PC-MO-231 and PC-AO-246 receive close signals upon a Group 6 isolation signal. When RPS B loses power, a full Group 6 isolation occurs. Therefore, PC-MO-231 and PC-AO-246 reposition. PC-MO-306 does not reposition, since it was already closed in the DW purge lineup.

Distracters:
.Answer A is plausible because MO-231 is the only motor operated valve of the three valves listed that is open during DW purge operations and because it is the inboard isolation valve. The unprepared applicant may select this answer because MO-231 is motor operated or because he believes only the inboard valve closes when RPS B loses power, since only one of two series valves in other isolation groups will close when one trip system trips, such as the Group 5 RCIC steam supply inboard/outboard valves. This answer is wrong because loss of RPS B bus results in a full Group 6 isolation; therefore both valves that were open for DW purge operations, MO-231 and AO-246, both close/reposition.

Answer B is plausible because AO-246 is the only air operated valve of the three valves listed that is open during DW purge operations and because it is the outboard isolation valve. The unprepared applicant may select this answer because AO-246 is air operated, and he believes it to fail closed, or because he believes only the outboard valve closes when RPS B loses power, since only one of two series valves in other isolation groups will close when one trip system trips, such as the Group 5 RCIC steam supply inboard/outboard valves. This answer is wrong because loss of RPS B bus results in a full Group 6 isolation; therefore both valves that were open for DW purge operations, MO-231 and AO-246, both close/reposition.

Answer D is plausible because if all three valves were open, all three would reposition closed upon a Group 6 isolation. The unprepared applicant who confuses DW venting operations with DW purge operations and believes MO-306 is open during DW purge operations would select this answer. It is wrong because MO-306 is a throttle valve opened during DW venting, but not during DW purging in preparation for startup. MO-306 would not reposition, because it would already be fully closed.

Technical References: Procedure 2.2.60 [Primary Containment Ventilation and Nitrogen Inerting System](Rev 95), Procedure 2.1.22 [Recovering from a Group Isolation](Rev 60), Lesson Plan COR002-29-02 [RPS](Rev 23), Procedure 4.5 [Reactor Protection/Alternate Rod Insertion Systems](Rev 32)

References to be provided to applicants during exam: none

Learning Objective: COR002-03-02 Obj LO-06c, 18c; COR002-21-02 Obj LO-08c

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X

Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X

10CFR Part 55 Content:	55.41(b)(9)

Level of Difficulty:	3
SRO Only Justification:	N/A
PSA applicability:	
Top !0 Risk Significant Systems – Primary Containment (Isolation)	

Examination Outline Cross-Reference	Level	RO
295022 Loss of CRD Pumps / 1 Ability to determine and/or interpret the following as they apply to LOSS OF CRD PUMPS : (CFR: 41.10) **AA2.01 Accumulator pressure**	Tier#	1
	Group#	2
	K/A #	295022 AA2.01
	Rating	3.5
	Revision	0
Revision Statement:		

Question 25

The plant is in Mode 2:

- Reactor power 3%
- Reactor pressure 500 psig
- CRD pump A in operation

Then, this alarm is received:

CRD PUMP A BREAKER TRIP	PANEL/WINDOW: 9-5-2/A-6

Then, this alarm is received due to pressure in **one** CRD HCU Accumulator for a control rod at position 48:

CRD ACCUM LOW PRESS OR HIGH LEVEL	PANEL/WINDOW: 9-5-2/G-6

(1) What is the minimum CRD HCU accumulator pressure required to satisfy LCO 3.1.5 [Control Rod Scram Accumulators]?

AND

(2) Which action is required IAW Alarm Cards if **neither** CRD pump can be immediately restarted?

A. (1) 900 psig
(2) Scram and enter Procedure 2.1.5 [Reactor Scram].

B. (1) 900 psig
(2) Monitor for rod drifts AND scram IF low pressure occurs for a second HCU.

C. (1) 940 psig
(2) Scram and enter Procedure 2.1.5 [Reactor Scram].

D. (1) 940 psig
(2) Monitor for rod drifts AND scram IF low pressure occurs for a second HCU.

Answer: C. (1) 940 psig (2) Scram NOW
Explanation: Alarm 9-5-2/A-6 reflects loss of the running CRD pump, which would cause charging water header pressure to drop below 940 psig. Per Alarm Card 9-5-2/G-6, the CRD HCU accumulator low pressure alarm setpoint is 960 psig. In this case, RPV pressure is 500 psig and at 3% power, multiple control rods are withdrawn. Alarm card 9-5-2/A-6 for CRD pump A trip states if RPV pressure is <900 psig with more than one Control Rod withdrawn, then immediately attempt to start a CRD pump. And if a CRD pump cannot be immediately restarted, then a scram must be manually inserted at that time. This reflects the <1hr TS action required per TS 3.1.5 Action D.1 for inability to meet Action C.1. For part 2, TS 3.1.5 CRD Scram Accumulator operability requires accumulator pressure to be ≥940 psig per SR 3.1.5.1
Distracters: Answer A part 1 is plausible because 900 psig is the LOWEST reactor pressure which will fully insert a control rod within the TS Allowable Control Rod Scram Time with NO CRD pumps available IAW Procedure 2.2.8 [Control Rod Drive Hydraulic System]. Part 2 is correct. Answer B part 1 is plausible and wrong for the same reasons as stated for distractor A.. Part 2 is plausible because it reflects the scram related actions required by TS 3.1.5 for reactor pressures above 900 psig. It also reflects monitoring for rod drifts, which is appropriate for re-starting a CRD pump, due to potential surge of cooling water header pressure. It is wrong because reactor pressure is < 900 psig, where CRD HCU accumulators are much more important due to the lower driving head available from RPV pressure via the CRD ball check valve. With RPV pressure <900 psig and CRD pumps unable to be started, TS requires an immediate scram if one CRD HCU accumulator associated with a withdrawn control rod falls below 940 psig. Alarm Card 9-5-2/A-6 requires conservative action to scram if no CRD pump can be immediately restarted. Answer D part 1 is correct. Part 2 is plausible and wrong for the same reason as stated for distractor B.
Technical References: Alarm cards 9-5-2/A-6 and 9-5-2/G-6 (Rev 46), TS 3.1.5, Procedure 2.2.8 [Control Rod Drive Hydraulic System](Rev 95)

References to be provided to applicants during exam: none		
Learning Objective: INT007-05-02 EO-1, 6b, 6c		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b)(6),(10)	
Level of Difficulty:	3	
SRO Only Justification:	N/A	
PSA applicability:		
N/A		

Examination Outline Cross-Reference	Level	RO
203000 RHR/LPCI: Injection Mode Knowledge of electrical power supplies to the following: (CFR: 41.7) **K2.03 Initiation logic**	**Tier#**	2
	Group#	1
	K/A #	203000 K2.03
	Rating	2.7
	Revision	0
Revision Statement:		

Question 26

What is the power supply to LPCI INITIATION LOGIC for RHR Pump D?

A. AA2

B. BB2

C. AA3

D. BB3

Answer: B. BB2

Explanation:
Div 2 LPCI initiation logic starts RHR Pump D in RHR Loop A and Pump C in RHR Loop B. Division 2 LPCI initiation logic is powered by 125 VDC panel BB2.

RHR Pumps A and B are powered by 4160V Bus 1F, which operates using 125 VDC power from AA3. Pumps C and D are powered by 4160V Bus 1G, which operates using 125 VDC power from BB3.

Distracters:
Answer A is plausible because AA2 powers Div 1 LPCI initiation logic for RHR Pump B, and like RHR Pump D, RHR Pump B is in RHR Loop B. It is wrong because AA2 power LPCI start logic for RHR Pumps A and B, not Pump D.

Answer C is plausible because AA3 provides breaker control power for RHR Pumps A and B. It is wrong because AA3 does not provide LPCI logic power at all, and it supplies no power related to RHR Pump D.

Answer D is plausible because BB3 provides breaker control power for RHR Pumps C and D. It is wrong because BB3 does not provide power to LPCI initiation logic.		
Technical References: GE dwgs 791E261 sheets 7, 8, 12; Procedure 2.2A_125DC.DIV2 [125 VDC Power Checklist (Div 2)](Rev 7)		
References to be provided to applicants during exam: none		
Learning Objective: COR002-23-02 Obj LO-02c		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b)(7)	
Level of Difficulty:	2	
SRO Only Justification:	N/A	
PSA applicability:		
Top 10 Risk Significant Systems – Emergency DC Power, RHR Top 10 Risk Significant Components – 125 VDC		

Examination Outline Cross-Reference	**Level**	RO
295036 Secondary Containment High Sump/Area Water Level / 5 Knowledge of the operational implications of the following concepts as they apply to SECONDARY CONTAINMENT HIGH SUMP/AREA WATER LEVEL : (CFR: 41.8 to 41.10) **EK1.02 Electrical ground/ circuit malfunction**	**Tier#**	1
	Group#	2
	K/A #	295036 EK1.02
	Rating	2.6
	Revision	0
Revision Statement:		

Question 27

Regarding the Maximum Safe Operating (MSO) value for Secondary Containment Water Level listed in EOP-5A Table 11,…

(1) What is the Maximum Safe Operating (MSO) water level value for the NW Quad basement area listed in EOP-5A Table 11?

AND

(2) What operational concern is precluded by maintaining water level in that area below the MSO value?

A. (1) 2.8 feet
(2) Electrical malfunctions of essential equipment due to submergence.

B. (1) 2.8 feet
(2) Hampered access due to radioactive water above the level of the floor.

C. (1) 9.5 feet
(2) Electrical malfunctions of essential equipment due to submergence.

D. (1) 9.5 feet
(2) Hampered access due to radioactive water above the level of the floor.

Answer: C. (1) 9.5 feet (2) Electrical malfunctions of essential equipment due to submergence.
Explanation: Per PSTG App. B bases for EOP-5A Table 11 maximum safe operating water level values, all Maximum Safe Operating water levels are consistent with the Stone and Webster CNS Internal Flooding Study. DCD-38, “Internal Flooding- Design Criteria Document” shows lowest essential equipment as 1.85 ft NE Quad, 1.563 ft NW quad, 1.813 ft SE Quad, and 1.625 ft SW Quad. The 1’-6” depth is conservatively used for all quad depths. Area water level is measured by sump level instrumentation. The

MSO Table 11 value is 9.5 ft for the NW quad area, relative to zero inches in the sump. (The MSO Table 11 value of 9.5 ft, Sump A indicated level on VBD-S, is equivalent to 1.5' above the floor of the NW Quad RHR pump room.) DCD-38 states the limiting operational concern for an internal flooding event is failure of electrical equipment subject to submergence.

Distracters:
Answer A part 1 is plausible because 2.8 feet equates to the setpoint for the sump A hi-hi level, 34 inches in the A sump, which is the Maximum Normal Operating value. It is wrong because MSO water level for the NW Quad is 9.5 ft. Part 2 is correct.

Answer B part 1 is plausible and wrong for the same reasons as given for distractor A. Part 2 is plausible because one of the objectives of EOP-5A is to limit water levels and radiation levels that could restrict personnel access required to mitigate an event. The unprepared applicant who does not understand the basis of MSO water level values or who only remembers indicated MSO level originates from sump level instrumentation may select this answer. It is wrong because the MSO water level value was selected based on the height above the Quad basement floor to protect the lowest essential component from submergence and consequential failure. MSO water level is 9.5' indicated sump level, which equates to 1.5' above basement floor level; therefore, having potentially contaminated water above the floor level is not prevented by maintaining water levels below their MSO value.

Answer D part 1 is correct. Part 2 is plausible and wrong for the same reasons as given for distractor B.

Technical References: EOP-5A [Secondary Containment Control] Table 11 (Rev 16), AMP-TBD00 [EOP/PSTG Technical Basis], Alarm card S-1/A-1 (Rev 24), DCD-38 [Internal Flooding Analysis]

References to be provided to applicants during exam: none

Learning Objective: INT008-06-17 EO-4

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b)(10)	
Level of Difficulty:	3	
SRO Only Justification:	N/A	

PSA applicability:
N/A

Examination Outline Cross-Reference	Level	RO
295034 Secondary Containment Ventilation High Radiation / 9 **2.4.31** Knowledge of annunciator alarms, indications, or response procedures. (CFR: 41.10)	Tier#	1
	Group#	2
	K/A #	295034 G2.4.31
	Rating	4.2
	Revision	0
Revision Statement:		

Question 28

The following annunciators are received while operating at 100% power:

RX BLDG VENT HI-HI RAD

PANEL/WINDOW:
9-4-1/E-4

(1) What is the LOWEST radiation level that caused this alarm?

AND

(2) What combination of Reactor Building Vent Exhaust Plenum Radiation Monitors indicating above Hi-Hi Rad setpoint will result in isolation of Secondary Containment?

A. (1) 5 mR/hr
(2) RMP-RM-452A and RMP-RM-452B

B. (1) 5 mR/hr
(2) RMP-RM-452A and RMP-RM-452C

C. (1) 10 mR/hr
(2) RMP-RM-452A and RMP-RM-452B

D. (1) 10 mR/hr
(2) RMP-RM-452A and RMP-RM-452C

Answer: C. (1) 10 mR/hr
(2) RMP-RM-452A and RMP-RM-452B

Explanation:
The setpoint for alarm 9-4-1/E-4 is 10 mR/hr. Therefore, the lowest radiation that will produce this alarm is 10 mR/hr. RMP-RM-452A/B/C/D input into Group 6 isolation logic. The logic arrangement for a high-high trip is (A or C) AND (B or D). Therefore, channels A and B high-high will cause a full Group 6 isolation, which necessitates performance of Procedure 2.1.22 Attachment 1 [Group Isolation Hard Card] to verify the isolation and SGT initiation occurred as designed.

Distracters:
Answer A part 1 is plausible because The setpoint for alarm 9-4-1/E-5 [Rx Bldg Vent High Rad] is 5 mR/hr, as well as the setpoints for various Area Radiation Monitors that input into RED tiled alarm 9-3-1/A-9 [Reactor Building High Rad]. The unprepared applicant may confuse the high and high-high setpoints, or he may confuse the RB Ventilation Rad High High alarm with the RB High Rad (ARM) alarm and choose this answer. It is wrong because the RB Vent Hi-Hi Rad setpoint is 10 mR/hr on RMP-RM-452A, B, C, or D. Part 2 is correct.

Answer B part 1 is plausible and wrong for the same reasons as stated for distractor A. Part 2 is plausible because Group 6 logic for RB Vent Exhaust Plenum Rad is one-out-of-two taken twice, so two channels must trip to cause a Group 6 isolation/initiation. It is wrong because the logic arrangement is (A or C) AND (B or D). Therefore, channels A and C high-high will NOT cause a Group 6 isolation, since neither channel B nor D is tripped.

Answer D part 1 is correct. Part 2 is plausible and wrong for the same reasons as stated for distractor B.

Technical References: Alarm Cards 9-4-1/E-4, 9-4-1/E-5 (Rev 56), 9-3-1/A-9 (Rev 37), Procedure 2.1.22 [Recovering from a Group Isolation](Rev 60), Procedure 4.7.5 [Reactor Building Vent Exhaust Radiation Monitoring System](Rev 18)

References to be provided to applicants during exam: none

Learning Objective: COR002-03-02 OPS Containment

21. Given plant conditions, determine if the following should have occurred:
    a. Secondary Containment isolation
    b. Any of the PCIS group isolations

Question Source:	Bank #	ILT 14-01 Audit #26 (audit for 12/2015 NRC)
(note changes; attach parent)	Modified Bank #	
	New	
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X

10CFR Part 55 Content:	55.41(b)(11)
Level of Difficulty:	3

SRO Only Justification:	
N/A	
PSA Applicability	
N/A	

Examination Outline Cross-Reference	**Level**	RO
205000 Shutdown Cooling Knowledge of the effect that a loss or malfunction of the SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE) will have on following: (CFR: 41.7) **K3.01 Reactor pressure**	**Tier#**	2
	Group#	1
	K/A #	205000 K3.01
	Rating	3.3
	Revision	0
Revision Statement:		

Question 29

Reference Provided

A scram from rated power occurred 36 hours ago.

The plant is now in Mode 4 with the following conditions:

- RPV water level +53 inches
- Reactor Recirc pump B in operation
- Reactor coolant temperature 140°F

A complete loss of Shutdown Cooling occurs.

IAW 2.4SDC [Loss of Shutdown Cooling], what is the SHORTEST time until reactor pressure will begin to rise?

A. 1 hour, 15 minutes

B. 1 hour, 30 minutes

C. 1 hours, 45 minutes

D. 2 hours, 50 minutes

Answer: A. 1 hour, 15 minutes
Explanation: Reactor pressure will begin to rise when coolant temperature reaches the boiling point, since 2.4SDC requires closing the reactor head vents before 212°F is reached. Plotting 36 hours after shutdown and interpolating halfway between the curves for

130°F and 150°F on the Hours After Shutdown curve of Figure 1, for water level at the high level trip, of procedure 2.4SDC [Loss of Shutdown Cooling], Attachment 5, the result obtained is approximately 1.25 hours, or 1 hour, 15 minutes.

Distracters:
Answer B is plausible because it reflects the time that would be obtained (2.5 hrs) by the applicant who makes the mistake of plotting 130°F initial water temperature instead of interpolating 140°F on the Hours After Shutdown curve of Figure 1 It is wrong because plotting 130°F would result in a non-conservative time estimate, so 140°F should be interpolated on Figure 1 Hours After Shutdown graph, which results in 1.25 hours to boiling.

Answer C is plausible because it reflects the time that would be obtained (~1.75 hrs) by the applicant who makes the mistake of using the Hours After Shutdown curve of Figure 2, for water level at the RPV flange, versus Figure 1, for water level at the high level trip. The curves look very similar and an attention to detail error could yield this result. It is wrong because Figure 1 Hours After Shutdown graph is the proper curve, and it results in 1.25 hours to boiling.

Answer D is plausible because it reflects the time that would be obtained (~2.8 hrs) by the applicant who makes the mistake of using the Days After Shutdown graph of Figure 1versus the Hours After Shutdown graph of Figure 1. The curves look very similar and an attention to detail error could yield this result. It is wrong because Figure 1 Hours After Shutdown graph is the proper curve, and it results in 1.25 hours to boiling.

Technical References: procedure 2.4SDC [Loss of Shutdown Cooling], Attachment 5 (Rev 15)

References to be provided to applicants during exam: procedure 2.4SDC [Loss of Shutdown Cooling], all of Attachment 5 (five pages)

Learning Objective: INT032-01-26 EO-L

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X

10CFR Part 55 Content:	55.41(b)(5),(10)
Level of Difficulty:	3

SRO Only Justification:	N/A
PSA applicability:	
Top 10 Risk Significant System –RHR	

Examination Outline Cross-Reference	**Level**	RO
205000 Shutdown Cooling Knowledge of shutdown cooling system (RHR shutdown cooling mode) design feature(s) and/or interlocks which provide for the following: (CFR: 41.7) **K4.02 High pressure isolation: Plant-Specific**	**Tier#**	2
	Group#	1
	K/A #	205000 K4.02
	Rating	3.7
	Revision	0
Revision Statement:		

Question 30

The plant is in Mode 3.

RHR pump A is operating in Shutdown Cooling (SDC) mode.

An instrument failure causes SDC **reactor** pressure instrumentation to sense 150 psig.

What is the status of the following valves 1 minute later?

(1) Pump A SDC Suction Valve MO-15A

AND

(2) Shutdown Cooling RHR Supply Outboard Valve MO-17

A. (1) MO-15A Open
(2) MO-17 Open

B. (1) MO-15A Open
(2) MO-17 Closed

C. (1) MO-15A Closed
(2) MO-17 Open

D. (1) MO-15A Closed
(2) MO-17 Closed

Answer: B. (1) MO-15A Open (2) MO-17 Closed
Explanation: The Reactor Pressure-High Function is provided to isolate the shutdown cooling portion of the Residual Heat Removal (RHR) System. This function is provided only for equipment protection to prevent an intersystem LOCA scenario. The Reactor Pressure-High signals are initiated from two pressure switches that are connected to the RPV steam dome. Two channels of Reactor Pressure-High Function are available. The Reactor Pressure High isolation is not considered part of PCIS Group 2, but PCIS

Group 2 reset switches must be operated to reset this logic. Reactor Pressure below the high setpoint, 72 psig, is a permissive for opening SDC suction isolation valves MO-17 and MO-18. MO-17 and MO-18 receive an automatic close signal from high reactor pressure. SDC Pump Suction Valve MO-15A does not auto close on a Reactor Pressure High signal, and it does not close if MO-17/18 close. Therefore, a sensed Reactor Pressure High signal >72 psig, will result in closure of MO-17, but MO-15A will remain open. It takes approximately 30 seconds (less than 1 minute) for MO-17 to fully close.

Distracters:
Answer A part 1 is correct. Part 2 is plausible for the unprepared applicant who does not remember the reactor pressure high setpoint for SDC isolation or does not remember which SDC valves are affected by the reactor pressure high signal. It is wrong because reactor pressure above 72 psig causes SDC suction valve MO-17 to close.

Answer C part 1 is plausible because MO-15A is, like MO-17, a SDC suction valve and because it is interlocked with other RHR valves. MO-15A cannot be opened unless MO-39A, MO-21A, and MO-13A are fully closed. It is wrong because MO-15A does not receive a close signal from reactor pressure high, nor does it automatically close due to isolation of SDC suction valves MO-17/18. Part 2 is plausible and wrong for the same reasons as stated for distractor A.

Answer D part 1 is plausible and wrong for the same reasons as stated for distractor C. Part 2 is correct.

Technical References: procedure 2.2.69.2 [RHR System Shutdown Operations](Rev 94), Operations instruction #8 [Guideline for Successful Transient Mitigation](Rev 14), Lesson Plan COR002-23-02 [RHR] (Rev 33)

References to be provided to applicants during exam: none

Learning Objective: COR002-23-02 Obj. LO-03j

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X

Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X

10CFR Part 55 Content:	55.41(b)(7)
Level of Difficulty:	2

SRO Only Justification:	N/A
PSA applicability:	
Top 10 Risk Significant System –RHR	

Examination Outline Cross-Reference	Level	RO
206000 HPCI Knowledge of the operational implications of the following concepts as they apply to HIGH PRESSURE COOLANT INJECTION SYSTEM : (CFR: 41.5) **K5.05 Turbine speed control: BWR-2,3,4**	**Tier#**	2
	Group#	1
	K/A #	206000 K5.05
	Rating	3.3
	Revision	0
Revision Statement:		

Question 31

Regarding manually starting HPCI IAW Procedure 2.2.33.1 [High Pressure Coolant Injection System Operations]…

(1) Which action results in initiation of the HPCI speed control ramp generator,

Opening INJECTION VALVE MO-19 **<u>or</u>** starting AUXILIARY OIL PUMP?

AND

(2) What is the potential consequence to HPCI if the ramp generator has NOT been reset when HPCI is started?

A. (1) Starting AUXILIARY OIL PUMP
(2) Overspeed trip

B. (1) Starting AUXILIARY OIL PUMP
(2) Failure to develop minimum speed

C. (1) Opening INJECTION VALVE MO-19
(2) Overspeed trip

D. (1) Opening INJECTION VALVE MO-19
(2) Failure to develop minimum speed

Answer: A. (1) Starting AUXILIARY OIL PUMP (2) Overspeed trip
Explanation: HPCI turbine speed demand originates from the lesser of either the flow demand from the flow controller automatic/manual setting or the output of a signal ramp generator. When in standby, the lesser signal is from the ramp generator. When the HPCI Aux Oil Pump is started, developing oil pressure causes the HPCI Turbine Stop Valve to

open. Limit switches on the Stop Valve send a signal to start the ramp generator as the Stop Valve comes off of its fully closed seat. The ramp generator output then increases to raise HPCI turbine speed in a controlled ramp, until the ramp generator signal exceeds the flow demand signal from the HPCI flow controller, which is based on its flow setting versus flow feedback. Once the HPCI flow controller is the minimum signal, the HPCI governor valve is unaffected by the ramp generator until it has been reset and once again becomes the minimum signal. The HPCI speed ramp generator is reset to its low, standby setting by fully closing the HPCI turbine stop valve, which is accomplished by pressing and, if the Auxiliary Oil Pump is still running, holding the turbine trip push button. With HPCI steam supply MO-14 closed and the turbine idle, the demand signal from HPCI flow controller will be great, since flow feedback would initially be nil. If the ramp generator has completed its ramp, such that the flow controller is the minimum signal to governor valve position demand, the HPCI turbine would accelerate excessively when steam supply valve MO-14 was opened and would overspeed in most cases.

Distracters:

Answer B part 1 is correct. Part 2 is plausible for the unprepared applicant who does not understand HPCI speed control and believes the ramp generator will not begin to ramp up if it has not been reset, preventing HPCI speed from rising. It is also plausible due to familiarity associated with the precaution and operator aid posted on panel 9-3 that states HPCI should not be allowed to operate at less than 2050 rpm. This answer is wrong because HPCI speed control is governed by a MIN gate demand signal. With the ramp generator ramping function complete, it would be the higher demand signal until reset, and therefore, would not influence HPCI speed. The flow demand signal would be high with HPCI idle, such that HPCI would likely overspeed upon steam admission to the turbine due to the time it would take for flow feedback to develop an begin reducing the flow demand.

Answer C part 1 is plausible because MO-19 is opened using its control switch on panel 9-3 immediately after the Aux Oil Pump is started and steam supply valve MO-14 is opened during a manual HPCI start. All of these actions precede the procedure step to verify HPCI Turbine accelerates properly. The unprepared applicant who confuses which valve provides the interlock for the HPCI ramp generator would choose this answer. It is wrong because MO-19 does not start or reset the HPCI ramp generator. Part 2 is correct.

Answer D part 1 is plausible and wrong for the same reasons as stated for distractor C. Part 2 is plausible and wrong for the same reasons as stated for distractor B.

Technical References: Procedure 2.2.33.1 [High Pressure Coolant Injection System Operations](Rev 37), Procedure 2.2.33 [High Pressure Coolant Injection System](Rev 78), Lesson Plan COR002-11-02 [HPCI](Rev 32)

References to be provided to applicants during exam: none		
Learning Objective: COR002-11-02 LO-08h		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b)(7),(8)	
Level of Difficulty:	4	
SRO Only Justification:	N/A	
PSA Applicability		
Top 10 Risk Significant System - HPCI		

Examination Outline Cross-Reference	**Level**	RO
206000 HPCI Knowledge of the effect that a loss or malfunction of the following will have on the HIGH PRESSURE COOLANT INJECTION SYSTEM : (CFR: 41.7) **K6.02 D.C. power: BWR-2,3,4**	**Tier#**	2
	Group#	1
	K/A #	206000 K6.02
	Rating	3.3
	Revision	0
Revision Statement:		

Question 32

HPCI is operating following a scram.

125 VDC panel AA2 loses power.

HPCI steam line area temperature rises to 200°F.

After 2 minutes, as a result of these conditions …

(1) What is the position of HPCI-MO-15, STM SUPP INBD ISOL VLV?

AND

(2) What is the position of HPCI-MO-16, STM SUPP OUTBD ISOL VLV?

A. (1) open
(2) open

B. (1) open
(2) closed

C. (1) closed
(2) open

D. (1) closed
(2) closed

Answer: B. (1) open (2) closed
Explanation:

MO-15 and MO-16 are the HPCI steam supply containment isolation valves and are open when HPCI is in standby or is operating. HPCI steam line area temp >195°F is a Group 4 isolation signal. HPCI isolation Logic A closes MO-15 and isolation Logic B closes MO-16. 125 VDC AA2 supplies some HPCI logic and instrumentation, including Logic A for Group 4 isolation. Group 4 isolation logic is energized to actuate, so loss of AA2 results in Logic A remaining un-tripped given an actual Group 4 isolation condition. Therefore, even though the motor for MO-15 is AC powered, it will not receive a close signal from Group 4 isolation and will remain open. Logic B is powered from 125 VDC panel BB2 and is unaffected by loss of AA2. MO-16 is powered from 125 VDC starter rack B. The Group 4 isolation signal will trip Logic B, which will cause MO-16 to close. Two minutes is stated in the stem to allow ample time for MO-16 to stroke.

Distracters:
Answer A part 1 is correct. Part 2 is plausible for the unprepared applicant who does not thoroughly understand HPCI logic arrangement and power distribution. The unprepared applicant would not recognize HPCI area temperature is above the Group 4 isolation setpoint or may believe both Logic A and B must trip to close either steam supply valve. This answer is wrong because MO-16 and its isolation logic are unaffected by loss of AA2; therefore, MO-16 closes due to area temperature >195°F.

Answer C part 1 and part 2 are plausible for the unprepared applicant who does not thoroughly understand HPCI logic arrangement and power distribution. That applicant may confuse which valve is associated with which logic and reverse the two. This answer is wrong because MO-15 isolation is defeated by loss of AA2, and MO-16 is unaffected by loss of AA2. Therefore, given a valid Group 4 isolation condition, MO-16 will close, but MO-15 will remain open.

Answer D part 1 is plausible because one steam supply isolation valve, MO-16, is unaffected by loss of AA2 and automatically closes due to a Group 4 condition. The majority of HPCI system is powered by Div 2 DC. The unprepared applicant who does not thoroughly understand HPCI logic arrangement and power distribution may believe loss of AA2 has no effect on component operation, since isolation Logic B is redundant to Logic A. The applicant may not realize the redundancy is provided via separate logics feeding separate components, MO-15 and MO-16. This answer is wrong because Group 4 isolation logic is energize to actuate, so loss of AA2 results in Logic A remaining un-tripped given an actual Group 4 isolation condition. Therefore, even though the motor for MO-15 is AC powered, it will not receive a close signal from Group 4 isolation and will remain open.

Technical References: GE drawings 791E271 sheets 2, 3, 4, 7

References to be provided to applicants during exam: none

Learning Objective: COR002-11-02 obj LO-10b

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b)(7),(8)	
Level of Difficulty:	4	
SRO Only Justification:	N/A	
PSA applicability:		
Top 10 Risk Significant system - HPCI		

Examination Outline Cross-Reference	Level	RO
209001 LPCS Knowledge of LOW PRESSURE CORE SPRAY SYSTEM design feature(s) and/or interlocks which provide for the following: (CFR: 41.7) **K4.04 Line break detection**	**Tier#**	2
	Group#	1
	K/A #	209001 K4.04
	Rating	3.0
	Revision	0
Revision Statement:		

Question 33

The plant is at 100% power when this alarm is received:

CORE SPRAY A BREAK DETECTION	PANEL/WINDOW: 9-3-1/A-8

Where in Core Spray A system is the break located? **Between …**

A. Injection Check Vlv CV-18 and the core shroud

B. CS Pump A and Outboard Injection Vlv MO-11A

C. Pump Torus Suction Vlv MO-7A and CS Pump A

D. Outboard Injection MO-11A and Inboard Injection Vlv MO-12A

Answer: A. Injection Check Vlv CV-18 and the core shroud

Explanation:
Core Spray pipe break detection is designed to sense a break in CS piping between the RPV penetration and the core shroud, in other words inside the downcomer region of the RPV. The high pressure side of DPIS-43A is connected to the SLC outer pipe, which senses pressure in the bypass region above the core plate. The low pressure side is connected downstream of the CS injection check valve CV-18 and manual isolation valve V-14A, both located in the drywell. At rated power, the dp will be approximately -3.5 psid. If a line break in the downcomer region occurs, dp will be approximately 7 psid, since the downcomer is at a lower pressure than the core region.

Distracters:
Answer B is plausible because it reflects the location of a pressure transmitter in the system, PT-38A. It is wrong because PT-38A is only used for CS A discharge pressure indication. It is not affiliated with the subject alarm.

Answer C is plausible because it reflects the location of a pressure transmitter in the system, PT-36A. It is wrong because PT-36A is only used for CS A suction pressure indication. It is not affiliated with the subject alarm.

Answer D is plausible because it reflects the location of a pressure transmitter in the system, PT-47A, which also supplies an alarm, CORE SPRAY A HIGH PRESSURE VALVE LEAK, 9-3-1-C-8. It is wrong because this alarm is to detect a high pressure condition in low pressure piping cause by injection check valve leakage. It is not affiliated with the subject alarm.

Technical References: Alarm Card 9-3-1/A-8 (Rev 37), P&ID drawings B&R 2026 sheet 1 [Nuclear Boiler Instrumentation], B&R 2045 sheet 1 [Core Spray System]

References to be provided to applicants during exam: none

Learning Objective: COR002-06-02 Obj LO-05d

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b)(3),(7)	
Level of Difficulty:	3	
SRO Only Justification:	N/A	

PSA applicability:
N/A

Examination Outline Cross-Reference	**Level**	RO
211000 SLC Ability to predict and/or monitor changes in parameters associated with operating the STANDBY LIQUID CONTROL SYSTEM controls including: (CFR: 41.5) **A1.08 RWCU system lineup**	**Tier#**	2
	Group#	1
	K/A #	211000 A1.08
	Rating	3.7
	Revision	0
Revision Statement:		

Question 34

Reactor Water Cleanup (RWCU) Pump B is in operation.

SLC Pump A control switch is placed in START on Panel 9-5.

Which one of the following describes how the Reactor Water Cleanup (RWCU) system is affected?

A. ONLY Inboard Isolation Vlv MO-15 closes, Pump B trips

B. ONLY Outboard Isolation Vlv MO-18 closes, Pump B trips

C. ONLY Inboard Isolation Vlv MO-15 closes, Pump B remains running

D. ONLY Outboard Isolation Vlv MO-18 closes, Pump B remains running

Answer: A. ONLY Inboard Isolation Vlv MO-15 closes, Pump B trips
Explanation: SLC Pump A control switch on Panel 9-5 isolates inboard isolation valve RWCU-MO-15. When MO-15 reaches the not fully open position, the running RWCU pump will trip. MO-15 is interlocked with both RWCU pumps, since it is a common suction line isolation valve.
Distracters: Answer B is plausible because either MO-15 or MO-18 automatically close when one SLC pump is started from panel 9-5. It is wrong because only SLC Pump B control switch closes MO-18.

Answer C is plausible because each SLC Pump control switch is associated with only one RWCU isolation valve. Also, other RWCU valves, such as Demin Suction Bypass Valve MO-74 and Return Line to Rx Valve MO-68 are not interlocked to directly trip RWCU pumps when they close. The unprepared applicant may believe SLC Pump A actuation is interlocked with only one RWCU pump. They may also not deduce a pump trip would later be generated due to low RWCU flow when the valve fully closed. It is wrong because the SLC Pump control switches are not directly associated with RWCU pump logic. Each RWCU pump control circuit contains series contacts for both MO-15 and MO-18, either of which will trip both RWCU pumps if not fully open. Therefore RWCU pump B will trip due to MO-15 closing upon SLC Pump A start.

Answer D is plausible and wrong for the same reasons given for distractors B and C.

Technical References: Procedure 2.2.66 [Reactor Water Cleanup](Rev 109), GE drawings 791E262 sheet 1, 791E263 sheet 1

References to be provided to applicants during exam: none

Learning Objective: COR002-29-02 obj LO-05f, 08f

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b)(6),(7)	
Level of Difficulty:	3	
SRO Only Justification:	N/A	

PSA applicability:
Top 10 Risk Significant System – Primary Containment Isolation

Examination Outline Cross-Reference	**Level**	RO
212000 RPS Ability to (a) predict the impacts of the following on the REACTOR PROTECTION SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5) **A2.16 Changing mode switch position**	**Tier#** **Group#** **K/A #** **Rating** Revision	2 1 212000 A2.16 4.0 0
Revision Statement:		

Question 35

Plant startup is in progress.

The Reactor Mode Switch is about to be placed to RUN.

(1) Which scram signal is enabled by placing the Reactor Mode Switch in RUN?

AND

(2) What is required to be independently verified IAW Procedure 2.1.1 [Startup Procedure] just before placing the Reactor Mode Switch in RUN?

A. (1) MSIV Closure
(2) Reactor Pressure > 835 psig

B. (1) MSIV Closure
(2) Reactor Water Level < 50 inches

C. (1) Turbine Stop Valve Closure
(2) Reactor Pressure > 835 psig

D. (1) Turbine Stop Valve Closure
(2) Reactor Water Level < 50 inches

Answer: A. (1) MSIV Closure (2) Reactor Pressure > 835 psig
Explanation: Placing the RMS to RUN enables RPS scram signals from APRM downscale with associated IRM upscale or inop and MSIV Closure. Procedure 2.1.1 step 4.23 requires the following independent verifications before placing the RMS to RUN:

APRM downscale lights OFF, APRMs reading between 3.0-11.5%, condenser vacuum > 20”, RPV pressure > 835 psig, and MSL Low Pressure annunciators clear.
Distracters: Answer B part 1 is correct. Part 2 is plausible because operators are required to insert a scram if level cannot be controlled below 50” IAW procedure 2.4RXLVL [RPV Water Level Control Trouble], since the main turbine trips on high water level, Level 8 at 53.5”, which would result in a scram if above 30% power, in RUN mode. It is wrong because procedure 2.1.1 does not require level to be independently verified < 50 inches. Answer C part 1 is plausible because the TSV Closure becomes active only in RUN mode, since it only becomes un-bypassed when turbine 1st stage pressure rises to the equivalent of about 30% power, which can only occur in RUN. It is incorrect because the bypass contact in RPS logic only relates to turbine 1st stage pressure and unaffected by RMS position. Since the transfer from STARTUP to RUN is performed before the turbine is placed in operation and well before 30% power is reached, transferring to RUN in no way results in TSV closure scram becoming active. Part 2 is correct. Answer D part 1 is plausible and wrong for the same reasons as stated for distractor C. Part 2 is plausible and wrong for the same reasons as stated for distractor B.
Technical References: GE dwgs 791E256 sheets 10 and 11; Procedure 2.1.1 [Startup Procedure] step 4.23 (Rev 185)
References to be provided to applicants during exam: none
Learning Objective: COR002-21-02 Obj LO-04e, 05c

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b)(6),(7)	
Level of Difficulty:	2	
SRO Only Justification:	N/A	
PSA applicability:		
Top 10 Risk Significant System - RPS		

Examination Outline Cross-Reference	Level	RO
215003 IRM Ability to monitor automatic operations of the INTERMEDIATE RANGE MONITOR (IRM) SYSTEM including: (CFR: 41.7) **A3.04 Control rod block status**	**Tier#**	2
	Group#	1
	K/A #	215003 A3.04
	Rating	3.5
	Revision	0
Revision Statement:		

Question 36

From IRM B on Panel 9-12:

The plant is in Mode 2.

IRM B is on Range 1.

Which of the following completes the statements below regarding generation of a Control Rod Withdrawal Block by IRM B for these conditions?

A rod block is present if __(1)__ light is ON.

A rod block ___(2)___ present if ONLY the INOP light is ON.

A. (1) Upscale Alarm, ONLY
(2) Is

B. (1) Upscale Alarm, ONLY
(2) Is NOT

C. (1) EITHER Upscale Alarm OR Downscale
(2) Is

D. (1) EITHER Upscale Alarm OR Downscale
(2) Is NOT

Answer: A. (1) Upscale Alarm, ONLY
(2) Is

Explanation:
The amber Upscale Alarm light is indicative of the IRM sensing flux at the control rod block setting of 108/125 divisions. That rod block is enabled in Mode 2 with the Reactor Mode Switch in START & HOT STBY. The IRM downscale rod block is automatically bypassed with IRM B on range 1.

Distracters:
Answer B part 1 is correct. Part 2 is plausible because it is not unusual for operators to see the INOP light ON during power operation, due to maintenance or other conditions, and no rod block be present. The IRM INOP rod block and trip are bypassed with the RMS in RUN. It is wrong because in Mode 2, the RMS is in STARTUP, and an INOP condition will generate a rod block..

Answer C part 1 is plausible because IRM downscale is a familiar rod block in Mode 2. It is wrong because downscale rod blocks are bypassed with the IRM on range 1. The either/or logic of this answer makes the answer wrong since the second part is wrong. Part 2 is correct.

Answer D part 1 is plausible and wrong for the same reason stated for distractor C. Part 2 is plausible and wrong for the same reason stated for distractor B.

Technical References: TRM 3.3.1 [Control Rod Block Instrumentation]

References to be provided to applicants during exam: none

Learning Objective: COR002-12-02 obj LO-05a, 09a

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b)(6)	
Level of Difficulty:	3	
SRO Only Justification:	N/A	

PSA applicability:
N/A

Examination Outline Cross-Reference	**Level**	RO
215004 Source Range Monitor Ability to manually operate and/or monitor in the control room: (CFR: 41.7) **A4.07 Verification of proper functioning / operability**	**Tier#**	2
	Group#	1
	K/A #	215004 A4.07
	Rating	3.4
	Revision	0
Revision Statement:		

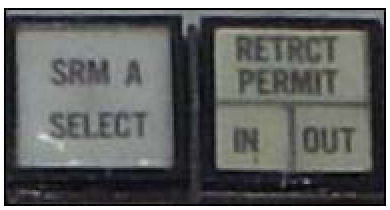
Question 37

All IRMs are on range 2 during STARTUP.

SRM A is being driven into the core and is:

- 9 inches above the core plate.
- 200 cps, slowly rising.

Which of the indicators below will be backlit ON for these conditions?



SRM A SELECT and …

A. IN, only

B. IN and OUT, only

C. RETRCT PERMIT, only

D. IN and OUT and RETRCT PERMIT

Answer: C. RETRCT PERMIT, only
Explanation: IN lit represents the SRM detector is at its fully inserted position, 18" above the core centerline. OUT lit represents the SRM detector is at its fully withdrawn position, 24"

below the core. SRM is in mid travel; therefore, neither IN nor OUT will be illuminated. Since SRM count rate is above 100 cps, Retract Permit will be backlit.

Distracters:
Answer A is plausible for the unprepared applicant who may believe IN represents the detector being *in* the core or driving *in*. The Retract Permit not being on is plausible to the unprepared applicant who cannot remember the setpoint or the logic. This answer is wrong because IN would not be lit since the detector is in mid travel, not fully inserted, and because Retract Permit would be enabled, since count rate is above the setpoint of 100 cps.

Answer B is plausible to the unprepared applicant who does not remember IN on represents fully inserted and OUT on represents fully withdrawn but is familiar with remote valve indication, where both red and green lights are on with the valve in mid travel. The Retract Permit not being on is plausible to the unprepared applicant who cannot remember the setpoint or the logic. This answer is wrong because neither IN nor OUT would be lit since the detector is in mid travel and because Retract Permit would be enabled, since count rate is above the setpoint of 100 cps.

Answer D is plausible to the unprepared applicant who might remember Retract Permit logic but confuses detector position indication, as described for distractor B. This answer is wrong because neither IN nor OUT would be lit since the detector is in mid travel.

Technical References: TRM 3.3.1 [Control Rod Block Instrumentation], Procedure 4.1.1 [Source Range Monitoring System](Rev 23)

References to be provided to applicants during exam: none

Learning Objective: COR002-30-02 obj LO-05f, 06d

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b)(6)	
Level of Difficulty:	3	
SRO Only Justification:	N/A	
PSA applicability:		

Examination Outline Cross-Reference	**Level**	RO
215005 APRM / LPRM **2.1.7** Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation. (CFR: 41.5)	**Tier#**	2
	Group#	1
	K/A #	215005 G2.1.7
	Rating	4.4
	Revision	0
Revision Statement:		

Question 38

The plant has entered the Stability Exclusion Region of the Power-to-Flow Map from 75% power due to a Reactor Recirc malfunction.

Which one of the following completes the statement below regarding indications of abnormal neutron flux oscillations as listed in Procedure 2.4RR [Reactor Recirculation Abnormal], Attachment 3 [Operation in the Stability Exclusion Region].

Abnormal neutron flux oscillations are indicated by:

LPRM upscale or downscale indications are alarming and clearing (annunciators or full core display indicators) with an annunciation period of ___(1)___ 3 seconds,

And/or

Sustained rising oscillations on APRMs reaching two or more times its initial peak-to-peak level _____(2)_________ .

A. (1) less than
(2) **after** the core flow reduction

B. (1) less than
(2) **before** the core flow reduction

C. (1) greater than
(2) **after** the core flow reduction

D. (1) greater than
(2) **before** the core flow reduction

Answer: A. (1) less than (2) **after** the core flow reduction
Explanation: 2.4RR Att. 3 is required to be performed if a core flow reduction results in entry into the Stability Exclusion Region of the Power-to-Flow Map. A note at Att. 3 step 1.2 lists the annunciation period of LPRM upscale and downscale alarms reflective of

abnormal neutron flux oscillations as less than 3 seconds. It also states the increase in peak-to-peak value of APRM oscillations reflective of abnormal neutron flux oscillations is two or more times the initial level of oscillations following the flow reduction, when flow first settled after entry into the region.

Distracters:
Answer B part 1 is correct. Part 2 is plausible because the normal peak-to-peak value of APRM oscillation is proportional to power. Oscillations are larger at 75% than at the power level following a flow reduction. The unprepared applicant might choose this answer believing oscillations would have to be larger than those *normally* seen to be considered abnormal. It is wrong because the intent is to recognize the incipient stages of limit cycle oscillations, which should be gauged against the amplitude of oscillations that are normal for the current (i.e. reduced) power/flow conditions.

Answer C part 1 is plausible to the unprepared applicant who does not understand the phenomenon of limit cycle oscillations or the strategy for detection. To that applicant, less than and greater than have equal attraction. Or, the unprepared applicant might choose this answer because he might conclude "greater" implies "worse". It is also plausible because limit cycle oscillations manifest in a discrete frequency range that has a lower limit. This answer is wrong because 2.4RR specifically defines the period as less than 3 seconds. Part 2 is correct.

Answer D part 1 is plausible and wrong for the same reasons as listed for distractor C. Part 2 is plausible and wrong for the same reasons as listed for distractor B.

Technical References: Procedure 2.4RR [Reactor Recirculation Abnormal], Attachment 3 [Operation in the Stability Exclusion Region](Rev 42)

References to be provided to applicants during exam: none

Learning Objective: INT032-01-24 EO-H

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X

Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	

10CFR Part 55 Content:	55.41(b)(5),(6),(10)
Level of Difficulty:	2
SRO Only Justification:	N/A

PSA applicability:
N/A

Examination Outline Cross-Reference	**Level**	RO
217000 RCIC Knowledge of the physical connections and/or cause/effect relationships between REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) and the following: (CFR: 41.2 to 41.9) **K1.02 Nuclear boiler system**	**Tier#**	2
	Group#	1
	K/A #	217000 K1.02
	Rating	3.5
	Revision	0
Revision Statement:		

Question 39

Regarding RCIC piping connections to the Nuclear Boiler System,…

(1) Which Main Steam Line supplies steam to drive the RCIC turbine?

AND

(2) Which Feedwater Line receives RCIC pump discharge?

A. (1) A
(2) A

B. (1) A
(2) B

C. (1) C
(2) A

D. (1) C
(2) B

Answer: C. (1) C
(2) A

Explanation:
The RCIC steam supply line branches off of MSL C in the drywell downstream of SRVs 71E and 71F and upstream of MSIV 80C. The RCIC injection line ties into FW line A, via a common pipe with RWCU return, between the outboard FW line A check valve RF-15CV and the drywell.

Distracters:

Answer A part 1 is plausible because RCIC injects to FW Line A. The unprepared applicant could confuse the MSL with the FW line associated with RCIC. It is wrong because steam is supplied from MSL C. Part 2 is correct. Answer B part 1 is plausible and wrong for the same reasons as stated for distractor A. Part 2 is plausible because HPCI injects via the B FW line. The unprepared applicant might confuse HPCI with RCIC. It is wrong because RCIC injects into the A FW line. Answer D part 1 is correct. Part 2 is plausible and wrong for the same reasons as stated for distractor B.		
Technical References: B&R dwgs 2041, 2043		
References to be provided to applicants during exam: none		
Learning Objective: COR002-18-02 obj LO-05b,05k, 05l		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b)(7)	
Level of Difficulty:	2	
SRO Only Justification:	N/A	
PSA applicability:		
Top 10 Risk Significant System - RCIC		

Examination Outline Cross-Reference	Level	RO
218000 ADS Knowledge of electrical power supplies to the following: (CFR: 41.7) **K2.01 ADS logic**	**Tier#**	2
	Group#	1
	K/A #	218000 K2.01
	Rating	3.1
	Revision	0
Revision Statement:		

Question 40

What is the backup power supply to Div 2 ADS Logic?

A. CPP

B. NBPP

C. 125 VDC Battery A

D. 125 VDC Battery B

Answer: C. 125 VDC Battery A

Explanation:
The normal power supply to Div 2 ADS logic is 125 VDC panel AA2. The normal power supply to Div 2 ADS logic is 125 VDC panel BB2. If Div 1 ADS logic loses power, there is no automatic transfer to another power source. If Div 2 ADS logic loses power, it will automatically transfer to its backup source, 125 VDC panel AA2, supplied from 125 VDC Battery A, via contact repositioning resulting from de-energization of relay 2E-K1B.

Distracters:
Answer A is plausible because Critical Power Panel (CPP) is supplied by safety related AC power and supplies some critical instrument and control power. The unprepared applicant might not know ADS Logic is DC powered and select this answer. This answer is wrong because Div 2 ADS logic is automatically backed up by power from 125 VDC battery A, via panel AA2.

Answer B is plausible because No Break Power Panel (NBPP), supplied by an inverter, supplies uninterruptible AC power to some important instrument and control circuits during a failure of normal AC power. The unprepared applicant might not know ADS Logic is DC powered and select this answer. It is wrong because 125 VDC

panel AA2, supplied from 125 VDC Battery A, is the backup power supply for Div 2 ADS Logic. Answer D plausible because there is no advantage to the unprepared applicant for selecting one DC power supply versus the other. The unprepared applicant who does not know the normal power supply to Div 2 ADS Logic is Battery B or who knows it is panel BB2 might conclude the backup power would come from another Div 2 125 VDC panel, such as BB3, in order to maintain divisional separation and, thereby, select this answer. This answer is wrong because Div 2 ADS logic is automatically backed up by power from 125 VDC battery A, via panel AA2.		
Technical References: GE drawings 791E253 sheets 01 and 02		
References to be provided to applicants during exam: none		
Learning Objective: COR002-16-02/ COR002-16-01 obj LO-02a, 08f		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b)(7)	
Level of Difficulty:	3	
SRO Only Justification:	N/A	
PSA applicability:		
Top 10 Risk Significant System – ADS/SRV Top 10 Risk Significant Components – 125 VDC Distribution Panels AA2/BB2		

Examination Outline Cross-Reference	**Level**	RO
223002 PCIS/Nuclear Steam Supply Shutoff Knowledge of the effect that a loss or malfunction of the PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF will have on following: (CFR: 41.7) **K3.20 Standby gas treatment system**	**Tier#**	2
	Group#	1
	K/A #	223002 K3.20
	Rating	3.3
	Revision	0
Revision Statement:		

Question 41

Standby Gas Treatment (SGT) A exhaust fan EF-R-1E control switch on VBD-K is in STANDBY.

SGT B exhaust fan B EF-R-1F control switch on VBD-K is in AUTO.

Wide Range reactor water level indicating switches NBI-LIS 57A and NBI-LIS-57B both sense -60 inches due to a leak on common instrument tubing.

What is the effect of this condition on SGT A and B exhaust fans?

A. SGT A starts immediately
SGT B does not start

B. SGT A does not start
SGT B starts immediately

C. SGT A does not start
SGT B starts after approximately 30 seconds

D. SGT A starts after approximately 30 seconds
SGT B starts immediately

Answer: B. SGT A does not start SGT B starts immediately
Explanation: SGT receive start signals from Group 6 isolation logic. The RPV water level low, Level 2, PCIS logic is arranged such that trip of LIS-57A or 58A combined with trip of LIS-57B or 58B initiates a Group 6 isolation in PCIS, which will cause SGT in the AUTO mode to start. Any SGT in STBY mode, in this case SGT A, does not immediately start. SGTs in STBY will auto start only if the following conditions are

met: a Group 6 isolation/initiation signal is received AND low flow (<800 scfm) in the opposite SGT subsystem AND ~30 second time delay. Since SGT B immediately starts, flow would be developed above its low flow setpoint before the 30 second timer elapsed; therefore, SGT A would remain idle. NBI-LIS 57A and NBI-LIS-57B share common reference leg and variable leg piping, so this is a credible failure mode.

Distracters:
Answer A is plausible to the unprepared applicant who confuses the effects of control switch position for SGT fans and reverses the AUTO and Standby functions. It is wrong because SGT B, in AUTO, starts immediately, and SGT A, in STANDBY, will not start at all.

Answer C is plausible to the unprepared applicant who does not understand SGT start logic and/or confuses the effects of control switch position for SGT fans and reverses the AUTO and Standby functions for SGT B. It is wrong because SGT B, in AUTO, starts immediately, and SGT A, in STANDBY, will not start at all.

Answer D is plausible to the unprepared applicant who does not anticipate the SGT that immediately starts will develop flow greater than the low flow setpoint before the 30 second timer has elapsed. It is wrong because SGT A, in STANDBY, will not start at all, since low flow in SGT B will not exist for 30 seconds following the initiation signal.

Technical References: Alarm card K-2/D-2 (Rev 8), Procedure 2.2.73 [Standby Gas Treatment System](Rev 52), GE dwg 791E266 sheets 05 and 06, Lesson Plan COR002-28-02 [SGT System](Rev 22)

References to be provided to applicants during exam: none

Learning Objective: COR002-28-02 obj LO-05g, 08a, 13a

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b)(7)	
Level of Difficulty:	3	
SRO Only Justification:	N/A	

PSA applicability:
N/A

Examination Outline Cross-Reference	Level	RO
239002 SRVs Knowledge of the operational implications of the following concepts as they apply to RELIEF/SAFETY VALVES: (CFR: 41.5) **K5.02 Safety function of SRV operation**	**Tier#**	2
	Group#	1
	K/A #	239002 K5.02
	Rating	3.7
	Revision	0
Revision Statement:		

Question 42

The plant is at rated power.

A pressure transient occurs, resulting in reactor pressure peaking at 1280 psig.

Low-Low-Set **fails** to arm.

(1) How many total SRVs/SVs open during this transient?
(Assume median Tech Spec allowable value for lift setpoints.)

AND

(2) At what RPV pressure will the last SRV/SV automatically close?

A. (1) 8
(2) 835-875 psig

B. (1) 8
(2) 960-1050 psig

C. (1) 11
(2) 835-875 psig

D. (1) 11
(2) 960-1050 psig

Answer: D. (1) 11 (2) 960-1050 psig
Explanation: There are eight Safety Relief Valves (SRV) that can operate either in electro-pneumatic assisted mode via panel 9-5 control switch, or in mechanical-only safety mode. Six of the SRVs can also operate automatically in electro-pneumatic assisted ADS mode. Two of the SRVs can also operate automatically in electro-pneumatic assisted Low-Low Set (LLS) mode. LLS is armed when one SRV opens, detected by its discharge pressure switch, coincident with RPV pressure ≥1050 psig. There are

also three Safety Valves (SV) that operate only in mechanical mode. SRV safety function opening setpoints are 1080+/-3%, 1090+/-3%, and 1100+/-3%. SV opening setpoints are 1240+/-3%. SVs have this highest setpoint, and the TS allowable value for the SV lift setpoint is 1240+/-37.2 psig. Since reactor pressure reaches 1280 psig in the stem, all SRVs and SVs will open, a total of 11 valves.

Regarding part 2, the stem states LLS fails to function; therefore, SRVs operate only in safety/mechanical mode. In safety mode, a SRV will close at a RPV pressure that is 30-120 psig below the SRV opening setpoint. The lowest SRV safety function lift setpoint median value is 1080 psig, applicable to two SRVs; therefore, the last two SRVs will close 30-120 psig below 1080 psig, or in the range of 960-1050 psig.

Distracters:
Answer A part 1 is plausible because there are control switches for 8 SRVs on panel 9-3, associated with the 8 SRVs that can function in either electro-pneumatic or safety mode. The unprepared applicant who does not consider operation of the safety valves or does not know the SV lift setpoints would choose this answer. It is wrong because the TS allowable value for the SV lift setpoint is 1240 psig. Since reactor pressure reaches 1280 psig in the stem, all SRVs and SVs will open, a total of 11 valves. Part 2 is plausible because it reflects the Low-Low Set reset setpoint allowable value range listed in TS Table 3.3.6.3-1. It is wrong because the stem states LLS failed to ARM; therefore, SRVs operate only in safety relief mode. The lowest SRV safety function lift setpoint median value is 1080 psig, applicable to two SRVs; therefore, the last two SRVs will close 30-120 psig below 1080 psig, or in the range of 960-1050 psig.

Answer B part 1 is plausible and wrong for the same reasons as given for distractor A. Part 2 is correct.

Answer C part 1 is correct. Part 2 is plausible and wrong for the same reasons as given for distractor A.

Technical References: Operations Instruction #8 [Guideline for Successful Transient Mitigation](Rev 14), TS 3.3.6.3 [LLS Instrumentation], TS 3.4.3 [SRVs and SVs], Lesson Plan COR002-16-02 [Nuclear Pressure Relief Systems](Rev 19), Procedure 2.2.1 [Nuclear Pressure Relief System](Rev 38)

References to be provided to applicants during exam: none

Learning Objective: COR002-16-02/ COR002-16-01 Obj LO-05j, 06c, 07d, 12c

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X

Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b)(7)	
Level of Difficulty:	3	
SRO Only Justification:	N/A	
PSA applicability:		
Top 10 Risk Significant System – ADS and Pressure Relief		

Examination Outline Cross-Reference	Level	RO
259002 Reactor Water Level Control Knowledge of the effect that a loss or malfunction of the following will have on the REACTOR WATER LEVEL CONTROL SYSTEM: (CFR: 41.7) **K6.02 A.C. power**	**Tier#**	2
	Group#	1
	K/A #	259002 K6.02
	Rating	3.3
	Revision	0
Revision Statement:		

Question 43

How is the Reactor Vessel Level Control System (RVLCS) affected by the loss of power panel CCP-1A?

A. Loss of two RVLCS HMIs on Panel 9-5 and Panel A

B. Loss of four RVLCS HMIs on Panel 9-5 and Panel A

C. Downscale failure of RPV level transmitter NBI-LT- 52A

D. Downscale failure of steam flow transmitter MS-FT-51A

Answer: A. Loss of two RVLCS HMIs on Panel 9-5 and Panel A

Explanation:
120 VAC power Panel CCP-1A is supplied by MCC-LX via distribution panel CDP1A. In the RVLCS, CCP-1A supplies power to HMIs RFC-CS-RFPT1A, RFPT-1A Primary Control Station HMI on Panel 9-5 (apron section) and RFC-CS-RFPT2A, RFPT-2A Secondary Control Station HMI on Panel A (vertical section). Loss of AC power to the two HMIs does not affect any other RVLCS function, only the operators' ability to use the particular HMIs for indication and control.

Distracters:
Answer B is plausible because four FW control HMIs are AC powered from CCP's. It is wrong because two HMIs, RFC-CS-RFPT1B and RFC-CS-RFPT2B, are powered from CCP-1B, not CCP-1A. CCP-1B is supplied by MCC-TX via distribution panel CDP1B.

Answer C is plausible because it is an electrically powered component of the RVLCS and, like the correct answer, is AC powered. It is wrong because NBI-LT- 52A is powered from NBPP, not CCP-1A.

Answer D is plausible because it is an electrically powered component of the RVLCS. It is wrong because MS-FT-51A is powered from DC distribution panel AA2 or BB2, not CCP-1A.

Technical References: GE Dwg 791E257 sheet 2, Procedure 2.2.23 [120/240 VAC Instrument Power Distribution System](Rev 62), Procedure 2.2A_125DC.DIV1 [125 VDC Power Checklist (Div 1)](Rev 7), Procedure 2.2A_125DC.DIV2 [125 VDC Power Checklist (Div 2)](Rev 7), Procedure 2.2A_120CRIT.DIV1 [120/240 VAC Critical Instrument Power Checklist (Div 1)](Rev 10), Procedure 2.2A_120CRIT.DIV2 [120/240 VAC Critical Instrument Power Checklist (Div 2)](Rev 25), Lesson Plan COR002-32-02 [RVLCS](Rev 19)

References to be provided to applicants during exam: none

Learning Objective: COR002-32-02 Obj LO-04a, 04b

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b)(7)	
Level of Difficulty:	3	
SRO Only Justification:	N/A	

PSA applicability:
N/A

Examination Outline Cross-Reference	**Level**	RO
259002 Reactor Water Level Control Ability to predict and/or monitor changes in parameters associated with operating the REACTOR WATER LEVEL CONTROL SYSTEM controls including: (CFR: 41.5) **A1.05 FWRV/startup level control position: Plant-Specific**	**Tier#**	2
	Group#	1
	K/A #	259002 A1.05
	Rating	2.9
	Revision	0
Revision Statement:		

Question 44

Plant startup is in progress.

The Feedwater Flow System is in Feedwater Sequence Mode 2.

Reactor Feed Pump (RFP) A is in service feeding through both Feedwater Startup Flow Control Valves (RF-FCV).

- RF-FCV-11AA is 30% open AND is selected as LEAD.
- RF-FCV-11BB is 30% open

How do Startup FCVs respond if RFP A speed demand is continually raised?

A. FCV-11AA and FCV-11BB will modulate closed simultaneously.

B. FCV-11AA will fully close, THEN FCV-11BB will modulate closed.

C. FCV-11BB will fully close, THEN FCV-11AA will modulate closed.

D. FCV-11BB will close to 20%, THEN FCV-11AA will modulate closed.

Answer: A. FCV-11AA and FCV-11BB will modulate closed simultaneously.

Explanation:
Startup FCVs are controlled by the RVLCS based on narrow range reactor water level. When two FCVs are in service, one is designated as Lead and the other as Follow. In Sequence Mode 2, Startup FCVs will modulate to maintain the reactor level setpoint. When RFP speed is raised, discharge pressure rises causing increased Feedwater flow to the RPV. The Startup FCVs both respond to the level rise by modulating closed.

When Startup FCVs are first placed into operation, the FCV designated a LEAD has a 15% bias to open. The LEAD valve opens to approximately 15%, then the other valve will begin to open. The 15% bias is ramped to 0% as the LEAD valve opens from 15% to 20%. With the FCVs initially at 30%, the bias has been ramped to 0%, so both valves respond simultaneously to the level demand.

Distracters:

Answer B is plausible because Startup FCVs have the capability to be biased such that one valve reacts to level changes before the other to allow for finer level control. When first placed into operation, the lead startup valve bias is set to 20%. When the lead startup valve reaches ~ 15% open the bias setting is ramped to 0%. This answer is wrong because the Startup FCVs are initially at 30%, where the bias setting has been set to 0%; therefore, both valves respond together.

Answer C is plausible because Startup FCVs have the capability to be biased such that one valve reacts to level changes before the other to allow for finer level control. When first placed into operation, the lead startup valve bias is set to 20%. When the lead startup valve reaches ~ 15% open the bias setting is ramped to 0%. This distractor is plausible, like distractor A, because the bias can be positive or negative, such that either the lead or non-lead valve responds first. This answer is wrong because the Startup FCVs are initially at 30%, where the bias setting has been set to 0%; therefore, both valves respond together.

Answer D is plausible because the lead Startup FCV has a 20% bias to open, which is ramped to 0% as the valve opens from 15% to 20%. The unprepared applicant may recognize 20% and select this answer. This answer is wrong because the Startup FCVs are initially at 30%, where the bias setting has been set to 0%; therefore, both valves respond together.

Technical References: Procedure 2.2.28 [Feedwater System Startup and Shutdown](Rev 105), Procedure 2.2.28.1 [Feedwater System Operation](Rev 93)

References to be provided to applicants during exam: none

Learning Objective: COR002-02-02 Obj. LO-10g; COR002-32-02 Obj. LO-07e

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X

Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X

10CFR Part 55 Content:	55.41(b)(4)

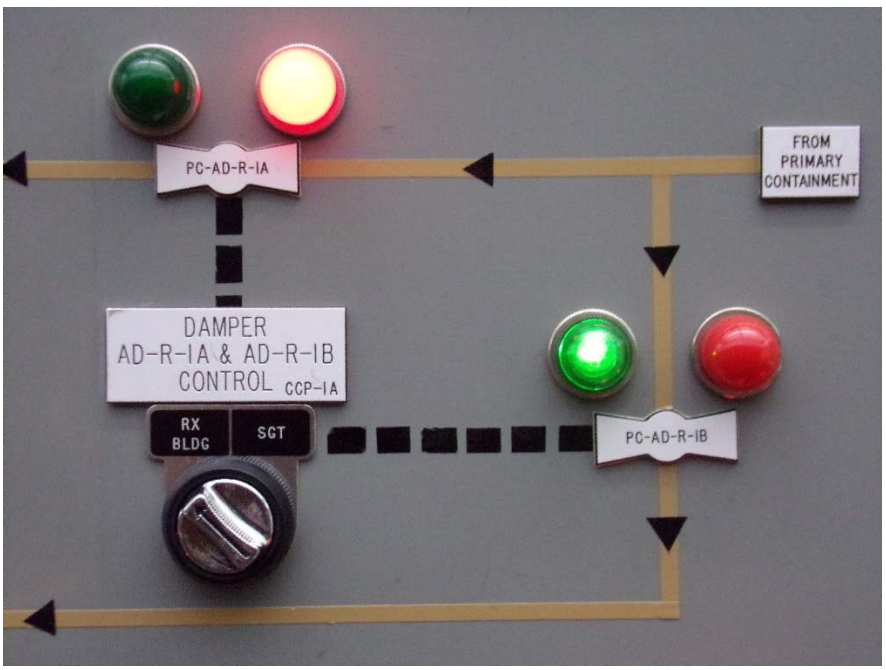
Level of Difficulty:	3
SRO Only Justification:	N/A
PSA applicability:	
N/A	

Examination Outline Cross-Reference	Level	RO
261000 SGTS Ability to (a) predict the impacts of the following on the STANDBY GAS TREATMENT SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5) **A2.11 High containment pressure**	**Tier#**	2
	Group#	1
	K/A #	261000 A2.11
	Rating	3.2
	Revision	0
Revision Statement:		

Question 45

Inerting of the primary containment is in progress during startup.

Containment ventilation is aligned as depicted below on VBD-K:



Then, a steam leak causes Drywell pressure to rise rapidly to 5 psig.

(1) What is the response of dampers PC-AD-R-1A and PC AD-R-1B, pictured above?

AND

(2) IF pressure on HV-DPR-835, RX BLDG/ATMOS DP (VBD-R), is being maintained at - 0.10" wg, what Standby Gas Treatment System (SGT) alignment is required for these conditions IAW Procedure 2.1.22 [Recovering from a Group Isolation]?

A. (1) PC-AD-R-1A closes, and PC AD-R-1B opens
(2) operate both SGT Exhaust Fans

B. (1) PC-AD-R-1A closes, and PC AD-R-1B opens
(2) operate ONLY the preferred SGT Exhaust Fan

C. (1) PC-AD-R-1A remains open, and PC AD-R-1B opens
(2) operate both SGT Exhaust Fans

D. (1) PC-AD-R-1A remains open, and PC AD-R-1B opens
(2) operate ONLY the preferred SGT Exhaust Fan

Answer: A. (1) PC-AD-R-1A closes, and PC AD-R-1B opens
(2) operate both SGT Exhaust Fans

Explanation:
The alignment of AD-R-1A and 1B pictured is normal during startup while PC inerting is in progress. Drywell Pressure High, 1.84 psig, is a Group 6 isolation and SGT auto start signal. Upon SGT auto start, AD-R-1B, SGT Suction from PC, automatically opens to enable venting PC through SGT, and AD-R-1A automatically closes.

Procedure 2.1.22 [Recovering from a Group Isolation] step 9.4 directs aligning SGT IAW Procedure 2.2.73 [Standby Gas Treatment System]. Procedure 2.2.73 section 4, for response to automatic initiation, states to stop the non-preferred SGT Exhaust Fan by placing it in STBY, but only if pressure on HV-DPR-835, RX BLDG/ATMOS DP (VBD-R), is being maintained ≤ **-** 0.25" wg. The procedure also requires monitoring RB pressure if one SGT is stopped, and if pressure rises above – 0.25” wg, directs restarting the idle SGT Exhaust Fan. In this case, -0.10” wg is above -0.25” wg, so both SGT subsystems are required to be operated.

Distracters:
Answer B part 1 is correct. Part 2 is plausible because Procedure 2.2.73 section 4, for response to automatic initiation, states to stop the non-preferred SGT Exhaust Fan by placing it in STBY, but only if pressure on HV-DPR-835, RX BLDG/ATMOS DP (VBD-R), is being maintained ≤ **-** 0.25" wg. It is wrong because RB pressure is – 0.10” wg, which is greater than – 0.25” wg.

Answer C part 1 is plausible because the control switch pictured is in RX BLDG position. The unprepared applicant may believe damper PC-AD-R-1A is controlled by switch position, only. It is wrong because the switch provides a permissive to open the damper, if a SGT initiation signal is absent. SGT initiation logic automatically closes PC-AD-R-1A. Part 2 is correct

Answer D part 1 is plausible and wrong for the same reasons as given for distractor C. Part 2 is plausible and wrong for the same reasons as given for distractor B.

Technical References: Procedure 2.1.22 [Recovering from a Group Isolation](Rev 60), Lesson Plan COR002-28-02 [SGT System](Rev 22)		
References to be provided to applicants during exam: none		
Learning Objective: COR002-28-02 Obj LO-08a, 07a		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b)(7)	
Level of Difficulty:	3	
SRO Only Justification:	N/A	
PSA applicability:		
Top 10 Risk Significant System – Primary Containment (Isolation)		

Examination Outline Cross-Reference	Level	RO
262001 AC Electrical Distribution Ability to monitor automatic operations of the A.C. ELECTRICAL DISTRIBUTION including: (CFR: 41.7) **A3.04 Load sequencing**	**Tier#**	2
	Group#	1
	K/A #	262001 A3.04
	Rating	3.4
	Revision	0
Revision Statement:		

Question 46

Which RHR pumps sequentially load at 5 seconds for a Loss of Offsite Power with LPCI initiation?

A. A and B

B. B and C

C. C and D

D. A and D

Answer: B. B and C
Explanation: With a loss of off-site power, pumps A and D start immediately upon restoration of power, and pumps B and C start 5 seconds after the restoration of power. This time delay prevents overloading the emergency diesels due to ECCS initiation.
Distracters: The group of answers involve each RHR pump twice, enhancing plausibility of distractors. Answer A is plausible because, like the correct answer, it lists two RHR pumps. It is wrong because RHR pump A has a sequence time of 0 seconds, not 5 seconds. Answer C is plausible because, like the correct answer, it lists two RHR pumps. It is wrong because RHR pump D has a sequence time of 0 seconds, not 5 seconds.

Answer D is plausible because, like the correct answer, it lists two RHR pumps. It is wrong because both RHR pumps A and D have a sequence time of 0 seconds, not 5 seconds.		
Technical References: Procedure 2.2.69 [Residual Heat Removal System](Rev 98)		
References to be provided to applicants during exam: none		
Learning Objective: COR002-23-02 Obj LO-03f		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b)(7)	
Level of Difficulty:	2	
SRO Only Justification:	N/A	
PSA applicability:		
Top 10 Risk Significant System - RHR		

Examination Outline Cross-Reference	Level	RO
262001 AC Electrical Distribution Ability to manually operate and/or monitor in the control room: (CFR: 41.7) **A4.05 Voltage, current, power, and frequency on A.C. buses**	**Tier#**	2
	Group#	1
	K/A #	262001 A4.05
	Rating	3.3
	Revision	0
Revision Statement:		

Question 47

A 4160 VAC emergency bus is being supplied **SOLELY** by its respective Diesel Generator during surveillance testing.

Which bus parameter is affected **most** as the result of placing the DG governor control switch to RAISE?

A. VARs

B. Voltage

C. Kilowatts

D. Frequency

Answer: D. Frequency
Explanation: With the diesel alone carrying its bus, the engine governor control will change the speed of the engine and thereby change the operating speed/frequency of the generator. There may theoretically be negligible effects on other parameters, but frequency is affected most.
Distracters: Answer A is plausible because if the DG was paralleled to the grid via its respective bus, VARs would be affected by raising the governor control, since that would raise load. It is wrong because, in this case, the DG is carrying the bus alone, separated from the grid. So raising DG speed raises frequency on the bus. Bus load is not affected, since component loads remain constant. Therefore, VARs is not affected.

Answer B is plausible because the unprepared applicant may confuse voltage regulator control switch with the governor control switch. It is wrong because the governor control switch changes engine speed, which directly affects generator speed, and therefore, frequency

Answer C is plausible because DG KW, not frequency, would be affected most if the DG was paralleled to the grid via its respective bus. Raising the governor control would raise DG load, since grid frequency would dominate. It is wrong because, in this case, the DG is carrying the bus alone, separated from the grid. So raising DG speed raises frequency on the bus. Bus load is not affected, since component loads remain constant.

Technical References: Procedure 2.2.18 [4160v Auxiliary Power Distribution System](Rev 208), Procedure 2.2.20.1 [Diesel Generator Operations](Rev 69)

References to be provided to applicants during exam: none

Learning Objective: COR002-08-02 LO-10d

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b)(8)	
Level of Difficulty:	3	
SRO Only Justification:	N/A	

PSA applicability:

Top 10 Risk Significant Systems – Emergency AC Power

Examination Outline Cross-Reference	Level	RO
262002 UPS (AC/DC) **2.1.27** Knowledge of system purpose and/or function. (CFR: 41.7)	**Tier#**	2
	Group#	1
	K/A #	262002 G2.1.27
	Rating	3.9
	Revision	0
Revision Statement:		

Question 48

What completes the statements below regarding the purpose of the Plant Management Information System Uninterruptible Power Supply (PMIS-UPS)?

The PMIS-UPS supplies electrical **output** at ___(1)____ to the PMIS-UPS Main Panel. If all AC power is lost, PMIS-UPS batteries are designed to power PMIS for a minimum of ___(2)____ .

A. (1) 125 VDC
   (2) 4 hours

B. (1) 125 VDC
   (2) 30 minutes

C. (1) 208 VAC
   (2) 4 hours

D. (1) 208 VAC
   (2) 30 minutes

Answer: D. (1) 208 VAC (2) 30 minutes
Explanation: Power to PMIS UPS Main panel is normally supplied from the 12.5 kV distribution system at the multi-purpose facility (MPF) panel MDP-2 through an automatic transfer switch to a battery charger. The battery charger's DC output is sent to a 75 kVA inverter and supplies a trickle charge to the battery. The inverter converts the DC voltage to 208 VAC and supplies PMIS UPS main panel. The PMIS-UPS assures a 30 minute source of power to the PMIS computer on a loss of an AC voltage. Normally the battery charger rectifies the AC power from normal source (MDP2) or the backup source (MCC-L) to supply DC to the battery to maintain it charged and to the inverter to be converted to AC to power the PMIS computers. But the battery charger would be de-energized in a loss of ALL AC power.

Distracters:
Answer A part 1 is plausible because the PMIS-UPS batteries are 125 VDC and because some computer electronics utilize DC power. (Two validators chose 125 VDC.) It is wrong because the PMIS-UPS output supplied to the Main Panel is 208 VAC. 125 VAC passes through a 75kva inverter to produce 208 VAC. The unprepared applicant who does not understand the purpose/function of a UPS might select this answer. Part 2 is plausible because Div 1 and 2 station batteries are rated to supply loads for a minimum of 4 hours. It is wrong because the PMIS-UPS batteries are rated for 30 minutes, since PMIS is not essential during a station blackout.

Answer B part 1 is plausible and wrong for the same reasons as stated for distractor A. Part 2 is correct.

Answer C part 1 is correct. Part 2 is plausible and wrong for the same reasons as stated for distractor A.

Technical References: Procedure 2.2.63 [PMIS Uninterruptible Power Supply System] Att. 2, step 1.2.1 (Rev 14)

References to be provided to applicants during exam: none

Learning Objective: COR-001-01-01 LO-02h

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b)(8),(10)	
Level of Difficulty:	3	
SRO Only Justification:	N/A	

PSA applicability:
N/A

Examination Outline Cross-Reference	Level	RO
263000 DC Electrical Distribution Knowledge of the physical connections and/or cause/effect relationships between D.C. ELECTRICAL DISTRIBUTION and the following: (CFR: 41.2 to 41.9) **K1.02 Battery charger and battery**	**Tier#**	2
	Group#	1
	K/A #	263000 K1.02
	Rating	3.2
	Revision	0
Revision Statement:		

Question 49

The plant is at rated power.

The AC INPUT breaker on 125V Charger 1A trips open due to a charger failure.

Electrical Maintenance has determined the fault is isolated to 125V Charger 1A.

(1) How does 125V DC BUS 1A voltmeter indication on panel-C respond when Charger 1A trips?

AND

(2) Which action is allowed to be performed IAW Procedure 2.2.25 [125 VDC Electrical System (Div 1)]?

A. (1) Remains constant
(2) Place 125V Charger 1C in service on 125V DC bus 1A.

B. (1) Remains constant
(2) Transfer 125 VDC Distribution Panel A to its emergency source.

C. (1) Lowers by approximately 5 VDC
(2) Place 125V Charger 1C in service on 125V DC bus 1A.

D. (1) Lowers by approximately 5 VDC
(2) Transfer 125 VDC Distribution Panel A to its emergency source.

Answer: C. (1) Lowers by approximately 5 VDC (2) Place 125V Charger 1C in service on 125V DC bus 1A.
Explanation: The in-service charger normally supplies charging current to the battery and to connected DC loads approximately 130.5 VDC, about 5 VDC greater than battery voltage when the battery is fully charged. Charger output voltage is slightly higher than battery voltage in order to supply current for the DC loads without resulting in

battery discharge. When the charger is de-energized by opening the input breaker, voltage displayed on panel-C drops to the existing battery voltage, which is approximately 5 VDC less by simulator modeling. With 125 VDC charger 1A faulted, 125 VDC bus 1A should be resupplied using swing 125V charger 1C; otherwise, battery 1A voltage will decay and the battery will become non-functional.

Distracters:
Answer A part 1 is plausible because the unprepared applicant might not understand the charger voltage has to be maintained higher than battery voltage in order to supply loads and maintain battery voltage; or he may believe the panel-C indication reflects only battery voltage; therefore, they may believe voltage indication on panel C would not change. It is wrong because the meter measures bus voltage and will drop from the charging level to the battery level, which is approximately 5-10 VDC lower. Part 2 is correct.

Answer B part 1 is plausible and wrong for the same reasons as stated for distractor A. Part 2 is plausible because there is transfer capability from the normal supply, 125 VDC battery/charger 1A, to the emergency supply, 125 VDC battery/charger 1B. It is wrong because, per 2.2.25.1 caution at step 34.1, this not allowed in Modes 1, 2, and 3 unless directed by EOPs or abnormal procedures, because it constitutes cross-connecting safety divisions.

Answer D part 1 is correct. Part 2 is plausible and wrong for the same reasons as stated for distractor B.

Technical References: procedure 2.2.25.1 [125 VDC ELECTRICAL SYSTEM (DIV 1)](Rev 21); Alarm Card C-1/C-2 (Rev 30)

References to be provided to applicants during exam: none

Learning Objective: COR002-07-02 LO-06c, 8e, 9b,

Question Source:	Bank #	new bank #107
(note changes; attach parent)	Modified Bank #	
	New	

Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X

10CFR Part 55 Content:	55.41(b)(8),(10)
Level of Difficulty:	3
SRO Only Justification:	N/A

PSA applicability:
Top 10 Risk Significant System – Emergency DC Power
Top 10 Risk Significant Component – 125 VDC Bus 1A

Examination Outline Cross-Reference	Level	RO
264000 EDGs Knowledge of the effect that a loss or malfunction of the EMERGENCY GENERATORS (DIESEL/JET) will have on following: (CFR: 41.7 / 45.4) **K3.03 Major loads powered from electrical buses fed by the emergency generator(s)**	**Tier#**	2
	Group#	1
	K/A #	264000 K3.03
	Rating	4.1
	Revision	0
Revision Statement:		

Question 50

A loss of offsite power has occurred.

Which major load would be de-energized as a result of a trip of DG2?

A. RHR Pump B

B. Reactor Equipment Cooling Pump B

C. Turbine Equipment Cooling Pump B

D. RHR Service Water Booster Pump B

Answer: D. RHR Service Water Booster Pump B

Explanation:
DG2 supplies 4160 VAC to bus 1G during a loss of offsite power. 4160 VAC Bus 1G supplies power to RHR SW Booster Pumps B and D. If DG2 tripped, 4160 VAC bus 1G and its supplied loads, including RHR SW Booster Pump B, would de-energize.

Distracters:
Answer A is plausible because RHR Pumps C and D are powered from bus 1G/DG2. It is wrong because RHR Pump B is powered from 4160 VAC bus 1F, which is not powered from DG2.

Answer B is plausible because it, like the correct answer, represents a “B” labeled major load. It is wrong because REC pump B is powered from 480 VAC MCC K via 480 VAC bus 1F, which is powered from 4160 VAC bus 1F and DG1, not 1G and DG2.

Answer C is plausible because it, like the correct answer, represents a "B" labeled major load. It is wrong because TEC pump B is powered from 480 VAC bus 1A, which is powered from 4160 VAC bus 1A, not 1G.		
Technical References: Procedure 2.2A.RHRSW.DIV2 [RHR Service Water Booster Pump System Component Checklist (Div 2)](Rev 8), Procedure 2.2A.REC.DIV1 [Reactor Equipment Cooling Water System Component Checklist (Div 1)](Rev 0), Procedure 2.2A_480.TG [480 VAC Turbine Building Breaker Checklist](Rev 26), Procedure 2.2A_4160.DIV1 [4160 VAC Auxiliary Power Checklist (Div 1)](Rev 1)		
References to be provided to applicants during exam: none		
Learning Objective: COR002-27-02 Obj LO-02b		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b)(8)	
Level of Difficulty:	2	
SRO Only Justification:	N/A	
PSA applicability:		
Top 10 Risk Significant System – RHR Service Water		

Examination Outline Cross-Reference	**Level**	RO
264000 EDGs Knowledge of EMERGENCY GENERATORS (DIESEL/JET) design feature(s) and/or interlocks which provide for the following: (CFR: 41.7) **K4.08 Automatic startup**	**Tier#**	2
	Group#	1
	K/A #	264000 K4.08
	Rating	3.8
	Revision	0
Revision Statement:		

Question 51

DG1 starting air receiver pressures are:

- Receiver 1A 0 psig
- Receiver 1B 205 psig

DG1 receives an automatic start signal.

According to Procedure 2.2.20 [Standby AC Power System (Diesel Generator)],…

(1) How many DG1 start attempts meeting DG1 design startup time requirements are assured to be available under these conditions?

AND

(2) How long will DG1 attempt to crank if an automatic start signal is received AND some failure prevents fuel from being supplied to the engine?

A. (1) One, ONLY
(2) 20 seconds, ONLY

B. (1) One, ONLY
(2) NOT time dependent. Until Starting Air is depleted OR Emergency Stop is depressed

C. (1) Two or more
(2) 20 seconds, ONLY

D. (1) Two or more
(2) NOT time dependent. Until Starting Air is depleted OR Emergency Stop is depressed

Answer: C. (1) Two or more
(2) 20 seconds, ONLY

Explanation:
Procedure 2.2.20 Attachment 3, step 1.3.2.5 states “the low pressure alarm value of 200 psig has been established as the lower limit in support of this design requirement and is an indicator of a degraded condition that if uncorrected, could affect diesel generator OPERABILITY. With pressure at least 200 psig in at least one starting air receiver, sufficient capacity for multiple DG start attempts exists. As long as the pressure is at least 125 psig in at least one starting air receiver, there is capacity for at least one start attempt.” “Multiple” is represented as “two or more” in the correct answer.

The DG trip logic includes an incomplete sequence trip, which is always active, even for emergency start conditions. If the DG has not reached 280 rpm within 20 seconds following a start signal, a DG trip will be generated. On a normal DG start, the DG should reach 280 rpm within 6-7 seconds. However, that relies on the DG receiving fuel. If no fuel is supplied to the DG, the Air Start subsystem will roll the engine to at least cranking speed, 40 rpm, for 20 seconds, then the incomplete sequence timer trips the engine.

Distracters:
Answer A part 1 is plausible because having between 125-200 psig in only one starting air receiver guarantees only one start attempt assured of cranking a DG within the design time requirement of 14 seconds. 205 psig is below the auto start setpoint of the air compressors, which the unprepared applicant may confuse with the low pressure alarm/operability/design requirement. It is wrong because the answer states one, ONLY, but above 200 psig “multiple” start attempts are assured. Part 2 is correct.

Answer B part 1 is plausible and wrong for the same reasons as stated for distractor A. Part 2 is plausible because some trips are bypassed under emergency start conditions, as stated in the stem. If the incomplete sequence trip was bypassed, the DG would attempt to crank (air roll) until starting/control air was depleted or the operator intervened to trip the engine. It is wrong because an incomplete sequence trip is always active on a DG start, and the DG would trip after rolling for 20 seconds if it did not reach 280 rpm, which it would not on starting air alone, without fuel.

Answer D part 1 is correct. Part 2 is plausible and wrong for the same reasons as stated for distractor B.

Technical References: Procedure 2.2.20 [Standby AC Power System (Diesel Generator)] Att. 3 (Rev 94)

References to be provided to applicants during exam: none		
Learning Objective: COR002-08-02 Obj LO-02, 06g, 09h, 14c		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b)(8)	
Level of Difficulty:	3	
SRO Only Justification:	N/A	
PSA applicability:		
Top 10 Risk Significant System – Diesel Generators		

Examination Outline Cross-Reference	**Level**	RO
300000 Instrument Air Knowledge of the operational implications of the following concepts as they apply to the INSTRUMENT AIR SYSTEM: (CFR: 41.5 / 45.3) **K5.01 Air compressors**	**Tier#**	2
	Group#	1
	K/A #	300000 K5.01
	Rating	2.5
	Revision	0
Revision Statement:		

Question 52

Station Air Compressors (SAC) are aligned as follows:

- SAC 1A 1st Backup
- SAC 1B Lead
- SAC 1C 2nd Backup

A leak has occurred on the instrument air supply header.

Instrument Air supply header pressure has fallen to 92 psig and has stabilized.

SAC 1B Lube Oil temperature is 150°F.

Which Station Air Compressor(s) is/are running loaded NOW?

A. 1A, only

B. 1B, only

C. 1A and 1B

D. 1A and 1C

Answer: C. 1A and 1B
Explanation: The lead SAC, 1B, is set to load between 100 to 110 psig. The 1st Backup SAC, 1A, is set to auto start at 93 psig and load between 93 to 105 psig. The 2nd Backup SAC, 1C, is set to auto start at 90 psig and load between 90 to 100 psig. Therefore, for IA pressure of 92 psig, only SAC 1A and 1B have a demand to be running loaded, since

92 psig is above the auto start and loading range of the 2nd backup compressor, SAC C.

SACs trip when Lube Oil temperature reaches 158°F. Since SAC 1B Lube Oil temperature is only 150°F, it will not be tripped.

Therefore, SACs 1A and 1B will be running loaded.

Distracters:
Answer A is plausible because high lube oil temperature is a SAC trip. 150°F was chosen because it is above the high lube oil temperature warning alarm setpoint, 149°F. The unprepared applicant who cannot remember the trip setpoint or confuses it with the alarm setpoint may believe SAC 1B will be tripped on high lube oil temperature and select this answer. It is wrong because SAC 1B lube oil temperature is listed as 150°F, which is below the trip setpoint of 158°F; therefore SAC 1B will be running loaded, as well as SAC 1A.

Answer B is plausible because each SAC will operate loaded for different IA pressure ranges. The unprepared applicant may believe only the compressor in Lead should operate at 102 psig IA pressure and select this answer. It is wrong because SAC 1A as 1st Backup is set to start and load in the range of 93-105 psig. Since the given IA pressure, 92 psig, is below 93 psig, SAC 1A will be running loaded, as well as SAC 1B.

Answer D is plausible because each SAC will operate loaded for different IA pressure ranges. The unprepared applicant may believe all SACs have a demand to operate loaded at 92 psig IA pressure. Also, because high lube oil temperature is a SAC trip, the unprepared applicant who cannot remember the setpoint may believe SAC 1B will be tripped on high lube oil temperature. These errors combined would result in the unprepared applicant selecting this answer. It is wrong because SAC 1B would still be running loaded as described for distractor A, and SAC 1C would not be running loaded, since IA pressure, 92 psig, has not yet lowered to its auto start setting of 90 psig.

Technical References: Procedure 2.2.59 [Plant Air System](Rev 74), Procedure 5.2AIR [Loss of Instrument Air](Rev 21)

References to be provided to applicants during exam: none

Learning Objective: COR0011701 Obj. LO-05a, 05b, 07b

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X

Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b)(4)	
Level of Difficulty:	3	
SRO Only Justification:	N/A	
PSA applicability:		
N/A		

Examination Outline Cross-Reference	**Level**	RO
400000 Component Cooling Water Knowledge of CCWS design feature(s) and or interlocks which provide for the following: (CFR: 41.7) **K4.01 Automatic start of standby pump**	**Tier#**	2
	Group#	1
	K/A #	400000 K4.01
	Rating	3.4
	Revision	0
Revision Statement:		

Question 53

The plant is at rated power when a loss of offsite power occurs.

How many REC pumps are running 1 minute later?

A. 1

B. 2

C. 3

D. 4

Answer: B. 2
Explanation: Per 2.2.65, there are normally 3 of 4 REC pumps in operation. Per 2.2.65 step 5.16, for each REC division, the selector switch for one running REC pump is in STANDBY and other selector switch is in NORMAL; therefore, 2 REC pumps will be in standby. Per 5.2REC step 2.1, REC pumps in STANDBY starts 20 seconds after 4160V Bus 1F and/or 1G re-energized by emergency power. For a loss of offsite power, DG1 and DG2 re-energize buses F and G within 15 seconds. Therefore, 2 REC pumps will be in operation after 1 minute.
Distracters: Answer A is plausible if only one REC pump was in standby, which the unprepared applicant may reason since normally one pump is idle. This answer is wrong because IAW 2.2.65, 2 REC pumps should be selected to standby, one in each division. Answer C is plausible if three REC pumps were in standby, which the unprepared applicant may reason since normally three pumps are operating. This answer is

wrong because IAW 2.2.65, 2 REC pumps should be selected to standby, one in each division. Answer D is plausible if all REC pumps were in standby, which the unprepared applicant may reason since loss of power to all running REC might seem to warrant starting all REC pumps to quickly restore cooling. Or, the unprepared applicant may believe all REC pumps would auto start due to sensing low REC system pressure, which is the case in other systems such as lube oil systems and DEH. This answer is wrong because IAW 2.2.65, two REC pumps should be selected to standby, one in each division and the auto start is based on electrical power availability.		
Technical References: Procedure 2.2.65 [Reactor Equipment Cooling Water System](Rev 65), Procedure 5.2REC [Loss of REC](Rev 17)		
References to be provided to applicants during exam: none		
Learning Objective: COR002-19-02 Obj LO-04c		
Question Source:	Bank #	New bank #66
(note changes; attach parent)	Modified Bank #	
	New	
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b)(4)	
Level of Difficulty:	3	
SRO Only Justification:	N/A	
PSA applicability:		
N/A		

Examination Outline Cross-Reference	Level	RO
201001 CRD Hydraulic Knowledge of the physical connections and/or cause-effect relationships between CONTROL ROD DRIVE HYDRAULIC SYSTEM and the following: (CFR: 41.2 to 41.9) **K1.07 Reactor protection system**	**Tier#**	2
	Group#	2
	K/A #	201001 K1.07
	Rating	3.4
	Revision	0
Revision Statement:		

Question 54

The plant was manually scrammed **one minute ago** from rated power; the scram has **NOT** been reset.

What is the effect of the scram on Control Rod Drive Hydraulic system parameters NOW relative to their values before the scram?

A. CRD System Total Flow **lower**
Charging Water Pressure **lower**

B. CRD System Total Flow **lower**
Charging Water Pressure **higher**

C. CRD System Total Flow **higher**
Charging Water Pressure **lower**

D. CRD System Total Flow **higher**
Charging Water Pressure **higher**

Answer: C. CRD System Total Flow **higher**
Charging Water Pressure **lower**

Explanation:
A scram causes CRD HCU accumulators to discharge through the scram inlet valves. CRD charging water header pressure rapidly lowers. CRD system flow, labeled Actual on panel 9-5, which senses nearly all system flow except ~20 gpm minimum flow to CST and ~2 gpm to Recirc and RWCU Pump seals, rises above 100 gpm, with one CRD pump operating, as flow directs toward the charging header. High system flow sensed on the CRD supply header upstream of the charging water branch signals the in-service CRD FCV to close to its minimum setting. Flows downstream of the FCV drop to near 0 gpm as downstream pressure falls to near RPV pressure. One minute after a scram, conditions have essentially stabilized, with nearly all CRD system flow directed to the charging header and routing into the RPV bottom head

region through seals of all of the CRDMs, since scram inlet valves are open. CRD Flow Control (Actual) remains higher (~100 gpm with one CRD pump running) than the normal 45 gpm since it reflects the high charging water flow. Charging header pressure has stabilized at a lower value since high charging flow exists.

Distracters:
Answer A is plausible because CRD Flow Control (Actual) indication is sensed downstream of some flow branches (e.g. CRC minimum flow to CST, Recirc pump seal purge, etc.). The unprepared applicant who believes CRD Flow Control (Actual) indication is also sensed downstream of the charging water branch would choose this answer. It is wrong because CRD Flow Control (Actual) is sensed upstream of the charging header and would, therefore, reflect the high charging water flow following a scram. Two validators mistook CRD Flow Control (Actual) to be sensed downstream of the CRD FCV and selected distractor A or B.

Answer B is plausible and wrong for the same reasons as given for distractor A. In addition, the unprepared applicant may believe that high charging water flow would result in a higher charging water pressure. This is wrong, since higher flow in the charging header results in lower charging header pressure. One validator chose this answer.

Answer D is plausible and wrong for the same reasons as given for distractor B relative to charging water pressure. The unprepared applicant who remembers the arrangement of CRD Flow Control (Actual) instrumentation but makes the errors described for distractor C would select this answer.

Technical References: Procedure 2.2.8 [Control Rod Drive Hydraulic System](Rev 95), B&R Drawing 2039 [CRD Hydraulic System], Lesson Plan COR002-04-02 [CRD System](Rev 24)

References to be provided to applicants during exam: none

Learning Objective: COR002-04-02 Obj 04m, 05b, 05c, 05f, 11f, 15b

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b)(6)	
Level of Difficulty:	3	

SRO Only Justification:	N/A
PSA applicability:	
N/A	

Examination Outline Cross-Reference	**Level**	RO
216000 Nuclear Boiler Inst. Knowledge of the effect that a loss or malfunction of the following will have on the NUCLEAR BOILER INSTRUMENTATION : (CFR: 41.7) **K6.01 A.C. electrical distribution**	**Tier#**	2
	Group#	2
	K/A #	216000 **K6.01**
	Rating	3.1
	Revision	0
Revision Statement:		

Question 55

Station Blackout conditions exist.

No operator action has been performed.

Which reactor water level indication is available?

A. NBI-LI-91B [Fuel Zone Level] on Panel 9-3

B. NBI-LI-85A [Wide Range Level] on Panel 9-5

C. NBI-LI-85B [Wide Range Level] on Panel 9-5

D. RFC-LI-94B [RX Narrow Range Level] on Panel 9-5

Answer: D. RFC-LI-94B [RX Narrow Range Level] on Panel 9-5

Explanation:
The stem states “No operator action has been performed” based on Ops validator and Ops reviewer comments that recent emphasis on new FLEX strategies and modifications may complicate an applicant’s view of this question. FLEX strategies resupply power to a wide range of control room indication; therefore, this statement has been added in this instance, only, to prevent the applicant from assuming time sensitive actions associated with FLEX strategies may have been accomplished.

Narrow Range level indicator RFC-LI-94B is powered from 125 VDC battery panel AA2; therefore, it would be available during a loss of all AC power (Station Blackout).

Distracters:
Answer A is plausible because Fuel Zone is an important, accident range level indication. This answer is wrong because Fuel Zone NBI-LI-91B is powered from

Critical Control Panel 1B (CCP-1B), which derives its 120 VAC power from 4160 VAC Bus 1G, which is de-energized during a station blackout.

Answer B is plausible because Wide Range channel A is an important, accident range level indication, independent from channel B. This answer is wrong because Wide Range NBI-LI-85A is powered from Critical Control Panel 1A (CCP-1A), which derives its 120 VAC power from 4160 VAC Bus 1F, which is de-energized during a station blackout.

Answer C is plausible because Wide Range channel B is an important, accident range level indication, independent from channel A. This answer is wrong because Wide Range NBI-LI-85B is powered from Critical Control Panel (CCP), which derives its 120 VAC power from 4160 VAC Bus 1G, which is de-energized during a station blackout.

Technical References: BR drawings 3010 sheets 04, 07, 08

References to be provided to applicants during exam: none

Learning Objective: COR002-15-02 Obj. LO-05k

Question Source:	Bank #	19198
(note changes; attach parent)	Modified Bank #	
	New	
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b)(7)	
Level of Difficulty:	3	
SRO Only Justification:	N/A	

PSA Applicability

Top 10 Risk Significant System – Nuclear Boiler Instrumentation

<table>
<tr><td>Examination Outline Cross-Reference</td><td>Level</td><td>RO</td></tr>
<tr><td rowspan="5">201003 Control Rod and Drive Mechanism
Knowledge of the operational implications of the following concepts as they apply to CONTROL ROD AND DRIVE MECHANISM : (CFR: 41.5)
K5.04 †Rod sequence patterns</td><td>Tier#</td><td>2</td></tr>
<tr><td>Group#</td><td>2</td></tr>
<tr><td>K/A #</td><td>201003 K5.04</td></tr>
<tr><td>Rating</td><td>3.1</td></tr>
<tr><td>Revision</td><td>0</td></tr>
<tr><td colspan="3">Revision Statement:</td></tr>
</table>

Question 56

The plant is in Mode 2.

- The Rod Worth Minimizer is manually BYPASSED.
- Banked Position Withdrawal Sequence (BPWS) requirements are in effect.
- The first three groups of control rods have been fully withdrawn.

According to Procedure 10.13 [Control Rod Sequence and Movement Control],…

(1) Which control rod in the fourth group will have the highest worth when banking between positions 08 and 12?

AND

(2) For which event is compliance with BPWS credited for mitigation in order to meet analytical assumptions for the event?

A. (1) Last
(2) Control Rod Drop Accident

B. (1) Last
(2) Control Rod Withdrawal Error

C. (1) First
(2) Control Rod Drop Accident

D. (1) First
(2) Control Rod Withdrawal Error

Answer: C. (1) First (2) Control Rod Drop Accident
Explanation:

Per procedure 10.13 Att. 7 section 1, the first two rod groups, constituting 25% of the control rods, are fully withdrawn to position 48. Then the third and fourth rod groups, constituting another 25% of the control rods, are banked out. Step 1.3 states the BPWS notch positions are 00, 04, 08,12, and 48. Procedure 10.13 step 3.1.2 states the first control rod in a RWM group should be treated with extra caution, since it is the highest worth rod in the group. Procedure 10.13 step 3.1.3 requires BPWS to be used when the RWM is bypassed below the Low Power Setpoint in order to meet the assumptions of the Control Rod Drop Accident (CRDA). USAR Section XIV-6.2 lists compliance with BPWS as a starting condition and assumption for the CRDA.

Distracters:
Answer A Part 1 is plausible to the unprepared applicant who confuses which end of a control rod group represents the highest worth. The unprepared candidate is as likely to choose this answer as the correct one (One validator selected this option). Part 2 is correct.

Answer B part 1 is plausible and wrong for the same reason as stated for distractor A. Part 2 is plausible because the Control Rod Withdrawal Error (CRWE) is an analyzed event related to control rod withdrawal. It is not unusual for unprepared applicants to confuse and intermingle the bases for the CRWE and CRDA. It is wrong because the requirement for BPWS compliance to meet assumptions of the CRDA is specifically noted in procedure 10.13 step 3.1. USAR section XII-7.5 ascribes IRM and Rod Block Monitor functions credit for meeting the CRWE analysis, not the CRDA analysis. In fact, BPWS noncompliance coupled with continuous control rod withdrawal would be classified as a CRWE, not a mitigation for one.

Answer D part 1 is correct. Part 2 is plausible and wrong for the same reasons as given for distractor B.

Technical References: Procedure 10.13 [Control Rod Sequence and Movement Control](Rev 72), USAR sections XIV-6.2, XII-7.5

References to be provided to applicants during exam: none

Learning Objective: INT032-01-17 EO-1a, COR002-20-02 LO-01s

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X

Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	

10CFR Part 55 Content:	55.41(b)(6),(10)

Level of Difficulty:	3
SRO Only Justification:	N/A
PSA applicability:	
N/A	

Examination Outline Cross-Reference	**Level**	RO
202001 Recirculation Knowledge of electrical power supplies to the following: (CFR: 41.7) **K2.02 MG sets: Plant-Specific**	**Tier#**	2
	Group#	2
	K/A #	202001 K2.02
	Rating	3.2
	Revision	0
Revision Statement:		

Question 57

What 4160 VAC bus supplies power to the drive motor for Reactor Recirc Motor Generator (RRMG) A?

A. 1A

B. 1B

C. 1C

D. 1D

Answer: C. 1C

Explanation:
4160 VAC bus 1C supplies RRMG A drive motor.

Distracters:
Answer A is plausible because it, like correct answer C, represents a 4160 VAC BOP bus. It is wrong because 4160 VAC bus 1C supplies RRMG A drive motor.

Answer B is plausible because it, like correct answer C, represents a 4160 VAC BOP bus. It is wrong because 4160 VAC bus 1C supplies RRMG A drive motor.

Answer D is plausible because it, like correct answer C, represents a 4160 VAC BOP bus that supplies RRMG B drive motor. It is wrong because 4160 VAC bus 1C supplies RRMG A drive motor.

Technical References: Procedure 2.2A.RR.DIV1 [Reactor Recirculation System Component Checklist (Div 1)](Rev 9)

References to be provided to applicants during exam: none		
Learning Objective: COR002-22-02 Obj LO-07a		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b)(3),(6)	
Level of Difficulty:	2	
SRO Only Justification:	N/A	
PSA applicability:		
N/A		

Examination Outline Cross-Reference	Level	RO
202002 Recirculation Flow Control Knowledge of RECIRCULATION FLOW CONTROL SYSTEM design feature(s) and/or interlocks which provide for the following: (CFR: 41.7) **K4.07 Minimum and maximum pump speed setpoints**	**Tier#** **Group#** **K/A #** **Rating** Revision	2 2 202002 K4.07 2.9 0
Revision Statement:		

Question 58

Reactor Recirc (RR) Pump A tripped.

- The cause of the pump trip has been corrected.
- RR Pump A is now being restarted at **30%** reactor power IAW Procedure 2.2.68.1 [Reactor Recirculation System Operations].
- The RRMG Set A GEN FIELD BKR GEN FIELD BKR has just closed.

This annunciator is in alarm:

RECIRC A FLOW LIMIT	PANEL/WINDOW: 9-4-3/D-3

(1) What is the highest RR pump A speed that can be attained with this annunciator sealed in?

AND

(2) Which one of the following will cause this annunciator to clear?

A. (1) 20 %
(2) Raising reactor power

B. (1) 20 %
(2) RR-MO-53A, PUMP DISCHARGE VLV fully open

C. (1) 45 %
(2) Raising reactor power

D. (1) 45 %

(2) RR-MO-53A, PUMP DISCHARGE VLV fully open

Answer: B (1) 20 % (2) RR-MO-53A, PUMP DISCHARGE VLV fully open		
Explanation: The subject annunciator can be caused by either a RVLC runback, low FW flow ≤20%, or pump discharge valve ≤90% open. For the conditions given, power ascension at 30% power, the FW flow and RPV water level would be above the low FW flow and RVLC runback water level setpoints. IAW procedure 2.2.68.1, when a RR pump is started while the other RR pump is operating, the pump discharge valve is placed in an automatic jog open mode, and it begins opening from the fully closed position, when the RR pump field breaker closes. In this case, when the discharge valve reaches fully open position, >90% open, the flow limit annunciator will clear, since low FW flow and RVLC runback signals are already clear.		
Distracters: Answer A part 1 is correct. Part 2 is plausible because low FW flow can also cause this annunciator, and raising reactor power raises FW flow. It is wrong because the low FW flow setpoint is ≤20%, and with power 30%, FW flow has to be above that setpoint. Therefore, the discharge valve opening to a position >90% open will cause the annunciator to clear, in this case. Answer C part 1 is plausible because a RVLC runback can cause this alarm, in which case a flow limit of ≤45% is enforced. It is wrong because, in this case, the RR pump discharge valve position interlock is being enforced, not RVLC runback, since a RVLC runback would not be present and because 20% pump speed is the highest speed that can be attained with the discharge valve not fully open, not 45% speed. Part 2 is plausible and wrong for the same reasons as stated for distractor A. Answer D part 1 is plausible and wrong for the same reasons as stated for distractor C. Part 2 is correct.		
Technical References: Procedure 2.2.68.1 [Reactor Recirculation System Operations](Rev 80), Alarm Card 9-4-3/D-3 (Rev 27)		
References to be provided to applicants during exam: none		
Learning Objective: COR002-22-02 LO-10a, 10m		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X

Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b)(3),(6)	
Level of Difficulty:	4	
SRO Only Justification:	N/A	
PSA applicability:		
N/A		

Examination Outline Cross-Reference	Level	RO
201002 RMCS Knowledge of the effect that a loss or malfunction of the REACTOR MANUAL CONTROL SYSTEM will have on following: (CFR: 41.7) **K3.03 Ability to process rod block signals**	**Tier#**	2
	Group#	2
	K/A #	201002 K3.03
	Rating	2.9
	Revision	0
Revision Statement:		

Question 59

The plant is in Mode 2.

Control rod 26-51 is being withdrawn using notch OVERRIDE mode.

The 28 VDC Rod Position Information System (RPIS) power supply in the Reactor Manual Control System (RMCS) fails.

What is the effect of this failure on the ability of RMCS to generate rod block signals?

A. Control rod will remain selected but rod movement stops at the next notch position due to Timer Malfunction rod block.

B. Control rod will remain selected and continues to withdraw. All Rod Worth Minimizer (RWM) rod insert AND withdrawal blocks are defeated.

C. Control rod will be automatically de-selected and control rods can NOT be withdrawn NOR inserted with RMCS due to a Select Block AND RWM Rod Block.

D. Control rod will be automatically de-selected and rod movement stops at the next notch position. Control rods can be inserted ONLY using RMCS via EMERGENCY IN.

Answer: C. Control rod will be automatically de-selected and control rods can NOT be withdrawn NOR inserted with RMCS due to a Select Block AND RWM Rod Block.
Explanation: RPIS function of RMCS provides positions of the 137 control rods for comparison with programmed rod pattern constraints. Permissive logic of the RMCS must be met to

enable control rod movement. With loss of rod position information, RMCS permissives for rod selection and pattern control cannot be met to allow rod movement. Loss of the 28 VDC RPIS power supply results in a control rod select block, which results in de-selection of the selected control rod, and a RWM rod insert/withdraw block. Since no control rod can be selected, rod motion is prevented except by scram.

Distracters:
Answer A is plausible because a Timer Malfunction does result in a rod block which causes the control rod that is being continuously withdrawn to be de-selected and latch at the next notch position. It is wrong because the Timer Malfunction is not involved in the subject scenario, since the timer starts only when the Override switch is released and the timer is powered from a separate source than that which was lost.

Answer B is plausible because there are some RMCS failure modes where the ability to process control rod blocks is defeated. Also, the unprepared applicant may confuse the effects of this failure during emergency control rod insertion, when rod motion is not immediately inhibited and subsequent control rods can be selected and inserted if a 3 second timer has not elapsed. This answer is wrong because the control rod is being withdrawn; therefore, the rod is immediate control rod de-selection and RWM rod block occurs.

Answer D is plausible because immediate control rod de-selection occurs and the rod stops at the next notch position. Also, if emergency rod insertion was involved, rather than control rod withdrawal, emergency rod insertion could continue under certain conditions. The unprepared applicant who confuses the constraints for this failure during emergency rod insertion versus control rod withdrawal would choose this answer. It is wrong because a control rod select block and RWM rod block would occur, preventing all rod withdrawal or insertion using RMCS

Technical References: Alarm Card 9-5-1/A-5 (Rev 34), Procedure 4.3 [Reactor Manual Control System and Rod Position Information System](Rev 28), Lesson Plan COR002-20-02 [RMCS](Rev 22)

References to be provided to applicants during exam: none

Learning Objective: COR002-20-02 Obj LO-05c, 05j, 08a, 08e, 15a, 15d

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X

Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	

10CFR Part 55 Content:	55.41(b)(6)
Level of Difficulty:	3
SRO Only Justification:	N/A
PSA applicability:	
N/A	

Examination Outline Cross-Reference	**Level**	RO
233000 Fuel Pool Cooling/Cleanup Ability to predict and/or monitor changes in parameters associated with operating the FUEL POOL COOLING AND CLEAN-UP controls including: (CFR: 41.5) **A1.06 System flow**	**Tier#**	2
	Group#	2
	K/A #	233000 **A1.06**
	Rating	2.5
	Revision	
Revision Statement:		

Question 60

Fuel Pool Cooling (FPC) system is being secured.

FPC -Pumps A and B are in service through FPC Filter Demineralizers (F/D) A and B.

FPC F/D A is being removed from service IAW Procedure 2.2.32 [Fuel Pool Cooling and Demineralizer System].

- FPC-FIC-69A, FPC F/D A EFFLUENT FLOW CONTROL output is being lowered in MANUAL.

(1) What is the effect on flow through FPC F/D **B** as FPC F/D A is removed from service?

AND

(2) What will be the effect on **Spent Fuel Pool (SFP)** level when the **last** FPC Pump is stopped by the operator? **(i.e. no other FPC Pump is operating.)**

A. (1) rises to twice the previous value
(2) rise slightly then stabilize

B. (1) rises to twice the previous value
(2) lower slightly then stabilize

C. (1) remains approximately the same
(2) rise slightly then stabilize

D. (1) remains approximately the same
(2) lower slightly then stabilize

Answer: D. (1) remains approximately the same

(2) lower slightly then stabilize

Explanation:
This question tests knowledge of FPC system flow characteristics in response to securing FPC F/Ds and pumps. FPC F/Ds are controlled from local panels in the Radwaste Building Control Room. FPC F/D A is removed from service by placing F/D A and B flow control in manual and lowering flow through the F/D while opening the F/D bypass valve to maintain nearly constant system flow. F/D B effluent flow is maintained constant (~475 gpm) during this process.

SPF level is controlled by adjustable weir skimmers. Water overflowing the weirs flows to the Skimmer Surge Tanks, which serve as the suction source for the FPC Pumps. The FPC Pumps drive water through the FPC Heat Exchangers and F/Ds and return it to the SFP. When the running FPC Pump is stopped, water is no longer being returned to the SFP. Water continues to overflow into the skimmers and SFP level lowers until pool level reaches the top of the weir, where it stabilizes.

Distracters:
Answer A part 1 is plausible to the unprepared applicant who does not know the procedural method for securing FPC F/Ds. The applicant who does not know F/D A flow is diverted to the F/D bypass line versus F/D B would select this answer. Part 2 is plausible because Skimmer Surge Tank level rises when the FPC Pump is stopped, because the tanks continue to receive overflow from the skimmers until pool level reaches the weirs. The unprepared applicant may confuse Skimmer Surge Tank response with SFP level response and choose this answer. It is wrong because SFP level will lower until it reaches the top of the skimmer weirs, which are the bottom of the skimmer openings.

Answer B part 1 is plausible and wrong for the same reasons as given for distractor A. Part 2 is correct.

Answer C part 1 is correct. Part 2 is plausible and wrong for the same reasons as given for distractor A.

Technical References: Procedure 2.2.32 [Fuel Pool Cooling and Demineralizer System](Rev 96)

References to be provided to applicants during exam: none

Learning Objective: COR001-06-01 Obj. LO-11b, 12e

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X

Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b)(4)	
Level of Difficulty:	3	
SRO Only Justification:		
N/A		
PSA Applicability		
N/A		

Examination Outline Cross-Reference	Level	RO
234000 Fuel Handling Equipment Ability to monitor automatic operations of the FUEL HANDLING EQUIPMENT including: (CFR: 41.7) **A3.02 †Interlock operation**	**Tier#**	2
	Group#	2
	K/A #	234000 A3.02
	Rating	3.1
	Revision	0
Revision Statement:		

Question 61

Which one of the following conditions, **ALONE**, will cause the Refueling Platform to stop traveling toward the core if it is already near or over core?
(Assume a control rod is selected.)

A. Reactor Mode Switch in STARTUP

B. Reactor Mode Switch in REFUEL AND Fuel Hoist loaded

C. Reactor Mode Switch in SHUTDOWN AND Fuel Hoist loaded

D. Reactor Mode Switch in REFUEL AND selected control rod not at 00

Answer: A. Reactor Mode Switch in STARTUP

Explanation:
The Refueling Platform traverses forward from the reactor vessel to the spent fuel pool or in reverse from the spent fuel pool to the reactor vessel. Refueling Platform travel toward the core (Reverse motion) is inhibited when the platform is near or over the core, as sensed by limit switches, when the RMS is in STARTUP. (A control rod must be selected to allow portions of the "all rods in" refueling interlock logic to function, so the stem states to assume a control rod is selected.)

Distracters:
Generation of a reverse motion inhibit signal requires either:
RMS in REFUEL **plus** more than one control rod not fully inserted, *OR*
RMS in REFUEL **plus** one control rod not fully inserted **plus** hoist loaded

Answer B is plausible because hoist loaded with one or more control rods not fully inserted inhibits bridge reverse motion if the RMS is not in REFUEL. It is wrong because the RMS is in REFUEL, so Refueling Platform reverse travel would not be inhibited.

Answer C is plausible because Refueling Platform reverse travel is inhibited regardless of RMS position if a hoist is loaded and one control rod is withdrawn. It is wrong because with the RMS in SHUTDOWN, a scram has been inserted and a rod withdrawal block exists; therefore, all control rods are fully inserted. With all rods in, Refueling Platform reverse motion is not inhibited.

Answer D is plausible because a combination of the RMS in REFUEL with control rods not fully inserted does cause Refueling Platform reverse travel, toward the core, to be inhibited. Also, if the RMS is in any position other than STARTUP or REFUEL, Refueling Platform reverse travel is inhibited if a hoist is loaded and only one control rod is withdrawn. It is wrong because in addition to the RMS being in REFUEL, **more than one** control rod would have to be withdrawn beyond position 00 to cause platform travel toward the core to be inhibited, In this answer, one rod is not fully inserted. The stem asks which of the listed conditions ALONE would stop platform travel. Therefore, Refueling Platform reverse travel would not be inhibited.

Technical References: Procedure 2.2.31 [Fuel Handling - Refueling Platform] (Rev 51)

References to be provided to applicants during exam: none

Learning Objective: COR001-21-01 Obj. LO-05a, 10a

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b)(13)	
Level of Difficulty:	3	
SRO Only Justification:	N/A	

PSA applicability:

N/A

Examination Outline Cross-Reference	Level	RO
241000 Reactor/Turbine Pressure Regulator Ability to (a) predict the impacts of the following on the REACTOR/TURBINE PRESSURE REGULATING SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5 / 45.6) **A2.07 Loss of condenser vacuum**	**Tier#**	2
	Group#	2
	K/A #	241000 A2.07
	Rating	3.7
	Revision	0
Revision Statement:		

Question 62

Reactor power is 20%.

2.4VAC [Loss of Condenser Vacuum] was entered due to air inleakage from the Turbine seals.

A Rapid Power Reduction has been performed resulting in the following:

- Delay Region of Condenser Pressure Trip Graph was entered 2 minutes ago
- Condenser vacuum is 23.2 inches Hg, lowering 0.2 inch Hg per minute.

(1) How many minutes remain before the Turbine **automatically** trips due to low vacuum?

AND

(2) What action(s) is/are required IAW Procedure 2.4VAC [Loss of Condenser Vacuum] for the current conditions?

A. (1) one
(2) Trip the Main Turbine, ONLY

B. (1) one
(2) Scram AND trip the Main Turbine

C. (1) three
(2) Trip the Main Turbine, ONLY

D. (1) three
(2) Scram AND trip the Main Turbine

Answer: C. (1) three
(2) Trip the Main Turbine, ONLY

Explanation:
Procedure 2.4VAC [Loss of Condenser Vacuum] Att. 3 Condenser Pressure Trip Graph depicts the vacuum limits that will cause an automatic turbine trip. The turbine trip on low condenser vacuum is a dynamically calculated value depicted by the 5 Minute Delay region. When operation degrades into the region bounded by the blue 5-Minute Delay line, a timer starts, and if the region is not exited, the Turbine trips 5 minutes following region entry. In this question, the 5 Minute Delay Region was entered 2 minutes ago; therefore, 3 minutes remain until the turbine automatically trips, since after a power reduction, vacuum is still lowering and will not improve, and power will not be raised to exit the delay region. The No Delay line of the Condenser Pressure Trip Graph at 22”Hg will not be reached, which would take 6 more minutes, before the 5 minute delay setpoint causes an automatic turbine trip, since vacuum is falling only 0.2”Hg/min.
A Turbine trip will not cause an RPS actuation when reactor power is below the TS&CV closure RPS trip bypass setpoint, 29.5% power. Therefore, procedure 2.4VAC does not require a reactor scram for the given condition, since power is within the capacity of Turbine Bypass Valves. From the conditions stated, it is apparent vacuum cannot be maintained above 23” Hg; therefore, Procedure 2.4VAC section 3.1.2 must be performed. Step 3.1.2.1 states IF Annunciator 9-5-2/C-4 [TSV & TCV CLOSURE TRIP BYP CHAN A/B] clear, THEN SCRAM and enter Procedure 2.1.5. This alarm is on when power is 20%; therefore, step 3.1.2.1 is not applicable and is not performed. Step 3.1.2.2 then directs manually tripping the Turbine.

Distracters:
Answer A part 1 is plausible because it reflects the time remaining until 23”Hg vacuum is reached for the given rate of fall, 0.2”Hg/min. 23”Hg is the manual Turbine trip criteria per 2.4VAC: If vacuum cannot be maintained ≥23”Hg, THEN a manual turbine trip, and depending on reactor power, a scram is required. It is wrong because the automatic turbine trip is dynamically calculated and will occur 5 minutes after the Delay Region has been entered. In this case, 3 minutes from the present, since the delay region was entered 2 minutes ago. Part 2 is correct.

Answer B part 1 is is plausible and wrong for the same reasons as stated for distractor A. Part 2 is plausible because a manual scram is required due to lowering vacuum is some cases. A scram is required when power is above the TS&CV RPS bypass setpoint, 29.5%, and scram is required if the cause of the vacuum leak is a condenser boot seal tear or if vacuum cannot be maintained above 12” Hg, regardless of power level. This answer is wrong because at 20% power, Annunciator 9-5-2/C-4 will be ON, indicating the RPS trip from TSV and TCV closure is bypassed; therefore, a scram is not manually inserted.

Answer D part 1 is correct. Part 2 is plausible and wrong for the same reasons as stated for distractor B.

Technical References: Procedure 2.4VAC [Loss of Condenser Vacuum](Rev 25), Alarm Card 9-5-2/C-4 (Rev 46)		
References to be provided to applicants during exam: none		
Learning Objective: INT032-01-32 Obj H, J, K		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b)(7),(10)	
Level of Difficulty:	3	
SRO Only Justification:	N/A	
PSA applicability:		
Top 10 Risk Significant System - RPS		

Examination Outline Cross-Reference	Level	RO
245000 Main Turbine Gen. / Aux. Ability to manually operate and/or monitor in the control room: (CFR: 41.7) **A4.02 Generator controls**	**Tier#**	2
	Group#	2
	K/A #	245000 A4.02
	Rating	3.1
	Revision	0
Revision Statement:		

Question 63

The plant is at rated power with the following Main Generator configuration:

Doniphan requests you to adjust Main Generator output to 100 MVARs **IN (LEADING)**.

How do you make this adjustment on Panel C?

A. Place GEN BASE ADJUST to RAISE

B. Place GEN BASE ADJUST to LOWER

C. Place GEN VOLTAGE ADJUST to RAISE

D. Place GEN VOLTAGE ADJUST to LOWER

Answer: D. Place GEN VOLTAGE ADJUST to LOWER
Explanation: Photos indicate the generator voltage regulator is in automatic and reactive load is positive (OUT) 72 MVARs. To adjust to 100 MVAR IN is to pick up negative MVARS. With the voltage regulator in automatic, GEN VOLTAGE ADJUST control switch adjusts voltage regulator output. To pick up negative (IN) MVARs, the switch is placed to LOWER.
Distracters: Answer A is plausible because GEN BASE ADJUST adjusts voltage regulator output if the voltage regulator is in MANUAL or TEST. RAISE direction is plausible because the target value of 100 is a larger number than the current MVAR value of 72. It is wrong because GEN VOLTAGE ADJUST control switch adjusts voltage regulator output with the voltage regulator in automatic, and MVAR IN is a leading power factor, or negative MVAR value, which would be obtained by lowering voltage regulator output from its current value. Answer B is plausible because GEN BASE ADJUST adjusts voltage regulator output if the voltage regulator is in MANUAL or TEST. It is wrong because GEN VOLTAGE ADJUST control switch adjusts voltage regulator output with the voltage regulator in automatic. Answer C is plausible because the target value of 100 is a larger number than the current MVAR value of 72. It is wrong because MVAR IN is a leading power factor, or negative MVAR value, which would be obtained by lowering voltage regulator output from its current value.
Technical References: Procedure 2.2.14 [22 KV Electrical System](Rev 83)
References to be provided to applicants during exam: none
Learning Objective: COR001-13-02 obj LO-06d

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	

	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b)(5),(10)	
Level of Difficulty:	3	
SRO Only Justification:	N/A	
PSA applicability:		
N/A		

Examination Outline Cross-Reference	Level	RO
286000 Fire Protection **2.1.32** Ability to explain and apply system limits and precautions. (CFR: 41.10)	Tier#	2
	Group#	2
	K/A #	286000 G2.1.32
	Rating	3.8
	Revision	0
Revision Statement:		

Question 64

IAW Procedure 2.2.2 [Carbon Dioxide Systems] Precautions and Limitations,…

(1) The Liquid CO_2 Storage Tank should be maintained below what pressure?

AND

(2) What would be the consequence of exceeding this pressure by 50 psig?

A. (1) 325 psig
(2) Loss of CO_2 inventory

B. (1) 325 psig
(2) Damage to O-rings in system pilot valves

C. (1) 400 psig
(2) Loss of CO_2 inventory

D. (1) 400 psig
(2) Damage to O-rings in system pilot valves

Answer: A. (1) 325 psig (2) Loss of CO_2 inventory
Explanation: Per procedure 2.2.2 step 2.1, the Liquid CO2 storage tank should be maintained <325 psig. Tank safety valves are set to open at 357 psig. Loss of CO2 inventory will result if this pressure is reached. Relief valves would protect against overpressurizing system piping and components.
Distracters: Answer B part 1 is correct. Part 2 is plausible because sufficiently elevated pressure could overpressure a system to the point of damaging system components. O-ring damage was chosen as a distractor because O-rings are associated with pressure boundaries within components such as valves. This answer is wrong because relief

valves would protect against overpressurizing system piping and components, but relief valve opening would cause loss of CO_2 inventory.

Answer C part 1 is plausible because it is the operability limit for maximum High Pressure CO_2 bottle outlet pressure listed in procedure 2.2.2 step 2.5. It is wrong because the Low pressure Liquid CO_2 storage tank is required to be maintained < 325 psig per step 2.1. Part 2 is correct.

Answer D part 1 is plausible and wrong for the reasons given for distractor C. Part 2 is plausible and wrong for the reasons given for distractorB.

Technical References: Procedure 2.2.2 [Carbon Dioxide Systems](Rev 39)

References to be provided to applicants during exam: none

Learning Objective: COR001-05-01 Obj LO-13h

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b)(4),(10)	
Level of Difficulty:	3	
SRO Only Justification:	N/A	

PSA applicability:

N/A

Examination Outline Cross-Reference	**Level**	RO
290002 Reactor Vessel Internals Knowledge of the physical connections and/or cause- effect relationships between REACTOR VESSEL INTERNALS and the following: (CFR: 41.2 to 41.9) **K1.15 Nuclear boiler instrumentation**	**Tier#**	2
	Group#	2
	K/A #	290002 K1.15
	Rating	3.4
	Revision	0
Revision Statement:		

Question 65

Which reactor water level instrument range indicates higher than actual due to operation of Reactor Recirc Pumps?

A. Fuel Zone

B. Wide Range

C. Narrow Range

D. Steam Nozzle

Answer: A. Fuel Zone

Explanation:
This question tests knowledge of the effects of the design of Fuel Zone level instrumentation, specifically variable leg arrangement, with respect to internal RR Jet Pumps. Fuel zone instruments tap off jet pump #6 and 16 lower taps. They are only accurate with no jet pump flow. The positive dynamic head at the variable tap resulting from the jet pump flow causes the differential pressure transmitter to indicate a high indicated fuel zone level relative to actual level.

Distracters:
Answer B is plausible because it is an instrument range with taps to the RPV, and its reference leg tap is at a common elevation with that of Fuel Zone, at 60 inches. . Also, Wide Range instruments are affected by Jet Pump flow. The reactor jet pump flow in the annulus region outside the core shroud and past the lower tap of the Wide Range instruments has a significant velocity head and some friction loss which reduces the pressure on the variable leg to the differential pressure (level) instrument, resulting in an indicated level lower than actual. It is wrong because Wide Range

instruments indicate lower than actual when RR Pumps are operating due to flow in the downcomer annulus creates a venturi effect, negative dynamic head, at the variable tap.

Answer C is plausible because it is another instrument range with taps to the RPV, and its reference leg tap is at a common elevation with that of Fuel Zone, at 60 inches. It is wrong because the variable tap for Narrow Range is at 0 inches, a height well above the influence of RR Pump flow. Narrow Range indication is not affected by RR Pump flow, so indicated is the same as actual level whether or not RR Pumps are in operation.

Answer D is plausible because it is another instrument range with taps to the RPV. The unprepared applicant who does not know the arrangement of reactor water level instrumentation or the effect of dynamic head produced by the Jet Pumps on Fuel Zone variable taps is as likely to choose this answer as any other. It is wrong because Steam Nozzle Range variable tap is at 0 inches, and is above the height where sensing line taps would be affected by RR Pump flow.

Technical References: Operations Instruction #8 [Guideline For Successful Transient Mitigation](Rev 14), Lesson Plan COR002-15-02 [Nuclear Boiler Instrumentation](Rev 26)

References to be provided to applicants during exam: none

Learning Objective: COR002-15-02 Obj. 04g, 04h,

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b)(2),(10)	
Level of Difficulty:	3	
SRO Only Justification:	N/A	

PSA applicability:

Top 10 Risk Significant System – Nuclear Boiler Instrumentation

Examination Outline Cross-Reference	Level	RO
1. Conduct of Operations 2.1.1 Knowledge of conduct of operations requirements. (CFR: 41.10)	Tier#	3
	Group#	
	K/A #	G2.1.1
	Rating	3.8
	Revision	
Revision Statement:		

Question 66

According to Procedure 2.0.3 [Conduct of Operations]…

Which one of the following activities is NOT allowed to be performed by the designated At-The-Controls-Operator (ATCO) during **normal** plant operation at rated power?

A. Silencing an alarm at the Fire Protection Panel.

B. Peer checking a switch operation for a surveillance at Panel C.

C. Obtaining a reading at Panel 9-21 for 0630 Control Room Data.

D. Communication with a building operator by telephone at the CRS desk.

Answer: C. Obtaining a reading at Panel 9-21 for 0630 Control Room Data.
Explanation: This question is phrased in the negative sense to make it more generic and, because procedure 2.0.3 specifically lists activities that are allowed more so than ones disallowed, this makes distractors more plausible. Procedure 2.0.3 step 2.9.4.2 states the control room operator designated as the ATCO is required to remain inside the At-The-Controls area, as defined in procedure 2.0.3 Attachment 1. Step 2.9.4.2 also lists an exception to this rule. In the event of an off-normal condition in the back panels, the ATCO is allowed to be absent from the At-The-Controls Area to verify the receipt of an alarm or initiate corrective action provided that at least one on-watch Licensed Operator (i.e., CRS or SM) remains within the At-The-Controls Area. Of the areas listed in all of the answers, Panel 9-21 is the only area not within the At-The-Controls-Area. Since plant conditions are given as normal, and no abnormal condition at back panel 9-21 is given, the ATCO would not be allowed to exit the ATC area to take a routine reading at panel 9-21. This answer may be overlooked by the unprepared applicant because procedure 2.0.3 step

2.9.4.3 specifically lists taking log readings as an allowed activity, but it applies only if from the front panels.

Distracters:
Answer A is plausible because the Fire Protection Panel is at the furthest point away from Panel 9-5 reactor controls area. The unprepared operator may believe it outside the ATC area or that facing away from panel 9-5 to depress the alarm acknowledge button is not allowed. This is wrong because the ATC Area is defined by procedure 2.0.3 as including the Fire Protection Panel. Step 2.9.4.3 states minor operational activities, specifically including alarm response within the ATC Area, are allowed. Silencing an alarm on the Fire Protection Panel would not be an activity that would degrade the performance of front panel monitoring, which is what is forbidden by step 2.9.4.3.

Answer B is plausible because panel C is at one outside corner of the ATC Area and because a peer check does require a finite amount of concentration. The unprepared applicant may believe a peer check at panel C exceeds what is allowed by procedure 2.0.3. It is wrong because Step 2.9.4.3 states minor operational activities, specifically including peer checks within the ATC Area, are allowed. Peer checking manipulation of one switch would be a minor operational activity.

Answer D is plausible because the CRS desk is at an outer boundary of the ATC Area and because telephone communication does require a finite amount of concentration. It is wrong because Step 2.9.4.3 states minor operational activities, specifically including telephone communications within the ATC Area, are allowed.

Technical References: Procedure 2.0.3 [Conduct of Operations](Rev 92)

References to be provided to applicants during exam: none

Learning Objective: INT032-01-03 EO-C1k

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b)(10)	
Level of Difficulty:	2	
SRO Only Justification:	N/A	

PSA applicability:
N/A

Examination Outline Cross-Reference	Level	RO
1. Conduct of Operations 2.1.37 Knowledge of procedures, guidelines, or limitations associated with reactivity management. (CFR: 41.1)	Tier#	3
	Group#	
	K/A #	G2.1.37
	Rating	4.3
	Revision	0
Revision Statement:		

Question 67

Power is being reduced from 100% to 60% on your shift to perform a planned control rod sequence exchange.

What is the minimum staffing requirement to fill the position of Reactivity Manager for this power change?

A. Additional SRO on shift with no concurrent duties

B. On shift SRO qualified WCO with concurrent duties

C. CRS with the Shift Manager present in the control room

D. CRS with a Reactor Engineer present in the control room

Answer: A. Additional SRO on shift with no concurrent duties

Explanation:
Procedure 2.0.3 [Conduct of Operations] section 2.5.1 lists the staffing requirements for Reactivity Manager. It states an additional SRO with no concurrent duties must be present in the control room and act as Reactivity Manager during significant power changes. It defines significant power changes as startups, shutdowns, and ≥25% power changes using control rods. It gives specific examples of power changes that are not considered "significant" power changes where the on shift CRS may act a Reactivity Manager. The specific exemptions include routine power changes via Recirc flow for surveillances or load adjustments, rapid power reductions IAW procedure 2.1.10 [Station Power Changes], and <25% power change in one shift using control rods. For the case given, power reduction from 100% to 60% would be a 40% change. In this case, the power change does not fall into the category of any of the exemptions allowing the CRS to serve as Reactivity Manager; Therefore, an additional SRO with no concurrent duties is required.

Distracters:
Answer B is plausible because an SRO qualified WCO is an extra SRO, since minimum staffing only requires two SROs, the CRS and the SM. It is wrong because the WCO position is required for minimum control room operator staffing, along with RO and BOP licensed operators. The Reactivity Manager can have no concurrent duties, so he cannot also serve as WCO.

Answer C is plausible because the SM does not normally have to be present in the control room. The unprepared applicant might select this answer because he may know the CRS can, at times, fill the position of Reactivity Manager but not remember the conditions when an additional, dedicated SRO is required to be Reactivity Manager, or he might interpret the presence of the SM in the control room as an additional SRO, This answer is wrong because the case stated is for a planned, non-transient power reduction and involves a 40% power change on one shift. It is, therefore, a significant reactivity manipulation for which the CRS may not serve as Reactivity Manager.

Answer D is plausible because normally a Reactor Engineer would be present in the control room to provide guidance and oversight during significant power changes using control rods. The unprepared applicant might interpret the expertise available from reactor engineering sufficient to allow the CRS to concurrently fill the position of Reactivity Manager. This answer is wrong for the same reasons as stated for distractor C.

Technical References: Procedure 2.0.3 [Conduct of Operations](Rev 92)

References to be provided to applicants during exam: none

Learning Objective: INT032-01-03 EO-C1a7

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X

Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X

10CFR Part 55 Content:	55.41(b)(10)
Level of Difficulty:	2
SRO Only Justification:	N/A

PSA applicability:
N/A

Examination Outline Cross-Reference	Level	RO
1. Conduct of Operations 2.1.45 Ability to identify and interpret diverse indications to validate the response of another indication. (CFR: 41.7)	Tier#	3
	Group#	
	K/A #	G2.1.45
	Rating	4.3
	Revision	0
Revision Statement:		

Question 68

Regarding requirements for use of diverse indications to validate response of another indication…

(1) According to Procedure 2.0.3 [Conduct of Operations], what is the MINIMUM number of independent indications required to be checked to validate a safety system initiation signal?

AND

(2) Is a control room annunciator allowed to be used as an independent indication?

A. (1) 2
(2) No

B. (1) 2
(2) Yes

C. (1) 3
(2) No

D. (1) 3
(2) Yes

Answer: B. (1) 2 (2) Yes
Explanation: Procedure 2.0.3 step 8.3.4 states Two independent indications of a parameter should be checked to validate an initiation signal. Control room annunciators are one form of indication to be used by the operator to determine plant conditions. Procedure 2.4RXLVL Att. 3 contains step by step guidance, by process of elimination, for determining what level instruments are indicating accurately in the event of multiple

inconsistent indications, as would occur during failure of reference leg 3A or 3B. Control room annunciators are specifically listed as diverse indications for comparison with other indicators.

Distracters:
Answer A part 1 is correct. Part 2 is plausible because, in practice, analog indicators are more commonly used to verify whether a safety system setpoint has been exceeded. An actuation annunciator is usually the first indication of spurious system actuation, and the operator typically checks the associated parameter via meter, recorder, or computer to assess system response. The unprepared applicant may not know which annunciators are independent from one another or may be habituated to only using analog indications and believe annunciators are not independent indications. This answer is wrong because procedures do not disallow use of annunciators for independent checks. In fact, procedures such as 2.4RXLVL rely on using annunciators for independent checks.

Answer C part 1 is plausible because the unprepared applicant may not consider the safety system initiation signal as one of the two required independent indications and believe two more independent indications are required. He would then incorrectly conclude the total required is three. This is wrong because procedure 2.0.3 step 8.3.4 states Two independent indications of a parameter should be checked to validate an initiation signal. More may be used, but two is the minimum required. Indication associated with safety related instrumentation that causes safety system initiation is considered a viable indication and is necessarily independent from the indication used to validate the initiation signal, since that is selected on the basis of its independence. Part 2 is plausible and wrong for the same reasons as stated for distractor A.

Answer D part 1 is plausible and wrong for the same reasons as stated for distractor C. Part 2 is correct.

Technical References: Procedure 2.0.3 [Conduct of Operations](Rev 92), Procedure 2.4RXLVL [RPV Water Level Control Trouble] Att. 3 (Rev 26),

References to be provided to applicants during exam: none

Learning Objective: INT032-01-03 EO-C1a10, INT032-01-35 EO-G

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X

Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	

10CFR Part 55 Content:	55.41(b)(10)
Level of Difficulty:	2
SRO Only Justification:	N/A
PSA applicability:	

Examination Outline Cross-Reference	Level	RO
2. Equipment Control 2.2.6 Knowledge of the process for making changes to procedures. (CFR: 41.10)	Tier#	3
	Group#	
	K/A #	G2.2.6
	Rating	3.0
	Revision	
Revision Statement:		

Question 69

While performing a Reference Use procedure, a step is encountered which contains an incorrect unit of measure (psia instead of psig). The unit of measure is correctly used earlier in the procedure.

A procedure change is required.

IAW Procedure 0.4 [Procedure Change Process],…

What is the **lowest** classification of procedure change required?

A. Intent change

B. Instant change

C. Editorial change

D. Pen-and-Ink change

Answer: D. Pen-and-Ink change
Explanation: Procedure 0.4 [Procedure Change Process] step 3.4.7 defines a pen-and-ink change:

PEN-AND-INK - A pen-and-ink correction may be made to a procedure if it is simple and obvious:

- Items normally considered simple and obvious:
  - Obvious typographical errors.
  - **<u>Obvious incorrect units of measure.</u>**
  - Obvious errors in step and table references.
  - Obvious equipment identification inconsistencies.
  - Obvious equipment location inconsistencies.
  - Changing values that are obviously incorrect and are referenced correctly elsewhere in the procedure.
- Items normally <u>not</u> considered simple and obvious:
  - Changes to setpoint values and tolerances.
  - Changes to commitments.
  - Changes in step sequence.
  - Changes to values that are not referenced correctly elsewhere in the procedure.
  - Addition or deletion of steps.
  - Changes to acceptance criteria.

Procedure 0.4 [Procedure Change Process] step 3.4.8 defines an editorial change:

1.1.1 EDITORIAL CHANGE - The TSG (or Procedure Owner) may approve the following types of changes as editorial changes:

- A procedure change that has no material change, other than to perhaps undo a change that should not have been made (the revision number should always increment forward; for instance, if returning a Revision 11 change to its Revision 10 state, the new revision number will be 12, not 10).
- A "Documentation Only" package that makes no change to the procedure.
- Deletion of a Vendor Procedure, Special Procedure, or change that was previously approved for a limited duration.
- Reinstituting a deleted Vendor Procedure that is still on the same revision (component classification or system must <u>not</u> be changed).
- For situations where an approved procedure revision is sitting on the shelf, awaiting pre-implementation actions, and there is a need for partial implementation (for example, only one division of a plant configuration change has been made and the procedure changes to support operation of that division need to be put into place).
- Any change that could be processed as an electronic file update.

This question involves a case of an obvious incorrect unit of measure, which is a specific example of a pen-and-ink correction listed in step 3.4.7.

Distracters:

Answer A is plausible because Intent Change is a type of procedure change defined in procedure 0.4. It is wrong because this correction to a unit of measure would pass

the Non-Intent change screening of procedure 0.4 Att. 6. The definition of Intent Change reflects a change that would not pass this screening Answer B is plausible because Instant Change is a type of procedure change defined in procedure 0.4. It is wrong because an Instant change requires preapproval by an ITR and a SRO and is not necessary for conditions that satisfy requirements for Pen-and-Ink changes. An Instant Change is a higher classification than Pen-and-Ink change. Answer C is plausible to the unprepared applicant who does not know the classifications of pen-and-ink or editorial as defined in procedure 0.4. The word “Editorial” may seem appropriate due to its commonly accepted usage. IT is wrong because this question involves a case of an obvious incorrect unit of measure, which is a specific example of a pen-and-ink correction listed in procedure 0.4 step 3.4.7 and does not meet the definition of an editorial change given in step 3.4.8.
Technical References: Procedure 0.4 [Procedure Change Process](Rev 65)
References to be provided to applicants during exam: none
Learning Objective: INT032-01-01 EO E.1.a

Question Source:	Bank #	Audit #69 for 4/2015 NRC
(note changes; attach parent)	Modified Bank #	
	New	
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b)(10)	
Level of Difficulty:	3	
SRO Only Justification:	N/A	
PSA applicability:		
N/A		

Examination Outline Cross-Reference	Level	RO
2. Equipment Control 2.2.14 Knowledge of the process for controlling equipment configuration or status. (CFR: 41.10)	Tier#	3
	Group#	
	K/A #	G2.2.14
	Rating	3.9
	Revision	0
Revision Statement:		

Question 70

A pipe cap painted yellow is installed directly downstream of a manual valve that has a single strand of wire with a lead seal attached.

What does this configuration represent?

A. Contaminated system

B. ECCS system high point vent

C. Operations test gauge connection

D. Primary Containment isolation boundary

Answer: D. Primary Containment isolation boundary
Explanation: Procedure 2.0.1 section 14 contains the requirements for PC manual isolation valve and cap administrative control. It states numerous primary containment manual isolation valves and associated caps throughout plant are administratively controlled for strict control over their manipulation. Valves are identified by a tag labeled PRIMARY CONTAINMENT BOUNDARY and caps are identified by being painted yellow. A single strand of wire with a single lead seal is used for certain sealed closed primary containment manual isolation valves. These are installed in such a manner as to prevent valve operation without destroying seal.
Distracters: Answer A is plausible because yellow catch basins are used to contain radioactively contaminated water from system leaks. It is wrong because the configuration

described in the stem is unique to control of PC manual isolation valves and pipe caps. Answer B is plausible because some PC manual isolation valves are ECCS system vent valves. It is wrong because not all ECCS system manual valves are PC manual isolation valves, and the configuration described in the stem is specifically established to control of PC manual isolation valves and pipe caps. Answer C is plausible because some PC manual isolation valves are system test gauge connection points. It is wrong because not all system test connection points are PC manual isolation valves, and the configuration described in the stem is specifically established to control of PC manual isolation valves and pipe caps.		
Technical References: Procedure 2.0.1 [Plant Operations Policy] section 14 (Rev. 63), Procedure 2.0.2 [Operations Logs and Reports] (Rev. 102), Procedure 9.EN-RP-108 [Radiation Protection Posting] (Rev. 8)		
References to be provided to applicants during exam: none		
Learning Objective: INT032-01-03 EO-A1p		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b)(10)	
Level of Difficulty:	3	
SRO Only Justification:	N/A	
PSA applicability:		
Top 10 Risk Significant Systems – Primary Containment (Isolation)		

Examination Outline Cross-Reference	Level	RO
2. Equipment Control 2.2.39 Knowledge of less than or equal to one hour Technical Specification action statements for systems. (CFR: 41.7 / 41.10)	Tier#	3
	Group#	
	K/A #	G2.2.39
	Rating	3.9
	Revision	0
Revision Statement:		

Question 71

Which of the following involves a “less than or equal to 1-hour” Tech Spec Required Action?

A. At 100% power, unidentified LEAKAGE is 10 gpm.

B. In Mode 2, DEH fault causes reactor pressure to rise to 1030 psig and stabilize there.

C. At 100% power, Average Planar Linear Heat Generation Rate (APLHGR) is not within the limit.

D. In MODE 2 <u>with</u> the reactor critical, operators determine that the Shutdown Margin is not within the limit.

Answer: B. In Mode 2, DEH fault causes reactor pressure to rise to 1030 psig and stabilize there.

Explanation:
This question tests generic knowledge of systems and TS by presenting a variety of situations that involve relatively short TS actions; therefore, it satisfies the “generic” intent of Tier 3 relating to TS knowledge.

Of the conditions listed, only reactor pressure above 1020 psig requires entry into a ≤1hr action, since it must be restored ≤1020 psig within 15 minutes IAW TS 3.4.10 Action A.1, or a TS shutdown statement must be entered. The TS action completion time for other conditions listed are in excess of 1 hour. TS 3.4.10 is applicable in Modes 1 and 2.

Distracters:

Answer A is plausible because high leakage in the drywell is a major concern and involves relatively quick TS action. The unprepared applicant may also confuse TS 3.4.4 requirements for unidentified leakage with that for pressure boundary leakage, which would require immediate entry into a TS shutdown statement, and select this answer. Leakage is required to be within limits in Modes 1, 2, and 3. This answer is wrong because TS 3.4.4 Action A.1 requires leakage to be restored ≤5 gpm within 4 hours, and if pressure boundary leakage is assumed, TS 3.4.4 Action C.1 requires being in Mode 3 within 12 hours; therefore, a “less than or equal to 1-hour” Tech Spec Required Action is not involved.

Answer C is plausible because violation of the APLHGR Thermal Limit is a significant event and involves relatively quick TS action. APLHGR is required to be in limits at 100% power. This answer is wrong because TS 3.2.1 requires actions to be performed within 2 hours; therefore, a “less than or equal to 1-hour” Tech Spec Required Action is not involved.

Answer D is plausible because failing to meet SDM is a significant concern and involves relatively quick TS action. Also, TS 3.1.1 contains actions that have immediate completion times. For example, if SDM is not met in Mode 3 or 4, action must be taken immediately to insert all insertable control rods IAW TS 3.1.1 Actions C.1 / D.1. SDM is required to be within limits at all times. This answer is wrong because TS 3.1.1 Action A.1 requires SDM to be restored within 6 hours if in Mode 2; therefore, a “less than or equal to 1-hour” Tech Spec Required Action is not involved.

Technical References: TS 3.4.10 [Reactor Steam Dome Pressure], TS 3.2.1 [APLHGR], TS 3.4.4 [RCS Operational Leakage], TS 3.1.1 [Shutdown Margin]

References to be provided to applicants during exam: none

Learning Objective: INT007-05-05 EO-10

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b)(10)	
Level of Difficulty:	3	
SRO Only Justification:	N/A	

PSA applicability:
N/A

Examination Outline Cross-Reference	Level	RO
3. Radiation Control 2.3.4 Knowledge of radiation exposure limits under normal or emergency conditions. (CFR: 41.12)	Tier#	3
	Group#	
	K/A #	G2.3.4
	Rating	3.2
	Revision	0
Revision Statement:		

Question 72

Regarding annual administrative dose guidelines of Procedure 9.ALARA.1 [Personnel Dosimetry and Occupational Radiation Exposure Program]…

What is the highest dose that can be received **on-site** under normal plant conditions by an individual in a calendar-year **without** having to acquire written authorization for a dose extension?

A. 1000 mrem

B. 2000 mrem

C. 3000 mrem

D. 4000 mrem

Answer: A. 1000 mrem

Explanation:
IAW procedure 9.ALARA.1 section 6.2, written approval to exceed 1000 mrem in a calendar year must be obtained from an individual's supervisor and the RP Technical Supervisor. Therefore, 1000 mrem is the highest dose that can be received annually without written authorization.

Distracters:
Answer B is plausible because it is a value listed in procedure 9.ALARA.1 section 6.2 that requires a level of written authorization to exceed. It is wrong because written authorization from the individual's supervisor and the RP Technical Supervisor would have already been required to reach 2000 mrem, since it is greater than 1000 mrem.

Answer C is plausible because it is a value listed in procedure 9.ALARA.1 section 6.2 that requires a level of written authorization to exceed. It is wrong because two levels

of written authorization would have already been required to reach 3000 mrem, since it is greater than 1000 mrem and 2000 mrem. Answer D is plausible because it is the overall administrative annual dose limit listed in procedure 9.ALARA.1 section 6.. It is wrong because three levels of written authorization would have already been required to reach 4000 mrem, and 4000 mrem is not to be exceeded.
Technical References: procedure 9.ALARA.1 section 6.2 (Rev 46)
References to be provided to applicants during exam: none
Learning Objective: INT032-01-15 EO-D1h

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b)(12)	
Level of Difficulty:	2	
SRO Only Justification:	N/A	
PSA applicability:		
N/A		

Examination Outline Cross-Reference	**Level**	RO
3. Radiation Control 2.3.12 Knowledge of radiological safety principles pertaining to licensed operator duties, such as containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc. (CFR: 41.12)	**Tier#**	3
	Group#	
	K/A #	G2.3.12
	Rating	3.2
	Revision	0
Revision Statement:		

Question 73

IAW Procedure 9.ALARA.4 [Radiation Work Permits],…

Which one of the following meets the minimum requirement when entry into a high-radiation area without an existing SWP is performed during emergency conditions?

A. Continuous Radiation Protection (RP) coverage for the entry, ONLY.

B. Anticipated stay time and exposure is documented on the trip ticket prior to entry, ONLY.

C. Revision of the RWP stating the reason for the emergency entry and the anticipated stay time and exposure.

D. Continuous Radiation Protection (RP) coverage for the entry and initiation of an SWP following entry with time and exposure documented.

Answer: D. Continuous Radiation Protection (RP) coverage for the entry and initiation of an SWP following entry with time and exposure documented.

Explanation:
Under normal conditions, a Special Work Permit (SWP) is required for entry into a high radiation area IAW Procedure 9.EN-RP-100 [Radiation Worker Expectations]. Per procedure 9.ALARA.4 section 2.10: In an emergency, Radiation Protection Technicians can provide continuous radiological protection coverage for entries into an area that would normally require a SWP. The procedure makes allowance for relaxing RWP/SWP requirements during an emergency. The SWP shall be completed following the entry to document the radiation protection measures taken and the radiation dose incurred. The CNS RP-1B, RWP Time and Dose Log, shall be used during these instances to record personnel time and dose in the SWP Area. Radiation Protection personnel shall ensure this data is entered in the Radiological Data Management System RDMS.

Distracters:
Answer A is plausible because the unprepared applicant may choose this option if they know that continuous RP coverage is required but do not know the requirements for documentation of the entry. It is incorrect because it does not also include the initiation of the SWP following the entry.

Answer B is plausible because the unprepared applicant who cannot recall the emergency requirements may default to the normal entry practices. It is incorrect because the trip ticket is not required in an emergency.

Answer C is plausible because RWPs can be revised once it is determined the current RWP does not meet the job requirements. The applicant who does not recall the emergency entry requirements may select this answer because RWP can be easily revised. It is incorrect because no RWP revision is needed in an emergency situation. Entry into a high radiation area requires a SWP not a RWP.

Technical References: Procedure 9.ALARA.4 [Radiation Work Permits](Rev 22), Procedure 9.EN-RP-100 [Radiation Worker Expectations](Rev 10)

References to be provided to applicants during exam: none

Learning Objective: INT032-01-15 EO-E2

Question Source:	Bank #	8/2014 NRC Q#73
(note changes; attach parent)	Modified Bank #	
	New	
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b)(12)	
Level of Difficulty:	2	
SRO Only Justification:	N/A	

PSA applicability:
N/A

Examination Outline Cross-Reference	Level	RO
4. Emergency Procedures / Plan 2.4.28 Knowledge of procedures relating to a security event (non-safeguards information). (CFR: 41.10 / 43.5 / 45.13)	Tier#	3
	Group#	
	K/A #	G2.4.28
	Rating	3.2
	Revision	0
Revision Statement:		

Question 74

IAW Procedure 5.5SECURITY [Security],…

(1) Which Security Code is more severe, Black OR Yellow?

AND

(2) For that level of security event, what CNS department other than Operations is responsible for performing Immediate Operator Actions to disable certain design features?

A. (1) Black
   (2) Security

B. (1) Black
   (2) Utility Fire Brigade

C. (1) Yellow
   (2) Security

D. (1) Yellow
   (2) Utility Fire Brigade

Answer: A. (1) Black (2) Security
Explanation: [redacted]

[redacted]

Distracters:
Answer B part 1 is correct. Part 2 is plausible because the Utility Fire Brigade is always on site, is prepared for rapid response, and is comprised of some Maintenance personnel. The Utility Fire Brigade performs some surveillances on plant equipment, such as Fire Protection system, whereas Security does not. [redacted]

Answer C part 1 is plausible because per Procedure 2.3.1 [General Alarm Procedure], yellow annunciators take precedence over Black annunciators. The window box assembly is a matrix of divided lamp windows with engraved legend plates and multi-colored window bezels. Each window has been given a "PRIORITY" signifying the importance of the alarm: Priority I - RED; alarms that alert of EOP entry conditions or conditions requiring or causing an automatic or manual plant shutdown, or significant system setpoints. Priority II - YELLOW; alarm conditions which may require or rapidly cause a plant shutdown or radiation release. Priority III - BLACK; alarms that indicate off normal plant conditions that affect plant or component operability but should not lead to plant shutdown or radiation release. [redacted]

Answer D part 1 is plausible and wrong for the same reasons as given for distractor C. Part 2 is plausible and wrong for the same reasons as given for distractor B.

Technical References: Procedure 5.5SECURITY [Security](Rev 40), GE drawing 793E253 sheet 01, Procedure 2.3.1 [General Alarm Procedure](Rev 63)

References to be provided to applicants during exam: none

Learning Objective: COR002-34-02 Obj. LO-02a, 10a; COR002-16-02/ COR002-16-01 Obj LO-02b, 03i, 08f

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X

Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	

10CFR Part 55 Content:	55.41(b)(10)	
Level of Difficulty:	2	
SRO Only Justification:	N/A	
PSA Applicability		
Top 10 Risk Significant System – ADS and Pressure Relief		

Examination Outline Cross-Reference	**Level**	RO
4. Emergency Procedures / Plan 2.4.39 Knowledge of RO responsibilities in emergency plan implementation. (CFR: 41.10)	**Tier#**	3
	Group#	
	K/A #	G2.4.39
	Rating	3.9
	Revision	0
Revision Statement:		

Question 75

An event occurred at 0425.

Based on that event, an Alert was declared at 0432.

Which one of the following is the **latest** time to begin the **initial** notification to state and local agencies that meets the requirements of Procedure 5.7.6 [Notification]?

A. 0439

B. 0442

C. 0454

D. 0520

Answer: B. 0442

Explanation:
Per Procedure 5.7.6 [Notification] Att. 1 Note 1 and Att. 3 Note 1, initial notifications to state and local agencies must be made within 15 minutes of declaration of an emergency. Since the emergency was declared at 0432, initial notification is required by 0447. Of the times listed in the answers, 0442 is the latest time that meets the requirement.

Distracters:
Answer A is plausible for the unprepared applicant who believes the 15 minute requirement starts at the time the event occurs. It is wrong because it starts at time of declaration, so 0439 is not the latest time allowed, 0442 is.

Answer C is plausible for the unprepared applicant who confuses the 15 minute requirement with the maximum possible time from event occurrence to initial notification. Declarations must be made within 15 minutes of occurrence of an event that exceeds an EAL, and notifications must be made within 15 minutes after that declaration. The unprepared applicant may believe initial notification within 30 minutes of event occurrence meets the requirement and select this answer, since 0454 is within 30 minutes of 0425. It is wrong because the 15 minute limit for initial notification begins at emergency declaration, regardless of the time of event occurrence. Answer D is plausible to the unprepared applicant who confuses the 60 minute requirement for follow-up notifications with the 15 minute limit for initial notifications. For Alerts or higher, follow-up notifications are required at least every 30 minutes, approximately. It is wrong since 0520 would be 33 minutes late.
Technical References: Procedure 5.7.6 [Notification](Rev 68)
References to be provided to applicants during exam: none
Learning Objective: GEN0030401B0B030B

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b)(10)	
Level of Difficulty:	2	
SRO Only Justification:	N/A	
PSA applicability:		
N/A		

Examination Outline Cross-Reference	Level	SRO
295003 Partial or Complete Loss of AC / 6 Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER : (CFR: 43.5) **AA2.03 Battery status: Plant-Specific**	Tier#	1
	Group#	1
	K/A #	295003 AA2.03
	Rating	3.5
	Revision	1
Revision Statement: Per Chief Examiner review comments - Simplified stem and removed subjective term "slightly." Added "up to" to distractors C and D, named 125A Station Battery consistently, added TS Bases to references, added title for TS 3.8.6 in references, added simplified 1-line DC distribution drawing to attached references, clarified in stem 6.EE.601 was performed due to loss of charger. In stem, changed rated power to 100% power for consistency.		

Question 76

Reference Provided

The plant is at 100% power.

AC power was lost to 125 VDC Battery Charger 1A for a period of time and has now been restored.

125A Station Battery is on float charge.

Due to loss of 125 VDC Battery Charger 1A, Surveillance 6.EE.601 [125V/250V Station and Diesel Fire Pump Battery 7 Day Check] has been performed for the 125A Station Battery with the following results for pilot cell #1:

- Electrolyte level is below the minimum mark but still above the top plates
- Voltage is 2.16 V.

Surveillance 6.1EE.602 [DIV 1 125V/250V Station Battery 92 Day Check] has been performed for 125A Station Battery with the following results:

- Average specific gravity is 1.220
- Cell #23 electrolyte specific gravity is 1.204
- Cell #23 voltage is 2.09 VDC

What is the status of 125A Station Battery with respect to TS requirements?

DIV 1 125V Station Battery…

A. is INOPERABLE.

B. remains OPERABLE for up to 24 hours regardless of individual cell voltage.

C. remains OPERABLE for up to 24 hours only if Pilot Cell voltage remains ≥2.10 VDC.

D. remains OPERABLE for up to 31 days only if Pilot Cell voltage remains ≥2.10 VDC.

Answer: A. is INOPERABLE.
Explanation: This question tests interpreting TS Table 3.8.6-1 and application of nesting with respect to TS Actions. 480 VAC MCC-LX supplies power to 125 VDC battery charger 1A. The stem reflects temporary loss of AC power to the charger, which would result in partial discharge of 125 VDC battery 1A. The Alarm Card C-1/C-2 for Battery Charger Trouble would have triggered performance of battery cell parameter checks per TS surveillance requirements. Normally, a pilot cell inspection would be performed, then contingent upon those results, an all cell inspection would be required. LCO 3.8.6 is not met due to a pilot cell's electrolyte level being below the minimum mark, the Category A limit listed in TS 3.8.6 Table 3.8.6-1. Therefore, Action A.1 and A.2 are required. Action A.1 is met because the pilot cell voltage is above the category C limit, 2.10 V. However, when the "all cell" surveillance is performed for Action A.2, Cell #23 voltage of 2.09 VDC is below the Category C limit of 2.10 VDC in TS 3.8.6 Table 3.8.6-1. This is failure to meet TS 3.8.6 Action A.2. Actions A.1 and A.2 and A.3 all have to be satisfied to satisfy Condition A; therefore, Condition A is not satisfied. Condition B states if Condition A is not satisfied, the associated battery must be declared inoperable immediately. Also, Condition B is not satisfied because cell parameters do not meet Category C limits; thus, Answer A is correct. Operations stated it would not be unusual to perform pilot cell checks per 6.EE.601 prior to performing all cell checks per 6.EE.602.
Distracters: Answers B, C, and D are all wrong because they list times and conditions that are in excess of the required time to immediately declare the battery inoperable. Answer B is plausible because Action A.2 has a completion time of 24 hours, which, according to TS bases, is granted to complete the initial all cell surveillance and assumes battery parameters are not severely degraded below Category C limits, not to perform charging or maintenance to restore the battery to within Category C limits. However, the stem gives results of the all cell surveillance that conclusively shows results below the TS Category C limit; therefore, the Action is not met at that time. Answer C is plausible because if nesting is not understood by the unprepared applicant, Action A.1 could be interpreted as independent from A.2 and operability inappropriately sustained by virtue of meeting A.1 alone. Answer D is plausible since the applicant that misreads TS Table 3.8.6-1 would not identify that a Category C limit is not met and only take action per TS 3.8.6 Action A.3, which this answer reflects.

Technical References: TS 3.8.6 [Battery Cell Parameters] and Bases, Alarm Card C-1/C-2 (Rev 30)		
References to be provided to applicants during exam: TS 3.8.6 [Battery Cell Parameters]		
Learning Objective: INT007-05-09 Enabling Obj 01, 03; COR002-07-02 Obj. LO-02		
Question Source:	Bank #	Audit#1 Q#76 for 12/2015 NRC
(note changes; attach parent)	Modified Bank #	
	New	
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.43(b)(2)	
Level of Difficulty:	3	
SRO Only Justification:		
This question tests application of nesting with respect to TS Actions.		
PSA applicability:		
Top 10 Risk Significant Systems – Emergency DC Power		

Examination Outline Cross-Reference	Level	SRO
295021 Loss of Shutdown Cooling / 4 **2.2.40** Ability to apply Technical Specifications for a system. (CFR:43.2 / 43.5)	**Tier#**	1
	Group#	1
	K/A #	295021 G2.2.40
	Rating	4.7
	Revision	1
Revision Statement: Per Chief Examiner review comments – added TS 3.9.7 Bases for SW requirements to attached references, removed "30 minutes" from explanation, added TS Bases for Note applicability to attached references and enhanced explanation of application rules for TS Notes, added title to TS 3.9.7 in references block. Enhanced explanation and part 1 distractor justification. Per Ops Mgmt review, added Reactor cavity is flooded to elevation 1001' as 1st bullet.		

Question 77

Reference Provided

The plant is at High Water Level during a refueling outage:

- Reactor cavity is flooded to elevation 1001'.
- RHR pump A has been continuously operating in Shutdown Cooling for 24 hours.
- RHR loop B is tagged out of service for maintenance.
- Recirc pumps A and B are stopped.

At 0800, SW-MO-89A, HX-A SW DISCH VLV fails closed, AND the crew secures RHR pump A.

(1) For TS 3.9.7 Action A.1, does the verification of an alternate decay heat removal system have to be by demonstration, or may it be by calculation?

AND

(2) What is the latest time allowed for coolant circulation to be re-established?

A. (1) MUST be by demonstration, ONLY
(2) 0900

B. (1) MUST be by demonstration, ONLY
(2) 1000

C. (1) May be by calculation OR demonstration
(2) 0900

D. (1) May be by calculation OR demonstration
(2) 1000

Answer: C. (1) May be by calculation OR demonstration
(2) 0900

Explanation:
Bases for TS 3.9.7 Condition A states the required cooling capacity of the alternate heat removal method may be by demonstration or calculation. Procedure 2.1.20.2 states demonstration of alternate Decay Heat Removal methods is allowed by LCO 3.4.7, 3.4.8, 3.9.7, and LCO 3.9.8. While these specifications allow demonstration, it should only be implemented during emergent conditions or when there is reasonable assurance the system(s) to be used will be adequate to remove decay heat. Engineering input should be utilized to aid in this determination.
LCO 3.9.7 Shutdown Cooling – High Water Level applies. With loss of service water flow through RHR A heat exchanger, SDC A is inoperable. TS 3.9.7 bases states SW is required for SDC operability. Therefore, LCO 3.9.7 is not met and applicable actions must be followed. TS 3.9.7 A applies, since the required SDC subsystem is inoperable. Therefore, an alternate method of decay heat removal must be verified available within 1 hour, by 0900. With RHR A secured, coolant circulation by SDC has stopped. Therefore, TS 3.9.7 C applies and requires verifying coolant circulation by a method other than SDC. There is a note for LCO 3.9.7 that allows an exception to the LCO requirement for the operable SDC subsystem to be in operation. TS 3.9.7 bases states a Note is provided to allow a 2 hour exception to shut down the operating subsystem every 8 hours. This note modifies the LCO requirement, only, and does not apply to the Action Statements. Since Conditions A and C are entered due to the LCO not met, the required Completion Times for the associated Required Actions must be met. It does not modify TS 3.9.7 Conditions A and C; therefore, it does not grant an exception to the 1 hour required Completion Time for those actions. TS is formatted such that Notes applicable to TS Actions are conveyed by Notes immediately preceding the Actions section. Notes modifying TS Actions are positioned below the TS Applicability section (i.e.“below the line”) and above the Actions table in individual specifications. (See TS 1.3 and 1.4 examples of how TS formatting arranges Notes to modify all or individual TS Actions and SR.) Since the LCO Note allowing exception to continuous SDC operation does NOT modify the Actions section, and the Actions section is entered due to not meeting the LCO, an alternate method of coolant circulation must be verified in service within 1 hour, by 0900.

Distracters:
Answer A Part 1 is plausible because TS bases allow demonstration of alternate decay heat removal methods to satisfy Action A.1, and the unprepared applicant may believe demonstration is the only viable method due to perceived uncertainties of calculational methods and may not be familiar with Procedure 2.1.20.2 [Cycle Specific Fuel Transfer and Alternate Cooling Guideline]. It is wrong because TS bases allows for verification by calculation, and Procedure 2.1.20.2 notes calculation is the preferred method of verification, and demonstration should only be performed during emergent conditions or when there is reasonable assurance the system(s) to be used will be adequate to remove decay heat. Engineering input should be utilized to aid in

this determination. Calculation, when available, would provide the timeliest verification to meet the 1 hour completion time of TS 3.9.7 A.1. Part 2 is correct.

Answer B Part 1 is plausible and wrong for the same reasons as stated for distractor A. Part 2 is plausible because the TS 3.9.7 LCO note allows the operating SDC subsystem to be secured for 2 hours in an 8 hour period. The unprepared applicant may reason that since RHR A was manually secured, and the SW valve failure does not alone prevent SDC recirculation flow, since the heat exchanger can be bypassed, the LCO note provides an exception to TS 3.9.7 C. This is wrong because LCO 3.9.7 is not met due to the required SDC being inoperable. TS 3.8.7 C then applies, since no SDC is in operation, and alternate coolant circulation is required to be established within 1 hour from the time RHR A was secured.

Answer D Part 1 is is correct. Part 2 is plausible and wrong for the same reasons as stated for distractor B.

Technical References: TS 3.9.7 [RHR – High Water Level] and bases, Procedure 2.1.20.2 [Cycle Specific Fuel Transfer and Alternate Cooling Guideline](Rev 20)

References to be provided to applicants during exam: TS 3.9.7 [RHR – High Water Level] (LCO and Actions)

Learning Objective: INT007-05-10 EO- 2, 12

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.43(b)(2)	
Level of Difficulty:	3	
SRO Only Justification:		

This question involves application of TS bases to determine TS actions.

PSA applicability:

Top 10 Risk Significant Systems – RHR

Examination Outline Cross-Reference	Level	SRO
295023 Refueling Acc / 8 Ability to determine and/or interpret the following as they apply to REFUELING ACCIDENTS: (CFR: 43.5) **AA2.04 †Occurrence of fuel handling accident**	Tier#	1
	Group#	1
	K/A #	295023 AA2.04
	Rating	4.1
	Revision	0
Revision Statement:		

Question 78

Reference Provided

Refueling is in progress.

A spent fuel bundle is dropped just as it passes through the reactor cavity gate into the spent fuel pool resulting in the following:

- The fuel bundle is damaged.
- The fuel pool liner is cracked 3 feet above the storage racks.
- Refuel floor personnel report pool level is just below the skimmers and slowly lowering.
- RMA-RA-1 Fuel Pool Area Radiation Monitor reading has risen to 200 mR/hr.
- RMP-RM-452 A-D Reactor Building Ventilation Exhaust Plenum Radiation Monitor readings are 50 mR/hr.

Which of the following identifies the HIGHEST required Emergency Classification declaration for this condition and why?

A. Alert based on radiation level

B. Alert based on pool water level

C. Unusual Event based on radiation level

D. Unusual Event based on pool water level

Answer: A. Alert based on radiation level
Explanation:

This question requires interpreting conditions resulting from a fuel handling accident for execution of the EALs. It is not a direct look-up because it requires knowledge of the Reactor Building Ventilation Exhaust Plenum Radiation Monitor high-high trip setpoint.
Answer A is correct because a fuel handling accident has occurred in Mode 5 and RMP-RM-452 A-D Reactor Building Ventilation Exhaust Plenum Radiation Monitor readings are 50 mr/hr, which is above the TS high-high setpoint of 49 mr/hr (actual high-high setpoint is 10 mr/hr). Thus EAL AA2.1 is met due to radiation level. (Also note RM-RA-1 at 50 mR/hr is below the Alert threshold of 50 **R**/hr)

Distracters:
Answer B is plausible because an unprepared applicant may extrapolate a reduction in pool level due to a liner leak into a condition that will inevitably result in fuel uncover, since there is no valve that can be closed to isolate the leak. There are mitigating actions in plant procedure to patch these types of leaks. There are also many methods of pool makeup to prevent the fuel from being uncovered. The implication in the stem is that the leak size is well within the capacity of makeup systems; therefore, EAL AA2.2 should not be chosen based on pool water level.

Answer C is plausible for the same reason as Answer A and also reflects EAL AU2.1. However the applicant could be attracted to attributing an EAL entry to radiation level if the applicant believes Reactor Building Ventilation Exhaust Plenum Radiation Monitor is elevated. the applicant may consider rad level more significant than a slight change in pool level as reflected in the stem.

Answer D is plausible because a less severe fuel handling accident might not cause high radiation levels. The unprepared applicant who does not know the Reactor Building Ventilation Exhaust Plenum Radiation Monitor high-high trip setpoint may believe rad levels are below the high-high setpoint, and since no information is given regarding rad monitor RMA-RA-2, concludes the only EAL exceeded is due to visual observation of lowered pool level, which is only a UE per EAL AU2.1.

Technical References: Procedure 5.7.1, Emergency Classification: EPIPEALCOLD (EAL Wall Chart – Modes 4,5))(Rev 13)

References to be provided to applicants during exam: EPIPEALCOLD (EAL Wall Chart – Modes 4,5

Learning Objective: GEN0030401C0C050E

Question Source:	Bank #	CNS NRC 12/15 #81
(note changes; attach parent)	Modified Bank #	
	New	

Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X

10CFR Part 55 Content:	55.43(b)(5) & (7)
Level of Difficulty:	3
SRO Only Justification:	
This requires event classification at the Emergency Director level.	
PSA applicability:	
N/A	

Examination Outline Cross-Reference	Level	SRO
295038 High Off-site Release Rate / 9 **2.4.18** Knowledge of the specific bases for EOPs. (CFR: 43.1)	**Tier#**	1
	Group#	1
	K/A #	295038 G2.4.18
	Rating	4.0
	Revision	1
Revision Statement: Per Chief Examiner review comments – rearranged stem conditions chronologically, simplified stem, added ">" to each of the answers. Per discussion with Chief Examiner, removed second portion of Part 2 in answers. Added "during the pressure reduction" to part 2 stem for bounding due to validator comments, since the RPV will always fully depressurize, eventually. (Kept "Rem Thyroid CDE" and "Approximately" in answers for readability based on validator comments.) Accentuated and enhanced clarity that HPCI is the only available injection system due to numerous validator comments.		

Question 79

An earthquake caused a LOCA and steam leak in the turbine building that can NOT be isolated.

- Reactor has been scrammed
- Reactor pressure 800 psig, stable
- Reactor water level -140 inches (Wide Range), stable
- HPCI is the **ONLY** injection system and is at rated injection flow
- **NO** other injection systems are available
- Offsite dose rates are rising

(1) What is the offsite dose **threshold value** at the site boundary that is a basis for transitioning to Emergency Depressurization IAW EOP-5A?

AND

(2) If Emergency Depressurization is required due to high offsite dose under these conditions, which one of the following represents the lowest reactor pressure to be attained during the depressurization?

A. (1) >1 Rem thyroid CDE
(2) Approximately 0 psig

B. (1) >1 Rem thyroid CDE
(2) Approximately 150 psig

C. (1) >5 Rem thyroid CDE
   (2) Approximately 0 psig

D. (1) >5 Rem thyroid CDE
   (2) Approximately 150 psig

Answer: D (1) >5 Rem thyroid CDE (2) Approximately 150 psig
Explanation: When a break in a primary system that cannot be isolated results in offsite doses that are projected to reach the GE level, EOP-5A step RR-3 requires emergency depressurization to be performed to reduce the driving head of the leak and the radioactive discharge rate due to the immediate threat to continued health and safety of the public. EALs provide the basis for the threshold of EOP-5A step RR-3. EAL AG1.2 lists the GE level for dose at the site boundary as either >1 Rem TEDE or > 5 Rem thyroid CDE. EOP-5A step RR-3 requires transitioning to Emergency Depressurization BEFORE the GE level is reached, so 1 Rem TEDE and 5 Rem thyroid CDE are the limiting values for offsite dose requiring Emergency Depressurization. Of those two values, 5 Rem thyroid CDE is the value listed as the correct answer to Part 1 of the question. (5 Rem TEDE is 5 times the GE level.) For Part 2, the PSTG basis for Emergency Depressurization states full depressurization should be prevented if that would result in loss of systems needed to provide adequate core cooling. PSTGs state loss of adequate core cooling would complicate the event and result in an increase in radioactivity released. In this case, the only injection system available is HPCI, and it is maintaining level stable by injecting rated flow. EOP-2A step RC/P-9 requires preventing full depressurization and controlling RPV pressure as low as practicable while maintaining injection. TS SR 3.5.1.8 requires HPCI be demonstrated to achieve rated flow at low reactor pressure, ≤165 psig. Surveillance procedure 6.HPCI.104 gives a target RPV pressure band of 145-150 psig for performing this test. Therefore, 150 psig is a known steam supply pressure at which rated HPCI flow can be maintained, and serves as a basis for limiting the RPV pressure reduction during Emergency Depressurization.
Distracters: Answer A Part 1 is plausible because a GE may be based on 1 Rem TEDE dose. It is wrong because 1 Rem thyroid CEDE is only 20% the offsite dose limit on which a GE is based, 5 Rem thyroid CEDE. Part 2 is plausible because the basis for emergency depressurization being required is to mitigate the immediate threat to continued health and safety of the public, and fully depressurizing would, in the short term, reduce the offsite release the most by eliminating the driving head of the leak. It is wrong because per PSTG bases for Contingency #2, Emergency RPV Depressurization states full depressurization should be prevented if that would result in loss of systems needed to provide adequate core cooling, which would complicate the event and result in an increase in radioactivity released overall. Therefore, sufficient steam supply pressure for full flow HPCI operation should be maintained, and approximately

150 psig is the pressure at which full HPCI flow is demonstrated IAW SR 3.5.1.8, not 0 psig.

Answer B Part 1 is plausible and wrong for the same reasons as stated for distractor A. Part 2 is correct.

Answer C Part 1 is correct. Part 2 is plausible and wrong for the same reasons as stated for distractor B.

Technical References: EOP-5A [Secondary Containment Control](Rev 16), EOP-1A [RPV Control](Rev 20), EOP-2A [Emergency Depressurization and Steam Cooling](Rev 19); AMP-TBD00 [EOP/PSTG Technical Basis] for Contingency #2 and step RR-3; Procedure 6.HPCI.104 [HPCI Cycle (≤ 165 psig) Test Mode Surveillance Operation](Rev 15), TS 3.5.1 [ECCS Operating](SR 3.5.1.8), **:** Procedure 5.7.1, Emergency Classification: EPIPEALHOT [EAL Wall Chart – Modes 1,2, or 3](Rev 13)

References to be provided to applicants during exam: none

Learning Objective: INT008-06-07 EO-1, 9

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X

10CFR Part 55 Content:	55.43(b)(5)
Level of Difficulty:	3

SRO Only Justification:

Part 1 of this question requires SRO knowledge of EALs.

PSA applicability:

N/A

Examination Outline Cross-Reference	Level	SRO
295031 Reactor Low Water Level / 2 Ability to determine and/or interpret the following as they apply to REACTOR LOW WATER LEVEL : (CFR: 43.5) **EA2.04 Adequate core cooling**	**Tier#**	1
	Group#	1
	K/A #	295031 EA2.04
	Rating	4.8
	Revision	1
Revision Statement: Per Chief Examiner review comments – simplified stem and removed second portion of each answer.		

Question 80

A LOCA and reactor scram have occurred with the following:

- RPV water level: -170 inches Corrected Fuel Zone, lowering 2 inches per minute
- RPV pressure: 750 psig, stable
- Suppression Pool temperature: 150°F, rising slowly
- Torus pressure: 20 psig, rising slowly

- RHR pumps A and C are the ONLY available pumps and are running on minimum flow.

CRS is implementing EOP-1A, and based on above conditions, is required to immediately transition to the ….

A. Drywell Spray leg of EOP-3A.

B. Steam Cooling leg of EOP-2A.

C. Torus Water Temperature leg of EOP-3A.

D. Emergency Depressurization leg of EOP-2A.

Answer: D. Emergency Depressurization leg of EOP-2A.
Explanation: This question implicitly tests understanding of adequate core cooling with respect to low RPV water level for proper EOP selection and prioritization and for how RHR pumps should be utilized. The main goal of EOPs is to prevent core damage by providing adequate core cooling. EOPs prioritize adequate core cooling, preferring submergence, over other functions. With level below TAF, -158”, and lowering,

maintaining adequate core cooling is not assured. For the conditions given, RPV level low and only RHR A/C pumps available, EOP-1A step RC/L-6 would have already directed aligning RHR A/C for injection. With RPV level -158" and lowering and RHR A/C available, EOP-1A steps RC/L-11, 12, and 14 are satisfied, and step RC/L-15 should be judged by the SRO as being met, level cannot be maintained above -183", since there is no injection currently. Therefore, Emergency Depressurization per EOP-2A is required, and available RHR pumps should be maintained aligned for injection.

Distracters:
Answer A is plausible because Torus pressure is above 10 psig, and EOP-3A step PC/P-3 requires DW Spray above 10 psig. It is wrong because adequate core cooling is prioritized over PC pressure. EOP-3A step DS-4 states start DW Sprays using only those RHR pumps not required for adequate core cooling. With RPV level below TAF and only RHR loop A available, pumps in RHR loop A must be used for LPCI injection.

Answer B is plausible because there is no injection currently since RPV pressure is above RHR pump shutoff head and level is below TAF and lowering. The unprepared applicant may incorrectly answer step RC/L-14 as no means of RPV injection is available, since RHR is not currently injecting, which would direct the applicant to Steam Cooling. This is wrong because RHR A/C are considered available if depressurizing will result in injection.

Answer C is plausible because Torus water temperature is well above 95°F, and EOP-3A step SP/T-2 and 3 require operation of Suppression Pool Cooling above 95°F. It is wrong because adequate core cooling is prioritized over PC pressure. EOP-3A step SP/T-3 states operate SP Cooling using only those RHR pumps not required for adequate core cooling. With RPV level below TAF and only RHR Loop A available, pumps in RHR loop A must be used for LPCI injection.

Technical References: EOP-1A [RPV Control](Rev 20), EOP-2A [Emergency Depressurization and Steam Cooling](Rev 19), EOP-3A [Primary Containment Control](Rev 17), AMP-TBD00 [EOP/PSTG Technical Basis]

References to be provided to applicants during exam: none

Learning Objective: INT008-06-09 EO-9

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X

Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X

10CFR Part 55 Content:	55.43(b)(5)
Level of Difficulty:	3
SRO Only Justification:	
This requires assessment of facility conditions and prioritization and selection of procedures and equipment utilization during emergency situations.	
PSA applicability:	
Top 10 Risk Significant System – Residual Heat Removal Top 10 Risk Significant Operator Action – failure to Emergency Depressurize with SRVs	

Examination Outline Cross-Reference	Level	SRO
600000 Plant Fire On Site / 8 **2.4.30** Knowledge of events related to system operation / status that must be reported to internal organizations or external agencies, such as the State, the NRC, or the transmission system operator. (CFR: 43.5)	**Tier#**	1
	Group#	1
	K/A #	600000 G2.4.30
	Rating	4.1
	Revision	0
Revision Statement:		

Question 81

Reference Provided

The plant is at 100% power.

At time zero (T=0), the following alarm is received:

SW PUMP ROOM ZONE 32	PANEL/WINDOW: FP-1/F-4

At time 17 minutes (T=17), responders have NOT yet reported findings, AND the following alarms are received:

SERVICE WATER PUMP B TRIP	PANEL/WINDOW: B-3/B-6
SERVICE WATER PUMP B OVLD/GROUND	PANEL/WINDOW: B-3/C-6

(1) What is the highest Emergency Classification for this event?

AND

(2) What is the MAXIMUM time limit from the time of emergency declaration to initial notification of the **NRC**?

A. (1) Unusual Event
(2) 15 minutes

B. (1) Unusual Event
(2) 1 hour

C. (1) Alert
(2) 15 minutes

D. (1) Alert
(2) 1 hour

Answer: D. (1) Alert (2) 1 hour
Explanation: The first alarm in the stem reflects at least one detector is in alarm (smoke and/or flame) in the fire detection zone for the SW Pump Room. Procedure 5.1INCIDENT Att. 2 step 5.1.1 requires assuming a fire exists based on a fire alarm unless disproved within 15 minutes, corresponding to EAL HU2.1, Unusual Event. Since the stem states 17 minutes elapses and no report of conditions has been received disproving a fire exists, at a minimum HU2.1 is met. With indication of failure of equipment within the affected area, trip of SW Pump B, EAL HA2.1, Alert, is met based on control room indication of degraded safety system performance associated with the affected area. An emergency declaration requires notification to state and local agencies and the NRC. Procedure 5.7.6 requires state and local initial notifications within 15 minutes of emergency declaration and NRC notification as soon as possible following state and local notifications, but no later than 1 hour after a declared emergency.
Distracters: Answer A Part 1 is plausible because 15 minutes elapses from receipt of a fire alarm and existence of a fire has not been disproved by personnel dispatched to the scene; therefore EAL HU2.1, Unusual Event is met. It is wrong because EAL HA2.1, Alert, is met due to control room indication of SW Pump B trip, and the stem asks for the highest emergency classification for the event. Part 2 is plausible because procedure 5.7.6 requires some notification to state and local agencies be made within 15 minutes of emergency declaration. It is wrong because the stem asks about NRC notification, which is required within a maximum of 1 hour IAW procedure 5.7.6. Answer B Part 1 is plausible and wrong for the same reasons as stated for distractor A. Part 2 is correct. Answer C Part 1 is correct. Part 2 is plausible and wrong for the same reasons as stated for distractor A.
Technical References: Procedure 5.7.1 [Emergency Classification](Rev 56), Procedure 5.7.2 [Emergency Director EPIP](Rev 34), Procedure 5.7.6 [Notification](Rev 68); Alarm cards FP-1/F-4(Rev 12) and B-3/B-6, B-3/C-6(Rev 37); EAL Chart EPIPEALHOT (Rev 13), Procedure 5.1INCIDENT [Site Emergency Incident](Rev 35)

References to be provided to applicants during exam: EPIPEALHOT (EAL Wall Chart Modes 1, 2, 3)		
Learning Objective: GEN020-04-04 EO-3		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.43(b)(5)	
Level of Difficulty:	2	
SRO Only Justification:		
This question requires knowledge of administrative procedures that require implementation and coordination of emergency implementing procedures.		
PSA applicability:		
Top 10 Risk Significant System – Service Water		

Examination Outline Cross-Reference	Level	SRO
700000 Generator Voltage and Electric Grid Disturbances / 6 Ability to determine and/or interpret the following as they apply to GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES: (CFR:43.5) **AA2.05 Operational status of offsite circuit**	Tier#	1
	Group#	1
	K/A #	700000 AA2.05
	Rating	3.8
	Revision	0
Revision Statement:		

Question 82

The plant is in Mode 1.

The SSST was declared inoperable at 0700 due to a voltage disturbance on the 161 kV line.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One offsite circuit inoperable.	A.1 Perform SR 3.8.1.1 for OPERABLE offsite circuit.	1 hour <u>AND</u> Once per 8 hours thereafter

IAW TS rules of application…

Which of the following times for initial and subsequent surveillance completion reflect the **LONGEST ALLOWED** intervals that will satisfy TS 3.8.1 Action A.1?

A. Initial performance at 0800;
Next performance at 1600

B. Initial performance at 0800;
Next performance at 1800

C. Initial performance at 0815;
Next performance at 1615

D. Initial performance at 0815;
Next performance at 1815

Answer: B. Initial performance at 0800; Next performance at 1800		
Explanation: This question tests interpretation of operational status (inoperable) of offsite power sources due to grid conditions with respect to required TS actions for that status. The subject TS Action involves a SR that verifies the condition of the remaining offsite circuits. There are two offsite AC sources credited for supplying CNS emergency buses. The Startup transformer (SSST), which is supplied by the 161kV line, and the Emergency transformer (ESST), which can be supplied via 161kV line or 69kV line. This question tests interpretation of TS requirements for an inoperable offsite power source. Per TS SR 3.0.2, surveillances with frequencies specified on a “once per” basis, the surveillance frequency extension of 1.25 times the specified frequency is allowed for subsequent performances following the initial performance, but not for the initial performance, which in the case given is 1 hour. Therefore, the initial performance must be completed no later than 0800. The subsequent performance may then be 8 X 1.25 = 10 hours later, at 1800. The bases SR 3.0.2 clearly states SR 3.0.2 is for operational flexibility.		
Distracters: Distracter A is plausible because no completion time extension may be applied for the initial performance. The unprepared applicant might not know the 25% frequency extension may be applied to subsequent performance and select this answer. It is wrong because the question asks for the longest allowed frequency. If the frequency extension of 25% is utilized for the second performance, it would be allowed to be delayed an additional 2 hours from 1600. Distractor C is plausible if the applicant mistakes which performance, initial or subsequent, the 25% frequency extension may be applied to. The unprepared applicant might apply the extension to the first performance and not to the next. Answer C is wrong because it reflects the 25% extension being applied to the first performance, which must be done by 0800, not 1000. Distractor D is plausible if the applicant mistakes how the 25% frequency extension may be applied. Answer D is wrong because it reflects the 25% extension being applied to both initial and subsequent performances, but it may only be applied to the subsequent.		
Technical References: TS 3.8.1 [AC Sources – Operating], SR 3.0.2 and bases, TS 1.3		
References to be provided to applicants during exam: none		
Learning Objective: INT00705010010700		
Question Source:	Bank #	

(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.43(b)(2)	
Level of Difficulty:	3	
SRO Only Justification:		
This requires knowledge of application of TS 3.0.2, described in TS 3.0.2 bases.		
PSA applicability:		
N/A		

Examination Outline Cross-Reference	Level	SRO
295014 Inadvertent Reactivity Addition / 1 **2.4.4** Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures.(CFR: 43.2)	**Tier#**	1
	Group#	2
	K/A #	295014 G2.4.4
	Rating	4.7
	Revision	1
Revision Statement: Per Chief Examiner review comments, replaced question due to potential K/A mismatch		

Question 83

The plant is at rated power.

SRV 71B is INOPERABLE.

HPCI initiates AND injects due to a spurious initiation signal that is sealed in.

(1) What procedure(s) is/are required to be entered?

AND

(2) As a result of immediate operator actions for this condition, what is the LATEST time allowed by TS to be in Mode 3? **(Assume reactor pressure is maintained above 500 psig.)**

A. (1) 2.4CSCS [Inadvertent CSCS Initiation], ONLY
(2) 13 hours

B. (1) 2.4CSCS [Inadvertent CSCS Initiation], ONLY
(2) 84 hours

C. (1) 2.4CSCS [Inadvertent CSCS Initiation] AND 2.4RXPWR [Reactor Power Anomalies]
(2) 13 hours

D. (1) 2.4CSCS [Inadvertent CSCS Initiation] AND 2.4RXPWR [Reactor Power Anomalies]
(2) 84 hours

Answer: A. (1) 2.4CSCS [Inadvertent CSCS Initiation], ONLY (2) 13 hours

Explanation:
HPCI suction is normally aligned to the CST, and it can take suction from the Suppression Pool. HPCI initiation at 100% power is a cold water injection transient, which causes reactor power to rise. Immediate operator actions of procedure 2.4CSCS places HPCI Aux Oil Pump in Pull-To-Lock, which makes HPCI inoperable. Per TS 3.5.1 Condition H, with one or more ADS valves and HPCI inoperable, TS 3.0.3 must be entered immediately. TS 3.0.3 requires placing the unit in a condition where the LCO is not required by being in Mode 2 within 7 hours, Mode 3 within 13 hours, and in Mode 4 within 37 hours, as necessary. HPCI is required to be operable in Mode 1 and in Modes 2 and 3 when reactor pressure is above 150 psig. Since the stem states assume pressure will remain above 500 psig, Mode 3 must be entered within 13 hours.

Distracters:
Answer B part 1 is correct. Part 2 is plausible because it reflects the time allowed for entry into Mode 3 for one ADS valve and one Core Spray pump inoperable. TS 3.5.1 Condition F requires restoring the ADS valve or CS pump operable within 72 hours, or Condition G must be entered. Condition G requires being in Mode 3 within 12 hours. [72 + 12 = 84]. It is wrong because TS 3.5.1 Condition H applies, which requires immediate entry into TS 3.0.3.

Answer C part 1 is plausible because HPCI initiation is a cold water injection transient, which causes reactor power to rise, and some power transients are an entry condition for 2.4RXPWR. It is wrong because 2.4RXPWR entry is only required for unexplained power transients. The given conditions do not result in an unexplained power transient, because 2.4CSCS Att. 5 states a power rise is expected for cold water injection. Part 2 is correct.

Answer D part 1 is plausible and wrong for the same reasons stated for distractor A. Part 2 is plausible and wrong for the same reasons stated for distractor B.

Technical References: Procedure 2.4CSCS [Inadvertent CSCS Initiation](Rev 9), Procedure 2.4RXPWR [Reactor Power Anomalies](Rev 7), TS 3.0.3, TS 3.5.1 [ECCS Operating]

References to be provided to applicants during exam: none

Learning Objective: INT007-05-01 EO-4

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X

Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X

10CFR Part 55 Content:	55.43(b)(2)
Level of Difficulty:	3
SRO Only Justification:	
This question requires application of TS 3.0.3.	
PSA Applicability	
Top 10 Risk Significant Systems – HPCI, ADS/SRV	

Examination Outline Cross-Reference	Level	SRO
295029 High Suppression Pool Wtr Lvl / 5	Tier#	1
Ability to determine and/or interpret the following as they apply to HIGH SUPPRESSION POOL WATER LEVEL : (CFR: 43.5)	Group#	2
	K/A #	295029 EA2.01
	Rating	3.9
EA2.01 Suppression pool water level	Revision	0
Revision Statement:		

Question 84

The plant is in Mode 3 preparing for startup when the following alarm is received:

SUPPR POOL NR/WR HIGH LEVEL	PANEL/WINDOW: 9-3-2/F-5

Actual Suppression Pool level is:

An acceptable risk evaluation for this condition has NOT been performed.

(1) Is entry into Mode 2 allowed by TS?

AND

(2) Is entry into Mode 4 allowed by TS?

A. (1) No
(2) No

B. (1) No
(2) Yes

C. (1) Yes
(2) No

D. (1) Yes
(2) Yes

Answer: B. (1) No (2) Yes
Explanation: TS 3.6.2.2 requires SP level to be within limits during Modes 1, 2, and 3. Failure to restore SP level to within limits would eventually lead to entry into TS 3.6.2.2 Condition B, which requires shut down to Mode 4. TS 3.0.4 only allows entry into a mode specified in the applicability if there is a stated exception to 3.0.4 in the TS, or if there is an acceptable risk evaluation performed, or if the associated TS Actions permit continued operation indefinitely. Since the stem states there is No acceptable risk evaluation, that allowance does not apply. Since TS 3.6.2.2 Actions do not permit continued operation indefinitely and there is no stated exception to the provisions of TS 3.0.4, TS 3.0.4 does not allow going from Mode 3 to Mode 2 with SP level out of limits without an acceptable risk evaluation. Also, TS 3.0.4 states shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit. So entry into Mode 4 is allowed.
Distracters: Answer A Part 1 is correct. Part 2 is plausible because TS 3.0.4 may prohibit changing to Mode 4 in some cases, such as from Mode 5 to Mode 4 with Shutdown Cooling systems inoperable. This is plausible to the unprepared applicant who does understand application of TS 3.0.4 or does not remember TS 3.6.2.2 would ultimately require plant shutdown. It is wrong because TS 3.6.2.2 is not applicable in Mode 4 and because going to Mode 4 would be part of a unit shutdown. Answer C Part 1 is plausible because TS 3.6.2.2 Actions do not permit continued operation indefinitely (i.e TS 3.0.4a), there is not an acceptable risk evaluation for changing modes (I.e. TS 3.0.4b), and there is no stated exception to the provisions of TS 3.0.4 (i.e. TS 3.0.4c). Therefore, TS 3.0.4 does not allow going from Mode 3 to Mode 2. Part 2 is plausible and wrong for the same reasons as stated for distractor A. Answer D Part 1 is plausible and wrong for the same reasons as stated for distractor C. Part 2 is correct.

Technical References: TS 3.6.2.2 [Suppression Pool Water Level], alarm card 9-3-2/F-5(Rev 32), TS 3.0.4		
References to be provided to applicants during exam: none		
Learning Objective: INT007-05-07 EO-1, INT007-05-01 EO-2		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.43(b)(2)	
Level of Difficulty:	3	
SRO Only Justification:		
This question requires application of TS 3.0.4.		
PSA applicability:		
Top 10 Risk significant Systems – Primary Containment		

Examination Outline Cross-Reference	Level	SRO
500000 High CTMT Hydrogen Conc. / 5 **2.4.6** Knowledge of EOP mitigation strategies. (CFR: 43.5)	**Tier#**	1
	Group#	2
	K/A #	500000 G2.4.6
	Rating	4.7
	Revision	0
Revision Statement:		

Question 85

LOCA conditions exist:

- Core Sprays at rated flow have stabilized RPV level at -190 inches Corrected Fuel Zone.
- Drywell Spray is operating.
- Torus pressure is 15 psig, lowering.
- Drywell H2 concentration is 1.1%, steady.
- PC radiation level is 50 R/hr, steady.
- Release rate from the ERP is at the ALERT level.

Which one of the following strategies is required for hydrogen control IAW EOP-3A [Primary Containment Control]?

A. Vent the PC IAW EOP 5.8.21 [PC Venting and Hydrogen Control (Less than Combustible Limits)].

B. Vent the PC IAW EOP 5.8.22 [PC Venting and Hydrogen Control (Greater than Combustible Limits)].

C. DO NOT vent the PC. Secure Drywell Spray IAW Procedure 2.2.69.3 [RHR Suppression Pool Cooling and Containment Spray].

D. DO NOT vent the PC. Monitor PC hydrogen and oxygen concentrations IAW Procedure 2.2.60.1 [Containment H_2/O_2 Monitoring System].

Answer: D. DO NOT vent the PC. Monitor PC hydrogen and oxygen concentrations IAW Procedure 2.2.60.1 [Containment H_2/O_2 Monitoring System].

Explanation:
PSTGs for PC gas control is limited to venting and purging to reduce gas concentrations, since large concentrations are not expected if the core remains adequately cooled. The conditions given reflect adequate core cooling exists, since CS is at rated flow and RPV level is above -209 inches; therefore, SAGs have not been entered, and EOP-3A is in use. DW hydrogen concentration is above 1%. EOP-3A step PC/G-3 is met. Step PC/G-4 directs venting the PC IAW EOP 5.8.21 [PC Venting and Hydrogen Control (Less than Combustible Limits)] only if PC radiation is less than 115 R/hr, which it is, and offsite dose levels are expected to remain below ODAM limits. In this case, release rate from the ERP is above ODAM limits, since it is given as being at the Alert level, which is above the ODAM instantaneous release limit (EAL AA1.1). Therefore, the PC should not be vented for hydrogen control at this time. EOP-3A step PC/G-1 requires monitoring PC hydrogen and oxygen concentrations IAW Procedure 2.2.60.1 [Containment H_2/O_2 Monitoring System] any time EOP-3A is entered.

Distracters:
Answer A is plausible because EOP 5.8.21 is the appropriate procedure for venting PC to lower H2 concentration per EOP-3A for H2 levels that may be experienced while EOP flowcharts are in use. It is plausible that release rates are below ODAM limits, thereby permitting PC venting, because the UE value for release from the ERP is only a fraction of the ODAM limit. It is wrong because the offsite release rate from RB Vent is above ODAM limits; therefore, venting is prohibited by EOP-3A steps PC/G-2 and PC/G-4.

Answer B is plausible because EOP 5.8.22 is an appropriate procedure for venting PC to lower H2 concentration. However, it is entered as directed from SAG-2, not EOP-3A, since it involves H2 concentrations above the flammability limit. It is plausible that release rates are below ODAM limits, thereby permitting PC venting, because the UE value for release from the ERP is only a fraction of the ODAM limit. This answer is wrong because the offsite release rate from RB Vent is above ODAM limits; therefore, venting is prohibited by EOP-3A steps PC/G-2 and PC/G-4 and because EOP 5.8.21 would be the correct procedure, not EOP 5.8.22, even if venting was allowed.

Answer C is plausible because PC venting is not allowed due to offsite dose rates above ODAM limits. Securing DW Spray now would be plausible to the unprepared applicant because DW pressure is lowering. The unprepared applicant may confuse the effect of spray operation on H2 control, incorrectly conclude it is detrimental, and deduce spray should be secured as early as possible. PSTG/SATGs state DW Spray is used for gas control when hydrogen concentration is above flammability limits, expected only during severe accident conditions. In this case, hydrogen is not above the flammability limit, 4%, so DW Spray is not required for hydrogen control, but it is not detrimental to H2 control and may be beneficial. DW Spray is required when Torus pressure exceeds 10 psig. Therefore, it should remain in service IAW PC pressure control strategies, and should operate to control DW pressure below 10 psig and above 0 psig.

Technical References: EOP-3A [Primary Containment Control](Rev 17), AMP-TBD00 [EOP/PSTG Technical Basis], EOP 5.8.21 [PC Venting and Hydrogen Control (Less than Combustible Limits)](Rev 18), Vent the PC IAW EOP 5.8.22 [PC Venting and Hydrogen Control (Greater than Combustible Limits)](Rev 18), Procedure 5.7.1 [Emergency Classification](Rev 56)		
References to be provided to applicants during exam: none		
Learning Objective: INT008-06-13 EO-4b		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.43(b)(5)	
Level of Difficulty:	4	
SRO Only Justification:		
This question requires diagnosing plant conditions and selecting the appropriate procedure section with which to proceed.		
PSA applicability:		
Top 10 Risk significant Systems – Primary Containment		

Examination Outline Cross-Reference	Level	SRO
203000 RHR/LPCI: Injection Mode Ability to (a) predict the impacts of the following on the RHR/LPCI: INJECTION MODE (PLANT SPECIFIC) ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: **A2.04 A.C. failures**	Tier#	2
	Group#	1
	K/A #	203000 A2.04
	Rating	3.6
	Revision	0
Revision Statement:		

Question 86

The following conditions exist during a large break LOCA from 100% power:

- EMERGENCY XFMR BKR **1GS** has failed in the OPEN position.
- DIESEL GEN 2 BKR **EG2** has failed in the OPEN position.
- Startup Transformer (SSST) is UNAVAILABLE.
- RHR Pump A is UNAVAILABLE.
- RPV Pressure is 35 psig and steady.

(1) Which RHR Loop is available for LPCI injection?

AND

(2) Which one of the following methods is required if additional injection is needed to assure adequate core cooling?

A. (1) Loop A
(2) Enter 5.3EMPWR [Emergency Power during Modes 1, 2, OR 3] and use the Supplemental Diesel Generator (SDG) to re-energize 4160V 1G Bus.

B. (1) Loop A
(2) Enter 5.3ALT-STRATEGY [Alternate Core Cooling Mitigating Strategies] and inject using Fire Protection to RHR.

C. (1) Loop B
(2) Enter 5.3EMPWR [Emergency Power during Modes 1, 2, OR 3] and use the Supplemental Diesel Generator (SDG) to re-energize 4160V 1G Bus.

D. (1) Loop B
(2) Enter 5.3ALT-STRATEGY [Alternate Core Cooling Mitigating Strategies] and inject using Fire Protection to RHR.

Answer: D. (1) Loop B (2) Enter 5.3ALT-STRATEGY [Alternate Core Cooling Mitigating Strategies] and inject using Fire Protection to RHR.	

Explanation:
The turbine generator is off line following a scram due to LOCA conditions, so the Normal Transformer (NSST) is unavailable. The SSST is given as unavailable, so the only two potential sources to critical AC buses F and G are the Emergency Transformer (ESST) and their respective DGs. Breaker 1GS is the ESST feeder to bus G and is unavailable. DG2 output breaker is also unavailable; therefore, bus G (Div 2) is de-energized. RHR Loop A consists of RHR pumps A & C. RHR Loop B consists of RHR pumps B & D. Loss of Div 2 4160V Bus 1G de-energizes RHR pumps C & D and AC valves in Loop B. The only pump with power is RHR Pump B (Loop B) via bus F, with RHR-MO-27B (Outboard Injection Valve) de-energized in the OPEN position provides LPCI flow to the RPV with RHR-MO25B (Inboard Injection Valve) being DC powered. No RHR pumps are available in RHR Loop A (RHR pump A unavailable and C with no power). Per 5.3ALT-STRATEGY, RCIC is preferred over fire water injection, but RPV pressure is too low to support RCIC operation. The connection of Fire Protection to SW Emergency Core Flooding IAW 5.3ALT-STRATEGY ties into RHR Loop A piping. This is accomplished by tying in Fire Protection to the suction side of the RHRSW Booster pump via a 3 inch fire hose. Since no RHR pumps are available in RHR loop A and since RPV pressure is below the fire water supply pressure, fire water injection is a viable option. (Fire pump discharge pressure is approximately 150 psig per Procedure 6.FP.102 [Station Fire Pump Surveillance Testing].)

Distracters:
Answer A Part 1 is plausible because the stem does not directly state anything is wrong with RHR pump C or the bus F supply from ESST or DG1. The unprepared applicant may confuse RHR pump power supplies and believe RHR pump C is available via RHR loop A. This is wrong because RHR pump C has no power due to bus G de-energized, and RHR pump A is given as unavailable. So no pump in RHR loop A is available. Only RHR pump B in RHR loop B is available, and the only other active component required for LPCI alignment in loop B is the DC powered injection valve, which is available. Part 2 is plausible because 5.3EMPWR allows for aligning the Supplemental DG to bus F or G if one is de-energized. It is wrong in this case because the SDG, like the ESST, also uses breaker 1GS to supply bus G, breaker 1GS must be functional and closed; however, 1GS has failed open and is unavailable.

Answer B Part 1 is plausible and wrong for the same reasons as given for distractor A. Part 2 is correct.

Answer C Part 1 is correct. Part 2 is plausible and wrong for the same reasons as given for distractor A.

Technical References: EOP-1A [RPV Control](Rev 20), Procedures 5.3ALT-STRATEGY [Alternate Core Cooling Mitigating Strategies](Rev 53), 5.3EMPWR

[Emergency Power during Modes 1, 2, OR 3](Rev 65), Procedure 6.FP.102 [Station Fire Pump Surveillance Testing](Rev 36)		
References to be provided to applicants during exam: none		
Learning Objective: COR0022302001080A		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	NRC 4/2015 #86
	New	
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.43(b)(5)	
Level of Difficulty:	3	
SRO Only Justification:		
Requires assessment of conditions and selection of procedure with which to proceed.		
PSA applicability:		
Top 10 Risk significant Systems – RHR/LPCI		

From ILT 1401 Retake exam 4/2015

Question: 86

The following conditions exist during a large break LOCA from rated power:

- No Off-Site is power available.
- DG1 is unavailable.
- RHR Pump D is unavailable.
- Core Spray Pump B is unavailable.
- Reactor Building is inaccessible.
- RPV Pressure is 35 psig and steady.

(1) Which RHR Loop is available for LPCI injection?

(2) What action is procedurally required if additional injection is required to assure adequate core cooling?

A. (1) Loop A
(2) Enter 5.3EMPWR (Emergency Power During MODES 1, 2, OR 3) and use the Supplemental Diesel Generator (SDG) to re-energize 4160V 1F Bus.

B. (1) Loop A
(2) Enter 5.3ALT-STRATEGY (Alternate Core Cooling Mitigating Strategies) and inject using Fire Protection to RHR.

C. (1) Loop B
(2) Enter 5.3ALT-STRATEGY (Alternate Core Cooling Mitigating Strategies) and inject using Fire Protection to RHR.

D. (1) Loop B
(2) Enter 5.3EMPWR (Emergency Power During MODES 1, 2, OR 3) and use the Supplemental Diesel Generator (SDG) to re-energize 4160V 1F Bus.

ANSWER:

A. (1) Loop A
(2) Enter 5.3EMPWR (Emergency Power During MODES 1, 2, OR 3) and use the Supplemental Diesel Generator (SDG) to re-energize 4160V 1F Bus.

Examination Outline Cross-Reference	Level	SRO
212000 RPS **2.2.25** Knowledge of the bases in Tech Specs for LCOs and Safety limits (CFR: 41.5 / 41.7 / 43.2)	**Tier#**	2
	Group#	1
	K/A #	212000 G2.2.25
	Rating	4.2
	Revision	0
Revision Statement:		

Question 87

Consider the following Condition from TS 3.3.1.1 [RPS Instrumentation]:

C. One or more Functions with RPS trip capability not maintained.	C.1 Restore RPS trip capability.	1 hour

According to TS Bases, …

(1) Which one of the following RPS instrumentation functions is **credited** in the safety analysis to ensure a Safety Limit is not exceeded?

AND

(2) For that function, which one of the following conditions requires entry into TS 3.3.1.1 Condition C during Mode 1?
(Assume no channels or trip systems are in trip.)

A. (1) APRM flux high
(2) APRM channels A, C, E inoperable

B. (1) APRM flux high
(2) APRM channels B, C, D inoperable

C. (1) MSIV closure
(2) All limit switches on MSIVs 80A, 80B, 80C, and 80D inoperable

D. (1) MSIV closure
(2) All limit switches on MSIVs 80A, 86A, 80B, and 86B inoperable

Answer: A. (1) APRM flux high (2) APRM channels A, C, E inoperable	

Explanation:
According to TS bases 3.3.1.1, function 2.c, APRM Flux High, along with SRVs, is credited in the safety analysis to terminate the pressure rise for the MSIV closure event and prevent the code allowable pressure limit from being exceeded. The bases for TS 3.3.1.1 function 5, MSIV Closure, explicitly states the MSIV closure scram is not credited in the safety analysis for overpressure protection and that APRM Flux High is.

The bases for TS 3.3.1.1 Condition C states the trip function is available if at least one channel in each RPS trip system is operable or in trip. APRM channels A, C and E supply RPS trip system A. If they are all inoperable and untripped, RPS A would not trip if a high flux event occurred; therefore, only RPS B would trip, and a full scram would not occur during a high flux event. With APRM channels A, C, and E inoperable and untripped, trip capability is lost and TS 3.3.1.1 Condition C is required to be entered.

Distracters:
Answer B Part 1 is correct. Part 2 is plausible because it lists three APRM channels, and each RPS trip system is supplied by three APRM channels. It is wrong because with Channels B,C, and D inoperable and untripped, RPS trip capability is maintained by channels A or E, and F. A full scram would occur if a high flux event occurred. Therefore, Condition C would not apply.

Answer C Part 1 is plausible because MSIV closure results in a RPV high pressure transient. It is wrong because TS bases states the APRM Flux High function, along with SRVs, is credited to prevent the code allowable pressure limit, the safety limit, from being exceeded during the design basis MSIV closure event, not the MSIV closure RPS trip function. Part 2 is plausible because it reflects all RPS channels for all four inboard MSIVs as inoperable and untripped, which numerically represents half of the channels for the MSIV Closure trip function. There is a MSIV limit switch input to each RPS trip system for each MSIV, 16 limit switches total. TS bases for TS 3.3.1.1 Condition C states for trip capability to be maintained for function 5, MSIV Closure, each RPS trip system must have an operable channel associated with a MSIV in three steam lines. This answer is wrong because trip capability is maintained via each RPS trip system having operable channels on each outboard MSIV in all four steam lines.

Answer D Part 1 is plausible and wrong for the same reasons as stated for distractor C. Part 2 is plausible because, for the MSIV Closure RPS trip function, the channels listed would require entry into TS 3.3.1.1 condition C. TS bases for TS 3.3.1.1 Condition C states for trip capability to be maintained for function 5, MSIV Closure, each RPS trip system must have an operable channel associated with a MSIV in three steam lines. This answer reflects RPS trip systems A and B only have operable channels associated with MSIVs in two steam lines, not three. Answer D is wrong because Part 1 is wrong.

Technical References: TS 3.3.1.1 [RPS Instrumentation] and bases

References to be provided to applicants during exam: none		
Learning Objective: INT007-05-04 EO-2		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.43(b)(2)	
Level of Difficulty:	3	
SRO Only Justification:		
This requires knowledge of TS bases for determining what constitutes entry into TS 3.3.1.1 Condition C, loss of RPS trip capability.		
PSA applicability:		
Top 10 Risk significant Systems – RPS		

Examination Outline Cross-Reference	**Level**	SRO
215005 APRM / LPRM Ability to (a) predict the impacts of the following on the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:(CFR: 41.5) **A2.08 Faulty or erratic operation of detectors / systems**	**Tier#**	2
	Group#	1
	K/A #	215005 A2.08
	Rating	3.4
	Revision	1
Revision Statement: Per Chief Examiner review comments – simplified stem and part 1 of answers, capitalized OPERABLE in part 1 of answers. Also, changed “should” to “is required to be” in stem.		

Question 88

Given the following:

- Rated power
- One B level LPRM detector fails to 0 watt/cm^2
- Associated APRM now indicates 97%

(1) Which of the following describes the operability of the associated APRM?

AND

(2) Which of the following procedures is required to be entered **FIRST** to correct this condition?

A. (1) INOPERABLE
   (2) Procedure 10.1 [APRM Calibration]

B. (1) INOPERABLE
   (2) Procedure 10.19 [LPRM and OD1 Performance Instructions]

C. (1) APRM remains OPERABLE if the gain is adjusted to within specification within 7 days
   (2) Procedure 10.1 [APRM Calibration]

D. (1) APRM remains OPERABLE if the gain is adjusted to within specification within 7 days
   (2) Procedure 10.19 [LPRM and OD1 Performance Instructions]

Answer: B. (1) INOPERABLE
(2) Procedure 10.19 [LPRM and OD1 Performance Instructions]

Explanation:
This question requires knowledge of TS SR 3.3.1.1.2, application of SR 3.0.1, and selection of plant procedures relative to a failed LPRM. SR 3.3.1.1.2 requires APRMs indicate within 2.0 of actual percent thermal power. In this case, failure of the LPRM results in the APRM reading 3% lower than actual power. SR 3.0.1 states failure to meet a SR, even if between performances of the associated surveillance procedure, constitutes failure to meet the LCO, at time of discovery. Therefore, the APRM is inoperable. Since failure of the APRM is due to failure of an individual LPRM, the first step in restoring the APRM operable is to bypass the affected LPRM. This is accomplished by procedure 10.19 [LPRM and OD1 Performance Instructions]. The sequence controlled by procedure 10.19 first bypasses the APRM to prevent spurious trips, then bypasses the inoperable LPRM, then adjusts APRM gain per procedure 10.1 [APRM Calibration] to satisfy SR 3.3.1.1.2, then unbypasses the APRM to return it to service.

Distracters:
Answer A Part 1 is correct. Part 2 is plausible because Procedure 10.1 [APRM Calibration] provides instructions for adjusting the APRM gain. The gain could be adjusted without first bypassing the LPRM. This is wrong because the LPRM has failed and is falsely biasing the APRM average low. Raising the APRM gain without first eliminating the bad signal would in essence mis-calibrate the APRM, accentuating flux response, which would be inconsistent with actual core conditions.

Answer C Part 1 is plausible because SR 3.3.1.1.2 is considered to be current (performed satisfactorily within the last 7 days) since the contrary is not stated in the stem. Since APRM signals oscillate slightly at rated power and gain adjustments are required relatively frequently, the unprepared applicant may assume the gain is only required by TS to be adjusted once per 7 days, or they may not understand the application of SR 3.0.1. This is wrong because SR 3.0.1 states failure to meet a SR, even if between performances of the associated surveillance procedure, constitutes failure to meet the LCO at time of discovery. Part 2 is plausible and wrong for the same reasons as stated for distractor A.

Answer D Part 1 is plausible and wrong for the same reasons as stated for distractor C. Part 2 is correct.

Technical References: SR 3.3.1.1.2 [RPS Instrumentation], SR 3.0.1, Procedure 10.19 [LPRM and OD1 Performance Instructions](Rev 45), Procedure 10.1 [APRM Calibration](Rev 49), Alarm Card 9-5-1/C-7 (Rev 34).

References to be provided to applicants during exam: none

Learning Objective: INT007-05-04 EO-3, INT007-05-01 EO-3

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.43(b)(2)	
Level of Difficulty:	3	
SRO Only Justification:		
This question requires knowledge of TS SR 3.3.1.1.2, application of SR 3.0.1, and selection of plant procedures relative to a failed LPRM.		
PSA applicability:		
N/A		

Examination Outline Cross-Reference	Level	SRO
218000 ADS **2.4.20** Knowledge of the operational implications of EOP warnings, cautions, and notes. (CFR: 41.10 / 43.5)	**Tier#**	2
	Group#	1
	K/A #	218000 G2.4.20
	Rating	4.3
	Revision	0
Revision Statement:		

Question 89

An ATWS is in progress:

- Reactor power 5%
- RPV water level -120 inches Wide Range, stable
- RPV pressure 900 psig, stable
- MSIVs closed
- One SRV open for pressure control
- Suppression Pool level 5.9 feet, lowering

Which of the following describes an allowable strategy for these conditions with respect to RPV pressure?

A. Emergency Depressurize using SRVs IAW EOP-6B.

B. Transition to RCIC for pressure control IAW EOP 5.8.2 [RPV Depressurization Systems (Table 2)], THEN close the SRV.

C. Close the SRV. Fully depressurize using MSL Drains and Bypass Valves IAW EOP 5.8.2 [RPV Depressurization Systems (Table 2)].

D. Close the SRV. Transition to MSL Drains and Bypass Valves for pressure control IAW EOP 5.8.12 [RPV Pressure Control Systems (Failure to Scram) (Table 12)].

Answer: C. Close the SRV. Fully depressurize using MSL Drains and Bypass Valves IAW EOP 5.8.2 [RPV Depressurization Systems (Table 2)].	

Explanation:
EOPs 5.8.12 [RPV Pressure Control Systems (Failure to Scram) (Table 12)] Caution #2 at step 4.1 and 5.8.2 [RPV Depressurization Systems (Table 2)] Caution #2 at step 6.1 state *If at any time suppression pool water level lowers to ≤ 6' (PNL 9-3/9-4, PC-LRPR-1A/1B, CONTAINMENT/TORUS PRESS & LEVEL RECORDER), STOP using SRVs for RPV Pressure Control.* Closing ADS/SRVs open for pressure control due to SP level lowering below 6 feet is not explicitly reflected in the EOP flowcharts. If SP level lowers below 6 feet, the top of the SRV discharge device, with an SRV open, steam could be introduced directly into the suppression chamber air space and result in containment failure. HPCI operation, if not needed for adequate core cooling, must be prevented per EOP-3A step SP/L-10 if SP level lowers below 11 feet, due to its exhaust uncovering, and EOP 5.8.2 requires SP level above 11 feet to use HPCI for pressure control. No such restriction exists for RCIC, since its exhaust is much smaller and because RCIC would trip on high exhaust pressure before containment increased to the point containment integrity would be challenged. Therefore, RCIC may and should be used for pressure control.

Distracters:
Answer A is plausible because SP level below 9.6 feet requires emergency depressurization IAW EOP-3A step SP/L-12. Also, applicants are most familiar with ED using ADS valves during simulator training. It is wrong because SP level below 6 feet exposes the ADS valve exhaust, which could lead to overpressurizing containment. EOP 5.8.12 and 5.8.2 Cautions disallow opening ADS valves with SP level <6 feet and require any ADS/SRVs open for pressure control to be closed. EOP-6B step FS/P-13 also allows ADS/SRV opening if SP level is above 6 feet.

Answer B is plausible because RCIC is a pressure control system and an alternate depressurization system, and RCIC is available with MSIVs closed. It is wrong because with SP level below 9.6 feet, emergency depressurization is required. RCIC alone is not able to depressurize the RPV with power at 5%, since rated RCIC steam flow only equates to approximately 0.2% power.

Answer D is plausible because SP level dictates the ADS valve must be closed, and Bypass valves and MSL drains would be effective in depressurizing the RPV. It is wrong because procedure 5.8.12 does not allow or provide steps necessary to defeat the MSIV Group 1 isolation signal and reopen MSIVs. Only procedure 5.8.2 provides steps for PTM installation and reopening MSIVs under the given conditions.

Technical References: AMP-TBD00 [EOP/PSTG Technical Basis] for EOP-6B step C2-1.3, EOP-6B [Emergency Depressurization (Failure-to-Scram)](Rev 20), Procedure 5.8.2 [RPV Depressurization Systems (Table 2)](Rev 42), Procedure 5.8.12 [RPV Pressure Control Systems (Failure To Scram) (Table 12)](Rev 28)

References to be provided to applicants during exam: none

Learning Objective: INT008-06-08 EO-4, INT008-06-13 EO-4

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.43(b)(5)	
Level of Difficulty:	3	
SRO Only Justification:		
This question requires interpreting plant conditions and knowledge and selection of EOP support procedures.		
PSA applicability:		
Top 10 Risk significant Systems – Primary Containment, ADS/Pressure Relief		

Examination Outline Cross-Reference	Level	SRO
300000 Instrument Air **2.4.45** Ability to prioritize and interpret the significance of each annunciator or alarm. (CFR: 41.10 / 43.5)	**Tier#**	2
	Group#	1
	K/A #	300000 G2.4.45
	Rating	4.3
	Revision	1
Revision Statement: Replaced question due to majority of Ops validators had difficulty understanding what was being asked without projecting other possible effects of lowering instrument air pressure. In stem, changed rated power to 100% power for consistency.		

Question 90

The plant is at 100% power.

The crew is having difficulty maintaining a Station Air Compressor in operation.

- Instrument Air AND Service Air pressures indicated on Panel A are EQUAL and lowering.
- Procedure 5.2AIR [Loss of Instrument Air] has been entered.

The following alarms are sealed in:

SERVICE AIR ISOLATION PCV-609	PANEL/WINDOW: A-4/B-4
INTAKE BLDG CONTROL AIR LOW PRESSURE	PANEL/WINDOW: A-4/G-4

(1) Which one of these Panel A-4 alarms is the highest priority?

AND

(2) Which procedure(s) is/are required to be entered <u>NOW</u> IAW Procedure 5.2AIR [Loss of Instrument Air]?

A. (1) B-4
(2) 2.1.5 [Reactor Scram], ONLY

B. (1) B-4
(2) 5.2AIR Attachment 2 [IA Pressure Loss] AND 2.1.5 [Reactor Scram]

C. (1) G-4

(2) 2.1.5 [Reactor Scram], ONLY

D. (1) G-4
(2) 5.2AIR Attachment 2 [IA Pressure Loss] AND 2.1.5 [Reactor Scram]

Answer: B. (1) B-4
(2) 5.2AIR Attachment 2 [IA Pressure Loss] AND 2.1.5 [Reactor Scram]

Explanation:
This question requires knowledge of alarm setpoints, AOP subsequent actions, and when to implement abnormal procedure attachments. Procedure 5.2AIR requires plant shutdown and concurrent performance of Att.2 when instrument air pressure lowers to 77 psig. Instrument air pressure is indicated PI-606 on Panel A and is sensed on the instrument air header just downstream of the air dryers. When pressure in the air distribution header downstream of the air receivers (upstream of Service Air isolation valve PCV-609) lowers to 77 psig as sensed by a low pressure switch, PCV-609 closes to isolate the service air header from the air receivers and instrument air header, in case the low air pressure is due to a downstream leak in service air piping. Alarm A-4/B-4 is generated from the same low pressure switch. In the case presented in the stem, Instrument Air and Service Air pressures are equal, which would be expected for loss of air compressors. Both instrument air and service air pressures lower at approximately the same rate due to normal air consumption with inadequate makeup, until PCV-609 closes. Therefore, instrument air pressure will be approximately 77 psig when alarm A-4/B-5 is received. Procedure 5.2AIR requires concurrent entry into procedure 2.1.5 to effect plant shutdown and concurrent performance of Att.2 when instrument air pressure lowers to 77 psig.

Alarm B-4 has a yellow border and background and is considered a yellow alarm. G-4 has a black border with white background. Alarms priorities are established based on their border color. Procedure 2.3.1 Att. 3 states Yellow alarms are priority II and Black alarms are priority III; therefore B-4 should be prioritized over G-4.

Distracters:
Answer A part 1 correct. Part 2 is plausible there are loss of instrument air scenarios where procedure 2.1.5 entry is required, but performance of Att. 2 is not required. Procedure 5.2AIR step 4.2 requires entry into procedure 2.1.5 if one or more control rods begin to drift, but 5.2AIR Att. 2 is not required for that reason. It is wrong because at ≤77 psig IA pressure, 5.2AIR step 4.9.1 requires entry into procedure 2.1.5 and step 4.9.7 requires performance of Att. 2.

Answer C part 1 is plausible because A-4/G-4 also requires entry into 5.2AIR for these conditions. Loss of air results in loss of Circulating Water seal pressure control and causes Service Water screens to go into continuous blowdown, which results in a slightly lower SW header pressure. The unprepared applicant may consider these

effects more important than those associated with an alarm associated with Service Air and select this answer. Part 2 is plausible and wrong for the same reasons as stated for distractor A. Answer D part 1 is plausible and wrong for the same reasons as stated for distractor C. Part 2 is correct.		
Technical References: Procedure 5.2AIR [Loss of Instrument Air](Rev 21); Alarm Cards A-4/A-5, A-4/G-4 (Rev 44)		
References to be provided to applicants during exam: none		
Learning Objective: INT032-01-36 EO-M,N		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.43(b)(5)	
Level of Difficulty:	4	
SRO Only Justification:		
This question requires knowledge of when to implement abnormal procedure attachments and how to coordinate them with procedure steps.		
PSA applicability:		
N/A		

Examination Outline Cross-Reference	Level	SRO
223001 Primary CTMT and Aux. **2.4.21** Knowledge of the parameters and logic used to assess the status of safety functions, such as reactivity control, core cooling and heat removal, reactor coolant system integrity, containment conditions, radioactivity release control, etc. (CFR: 41.7 / 43.5)	**Tier#** **Group#** **K/A #** **Rating** Revision	2 2 223001 G2.4.21 4.6 0
Revision Statement:		

Question 91

Reference Provided

The plant is at 100% power.

Reactor Water Clean Up is operating.

At 0100 on 4/1, RWCU-MO-18, OUTBD ISOL VLV loses power due to a blown fuse.

At 0200 on 4/1, 480 VAC bus 1F is declared inoperable due to a parts qualification issue.

If conditions do not change…

What is **latest** time allowed per TS for entry into Mode 3?

A. 1400

B. 1500

C. 1700

D. 2200

Answer: B. 1500
Explanation: This question involves a Safety Function Determination regarding the containment safety function. RWCU valves MO-18 and MO-15 are series containment isolation valves in the common supply piping to RWCU. Both valves are required operable per TS 3.6.1.3. MO-15 is the inboard isolation valve, and MO-18 is the outboard isolation

valve. Both valves are open when RWCU is operating and are required to automatically close upon a Group 3 signal. Loss of power to MO-18 would prevent it from closing if containment isolation was required. MO-15 is powered from MCC-R, which is supplied by MCC-K via 480VAC bus 1F. Bus 1F is required operable per TS 3.8.7 and is a support system for MO-15. Procedure 2.0.11.1 [Safety Function Determination Program], which implements TS 5.5.11, requires a Safety Function Determination be performed when a support system is inoperable and a supported system is inoperable. For 480 VAC bus 1F inoperable, TS 3.0.6 allows not entering all the supported system required actions, since TS 3.8.7 provides specific actions for a required electrical distribution system inoperable. However, with the redundant isolation valve, MO-18 inoperable, a loss of safety function exists, as reflected by procedure 2.0.11.1 Att. 2. In this case, the supported system/component is inoperable and its LCO actions are required to be entered IAW TS 3.0.6. With MO-18 inoperable, inoperability of support system 480 VAC bus 1F, even though bus 1F remains energized, results in inoperability of MO-15 and loss of the valves' containment isolation safety function to isolate the associated penetration. With both MO-18 and MO-15 inoperable, TS 3.6.1.3 Required Action B.1 requires isolating the penetration within 1 hour. When that is not met, Action E.1 requires being in Mode 3 within 12 hours.
Distracters: Answer A is plausible because it reflects the 12 hour time of TS 3.6.1.3 Condition E for entry into Mode 3 beginning at the inoperability of the support system, at 0200. It would not be unusual for the unprepared applicant to confuse application of LCO 3.0.6 and time entry of the Condition associated with loss of safety function to the time of the first inoperability. It is wrong because it does not include the 1 hour to isolate the penetration as allowed for Condition B, and because the stem asks for the latest time allowed by TS. Answer C is plausible for the unprepared applicant who does not recognize a loss of safety function is created by the inoperability of 480 VAC bus 1F. This could occur due to the applicant not knowing the power supply to MO-15 or the MCC-R link to 480 VAC bus 1F or due to applicant not understanding the safety function concept. This answer would be chosen by the applicant who applies TS 3.6.1.3 Action B.1 at 0100, which requires isolating the penetration within 4 hours, by 0500, and then applies Action E.1 at 0500 to be in Mode 3 within 12 hours, by 1700. It is wrong because Action B.1 is required at 0200 due to loss of safety function, then Action E.1 is entered at 0300; therefore, Mode 3 must be entered 12 hours after 0300, which is 1500. Answer D is plausible because it reflects the time Mode 3 would be required to be entered if only inoperability of 480 VAC bus 1F was considered. This answer would result if the applicant applies the 8 hour time limit of TS 3.8.7 Action A.1 plus the 12 hour time limit of TS 3.8.7 Action C.1, for a total time of 20 hours, beginning at the time of inoperability of bus 1F, 0200, for a time of 2200 for Mode 3. It is wrong because Mode 3 must be entered earlier, by 1500.

Technical References: TS 3.0.6, TS 3.6.1.3 [Primary Containment Isolation Valves], TS 3.8.7 [Distribution Systems – Operating], Procedure 2.0.11.1 [Safety Function Determination Program](Rev 8)		
References to be provided to applicants during exam: LCO and actions for TS 3.6.1.3 and TS 3.8.7		
Learning Objective: INT007-05-01 EO-2		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.43(b)(2)	
Level of Difficulty:	3	
SRO Only Justification:		
This question requires determination of safety function per TS 3.0.6, TS 5.5.6, and procedure		
PSA applicability:		
Top 10 Risk Significant Systems – Primary Containment Isolation		

Examination Outline Cross-Reference	Level	SRO
239001 Main and Reheat Steam Ability to (a) predict the impacts of the following on the MAIN AND REHEAT STEAM SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5 / 45.6) **A2.10 Closure of one or more MSIV's at power**	**Tier#**	2
	Group#	2
	K/A #	239001 A2.10
	Rating	3.9
	Revision	1
Revision Statement: Per Chief Examiner review comments – simplified first part of stem and changed stem to also directly ask Part 1, what is the impact of closure of two MSIVs. As a result, added effects to each distractor. Modified second part of distractor C to be more plausible with addition of first part. In stem, changed rated power to 100% power for consistency. Reworded stem for readability per Ops Mgmt review.		

Question 92

With the plant at 100% power, MSIVs 80B AND 80C simultaneously FAST close.

What is the effect of this condition AND which of the following mitigation strategies is required to be the initial priority of the CRS?

A. Power rises. Enter Procedure 2.4MSIV [Inadvertent MSIV Closure] and perform a rapid power reduction to below 70%.

B. Reactor scram ONLY. Enter Procedure 2.1.5 [Reactor Scram], ONLY, and direct level control using Feedwater and pressure control using Bypass Valves.

C. Reactor scram AND Group 1 isolation. Enter EOP-1A AND Procedure 2.1.5 [Reactor Scram], and direct level control using Feedwater and cooldown using Bypass Valves.

D. Reactor scram AND Group 1 isolation. Enter EOP-1A AND Procedure 2.1.5 [Reactor Scram], and transition level and pressure control to HPCI and RCIC IAW EOP 5.8.1 [RPV Pressure Control Systems (Table 1)].

Answer: D. Reactor scram AND Group 1 isolation. Enter EOP-1A AND Procedure 2.1.5 [Reactor Scram], and transition level and pressure control to HPCI and RCIC IAW EOP 5.8.1 [RPV Pressure Control Systems (Table 1)].
Explanation: When an MSIV closes, steam flow in the open steam lines increases to accommodate the steam flow lost from the line that closed. The Group 1 isolation setpoint for MSL

Flow High is 142.7% of rated steam flow. When two MSIVs close and all flow from those steam lines must redirect through the two lines that remain open, the steam flow in each of the open steam lines approaches 200% of their normal rated flow. Therefore, a Group 1 isolation occurs, resulting in closure of all eight MSIVs and loss of the main condenser as a heat sink, shortly followed by loss of Feedwater due to loss of steam to the RFPTs (loss of steam to RFPTs is not immediate due to remaining stored energy in main steam system). A reactor scram occurs almost immediately due to Reactor Pressure High or APRM Flux High. The effect of level shrink from void collapse, compounded by the effect of high pressure from MSIV closure, causes level to fall below +3 inches, requiring entry into EOP-1A. With MSIVs closed, SRVs cycle automatically in Low-Low Set mode to control reactor pressure. Since SRVs are cycling and Feedwater becomes unavailable, the applicant should transition to HPCI/RCIC for level and pressure control. HPCI/RCIC are preferred for pressure control over SRVs, since that results in less energy addition to the primary containment.

Distracters:

Answer A is plausible because one MSIV closure would cause power to rise slightly due to the resulting pressure transient, but it would not cause a scram. Single MSIV closure is an entry condition for procedure 2.4MSIV, and rapid power reduction is directed by 2.4MSIV for closure of one MSIV. It is also plausible because at low power, two MSIVs closing might not cause a reactor scram. It is wrong because at 100% power, closure of 2 MSIVs would result in a reactor scram and closure of all MSIVs on high steam flow Group 1 isolation signal. EOP-1A entry would be required due to high RPV pressure or high APRM flux, and low water level. EOP-1A takes precedence over 2.4MSIV and contains the actions to mitigate these conditions, not 2.4MSIV. 2.4MSIV states it is only intended to be used for events where one or more MSIVs close, but no scram results.

Answer B is plausible for the unprepared applicant who anticipates a reactor scram would occur due to more than one MSIV closure causing high flux but fails to recognize this would also result in closure of all MSIVs on high steam flow and EOP-1A entry on high RPV pressure and/or low water level due to shrink from the scram and MSIV closure. If only a scram occurs and low level, Level 3 does not, only procedure 2.1.5 entry is required. It is wrong because EOP-1A entry is required, and closure of all MSIV would result in loss of the condenser as a heat sink and RFP supply steam, necessitating the use of RCIC/HPCI for level and pressure control.

The first portion of Answer C is correct. The second portion of the answer is wrong because closure of all MSIV would result in loss of the condenser as a heat sink and eventual loss of RFPT supply steam, necessitating the use of RCIC/HPCI for level and pressure control. Eventually, MSIVs could be reopened in order to use Bypass valves IAW EOP 5.8.1, but the initial priority would be to use HPCI/RCIC to stabilize pressure. An aggressive cooldown would be inappropriate, since no leak exists.

Technical References: EOP-1A [RPV Control](Rev 20), Procedure 2.4MSIV [Inadvertent MSIV Closure](Rev 9), procedure 2.1.5 [Reactor Scram](Rev 73), Operations Instruction #8 [Guidelines for Successful Transient Mitigation](Rev 16)		
References to be provided to applicants during exam: none		
Learning Objective: INT032-01-25 EO-I,J		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.43(b)(5)	
Level of Difficulty:	3	
SRO Only Justification:		
This question requires diagnosing plant conditions and selection of the appropriate mitigating procedures with which to proceed.		
PSA applicability:		
N/A		

Examination Outline Cross-Reference	Level	SRO
259001 Reactor Feedwater Ability to (a) predict the impacts of the following on the REACTOR FEEDWATER SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5 / 45.6) **A2.07 Reactor water level control system malfunctions**	**Tier#**	2
	Group#	2
	K/A #	259001 A2.07
	Rating	3.8
	Revision	0
Revision Statement:		

Question 93

The plant is at 100% power.

At 1200 on 9/1, Feedwater Control level transmitter NBI-LT-94A fails to 15 inches.

At 1500 on 9/1, NBI-LT-94C fails to 55 inches.

Consider the following from TS 3.3.2.2 [Feedwater and Main Turbine High Water Level Trip Instrumentation]:

NOTE
Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One feedwater and main turbine high water level trip channel inoperable.	A.1 Place channel in trip.	7 days
B. Two or more feedwater and main turbine high water level trip channels inoperable.	B.1 Restore feedwater and main turbine high water level trip capability.	2 hours

(1) Which of the following describes the impact of these conditions on the Reactor Feedwater System **high level trip capability** relative to Action B.1, above?

AND

(2) According to Procedure 2.0.11 [Entering and Exiting Technical Specification/TRM/ODAM LCO Condition(s)], how is Condition A for the inoperable channels tracked?

A. (1) Feedwater high level trip capability is maintained.
   (2) together, with a mutual start time of 1200

B. (1) Feedwater high level trip capability is maintained.
   (2) separately, with respective start times of 1200 and 1500

C. (1) Feedwater high level trip capability is NOT maintained.
   (2) together, with a mutual start time of 1200

D. (1) Feedwater high level trip capability is NOT maintained.
   (2) separately, with respective start times of 1200 and 1500

Answer: B. (1) Feedwater high level trip capability is maintained.
(2) separately, with respective start times of 1200 and 1500

Explanation:
With Channel A of the Feedwater System high level trip failed at 15 inches, it would not produce a trip signal if an actual Level 8 were to occur. Entry into LCO 3.3.2.2 Condition A is required for Channel A. When channel C fails to 55 inches, it would result in a trip signal sealed in, since the trip setpoint is 54 inches (TS setting per SR 3.3.2.2.2, the actual setting is 52.5 inches). Entry into TS 3.3.2.2 Condition A is also required for Channel C. Entry into Condition B is also required, since 2 of 3 channels are inoperable. However, with Channel C tripped, trip capability is maintained. The remaining operable channel, B, would trip, resulting in a RFP/Main Turbine trip if an actual Level 8 were to occur, since the logic is 2 out of 3. Therefore, Action B.1, which requires restoring trip capability within 2 hours when 2 channels are inoperable is already satisfied, because trip capability is maintained. Part 2 of the question tests the applicant's understanding of basic TS application rules and procedure 2.0.11 requirements. TS 3.3.2.2 action statements are preceded by a note which allows separate condition entry for each channel. This means, when more than one channel is inoperable, the completion time for Condition A is allowed to be tracked separately for each inoperable channel, since the condition is for "a" channel inoperable. This contrasts with TS which do not have this allowance and would require all inoperable components to be restored operable based on the completion time clock for the first inoperable component. Using this allowance for separate condition entry provides greater operational flexibility. Procedure 2.0.11 step 2.5 states that in order to make use of this allowance, the inoperable channels should be tracked separately. This means each channel with its specific LCO start time in eSOMS to track TS 3.3.2.2 Condition A requirements.

Distracters:
Answer A Part 1 is correct. Part 2 is plausible because separate tracking is not allowed/used for all LCOs where multiple inoperabilities exist, for example, when 2 LP ECCS subsystems are inoperable. It is also plausible to the unprepared applicant who does not understand application of separate condition entries. It is wrong because Procedure 2.0.11 step 2.5 specifically states the inoperable channels should

be tracked separately. That is to provide operational flexibility by having the full span of completion time for each inoperable channel.

Answer C Part 1 is plausible because TS 3.3.2.2 Condition B is applicable with 2 channels inoperable, and, to the unprepared applicant, Action B.1 might imply trip capability must have been lost. Condition B must be entered and Action B.1 met within 2 hours. For the case given, Action B.1 is already satisfied when Condition B is entered, which might confuse the unprepared applicant. It is wrong because the subject logic is 2 out of 3, and since channel C failed to above the trip setpoint, an actual level transient above Level 8 would result in trip of Channel B, which would cause the RFP trip function. TS 3.3.2.2 bases for Condition B states trip capability is maintained if 2 channels are either operable or in trip. Part 2 is plausible and wrong for the same reasons as stated for distractor A.

Answer D Part 1 is plausible, but wrong for the same reasons as stated for distractor C. Part 2 is correct.

Technical References: TS 3.3.2.2 [Feedwater and Main Turbine High Water Level Trip] and bases, procedure 2.0.11 [Entering and Exiting Technical Specification/TRM/ODAM LCO Condition(s)](Rev 41)

References to be provided to applicants during exam: none

Learning Objective: INT007-05-04 EO-2,3

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.43(b)(2)	
Level of Difficulty:	3	

SRO Only Justification:

This question requires application of TS bases information and knowledge of administrative requirements for tracking TS required actions.

PSA applicability:

N/A

Examination Outline Cross-Reference	Level	SRO
1. Conduct of Operations 2.1.4 Knowledge of individual licensed operator responsibilities related to shift staffing, such as medical requirements, "no-solo" operation, maintenance of active license status, 10CFR55, etc. (CFR: 43.2)	Tier#	3
	Group#	
	K/A #	G2.1.4
	Rating	3.8
	Revision	0
Revision Statement:		

Question 94

A CRS, who is also qualified as Shift Manager and as STE, began reactivation as SRO on May 1st, after an extended absence.

The SRO signed into eSOMS and completed Procedure 2.0.7 [Licensed Operator Active/Reactivation/Medical Status Maintenance Program], Att. 4 [SRO Reactivation] per the following:

May 1	May 2	May 3	May 4	May 5	May 6	May 7
12 hours under instruction (CRS initialed as coach)	12 hours under instruction (CRS initialed as coach)	12 hours under instruction (Shift Mgr initialed as coach)			12 hours as on-shift STE	12 hours as on-shift STE
May 8	May 9	May 10	May 11	May 12	May 13	May 14
		12 hours under instruction (Shift Mgr initialed as coach)			12 hours under instruction (CRS initialed as coach)	12 hours under instruction (CRS initialed as coach)

When did the SRO complete the **minimum** proficiency time for reactivation IAW Procedure 2.0.7 [Licensed Operator Active/Reactivation/Medical Status Maintenance Program]?

A. May 6

B. May 7

C. May 10

D. May 14

Answer: C. May 10		
Explanation: IAW Procedure 2.0.7 [Licensed Operator Active/Reactivation/Medical Status Maintenance Program], step 2.8.5 specifies a minimum of four 12-hours shifts to meet the NRC required 40 hours for reactivation of an SRO license. Att. 4, SRO Reactivation, specifies 40 hours of under instruction proficiency time is required for reactivation, and SRO Licensees proficiency time is performing on-shift license authorized (CRS and/or SM) position tasks under active SRO (Coach) direction, so under instruction of either CRS or SM counts toward the required proficiency time. Time acting as STE, not under instruction for proficiency, does not count. Therefore, the four 12-hour shift (40 hour NRC) requirement would be completed after the fourth 12 hour day under instruction of either the CRS or SM, which is on May 10.		
Distracters: Answer A is plausible because it reflects the fourth day of work after beginning reactivation, so 40 work hours would have elapsed. It is wrong because only 36 hours of under instruction time for proficiency would have been reached, which is less than the required 40 hours. Answer B is plausible because it reflects the fifth day of work and 60 total hours, the normal period required in a quarter for maintaining proficiency. The unprepared applicant may confuse the proficient requirement of 40 hours with a familiar limit of 60 hours and select this answer. It is wrong because only 36 hours of under instruction time for proficiency would have been reached, which is less than the required 40 hours. Answer D is plausible because it reflects the day 40 hours of U/I time under the CRS is reached. The unprepared applicant may believe, since he/she is primarily a CRS, only U/I under the CRS counts toward his/her proficiency. It is wrong because procedure 2.0.7 Att 4 states U/I time to regain proficiency may be under CRS or SM; therefore, the minimum requirement is met earlier than this answer reflects.		
Technical References: Procedure 2.0.7 [Licensed Operator Active/Reactivation/Medical Status Maintenance Program](Rev 10)		
References to be provided to applicants during exam: none		
Learning Objective: INT032-01-01 EO-S1e		
Question Source:	Bank #	

(note changes; attach parent)	Modified Bank #	GGNS 2015 NRC#94
	New	
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.43(b)(1)	
Level of Difficulty:	3	
SRO Only Justification:		
This question tests knowledge of requirements for maintenance of active license status related to shift staffing requirements of the Operating License that is specific to SROs.		
PSA applicability:		
N/A		

GGNS 2015 NRC SRO Q#94

Question 94

An on-shift SRO, qualified as an STA also, has been on short term disability.

Total time away from work was 5 months (April thru August)

He has completed his STA proficiency.

He is scheduled to work the following hours:

Sunday 1	Monday 2	Tuesday 3	Wednesday 4	Thursday 5	Friday 6	Saturday 7
12 hrs parallel as CRS	12 hrs parallel as CRS	12 hrs as STA/FSS			12 hrs as STA/FSS	12 hrs as STA/FSS
Sunday 8	Monday 9	Tuesday 10	Wednesday 11	Thursday 12	Friday 13	Saturday 14
12 hrs as STA/FSS				12 hrs parallel as CRS	12 hrs parallel as CRS	12 hrs parallel as CRS

Which of the following describes when the SRO has completed the watch standing proficiency for a CRS?

A. Completion on Sunday the 1st

B. During the shift on Friday the 6th

C. During the shift on Friday the 13th

D. Completion on Saturday the 14th

Answer: C

Examination Outline Cross-Reference	Level	SRO
1. Conduct of Operations 2.1.36 Knowledge of procedures and limitations involved in core alterations. (CFR: 43.6)	Tier#	3
	Group#	
	K/A #	G2.1.36
	Rating	4.1
	Revision	0
Revision Statement:		

Question 95

Which activity REQUIRES Refuel Floor Supervisor permission during Core Alterations?

A. Shifting shutdown cooling trains

B. Suspending fuel handling operations

C. Allowing access to the fuel handling area on the refuel floor

D. Using greater than 10 gallons of demineralized water on the refuel floor

Answer: C. Allowing access to the fuel handling area on the refuel floor

Explanation:
Movement of fuel within the RPV is a Core Alteration. Refuel SRO responsibilities are, by nature, generic. Procedure 2.2.31 [Fuel Handling – Refueling Platform] step 2.10 states access to fuel handling area on refueling floor and to overhead bridge crane when fuel handling is in-progress shall be limited to authorized personnel, and that individual authorization is determined by the Refuel Floor Supervisor. Step 3.3.1 states the Refuel Floor Supervisor must be a SRO when Core Alterations are in progress.

Distracters:
Answer A is incorrect because Refuel Floor Supervisor permission is not required to allow shutdown cooling operations. This answer is plausible because shifting shutdown cooling trains can affect water clarity, which could cause the Refuel Floor Supervisor to delay fuel handling.

Answer B is plausible because Refuel Floor Supervisor permission is required to re-commence fuel handling operations IAW Attachment 4 (Reset Checklist) which shall be used each time the normal fuel handling process is stopped/interrupted. It is

wrong because fuel handling may be immediately suspended due to a variety of reasons, such as equipment failure, without Refuel Floor Supv permission. Answer D is incorrect because Refuel Floor Supervisor permission is not required to use greater than 10 gallons of demineralized water on the refuel floor. This choice is plausible due to the Refuel floor SRO is required to brief available refueling floor personnel on limiting demineralized water usage and requirement to notify Control Room if using > 50 gallons demineralized water each shift. The applicant who confuses briefing vs. giving permission would choose this answer.		
Technical References: Procedure 2.2.31 [Fuel Handling – Refueling Platform](Rev 51), TS 1.1 [Definitions]		
References to be provided to applicants during exam: none		
Learning Objective: INT0231002001160A		
Question Source:	Bank #	Audit #1 Q#95 for 12/2015 NRC
(note changes; attach parent)	Modified Bank #	
	New	
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.43(b)(7)	
Level of Difficulty:	3	
SRO Only Justification:		
This question requires knowledge of Refuel Floor SRO responsibilities during refueling activities.		
PSA applicability:		
N/A		

Examination Outline Cross-Reference	**Level**	SRO
2. Equipment Control 2.2.20 Knowledge of the process for managing troubleshooting activities. (CFR: 43.5)	**Tier#**	3
	Group#	
	K/A #	G2.2.20
	Rating	3.8
	Revision	1
Revision Statement: Per Chief Examiner review comments – simplified stem and corrected punctuation		

Question 96

An I&C troubleshooting activity involving installation of a temporary jumper is planned.

The degree of rigor assigned to the troubleshooting plan is "Category A."

IAW Procedure 7.0.1.7 [Troubleshooting Plant Equipment] for this level of troubleshooting activity…

Who, by title, is required to sign the **Complex Troubleshooting Plan Form** as the "Approval Authority" before the Shift Manager gives his authorization to execute the plan?

A. I&C Supervisor

B. Maintenance Manager

C. Work Control Center Administrator

D. General Manager of Plant Operations

Answer: D. General Manager of Plant Operations
Explanation: Procedure 7.0.1.7 step 2.8.1.2 states the GMPO is the approval authority for troubleshooting activities requiring Category A controls. The applicant should recognize Category A controls provide the highest degree of rigor and must remember the GMPO is the person with which the SM must concur to authorize these troubleshooting plans. As the final authorization for execution of the Complex Troubleshooting Plan, the SM/SRO must be able to verify the proper authority has approved the plan, based on the degree of rigor required, since the Approval Authority varies for different degrees of rigor.

Distracters:
All distractors are plausible because they reflect approval authorities for troubleshooting activities where other degrees of rigor may be applied. They are all wrong because Category A is the highest level of rigor, and the activity can only be approved by the GMPO with SM concurrence.

Answer A reflects the approval authority for Category C, the third highest of rigor for a formal troubleshooting plan.

Answer B reflects the approval authority for Category B, the second highest degree of rigor for a formal troubleshooting plan.

Answer C reflects the approval authority for Category D, the lowest degree of rigor for a formal troubleshooting plan.

Technical References: Procedure 7.0.1.7 [Troubleshooting Plant Equipment](Rev 15)

References to be provided to applicants during exam: none

Learning Objective: SKL011-01-1 EO-37b, SKL011-01-02 performance objective task 200300W0203

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.43(b)(3)	
Level of Difficulty:	3	

SRO Only Justification:
This question requires knowledge of a specific SM(SRO) responsibility pertaining to authorization of troubleshooting on plant equipment requiring plant modification.

PSA applicability:
N/A

Examination Outline Cross-Reference	Level	SRO
2. Equipment Control 2.2.38 Knowledge of conditions and limitations in the facility license. (CFR: 43.1)	Tier#	3
	Group#	
	K/A #	G2.2.38
	Rating	4.5
	Revision	0
Revision Statement:		

Question 97

The plant is in Mode 1.

The number of non-licensed operators on site is one less than required by TS Section 5, Administrative Controls due to an unexpected illness.

IAW TS 5.2.2c, …

(1) What is the maximum time the position is allowed to remain unfilled?

(2) When does TS specify action must be taken to fill the vacancy of the required non-licensed operator?

A. (1) One hour
(2) Immediately

B. (1) One hour
(1) Within 1 hour

C. (1) Two hours
(2) Immediately

D. (1) Two hours
(1) Within 1 hour

Answer: C. (1) Two hours (2) Immediately
Explanation: TS 5.5.2.c states Shift crew composition may be less than the minimum requirement of 10CFR 50.54(m)(2)(i) and Specification 5.2.2.a for a period of time not to exceed **2 hours** in order to accommodate unexpected absence of on-duty shift crew members provided **immediate** action is taken to restore the shift crew composition to within the minimum requirements.
Distracters:

Answer A Part 1 is plausible because other important actions, such as TS actions and event reporting requirements have 1 hour time limits. It is wrong because the stem asks for the maximum time allowed, which is 2 hours. Part 2 is correct. Answer B Part 1 is plausible and wrong for the same reasons as given for distractor A. Part 2 is plausible, because like the correct answer, 1 hour is a very restrictive time limit and because some conditions required to be immediately reported to the NRC, such as EAL declaration, actually have a concrete limit of 1 hour. It is wrong because TS 5.2.2.c *specifies* action must be taken *immediately* to fill the vacancy. Answer D Part 1 is correct. Part 2 is plausible and wrong for the same reasons as given for distractor B.		
Technical References: TS Admin Controls section 5.2.2.c		
References to be provided to applicants during exam: none		
Learning Objective: INT007-05-13 EO-2		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.43(b)(1)	
Level of Difficulty:	3	
SRO Only Justification:		
This question requires knowledge of TS Administrative Controls (TS Section 5) requirements for filling on-shift staffing vacancies.		
PSA applicability:		
N/A		

Examination Outline Cross-Reference	Level	SRO
3. Radiation Control 2.3.14 Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities. (CFR: 43.4)	Tier#	3
	Group#	
	K/A #	G2.3.14
	Rating	3.8
	Revision	0
Revision Statement:		

Question 98

Which one of the following completes the statement below regarding mitigating the offsite radiation/contamination hazard during emergency conditions?

The **standard** Protective Action Recommendation (PAR) for a General Emergency declared due to loss of fission product boundaries during a slowly progressing event requires evacuating **downwind sectors** out to a minimum of__________ miles, if conditions permit.

A. 2

B. 5

C. 10

D. 15

Answer: B. 5

Explanation:
IAW Procedure 5.7.20 Att. 1 [Protective Action Recommendation Flowchart], the standard PAR for declaration of a General Emergency that is not a "rapidly progressing severe accident" is to evacuate a 2 mile radius and downwind (3 or 4) sectors for 5 miles.

Distracters:
Answer A is plausible because all sectors in a 2 mile radius is required to be evacuated for the standard PAR. It is wrong because the stem asks for the minimum distance for evacuation of downwind sectors, which is 5 miles.

Answer C is plausible because, per procedure 5.7.20 Att. 1 step 2.1.2, downwind sectors would be evacuated out to 10 miles, if dose at 5 miles exceeds 1 Rem TEDE. It is wrong because the stem asks only regarding the standard PAR associated with declaration of a GE and does not provide any dose data. This answer is greater than the requested "minimum" and is, therefore, wrong.

Answer D is plausible because, per procedure 5.7.20 Att. 1 step 2.1.3, downwind sectors would be evacuated out to 15 miles, if dose at 10 miles exceeds 1 Rem TEDE. It is wrong because the stem asks only regarding the standard PAR associated with declaration of a GE and does not provide any dose data. This answer is greater than the requested "minimum" and is, therefore, wrong.

Technical References: procedure 5.7.20 [Protective Action Recommendations](Rev 28)

References to be provided to applicants during exam: none

Learning Objective: GEN0030401B0B030B

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.43(b)(5)	
Level of Difficulty:	2	
SRO Only Justification:		

This question requires knowledge of EPIP procedures and Protective Action Recommendations for potential radioactivity releases.

PSA applicability:

N/A

Examination Outline Cross-Reference	**Level**	SRO
4. Emergency Procedures / Plan 2.4.16 Knowledge of EOP implementation hierarchy and coordination with other support procedures or guidelines such as, operating procedures, abnormal operating procedures, and severe accident management guidelines. (CFR: 43.5)	**Tier#**	3
	Group#	
	K/A #	G2.4.16
	Rating	4.4
	Revision	1
Revision Statement: Per Chief Examiner review comments – simplified stem		

Question 99

EOPs and EPs are in use during a LOCA.

The CRS determines core damage is occurring due to loss of core cooling.

Which of the following completes the statement below regarding EOP implementation hierarchy and coordination with other procedures for this condition?

The CRS is required to transition out of all __(1)____ directly to __(2)____.

A. (1) EOP flowcharts, only
(2) Severe Accident Guideline (SAG) 1

B. (1) EOP flowcharts, only
(2) Severe Accident Guideline (SAG) 2

C. (1) EOPs AND EPs
(2) Severe Accident Guideline (SAG) 1

D. (1) EOPs AND EPs
(2) Severe Accident Guideline (SAG) 2

Answer: A. (1) EOP flowcharts, only
(2) Severe Accident Guideline (SAG) 1

Explanation:
If it is determined that core damage is occurring, it must be assumed that available injection is insufficient to cool the core. Severe accident management strategies must then be initiated in accordance with SAG 1. When it is determined that core damage is occurring due to a loss of adequate core cooling, the core is submerged by flooding the primary containment. If primary containment flooding is required, all PSTG parameter control paths, delineated by EOP flowcharts, transfer to the SATGs. The EOP Flowcharts are exited, and SAG 1, RPV and Primary Containment Flooding is entered. SAG 1 then provides necessary instructions for entry into SAG 2.

Distracters:
Answer B Part 1 is correct. Part 2 is plausible because adequate injection has been lost, and SAG 2 contains instructions for use of injection systems for RPV and Containment flooding. It is wrong because SAG 1 must be entered directly from EOPs, because it contains the logic for entry into the appropriate flowchart of SAG 2, based on conditions such as whether the core has breached the RPV and whether the containment is in jeopardy.

Answer C Part 1 is plausible because if PC Flooding is required, EOP steps RC/F-1 and FS/F-1 state exit Exit EOPs. The unprepared applicant may believe this means to exit Emergency Procedures and all EOP support procedures as well. It is wrong because PSTGs specifically state EOP parameter control paths, synonymous to EOP flowcharts, must be exited, and because use of certain Emergency Procedures and EOP support procedures is prescribed within the SAGs. Part 2 is correct.

Answer D Part 1 is plausible and wrong for the same reasons as stated for distractor C. Part 2 is plausible and wrong for the same reasons as stated for distractor B.

Technical References: EOP-1A [RPV Control](Rev 20), EOP-2B [RPV Flooding](Rev 19), EOP-7B [RPV Flooding, Failure-to-Scram](Rev 20), AMP-TBD00 [EOP/PSTG Technical Basis]

References to be provided to applicants during exam: none

Learning Objective: INT008-06-11 Obj. LO-10**,** INT008-06-12 Obj. LO-09

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.43(b)(5)	
Level of Difficulty:	2	

SRO Only Justification:
This question requires knowledge of implementation hierarchy and coordination between EOPs and SAGs.

PSA applicability:
N/A

Examination Outline Cross-Reference	Level	SRO
4. Emergency Procedures / Plan 2.4.38 Ability to take actions called for in the facility emergency plan, including supporting or acting as emergency coordinator if required. (CFR: 41.10 / 43.5 / 45.11)	Tier#	3
	Group#	
	K/A #	G2.4.38
	Rating	4.4
	Revision	0
Revision Statement:		

Question 100

According to Procedure 5.7.2 [Emergency Director EPIP], which one of the following Emergency Director responsibilities may **NOT** be delegated?

A. Completing the Emergency Director turnover checklist.

B. Initial activation of the CNS Automated Notification System (ANS) for a declared emergency.

C. Recommending protective actions to authorities responsible for off-site emergency measures.

D. Ensuring adequate technical and logistical support is available to the station emergency organization.

Answer: C. Recommending protective actions to authorities responsible for off-site emergency measures.

Explanation:
Procedure 5.7.2 explicitly states recommending protective actions to authorities responsible for off-site emergency measures may not be delegated. Other responsibilities that may not be delegated are:

- Event Declaration.
- The decision to notify authorities responsible for off-site emergency measures.
- Authorize emergency workers to receive dose in excess of 10CFR20 occupational limits.
- Authorize KI for emergency workers.

Distracters:

Answer A is plausible because it is part of the Emergency Director flowchart. It is wrong because delegation of filling out data on the ED turnover checklist is not explicitly forbidden and may be accomplished by other ERO personnel.

Answer B is plausible because it is part of the Emergency Director flowchart. It is wrong because delegation of activating CNS ANS is not explicitly forbidden and may be accomplished by other ERO personnel.

Answer D is plausible because it is an ED responsibility listed in procedure 0-EP-01 [Emergency Response Organization Responsibilities], after ED responsibilities are transferred to the EOF. It is wrong because delegation of aligning adequate technical and logistical support is not explicitly forbidden and may be accomplished by other ERO personnel, including the Logistics Coordinators.

Technical References: Procedures 5.7.2 [Emergency Director EPIP](Rev 34), Procedure 0-EP-01 [Emergency Response Organization Responsibilities](Rev 29)

References to be provided to applicants during exam: none

Learning Objective: GEN003-04-01 EO-EP04-1

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.43(b)(5)	
Level of Difficulty:	2	
SRO Only Justification:		

This question requires knowledge of EPIP procedures and Emergency Director responsibilities during emergency conditions.

PSA applicability:

N/A