GGNS LOT 12/2017 **NRC** INITIAL LICENSED OPERATOR WRITTEN EXAMINATION

RO EXAM

ANSWER KEY

1	C		26	A		51	D
2	A		27	C		52	A
3	D		28	B		53	D
4	D		29	D		54	D
5	B		30	A		55	D
6	B		31	B		56	C
7	D		32	D		57	D
8	A		33	A		58	C
9	A		34	A		59	B
10	C		35	B		60	B
11	B		36	A		61	A
12	B		37	C		62	D
13	C		38	D		63	B
14	B		39	C		64	D
15	D		40	C		65	A
16	C		41	A		66	B
17	C		42	A		67	A
18	D		43	C		68	B
19	D		44	D		69	A
20	B		45	C		70	D
21	A		46	B		71	B
22	C		47	C		72	B
23	B		48	B		73	C
24	A		49	D		74	D
25	C		50	A		75	A

<table>
<tr><td>Examination Outline Cross Reference</td><td>Level</td><td>RO</td></tr>
<tr><td rowspan="6">295001 Partial or Complete Loss of Forced Core Flow Circulation

AK1 Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION :

AK1.02 Power/flow distribution</td><td>Tier</td><td>1</td></tr>
<tr><td>Group #</td><td>1</td></tr>
<tr><td>K/A</td><td>295001</td></tr>
<tr><td>Rating</td><td>3.3</td></tr>
<tr><td>Revision</td><td>3</td></tr>
<tr></tr>
<tr><td colspan="3">Revision Statement:

Rev 1: Added revision numbers to handout material, added space between bullets

Rev 2: Editorial changes to stem and answers

Rev 3: Minor editorial changes</td></tr>
</table>

Question: 1

HANDOUT PROVIDED

Plant is operating at rated thermal power.

All OPRMs are INOP.

Then:

- Both Reactor Recirc pumps unexpectedly downshift to slow speed
- Core flow stabilizes at 40 mlbm/hr
- Reactor power stabilizes at 52 percent of rated thermal power

Which of the following describes the required immediate operator action?

A. Begin inserting control rods.

B. Immediately shift Recirc pumps to fast speed.

C. Place the Reactor Mode Switch in SHUTDOWN.

D. Open Reactor Recirc pump flow control valves to maximum position.

Answer: **C**

Explanation:

Using Figure 1 from 05-01-02-III-3, Power-flow map, the applicant must plot the current position on the figure. Recognizing that the core is operating in the scram region of the power-flow curve with OPRMs inoperable, step 4.1 of 05-01-02-III-3, Reduction in Recirculation Flow Rate ONEP, requires an immediate scram.

Distracters:

A is wrong. Would be correct if in the controlled entry region of the power-flow curve per 05-01-02-III-3, Reduction in Recirculation Flow Rate ONEP, step 5.3.1. Plausible if the applicant incorrectly plots on the PF Map.

B is wrong. Caution prior to step 5.3.1 states, "Do NOT restart OR upshift a Reactor Recirc pump to exit the Controlled Entry Region."

D is wrong. Would be correct if the pumps are manually downshifted, per SOI when the Recirc pumps are manually downshifted fully open the flow control valves.

K/A Match

Knowledge of the operational outcome of a lowered core flow by using the Power / Flow map to determine the next operator actions.

Technical References:

05-01-02-III-3, Reduction in Recirculation Flow Rate ONEP, Rev. 115

Handouts to be provided to the Applicants during exam:

Figure 1 from 05-01-02-III-3, Reduction in Recirculation Flow Rate ONEP, Rev. 115, Power-flow map

Learning Objective:

GLP-OPS-ONEP OBJ. 1 & 35
GLP-OPS-B3300 OBJ. 31.3

Question Source:	**Bank # 1042**	2014 NRC Exam Q# 1
(note changes and attach parent)	**Modified Bank #**	
	New	
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(10)	
Level of Difficulty:	3.0	
PRA Applicability:		

Placing the Mode switch to SHUTDOWN is a RPS actuation. RPS is a top 10 risk significant system.

Examination Outline Cross Reference	Level	RO
295003 Partial or Complete Loss of AC AK3 Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER : AK3.01 Manual and auto bus transfer	Tier	1
	Group #	1
	K/A	295003
	Rating	3.3
	Revision	**3**
Revision Statement: **Per NRC: Reworded question stem and part 2 of answers. Added rev numbers to reference material** **Rev 2: Added name to reference material.** **Rev 3: Minor editorial changes**		

Question: 2

A loss of Service Transformer 21 has occurred.

Which buses automatically regain power and why?

A. 16AB and 17AC
Bus loads are required for safe shutdown

B. 11HE and 14AE
Ensure LSS is able to perform its design function

C. 16AB and 17AC
Ensure LSS is able to perform its design function

D. 11HE and 14AE
Bus loads are required for safe shutdown

Answer: A
Explanation: Based on System Operating Instructions for normal electrical lineup on major AC buses, the applicant must be able to determine this from memory. • 04-1-01-R21-16 SOI for 16AB • 04-1-01-R21-17 SOI for 17AC A loss of Service Transformer 21 will cause a loss of power to buses 11HE, 14AE, 16AB and 17AC. The 16 and 17 buses are ESF AC and are automatically restored by diesel generators. Bus 17AC does not have a LSS system. The 11 and 14 buses are BOP AC and must be manually transferred to an alternate

source.

The ESF Power System provides electrical power to systems and components required for reactor shutdown and prevent the release of radioactive material following a design basis accident. The ESF power system and buses must maintain a reliable source of power automatically and without operator action.

Distracters:

All distracters are plausible if the applicant can't recall which 4160 electrical buses are ESF and will regain power and the difference between function of LSS and the reason for maintaining power on ESF electrical buses.

B is wrong. 11 and 14 buses are not automatically restored and the LSS system is used to prevent excessive loading of the diesel generators not a reason for auto restoration.

C is wrong. The LSS system is used to prevent excessive loading of the diesel generators not a reason for auto restoration. Bus 17AC does not have a LSS system.

D is wrong. 11 and 14 buses are not automatically restored

K/A Match

This question ensures the applicant has the knowledge of the reason for automatic restoration of the ESF buses. It requires the applicant to know what are the preferred power sources for major buses within the plant and then the reason that only ESF buses have the ability to be automatically restore power from another source.

Technical References:

04-1-01-R21-16, ESF BUS 16AB SOI, Rev. 33

04-1-01-R21-17, ESF BUS 17AC SOI, Rev. 10

GLP-OPS-R2100, Load Shedding & Sequencing System and ESF AC Power Distribution System, Rev. 19, Lesson Plan

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-R2100 OBJ. 1, 3.2, & 4.4

Question Source:	**Bank #**	
(note changes and 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	
PRA Applicability:		
Failure to align alternate power to AC buses is ranked #6 for PRA Operator Action Importance To CDF.		

Examination Outline Cross Reference	Level	RO
295004 Partial or Total Loss of DC Pwr AK2 Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF D.C. POWER and the following: AK2.03 D.C. bus loads	Tier	1
	Group #	1
	K/A	295004
	Rating	3.3
	Revision	1

Revision Statement:

Added current plant conditions
Re-ordered answers short to long
Corrected Technical references.

Question: 3

The plant is operating at rated thermal power.

Which of the following describes an impact of a loss of bus 11DA?

A. SLC Train ‘A’ squibs cannot be fired.

B. SGTS Train ‘A’ logic cannot automatically initiate.

C. RCIC steam supply line cannot automatically isolate.

D. Backup Scram Valves cannot depressurize the scram air header.

Answer: D

Explanation:

11DA (Div 1) is the ESF DC Division 1 power and it supplies power to the normally de-energized solenoid for one of the 3 Backup Scram Valves (the other two are Div 2 powered from 11DB). Without 11DA, the Div 1 valve cannot reposition. All 3 Backup Scram Valves must reposition in order to depressurize the scram air header via the Backup Scram Valves.

Distracters:

A is wrong. Because the SLC squibs are 120 VAC powered via the 480 VAC MCC breaker powering the associated SLC pump; they have no reliance on DC power, either directly or indirectly. Plausible because SLC Squib valve power available is indicated by a white status light on P601. Other systems with white status lights on P601, such as for ECCS logic status are powered from DC.

B is wrong. Because, although SGTS ‘A’ initiation logic is DC-powered from 11DA, the logic is de-energize-to-function…it initiates upon loss of the 11DA bus.

C is wrong. RCIC steam line has two supply isolations, E51-F064 (Div 1) which uses 11DA for Div 1 isolation logic, and E51-F063 (Div 2) which uses 11DB for Div 2 isolation logic. So long as 11DB is available, the F063 valve will close to effect a successful RCIC steam line isolation. Plausible because this is an actual load from the DC system but due to redundancy the system will isolate.

K/A Match

Knowledge of the DC loads that are supplied by a specific bus.

Technical References:

E-1023, One Line Meter & Relay Diagram 125V DC Buses 11DA, 11DB, & 11DC, Rev. 37

E-1173-014, RPS system Rev. 14

E-1173-021, RPS system, Rev. 19

GLP-OPS-C111A, Control Rod Drive Lesson Plan, Rev. 10

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-L1100, OBJ. 17

Question Source:	**Bank #429**	2010 NRC Exam Q# 11
(note changes and attach parent)	**Modified Bank #**	
	New	
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(7)	
Level of Difficulty:	2.5	

PRA Applicability:

125 VDC ESF is a top 10 risk significant system.

Examination Outline Cross Reference	Level	RO
295005 Main Turbine Generator Trip AA1 Ability to operate and/or monitor the following as they apply to MAIN TURBINE GENERATOR TRIP : AA1.05 Reactor/turbine pressure regulating system	Tier	1
	Group #	1
	K/A	295005
	Rating	3.6
	Revision	2

Revision Statement:

Rev 1: New Question

Rev 2: Per discussion with chief on 11/16

- **moved “to control reactor pressure” to stem**
- **Deleted “bypass” in answer A**
- **Added “throttle” in answer A**
- **Deleted “because” statement in answer B and added “#3 remains closed”**
- **Fixed plausibility statements**

Question: 4

The plant is operating at 20%

Main turbine trip occurs.

How does the pressure regulating system operate the Turbine Bypass Valves (TBVs) to control reactor pressure?

A. All 3 valves will throttle open together

B. Only the #1 and #2 valves throttle open and #3 remains closed

C. After the #1 valve is about 50% open then all 3 valves throttle open together

D. After the #1 valve is about 2% open then all 3 valves throttle open together

Answer: D

Explanation:

At only 20% power (well below the Turbine Trip Reactor Scram enabled setpoint of 35.4%; see RPS Instrumentation LCO Table 3.3.1.1-1, Functions 9 and 10), the reactor remains operating producing about 20% steam. GGNS total bypass capacity is about 35% steam. When the turbine trips (i.e., Turbine Stop Valves and Turbine Control Valves close), the EHC system’s Initial Pressure Controller (IPC) will begin to use the Bypass Control Units (BCUs) to maintain reactor pressure at its original setpoint. The 3 BCUs (one for each TBV) at GGNS are designed to throttle the 3 TBVs in unison, rather than in sequence, to control reactor pressure…with one exception, TBV #1 is designed to open to about 2% open before #2 and #3 throttle open, but then all 3 uniformly position to control pressure.

Distracters:

A is wrong. TBV #1 is designed to open to about 2% open before #2 and #3 throttle open, but then all 3 uniformly position to control pressure. Plausible due to all three will eventually uniformly all open together but A valve opens first.

B is wrong but plausible. The temperature control system for the 2nd stage reheaters controls heatup in this manner, sequential opening of 4 valves small to large. Plausible due to total bypass steam flow equals 30% power, if each valve controlled 10% each then only 2 would be required at 20%.

C is wrong. This choice suggests that one valve will open to 50% then the others will open to control pressure; this is not the GGNS design (as already described). Plausible because the Seal Steam control system works in this manner, after the first valve reaches 50% then the other will begin to throttle.

K/A Match

This question requires the applicant to process a turbine trip and determine the status of the pressure control system for the reactor by monitoring control valves.

Technical References:

GLP-OPS-N32-02, Main Turbine EHC Control System, Rev. 11, Page 31 of 56

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-N3202, OBJ. 9.1 & 9.2

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank # 97**	NRC 2012 Q# 5
	New	
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(5)	
Level of Difficulty:	3	
PRA Applicability:		

Examination Outline Cross Reference	Level	RO
295006 SCRAM AK3 Knowledge of the reasons for the following responses as they apply to SCRAM : AK3.04 Reactor water level setpoint setdown: Plant-Specific	**Tier**	1
	Group #	1
	K/A	295006
	Rating	3.1
	Revision	2
Revision Statement: **Rev 1: Removed information from each question and put in stem.** **Added plausibility statements** **Rev 2: editorial changes to answers**		

Question: 5

Which of the following is the reason for the Feedwater Control Setpoint Setdown function?

Assists with overcoming the effects of RPV level:

A. swell after a Lo-Lo Set actuation.

B. shrink and swell after a Reactor scram.

C. shrink after a Reactor Feedpump Turbine trip.

D. shrink and swell after a Reactor Recirc Pump speed shift.

Answer: B
Explanation: Per GLP-OPS-C3400; page 35 of 57, Following a Reactor scram, the Setpoint Setdown feature automatically operates to overcome the effects of shrink and swell on RPV water level control.
Distracters: A is wrong. Lo-Lo Set is a SRV actuation that helps control reactor pressure and this is also a parameter that is controlled post scram like level. Setpoint setdown is designed to assist with post scram level control not a lo lo set actuation. Plausible if the applicant confuses lo-lo set with setpoint setdown C is wrong. Setpoint setdown is designed to assist with post scram level control not any feedpump trip. Plausible due to the level swings that occur when a feedpump trips and a Reactor Feedpump trip is

associated with a FCV runback not Setpoint Setdown.

D is wrong. Setpoint setdown is designed to assist with post scram level control not an auto transfer to slow speed on the recirc pumps. Plausible due to the level swing that occurs when shifting Reactor Recirc pumps.

K/A Match

This question requires the applicant to have the knowledge for the reason on the Setpoint Setdown part of the Feedwater Level Control System

Technical References:

GLP-OPS-C3400, Rev 17, Digital Feedwater Control System (DFCS) Lesson Plan

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-C3400, OBJ. 12

Question Source:	**Bank #**	May 2017 NRC Exam Q# 5
(note changes and attach parent)	**Modified Bank #**	
	New	
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.41(b)(7)	
Level of Difficulty:	2	

PRA Applicability:

Examination Outline Cross Reference	Level	RO
295016 Control Room Abandonment Ability to determine and/or interpret the following as they apply to CONTROL ROOM ABANDONMENT : AA2.02 Reactor water level	Tier	1
	Group #	1
	K/A	295016
	Rating	4.2
	Revision	1
Revision Statement: **Changed inch symbol to "inches", Changed wording in distracter description to "indicated level"**		

Question: 6

HANDOUT PROVIDED

Due to a fire, operators have abandoned the control room and have manned the Remote Shutdown Panels.

- Reactor pressure 700 psig
- Reactor water level +50 inches

What is actual reactor water level?

A. +32 inches

B. +40 inches

C. +45 inches

D. +47 inches

Answer: B
Explanation: Applicants will be provided with Attachment II (Wide Range Level vs. Actual Level) found in the Shutdown from the Remote Shutdown Panel ONEP (05-1-02-II-1). Applying Attachment II, with reactor pressure at 700 psig, and Wide Range at +50", actual Level is +40". The Applicant must interpolate between the 600 psig and 800 psig lines. Due to the need for accurate level the applicant must interpolate and not go to the most conservative indication as in other graphs used by licensed operators. Shutdown from the Remote Shutdown Panel ONEP (05-1-02-II-1) states "correlated with Attachment II", nothing is stated to use conservative indications for plots.

Distracters:

A is wrong. This is the Actual Level (with indicated level at +50”) derived from inappropriately interpolating Attachment II for a 500 psig reactor pressure, the applicant must interpolate between two pressure lines. This answer could be selected if the applicant uses the wrong two lines. Plausibility is premised on the Applicant applying attention-to-detail.

C is wrong. This is the Actual Level (with Wide Range Level at +50”) derived from inappropriately interpolating Attachment II for an 800 psig reactor pressure. Plausibility is premised on the Applicant applying attention-to-detail.

D is wrong. This is the level derived from inappropriately using the wrong axis on Attachment II. Plausibility is premised on the Applicant applying attention-to-detail.

K/A Match

This question requires the applicant to have the knowledge and ability to determine actual reactor water level with given indicated reactor water level and reactor pressure.

Technical References:

05-1-02-II-1, Shutdown from the Remote Shutdown Panel ONEP, Rev. 49

Handouts to be provided to the Applicants during exam:

05-1-02-II-1, Shutdown from the Remote Shutdown Panel ONEP, Attachments I and II

Learning Objective:

GLP-OPS-ONEP, OBJ. 2

Question Source:	**Bank # 109**	2012 NRC Exam Q# 17
(note changes and attach parent)	**Modified Bank #**	
	New	
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(10)	
Level of Difficulty:	3	
PRA Applicability:		

Examination Outline Cross Reference	Level	RO
295018 Partial or Total Loss of CCW AA1 Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER : AA1.02 System loads	Tier	1
	Group #	1
	K/A	295018
	Rating	3.3
	Revision	1
Revision Statement: **Changed to operating pump** **Changed stem verb to identifies and wording to without CCW flow** **Swapped answers B and C** **Changed LOD to 4**		

Question: 7

Plant is operating at rated thermal power.

One operating CCW pump trips, AND the standby CCW pump can not be started.

All subsequent actions from the Loss of CCW ONEP for a partial loss of CCW have been completed.

Which of the following identifies the components without CCW flow?

A. Reactor Recirculation Pumps and Control Rod Drive Pump coolers

B. Control Rod Drive Pump coolers and RWCU Non-Regen Heat Exchangers

C. Reactor Recirculation Pumps and Fuel Pool Cleaning and Cleanup Heat Exchangers

D. Fuel Pool Cleaning and Cleanup Heat Exchangers and RWCU Non-Regen Heat Exchangers

Answer: **D**
Explanation: Per 05-1-02-V-1, Loss of CCW ONEP, If only one pump is running, then isolate the CCW to Fuel Pool cooling HT EX. and isolate CCW to RWCU non-regen Ht EX. Also, if only one pump is running, insufficient flow will be provided to the Fuel Pool Heat exchangers and they will auto isolate due to low CCW flow. Reactor Recirc Pmps and CRD coolers have manual isolation valves and will not isolate and there is no

direction in ONEP to isolate CCW from these components.

Distracters:

A is wrong. These are two components that will always maintain CCW system flow.

B is wrong. Reactor Recirc system will always maintain CCW system flow due to only manual valves are used, no auto isolating valves supplying this component and there is no direction in ONEP to isolate CCW from these components.

C is wrong. CRD system will always maintain CCW system flow due to only manual valves are used, no auto isolating valves supplying this component and there is no direction in ONEP to isolate CCW from these components.

K/A Match

This question requires the applicant to have the knowledge of CCW system loads and which are isolated by the use of the ONEP and knowledge of overall mitigating strategy of the ONEP procedure.

Technical References:

GLP-OPS-P4200, Rev. 15, Component Cooling Water Lesson Plan

GLP-OPS-G3336, Rev. 17, Reactor Water Cleanup Lesson Plan

05-1-02-V-1, Loss of CCW ONEP, Rev. 24

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-P4200, OBJ. 10

Question Source:	**Bank # 1145**	X
(note changes and attach parent)	**Modified Bank #**	
	New	
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(4)	
Level of Difficulty:	4	

PRA Applicability:

Examination Outline Cross Reference	Level	RO
295019 Partial or Total Loss of Inst. Air 2.2.44 Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions.	Tier	1
	Group #	1
	K/A	295019
	Rating	4.2
	Revision	1
Revision Statement: **Defined PAC in stem** **Editorial changes in answers**		

Question: 8

The plant is operating at rated thermal power with the following lineup for the Plant Air Compressors (PAC):

- 'A' PAC is running
- 'B' PAC is in STANDBY
- 'C' PAC is Tagged out for maintenance

'A' PAC trips.

The RED light indication for the 'B' PAC begins to blink with the GREEN indication on solid.

(1) What is the current status of PAC 'B'?

(2) IAW Loss of Instrument Air ONEP, which condition requires immediate plant shutdown?

A. (1) In process of automatically starting
(2) Control Rods drifting

B. (1) Has trip signal present
(2) P43-F289, TBCW Air Compressor Return Valve, closes

C. (1) In process of automatically starting
(2) P43-F289, TBCW Air Compressor Return Valve, closes

D. (1) Has trip signal present
(2) Control Rods drifting

Answer:	A

Explanation:

Per GLP-OPS-P5100, when the Standby Air Compressor receives an auto start signal the Red indication will begin to blink showing auto start process is in progress. The Green indication will stay on until the compressor starts then the Green light will extinguish and the Red light will go solid.

Also, Per ONEP, 05-1-02-V-9, Loss of Instrument Air, Step 3.1, Monitor for Control Rod drifts. If a Control Rod is drifting, then take action of ONEP 05-1-02-IV-1, Control Rod/Drive Malfunctions. Step 2.3, Multiple control rod drifting Manually scram the reactor.

Distracters:

B is wrong, but plausible. Indication of an auto start process is shown not a trip signal. Also, if the P43-F289 fails closed the actions are to place the Air Compressors on SSW for cooling per 05-1-02-V-9, Loss of Instrument Air ONEP step 3.3 and if no cooling water available, perform a plant shutdown due to loss of IA.

C is wrong, but plausible. If the P43-F289 fails closed the actions are to place the Air Compressors on SSW for cooling per 05-1-02-V-9, Loss of Instrument Air ONEP step 3.3 and if no cooling water available perform a plant shutdown due to loss of IA.

D is wrong, but plausible. Indication of an auto start process is shown, not a trip signal.

K/A Match

This question requires the applicant to interpret given control room air compressor indications and evaluate plant status for possible operator actions. Also, should have the knowledge of Loss of Instrument air, the auto start indication of the Standby compressor and understanding of actions and processes when using an ONEP and the effects on the plant.

Technical References:

GLP-OPS-P5100, Rev. 8, Plant Air System Lesson Plan, page 14 of 39

ONEP, 05-1-02-V-9, Loss of Instrument Air ONEP Rev. 45, Step 3.1

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-P5100, OBJ. 11

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	X

	Comprehensive / Analysis	
10CFR Part 55 Content:	55.41(b)(4)	
Level of Difficulty:	3	
PRA Applicability:		

Failure to align SSW ‘B’ to cool IA compressors is listed as a top 10 “Operator Action Importance to CDF.

Plant Air System is listed as #15 of System Importance to CDF.

<table>
<tr><td>Examination Outline Cross Reference</td><td>Level</td><td>RO</td></tr>
<tr><td rowspan="5">295021 Loss of Shutdown Cooling

AA2 Ability to determine and/or interpret the following as they apply to LOSS OF SHUTDOWN COOLING :

AA2.02 RHR/shutdown cooling system flow</td><td>Tier</td><td>1</td></tr>
<tr><td>Group #</td><td>1</td></tr>
<tr><td>K/A</td><td>295021</td></tr>
<tr><td>Rating</td><td>3.4</td></tr>
<tr><td>Revision</td><td>3</td></tr>
<tr><td colspan="3">Revision Statement:

Rev 1: Used bullets in stem
Minor editorial edits

Rev 2: Editorial changes in stem

Rev 3: Minor editorial changes</td></tr>
</table>

Question: 9

Plant is in refueling outage:

- RHR ‘B’ operating in Shutdown Cooling
- Return flowpath through E12-F037B, RHR B TO CTMT POOL
- CRO notices RHR Pump ‘B’ flow indication is lowering with pump still running

Which of the following Group 3 isolation signals occurred?

A. Drywell pressure >1.23 psig

B. RHR room temp >165°F

C. Reactor water level <11.4 inches

D. Reactor pressure >135 psig

Answer: A		
Explanation: The applicant must interpret the given information and determine cause for the given indicaitons. Group 3 isolation signal isolates the RHR Shutdown Cooling system. A group 3 isolation includes drywell pressure of >1.23 psig but will only isolate the E12-F037A/B, RHR A/B TO CTMT POOL . These are the only valves that will close. No trip signals are sent to the pump therefore the pump would go to min flow and the flow indicator would reduce to zero. All other Group 3 signals would cause a pump trip due to loss of suction path.		
Distracters: B is wrong, but plausible. This would cause a full Group 3 isolation, the suction valves would close and the RHR pump would trip. C is wrong, but plausible. This would cause a full Group 3 isolation, the suction valves would close and the RHR pump would trip. D is wrong, but plausible. This would cause a full Group 3 isolation, the suction valves would close and the RHR pump would trip, also this would not affect the F037A/B valves. IAW with 03-1-01-5, Refueling IOI, this isolation is bypassed.		
K/A Match This question requires the applicant to interpret RHR system flow indication and ascertain the cause for the reduction in flow.		
Technical References: 05-1-02-III-5, Automatic Isolations ONEP, Rev. 49		
Handouts to be provided to the Applicants during exam: NONE		
Learning Objective: GLP-OPS-E1200, OBJ. 12.2		
Question Source:	**Bank # 648**	X
(note changes and attach parent)	**Modified Bank #**	
	New	
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(8)	

Level of Difficulty:	3	
PRA Applicability:		
RHR is #3 of System Importance to CDF.		

Examination Outline Cross Reference	Level	RO
295023 Refueling Acc AK2 Knowledge of the interrelations between REFUELING ACCIDENTS and the following: AK2.06 Containment ventilation: Mark-III	Tier	1
	Group #	1
	K/A	295023
	Rating	3.4
	Revision	1
Revision Statement: **Use of capitals and use of all caps**		

Question: 10

The plant is in a refueling outage with fuel movement in progress.

Containment ventilation is operating in Normal mode:

- One Supply fan running
- One Exhaust filter train
- One Recirc filter train
- Two Containment coolers

A spent fuel bundle on the Refueling Platform is raised too high and causes Containment Ventilation Exhaust Radiation monitors to reach 4.0 mR/hr.

Containment ventilation system will___________.

A. auto transfer into Containment Purge mode.

B. trip and require manual restart in Cleanup mode ONLY.

C. continue to operate with Recirc filter train and Containment coolers ONLY.

D. continue to operate with all components, but required to be immediately shutdown.

Answer: **C**		
Explanation: At a setpoint of 3.6 mR/hr on the Containment Exhaust Rad monitor causes the following: • Containment Vent Supply fans to trip • Containment Exhaust Charcoal Filter Train Fans to trip • Drywell / Containment Purge Fans will trip However, the Recirc Charcoal Filter Trains and the Containment coolers will continue to operate which is considered the containment cleanup mode. Therefore, the Containment ventilation continues to operate with Recirc filter train and Containment coolers only.		
Distracters: A is wrong, but plausible. Containment Purge mode is unavailable due to auto valve isolations. B is wrong, but plausible. The Coolers and Recirc charcoal filter train will continue to operate. D is wrong, but plausible. Will not continue to operate due to system isolation.		
K/A Match This question requires the applicant to have the knowledge of interrelationship between Containment ventilation and a refueling accident. If a fuel bundle is raised too high causing high radiation would be considered a refueling accident.		
Technical References: 04-1-01-M41-1, Rev. 114, Containment Cooling System SOI GLP-OPS-M4100, Rev. 9, Containment Cooling System Lesson Plan		
Handouts to be provided to the Applicants during exam: NONE		
Learning Objective: GLP-OPS-M4100, OBJ. 12.1		
Question Source:	**Bank #**	
(note changes and 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	
PRA Applicability:		

Examination Outline Cross Reference	Level	RO
295024 High Drywell Pressure 2.2.37 Ability to determine operability and/or availability of safety related equipment.	Tier	1
	Group #	1
	K/A	295024
	Rating	3.6
	Revision	2

Revision Statement:

Rev 1: Used the word “status” instead of “availability”

Rev 2: Removed ‘The system’ in each answer and reworded answer D.

Question: 11

HPCS has automatically started on low reactor water level and has restored level to +34” and rising.

Drywell pressure is 2.5 psig and stable.

To control water level, a RO stops the HPCS Pump using its handswitch.

The RO then depresses the HPCS INIT RESET pushbutton.

What is the status of the High Pressure Core Spray system?

A. Will immediately restart and inject.

B. Will auto start when Reactor water level reaches -41.6”.

C. Can ONLY be started using the Manual Initiation pushbutton.

D. Can ONLY be started manually using respective handswitches.

Answer: **B**

Explanation:

When the RO secures the HPCS pump and closes the injection valve, both components receive an override condition. An alarm is also received to verify the override condition.

Once generated, the HPCS initiation signal is sealed-in and must be manually reset by depressing the HPCS INIT RESET pushbutton. If reactor water level has been restored, but a high drywell pressure signal is still present, depressing the HPCS INIT RESET pushbutton removes the high drywell pressure initiation signal from the automatic start sequence.

When the system is shutdown, with auto signals present, it will go into an override condition. With only drywell pressure initiation signal currently present, when the INIT RESET pushbutton is depressed all initiation signals are now gone, and the override condition is reset. The HPCS pump and injection valve return to normal automatic lineup and all override alarms clear.

Distracters:

A is wrong, but plausible. Without the knowledge that the reset pushbutton overrides the drywell pressure even if it's above the setpoint, the applicant might believe the system will auto restart immediately

C is wrong, but plausible. The system will auto start on level, manual action will start the system, but not the only way that the system can be started.

D is wrong, but plausible. The system can also be manually started using respective handswitches but not required to start the system. HPCS will auto start on level if it reaches the start setpoint. Plausible if the applicant believes that the override is still in effect.

K/A Match

This question requires the applicant to have the knowledge of availability of the HPCS system (safety related) equipment with a high drywell pressure.

Technical References:

04-1-01-E22-1, Rev. 123, High Pressure Core Spray System SOI

GLP-OPS-E2201, Rev. 10, High Pressure Core Spray System Lesson Plan

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-E2201, OBJ. 11 & 12

Question Source:	**Bank #**	
(note changes and 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	
PRA Applicability:		

HPCS is #16 system on list of System Importance To CDF.

Rx Rupture is listed at .037% in the chart of Accident Type Contribution to CDF.

Examination Outline Cross Reference	Level	RO
295025 High Reactor Pressure EK1 Knowledge of the operational implications of the following concepts as they apply to HIGH REACTOR PRESSURE : EK1.03 Safety/relief valve tailpipe temperature/pressure Relationships	Tier	1
	Group #	1
	K/A	295025
	Rating	3.6
	Revision	2

Revision Statement:

Rev 1: Reworded stem and #2 question for clarity

Rev 2: Minor editorial change & Added reference material for discussion on actuation of lo-lo set.

Question: 12

A transient occurred causing Reactor pressure to rise just high enough to initiate the Lo-Lo Set subsystem of SRVs.

(1) How many SRVs initially opened?

(2) What is the peak SRV tail pipe temperature?

A. (1) One
(2) 411°F

B. (1) Two
(2) 411°F

C. (1) One
(2) 558°F

D. (1) Two
(2) 558°F

Answer: **B**

Explanation:

Lo-Lo set system initiates at 1103 psig. Convert 1103 psig to psia by adding 14.7, which is approximately 1118 psia.

Once initiated one SRV lowers its opening setpoint to 1033 psig and another lowers its opening setpoint to 1073, therefore 2 SRVs will initially be open. The other SRVs open at 1113 psig or 1123 psig.

The point of initiation is 1103 psig or 1118 psia reactor pressure; however, tailpipe pressures average about 25% of reactor steam dome pressure for a full valve lift at rated conditions.

1118 X 25% = 279.5 psia.

Saturated temperature for 279.5 psia is 411°F.

Distracters:

A is wrong. Two SRVs will be open, not one. Plausible if the applicant doesn't remember that when LO-LO-set initiates at 1103 psig the opening setpoint of 2 SRVs lowers to 1033 and 1073, therefore, 2 SRVs will immediately open.

C is wrong. Temperature is for 1118 psia and two SRVs will be open, not one. Plausible if the applicant doesn't remember that when LO-LO-set initiates at 1103 psig the opening setpoint of 2 SRVs lowers to 1033 and 1073, therefore, 2 SRVs will immediately open. Also, if the applicant doesn't recall the process of steam going through a SRV at rated conditions.

D is wrong. Temperature is for 1118 psia. Plausible if the applicant doesn't recall the process of steam going through a SRV at rated conditions.

K/A Match

This question requires the applicant to have the knowledge of Safety/relief valve tailpipe temperature/pressure Relationships and the operational implications of the indication.

Technical References:

GLP-OPS-E2202, Rev. 9, Automatic Depressurization System (ADS) Lesson Plan pages 21 and 34 of 45

Handouts to be provided to the Applicants during exam:

Steam Tables

Learning Objective:

GLP-OPS-E2202, OBJ. 18.0

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(14)	
Level of Difficulty:	3	
PRA Applicability:		

Examination Outline Cross Reference	Level	RO
295026 Suppression Pool High Water Temp. EA2 Ability to determine and/or interpret the following as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE: EA2.03 Reactor pressure	Tier	1
	Group #	1
	K/A	295026
	Rating	3.9
	Revision	0
Revision Statement:		

Question: 13

HANDOUT PROVIDED

An ATWS is in progress.

- Suppression pool water level is 18.0 ft.
- Suppression pool water temperature is 150°F

Assume these parameters cannot be restored or maintained.

Which of the following describes the LOWEST Reactor pressure at which an Emergency Depressurization is required?

A. 1100 psig

B. 1050 psig

C. 940 psig

D. 760 psig

Answer: C
Explanation: Per EP3 "If SP temp and RPV pressure cannot be restored and maintained within the Safe zone of the HCTL, Emergency Depressurization is Required. Using the giving information, IAW 02-S-01-43, Transient Mitigation Strategy, step 6.1.6, with SP level at 18.0 ft. the 16.50 ft. line should be used to be conservative and prevent interpolation and SP temp of 150°F the RPV pressure that would require an ED is 940 psig.
Distracters:

All distracters are plausible if the applicant can't execute the use of the HCTL curve.

A is wrong. This is the RPV pressure if the student uses 18.34 ft. line.

B is wrong. this is the RPV pressure if the student interpolates the 18.0 ft (Bases does not allow for interpolation on this figure).

D is wrong. This is the RPV pressure if the student uses 14.5 ft. line.

K/A Match

This question requires the applicant to have the ability to determine and/or interpret Reactor pressure where actions are required when given suppression pool level and temperature.

Technical References:

GLP-OPS-EP3, Rev. 2, Emergency Procedure (EP-3)

02-S-01-43, Transient Mitigation Strategy, Rev. 3

Handouts to be provided to the Applicants during exam:

05-S-01-EP-1, FIGURES HCTL curve

Learning Objective:

GLP-OPS-EP3, OBJ. 8.0

Question Source:	**Bank # 631**	X
(note changes and attach parent)	**Modified Bank #**	
	New	
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(5)	
Level of Difficulty:	2	

PRA Applicability:

Failure to manually depressurize with ADS/SRVs. Number 1 on list of Operator Action Importance to CDF.

Failure to start suppression pool cooling. Number 10 on list of Operator Action Importance to CDF.

Examination Outline Cross Reference	Level	RO
295027 High Containment Temperature 2.1.25 Ability to interpret reference materials, such as graphs, curves, tables, etc.	Tier	1
	Group #	1
	K/A	295027
	Rating	3.9
	Revision	3
Revision Statement: **Rev 1: Added "current data points"** **Changed answer 'A' from Fuel Zone Range only.** **Make 'only" All Caps** **Rev 2: changes inch symbol to inches in stem** **Rev 3: changed LOD from 2 to 3**		

Question: 14

HANDOUT PROVIDED

A LOCA is in progress with the following current data points:

- Wide Range level is -10 inches
- Fuel Zone level is -25 inches
- Upset Range level is 5 inches
- Shutdown Range level is 10 inches
- RPV pressure is 50 psig
- Drywell temperature (166 ft) = 220°F; (139 ft) = 190°F
- CTMT temperature (166 ft) = 155°F; (139 ft) = 150°F

Which of the reactor water level instrument(s) **are** usable?

A. Upset Range and Shutdown Range ONLY

B. Fuel Zone Range and Wide Range ONLY

C. Fuel Zone Range, Wide Range, and Upset Range, ONLY

D. Fuel Zone Range, Wide Range, Upset Range, and Shutdown Range

Answer: B

Explanation:

See EP-1 CAUTION 1. The DW and CTMT temperatures in the stem fall within the “safe zone” of the RPVST curve (Figure 2); therefore, there are no possible boiling concerns. This makes the Fuel Zone Range instrument completely valid and usable per Caution 1.1. Per Caution 1.2, a Wide Range, Upset Range, or Shutdown Range instrument may not be used if BOTH 1) indicated level is below a certain limit AND 2) DW or CTMT temperature at a specified elevation is above a certain limit. The indicated level for Wide Range (-10”) is above the specified limit (-131”); therefore, Wide Range is usable.

Distracters:

All distracters are wrong but are plausible based on the Applicant’s need to apply Caution 1 as already described.

The indicated level for Upset Range (5”) is below its limit (159”) AND the stem’s given DW temperature at the 166 ft elevation (220°F) is above the associated limit (195°F); therefore, Upset Range is not usable. The indicated level for Shutdown Range (10”) is below its limit (139”) AND the stem’s given DW temperature at the 166 ft elevation (220°F) is above the associated limit (66°F); therefore, the Shutdown Range is not usable.

K/A Match

This question requires the applicant to have the ability to interpret reference materials, such as graphs and tables (i.e. EP Caution 1) to determine if reactor water level indication is valid and useable.

Technical References:

EP-1, CAUTION 1, 05-S-01-EP-1, Emergency/Severe Accident Procedure Support Documents, Rev. 36

Handouts to be provided to the Applicants during exam:

EP-1, CAUTION 1

Learning Objective:

GLP-OPS-EP02, OBJ 8

Question Source:	**Bank # 364**	2015 NRC Q #13
(note changes and attach parent)	**Modified Bank #**	
	New	

Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(10)	
Level of Difficulty:	3	
PRA Applicability:		

Examination Outline Cross Reference	Level	RO
295028 High Drywell Temperature EK2 Knowledge of the interrelations between HIGH DRYWELL TEMPERATURE and the following: EK2.03 Reactor water level indication	Tier	1
	Group #	1
	K/A	295028
	Rating	3.6
	Revision	1
Revision Statement: **Moved 'go" to the stem and deleted from each answer**		

Question: 15

A steam leak has occurred in the upper elevations of the Drywell.

Both CRD pumps have tripped.

Reactor water level condensing pot reference leg begins to boil.

Actual water level remains constant.

Initially, differential pressure output of the level transmitter will go _____(1)______ and **indicated** level will go ____(2)______.

A. (1) down
   (2) down

B. (1) up
   (2) up

C. (1) up
   (2) down

D. (1) down
   (2) up

Answer:	**D**
Explanation: In a normal level d/p cell reference leg pressure is the high side and actual level pressure is the low side. As temperature around the reference leg rises and reaches saturation it begins to boil away. This will cause pressure on the high side to reduce causing a lower d/p; therefore, d/p will go down. This will cause indicated level to actually rise so d/p goes down and level goes up.	
Distracters: All distracters are wrong, but are plausible based on the applicant's understanding of a d/p level	

instrumentation.

K/A Match

This question requires the applicant to have the knowledge of the fundamental operation of a level d/p cell and the interrelationship with high drywell temperatures.

Technical References:

GLP-OPS-B2101, Rev. 10, RPV Level Instrumentation Lesson Plan

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-B2101, OBJ. 12.2

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.41(b)(5)	
Level of Difficulty:	2	
PRA Applicability:		

Examination Outline Cross Reference	Level	RO
295030 Low Suppression Pool Wtr Lvl **EA1 Ability to operate and/or monitor the following as they apply to LOW SUPPRESSION POOL WATER LEVEL:** **EA1.03 HPCS: Plant-Specific**	Tier	1
	Group #	1
	K/A	295030
	Rating	3.4
	Revision	1
Revision Statement: **Changed in answer D, FT to ft.**		

Question: 16

IAW 04-1-01-E22-1, HPCS SOI:

To ensure adequate NPSH, the HPCS pump should not be run with suction from the suppression pool if level is ≤ _________, except during an emergency.

A. 18.34 ft.

B. 17.5 ft.

C. 14.5 ft.

D. 10.5 ft.

Answer:	**C**
Explanation: Per 04-1-01-E22-1, HPCS SOI, step 3.20, To ensure adequate NPSH, the HPCS pump should not be run with suction from the suppression pool if level is ≤ 14.5’, except during an emergency.	
Distracters: A is wrong, but plausible. This is the low level suppression pool EP3 entry and Tech Spec entry. B is wrong, but plausible. This is the auto initiation signal for Suppression pool makeup system D is wrong, but plausible. This is the required level to be above to emergency depressurize the RPV.	
K/A Match This question requires the applicant to have the knowledge of the ability to operate the HPCS system with low suppression pool level.	

Technical References: GLP-OPS-E2201, Rev. 10, High Pressure Core Spray Lesson Plan 04-1-01-E22-1, High Pressure Core Spray SOI, Rev. 123, Step 3.20
Handouts to be provided to the Applicants during exam: NONE
Learning Objective: GLP-OPS-E2201, OBJ. 15

Question Source:	**Bank #**	
(note changes and 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	

PRA Applicability:
HPCS is #16 system on list of System Importance To CDF.

Examination Outline Cross Reference	Level	RO
295031 Reactor Low Water Level EA2 Ability to determine and/or interpret the following as they apply to REACTOR LOW WATER LEVEL : EA2.04 Adequate core cooling	Tier	1
	Group #	1
	K/A	295031
	Rating	4.6
	Revision	0
Revision Statement:		

Question: 17

Which of the following is the **lowest** reactor water level that adequate core cooling is assured **without** RPV injection?

A. -167 inches

B. -191 inches

C. -204 inches

D. -217 inches

Answer: C

Explanation:

Per 02-S-01-43, Transient Mitigation Strategy, step 5.1, Adequate core cooling is assured whenever any of the following conditions exist.

- Reactor water level is at or above -167 inches
- Reactor water level is at or above -191 inches
- Reactor water level is at or above -204 inches without RPV injection
- HPCS or LPCS flow above 7000 gpm and reactor water level above -217 inches.

Distracters:

All distracters are other levels used in the definition of adequate core cooling with certain conditions existing as stated above.

K/A Match

This question requires the applicant to have the knowledge of adequate core cooling and the RPV levels required to achieve.

Technical References:

02-S-01-43, Transient Mitigation Strategy, Rev. 3, step 5.1		
Handouts to be provided to the Applicants during exam: NONE		
Learning Objective: GLP-OPS-PROC, OBJ. 15		
Question Source:	**Bank #**	
(note changes and 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	
PRA Applicability:		

Examination Outline Cross Reference	Level	RO
295037 SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown Knowledge of the reasons for the following responses as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN : EK3.01 Recirculation pump trip/runback: Plant-Specific	Tier	1
	Group #	1
	K/A	295037
	Rating	4.1
	Revision	1
Revision Statement: **Changed answers from short to long** **Added CAPS to component in stem**		

Question: 18

During an ATWS, what is the reason for verifying Reactor Recirc pumps have transferred to LFMG prior to initiating ATWS/ARI?

A. Prevent 6.9 KV BOP bus power fluctuations

B. Prevent Thermal Hydraulic Instabilities (THI)

C. Prevent level perturbations during level reduction

D. Prevent tripping the Main Turbine and RFPT turbine

Answer: **D**

Explanation:

An immediate and rapid reactor power reduction may be effected by reducing reactor coolant recirculation flow rate. The most rapid flow rate reduction and, consequently, the most rapid power reduction, is achieved by tripping the recirculation pumps to OFF. However**, if the pump trip to OFF is initiated from a high power level, the resulting plant transient may trip the main turbine, Feedwater, or RCIC due to the rapid change in RPV water level. If the turbine trips and reactor power exceeds the turbine bypass valve capacity, RPV pressure will increase until one or more SRVs open. Heatup of the suppression pool would then begin and boron injection might ultimately be required. A trip of Feedwater or RCIC could complicate RPV water level control actions.**

To effect a more controlled power reduction and avoid undesirable RPV water high level trips, the recirculation pumps are transferred to the LFMGs before the pumps are tripped to OFF. If the transfer has not occurred automatically, the action should be performed manually.

Since the ARI logic trips the recirculation pumps to OFF at GGNS, the pumps are transferred to the LFMGs before ARI initiation is verified. Initiating ARI first would immediately trip the recirculation pumps to OFF, increasing the likelihood of RPV water high level trips of the main turbine, Feedwater, and RCIC.

Distracters:

A is wrong, but plausible. When starting the Reactor Recirc pumps, taps are adjusted on the 6.9 KV buses to prevent voltage fluctuations, but not for tripping a recirc pump.

B is wrong, but plausible. Reducing recirculation flow may place the plant in a high power-to-flow condition where thermal-hydraulic instabilities are more likely. The actions remain appropriate. This is not the reason for going to LFMG prior to initiating ATWS/ARI.

C is wrong, but plausible. Down shifting to LFMG and then tripping the recirc pumps to OFF is for an immediate and rapid power reduction. These steps are done prior to reduction of level.

K/A Match

This question requires the applicant to have the knowledge of the reason for downshifting the recirc pumps prior to tripping to OFF during an ATWS condition.

Technical References:

02-S-01-40, EP Technical Bases, Rev. 8, page 4 of 58 Attachment V

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-EP02A, OBJ. 7

Question Source:	**Bank #**	
(note changes and 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	
PRA Applicability:		

Examination Outline Cross Reference	Level	RO
600000 Plant Fire On Site AK1 Knowledge of the operation applications of the following concepts as they apply to Plant Fire On Site: AK1.02 Fire Fighting	Tier	1
	Group #	1
	K/A	600000
	Rating	2.9
	Revision	1

Revision Statement:

Changed verb to identifies in stem
Editorial changes in stem and answers
ALL CAPS on “only”.

Question: 19

Heavy smoke and flames are being reported coming from load center 15B51.

Per 10-S-03-2, Response to Fires, which of the following identifies the <u>required</u> response of the Control Room crew?

A. Dispatch 5 Man Fire Brigade, ONLY

B. Dispatch Claiborne County Fire Department, ONLY

C. Dispatch 3 Man Fire Brigade with Claiborne County Fire Department for backup support

D. Dispatch 5 Man Fire Brigade with Claiborne County Fire Department for backup support

Answer: **D**

Explanation:

Per 10-S-03-2, Response to fires, 6.2.3.e, The Operations 5 man Fire Brigade is the Primary Responder for all fires in: U1 and U2 Power Block, Diesel Generator Bays, **SSW A and B**, Independent Fuel Storage, ESF Transformers (11, 21, 12), Fire Water Pump House, and all other areas of the plant containing structures, systems or components important to safety (manholes, tanks, etc.). **Claiborne County will provide backup support**.

Distracters:

A is wrong, but plausible. Procedure states Claiborne County **<u>will</u>** provide backup support

B is wrong, but plausible. See explanation above

C is wrong, but plausible. See explanation above		
K/A Match This question requires the applicant to have the knowledge of the application of the operations crew during a fire fighting event.		
Technical References: 10-S-03-2, Response to Fires, Rev. 28, step 6.2.3.e		
Handouts to be provided to the Applicants during exam: NONE		
Learning Objective: GLP-OPS-PROC, OBJ. 58.2 & 58.3		
Question Source:	**Bank # 655**	X
(note changes and attach parent)	**Modified Bank #**	
	New	
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(10)	
Level of Difficulty:	3	
PRA Applicability:		

Examination Outline Cross Reference	Level	RO
700000 Generator Voltage and Electric Grid Disturbances AK3 Knowledge of the reasons for the following responses as they apply to GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES: AK3.02 Actions contained in abnormal operating procedure for voltage and grid disturbances	Tier	1
	Group #	1
	K/A	700000
	Rating	3.6
	Revision	1
Revision Statement: **Editorial changes in stem and answers**		

Question: 20

Loss of AC Power ONEP states for grid instability, “Maintain Main Generator MVARs between +297 and -253 MVARs.”

What is the reason for this step?

A. Prevent a reverse power condition

B. Prevent Generator stator bars from overheating

C. Ensure ESF bus voltages remain within Tech Spec limits

D. Ensure Generator output voltage is maintained at 22.5 KV

Answer: **B**

Explanation:

Per ONEP, section 3.4.3 NOTE:

Maintaining Generator MVARS within these limits Should prevent Generator stator bars from overheating AND ensure the reverse power relay recognizes a reverse power condition.

Distracters:

A is wrong, but plausible. Per the above explanation, this will ensure the reverse power relay recognizes a reverse power condition not prevent one.

C is wrong, but plausible. Maintaining MVARs on the Generator has no control of Grid parameters.

D is wrong, but plausible. This is performed by the Voltage Regulator System.

K/A Match This question requires the applicant to have the knowledge of the reason for steps in the Loss of Power ONEP.		
Technical References: 05-1-02-I-4, Loss of AC Power, Rev. 50, step 3.4.3 CAUTION		
Handouts to be provided to the Applicants during exam: NONE		
Learning Objective: GLP-OPS-ONEP, OBJ. 19 & 52		
Question Source:	**Bank #**	
(note changes and 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	
PRA Applicability:		
Offsite power is #12 on the list of System Importance to CDF.		

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

Revision Statement:

Rev 1Per NRC: Changed stem and reworded answer 'B', added noun names and rev numbers on all Reference Material.

Rev 2 Changed 10CFR number to 3, MSIVs are a component of the primary system

Question: 21

Given the following conditions:

- Plant in MODE 2
- Reactor power is 5%
- Reactor pressure is 900 psig
- MSIVs are open

A leak occurs causing main condenser vacuum to drop to 8" Hg Vac.

Without operator action, what is the expected plant response?

A. REACTOR HIGH PRESSURE scram

B. Reactor power remains constant

C. MSIV CLOSURE scram

D. SRM UPSCALE scram

Answer: A
Explanation: The MSIV CLOSURE scram is bypassed in MODE 2 (not in RUN), the MSIVs still auto-closure when vac drops below 9" Hg vac. When they do, and without operator action, the 5% steam being produced will have no place to go, the implication of loss of heat sink would be rapid reactor pressure rise to the REACTOR HIGH PRESSURE scram setpoint.

Distracters:

B is wrong, but plausible. Reactor will scram on High Pressure (never bypassed)

C is wrong, but plausible. MSIV CLOSURE scram is bypassed in MODE 2 (not in RUN)

D is wrong, but plausible. SRM upscale will not cause a scram due to shorting links installed, but plausible due to a rise in reactor pressure will cause a rise in reactor power.

K/A Match

This question requires the applicant to have the knowledge of the operational implications due to a loss of the main heat sink (main condenser) when the MSIVs close on low vacuum.

Technical References:

GLP-OPS-M7100, Containment And Drywell Instrumentation And Control System Lesson Plan Rev. 14

GLP-OPS-C7100, Reactor Protection System (RPS) Lesson Plan, Rev 13

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-C7100, OBJ. 9 & 10
GLP-OPS-M7100, OBJ 7.1

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

PRA Applicability:

RPS is #5 on the list of System Importance to CDF.

Examination Outline Cross Reference	Level	RO
295009 Low Reactor Water Level AK2 Knowledge of the interrelations between LOW REACTOR WATER LEVEL and the following: AK2.02 Reactor water level control	Tier	1
	Group #	2
	K/A	295002
	Rating	3.6
	Revision	1
Revision Statement: **Editorial changes in stem and answers** **Added “in inches” to stem** **Changed inch symbol to inches in stem** **Removed inch symbol in all answers**		

Question: 22

The plant is operating at rated conditions.

An event occurs causing reactor water level to lower.

Feedwater system is unable to restore reactor water level.

Over a time of 5 minutes reactor water level continues to lower until Level 2, -41.6 inches is reached.

Reactor water level currently is -30 inches and rising.

(1) What reactor water level setpoint (in inches) initiates Setpoint Setdown in the reactor water level control system?

(2) What is the current setpoint (in inches) of the Master Level Controller?

A. (1) +11.4
(2) +54

B. (1) -41.6
(2) +54

C. (1) +11.4
(2) +4

D. (1) -41.6
(2) +4

Answer: C

Explanation:

When the selected level channel senses RPV level at less than +11.4 inches NR, the following actions occur:

1. The Master Level Control Station shifts into the AUTO-TRACK mode and immediately receives a new setpoint of +54 inches, generated by the INFI-90 software, resulting in an immediate increase in feedwater flow to overcome the initial shrink that follows a scram.

2. Ten seconds later or when the selected level channel senses RPV water level at greater than +12.4 inches NR, whichever occurs first, the Master Level Control setpoint is automatically changed to +4 inches.

Distracters:

A is wrong, but plausible. +11.4 inches is correct, however +54 inches is only used for approximately 10 seconds then +4 inches is used.

B is wrong, but plausible. Setpoint setdown will initiate at +11.4 inches not -41.6 inches. -41.6 inches is the level that will initiate RCIC, HPCS and various isolations but nothing dealing with level control system, and +54 inches is only used for approximately 10 seconds then +4 inches is used.

D is wrong, but plausible. Setpoint setdown will initiate at +11.4 inches not -41.6 inches. -41.6 inches is the level that will initiate RCIC, HPCS and various isolations but nothing dealing with level control system and +4 inches is correct.

K/A Match

This question requires the applicant to have the knowledge of the interrelations of Reactor water level low and reactor water level control system.

Technical References:

GLP-OPS-C3400, Digital Feedwater Control System (DFCS) Lesson Plan, Rev. 17

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-C3400, OBJ. 12

Question Source:	**Bank #**	
(note changes and 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	
PRA Applicability:		

Examination Outline Cross Reference	Level	RO
295011 High Containment Temp AK3 Knowledge of the reasons for the following responses as they apply to HIGH CONTAINMENT TEMPERATURE (MARK III CONTAINMENT ONLY): AK3.01 Increased containment cooling: Mark-III	**Tier**	1
	Group #	2
	K/A	295011
	Rating	3.6
	Revision	1
Revision Statement: **Made question part of stem 2 parts** **Made answers 2X2** **Added space in bullets**		

Question: 23

The following conditions exist following a scram:

- Containment temperature 160°F, rising 2°F/min
- Containment pressure 2.5 psig, rising 0.1 psig/min
- All containment coolers operating

If these conditions continue;

(1) Which of the following is the next action required IAW EP-3?

(2) What is the reason for this action?

A. (1) Initiate Containment Spray
(2) to prevent exceeding containment design **pressure** limit

B. (1) Initiate Containment Spray
(2) to prevent exceeding containment design **temperature** limit

C. (1) Place Containment Ventilation in HIGH Volume Purge Mode
(2) to prevent exceeding containment design **pressure** limit

D. (1) Place Containment Ventilation in HIGH Volume Purge Mode
(2) to prevent exceeding containment design **temperature** limit

Answer: **B**

Explanation:

Entering EP-3 for containment temperature, the first step is to Operate all CTMT cooling (Step CNT-2). If temperature continues to increase, the next step is to provide any type of cooling. Before CTMT temperature reaches 185°F, initiate Containment Spray per step CNT-6. With temperature rising at 2 degrees per minute the concern is with temperature not pressure in the containment.

This action will prevent exceeding the containment design temperature.

Distracters:

A is wrong, but plausible. The concern is with containment temperature not pressure, containment temperature will exceed design limits before pressure.

C is wrong, but plausible. The concern is with containment temperature not pressure. High volume purge is part of containment ventilation but not used for cooling nor is it mentioned in EP3.

D is wrong, but plausible. High volume purge is part of containment ventilaiton but not used for cooling nor is it mentioned in EP3.

K/A Match

This question requires the applicant to have the knowledge of the reason for increased cooling for the containment.

Technical References:

02-S-01-40, EP Technical Bases Rev. 8, Attachment VI, page 14 - 18 of 37

05-S-01-EP-3, Rev. 29

GLP-OPS-EP3, Rev 2

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-EP3, OBJ. 7

Question Source:	**Bank #**	
(note changes and 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:	2	
PRA Applicability:		
Failure to initiate SPC and Containment Spray is #2 on list of Operator Action Importance to CDF.		

Examination Outline Cross Reference	Level	RO
295014 Inadvertent Reactivity Addition AA1 Ability to operate and/or monitor the following as they apply to INADVERTENT REACTIVITY ADDITION: AA1.06 Reactor/turbine pressure regulating system	Tier	1
	Group #	2
	K/A	295014
	Rating	3.3
	Revision	1
Revision Statement: **Changed LOD to 3**		

Question: 24

A plant power ascension is in progress with current reactor power at 20% with the main Turbine/Generator synchronized to the grid.

A feedwater heating problem causes an <u>inadvertent</u> reactor power increase.

The plant is now stable, but the Bypass Valves are slightly open as a result of the power excursion.

(1) Per the Precautions/Limitations of IOI-2 (Power Operations), control room operators must ensure the Bypass Valves are re-closed prior to withdrawing control rods to above _____ reactor power.

(2) The RO will use the ________________ pushbutton to close the Bypass Valves.

A. (1) 35%
(2) LOAD DEMAND RAISE

B. (1) 25%
(2) LOAD DEMAND LOWER

C. (1) 25%
(2) LOAD DEMAND RAISE

D. (1) 35%
(2) LOAD DEMAND LOWER

Answer: A
Explanation: Depressing the Load Demand RAISE pushbutton results in a throttling open of the TCVs (in an effort to pick up more generator load) with a complimentary throttling <u>closed</u> of the BPVs.

03-1-01-2, Power Operations, P/L 2.15 prohibits control rod withdrawal above 35% with open Bypass Valves.

Distracters:

B & D are wrong. Lowering the Load Demand setpoint will result in a throttling close of the Turbine Control Valves and a complimentary throttling open of the Bypass Valves, as well. Their plausibility is based on the Applicant's need to understand this TCV/BPV relationship with the Load Demand setpoint.

C is wrong because the power level is 35% not 25% as suggested in this answer choice. Plausibility of this 25% value is based on the applicant's recall that another P&L states that sustained power operation at 21 to 25 % power induces high vibrations on the Main Steam Lines and attachment V is "Shutdown by Scram from 25-30% Reactor Power."

K/A Match

This question requires the applicant to have the ability to monitor the Reactor/turbine pressure regulating system during an inadvertent reactivity addition

Technical References:

03-1-01-2, Power Operations IOI, Rev. 170

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-N3202, Objective 13

Question Source:	**Bank # 359**	2011 NRC Exam Q# 52
(note changes and 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	
PRA Applicability:		

Examination Outline Cross Reference	Level	RO
295020 Inadvertent Cont. Isolation AA2 Ability to determine and/or interpret the following as they apply to INADVERTENT CONTAINMENT ISOLATION: AA2.03 Reactor power	Tier	1
	Group #	2
	K/A	295020
	Rating	3.7
	Revision	**2**

Revision Statement:
Rev 1 Per NRC; Re-ordered answers short to long, added rated thermal power, changed answers to a 2X2, added bases for answers from simulator operation, added titles and rev numbers to references

Rev 2 Created two line answers

Question: 25

Plant is operating at rated thermal power.

An inadvertent containment isolation has occurred due to loss of power to Bus 15AA.

With no operator action, which of the following describes the effect on Reactor power?

A. Reactor power will remain stable for approximately 5 to 10 minutes;
then power will begin to lower.

B. Reactor power will remain stable for approximately 5 to 10 minutes;
then power will rapidly go to zero.

C. After approximately 2 minutes, Reactor power will begin to lower;
after approximately 5 to 10 minutes, power will rapidly go to zero.

D. After approximately 2 minutes, Reactor power will begin to lower;
after approximately 5 to 10 minutes, power will stabilize at a lower level.

Answer: C

Explanation:

The loss of Bus 15AA results in an immediate inadvertent isolation of instrument air to containment since P53-F001, Instrument Air Containment isolation is an AOV that will fail closed on a loss of power to its solenoid. The stem states no operator action, so if this valve remains closed for approximately 2.5 minutes, control rods will drift in when scram valves open on low Instrument Air pressure resulting in a

lowering of power. Plant data and simulator operation, after approximately 2.5 minutes the control rods will begin to drift IN. After 5 to 10 minutes MSIVs will begin to close on loss of Instrument Air and a reactor scram will initiate on MSIV closure scram signal. Plant data and simulator operation, after approximately 8 minutes the Inboard MSIVs will close on loss of instrument air.
Distracters: A is wrong, but plausible. Reactor power will begin to lower not remain stable when control rods begin to drift in and/or individually scram. B is wrong, but plausible. Reactor power will begin to lower not remain stable when control rods begin to drift in and/or individually scram. D is wrong, but plausible. First part is correct, however, power will rapidly go to zero not stabilize due to reactor scram.
K/A Match This question requires the applicant to have the knowledge to monitor reactor power during an inadvertent containment isolation.
Technical References: 05-1-02-V-9, Loss of Instrument Air ONEP, Rev. 45 05-1-02-III-5, Automatic Isolations ONEP, Rev. 49 GLP-OPS-M7100, Containment And Drywell Instrumentation And Control System Lesson Plan, Rev. 14
Handouts to be provided to the Applicants during exam: NONE
Learning Objective: GLP-OPS-M7100, OBJ. 8

Question Source:	**Bank #**	
(note changes and 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:	3	
PRA Applicability:		
Failure to reinstate IA to containment following isolation is #8 on list of Operator Action Importance to CDF.		

Examination Outline Cross Reference	Level	RO
295029 High Suppression Pool Wtr Lvl 2.4.4 Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures.	Tier	1
	Group #	2
	K/A	295029
	Rating	4.5
	Revision	1
Revision Statement: **Editorial changes in stem and answers**		

Question: 26

Which of the following corresponds to an EP entry condition?

A. HPCS in surveillance test (return to suppression pool) with suction pressure lower than normal

B. RCIC in **normal** standby lineup with pump suction pressure higher than normal

C. RHR PMP A DISCH PRESS ABNORMAL alarm

D. RCIC TURB EXH LINE DRAIN TROUBLE alarm

Answer: **A**

Explanation:

When HPCS system is run for surveillance testing it must be aligned to the Suppression Pool and test returned to the suppression pool. If suction pressure is abnormally low could be an indication of low suppression pool level and possible entry into EP-3 due to low suppression pool level. Added also that there is no low suppression pool level alarm for the HPCS system.

Distracters:

B is wrong, but plausible. RCIC normal lineup is to the CST not the suppression pool, therefore this would indicate a high level in the CST and no EP concerns.

C is wrong, but plausible. This alarm indicates a jockey pump problem or high system pressure, nothing to do with the EPs.

D is wrong, but plausible. This alarm indicates a drain trap level valve is either open too long due to excessive condensation, there are no EP concerns with this alarm.

K/A Match

This question requires the applicant to have the ability to recognize abnormal system indications that could possibly be entry into EOPs.

Technical References:

04-1-01-E22-1, High Pressure Core Spray SOI, Rev. 123

06-OP-1E22-Q-0005, HPCS Quarterly Functional Test, Rev. 124

04-1-01-E51-1, Reactor Core Isolation Cooling SOI, Rev. 136

04-1-02-1H13-P601-21A-G5, RCIC TURB EXH LINE DRAIN TROUBLE, Alarm Response Instruction, Rev. 144

04-1-02-1H13-P601-20A-C4, RHR PMP A DISCH PRESS ABNORMAL, Alarm Response Instruction, Rev. 102

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-EP3, OBJ. 5

Question Source:	**Bank #**	
(note changes and 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:	3	

PRA Applicability:

HPCS is #16 on list of System Importance to CDF.

Examination Outline Cross Reference	Level	RO
295033 High Secondary Containment Area Radiation Levels EK2 Knowledge of the interrelations between HIGH SECONDARY CONTAINMENT AREA RADIATION LEVELS and the following: EK2.01 Area radiation monitoring system	Tier	1
	Group #	2
	K/A	295033
	Rating	3.8
	Revision	1
Revision Statement: **Editorial changes in stem and answers**		

Question: 27

A steam leak has occurred on RCIC near the E51-F045 Steam Supply Valve.

- Steam supply isolation valves failed to close
- RCIC area radiation monitor reading above MAX SAFE value
- RCIC room temperature above MAX SAFE value
- CRS orders Mode Switch to SHUTDOWN

Which of the following secondary containment area radiation monitors exceeding a Max Safe Value will cause the CRS to order an Emergency Depressurization?

A. CCW HX

B. CTMT PERS HATCH

C. SGTS FLTR TRAIN A

D. SPENT FUEL STOR POOL

Answer: C

Explanation:

Per EP-4, "If two or more area temperatures, radiation levels, or water levels are above max safe values" with a non-isolable leak from the RPV discharging outside the primary containment, then Emergency depressurize.

Only SGTS above the Max Safe as shown on Table SC-1. Max safe value is 8 X 10^2 mr.

The CRS will direct one RO to monitor EP-4 parameters; the RO should be able to recognize when a Max Safe parameter has been exceeded and which ARM should be monitored.

Distracters: All distractors are plausible as they refer to ARMs in the Secondary Containment that could be in alarm due to the initiating event, but are not listed in EP-4 table SC-1.
K/A Match This question requires the applicant to have the knowledge of the interrelationship between high radiation and the Area Radiation Monitoring system.
Technical References: 05-S-01-EP4, Auxiliary Building Control, Rev. 29
Handouts to be provided to the Applicants during exam: NONE
Learning Objective: GLP-OPS-EP4, OBJ. 5

Question Source:	**Bank #**	
(note changes and 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:	3	
PRA Applicability:		
RCIC is #17 on list of System Importance to CDF.		

Examination Outline Cross Reference	Level	RO
203000 RHR/LPCI: Injection Mode K4 Knowledge of RHR/LPCI: INJECTION MODE (PLANT SPECIFIC) design feature(s) and/or interlocks which provide for the following: K4.07 Emergency generator load sequencing	Tier	2
	Group #	1
	K/A	203000
	Rating	3.7
	Revision	0
Revision Statement:		

Question: 28

A Loss of Offsite Power with a LOCA has occurred.

All Emergency Diesel Generators start and restore power to respective buses.

Which of the following describes the correct sequencing for all of the RHR/LPCI pumps?

A. RHR ‘A’ - 0 seconds
RHR ‘B’ - 5 seconds
RHR ‘C’ - 5 seconds

B. RHR ‘A’ - 5 seconds
RHR ‘B’ - 5 seconds
RHR ‘C’ - 0 seconds

C. RHR ‘A’ - 0 seconds
RHR ‘B’ - 0 seconds
RHR ‘C’ - 5 seconds

D. RHR ‘A’ - 5 seconds
RHR ‘B’ - 5 seconds
RHR ‘C’ - 5 seconds

Answer: **B**

Explanation:

Per 04-1-01-R21-1, Load Sheading and Sequencing SOI, Sequencing times in seconds for the following:

- RHR ‘A’ - 5 seconds
- RHR ‘B’ - 5 seconds
- RHR ‘C’ - 0 seconds

Distracters:

All distractors are combinations of incorrect sequencing times and all are plausible.

K/A Match

This question requires the applicant to have the knowledge of the sequencing times for the RHR/LPCI pumps.

Technical References:

04-1-01-R21-1, Load Sheading and Sequencing SOI, Rev. 106

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-R2100, OBJ. 17

Question Source:	**Bank #**	
(note changes and 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:	2	

PRA Applicability:

RHR is #3 on list of System Importance to CDF.

ESF (R20) is #4 on list of System Importance to CDF.

R21 ESF is #8 on list of System Importance to CDF.

Div. 1 & 2 EDGs is #11 on list of System Importance to CDF.

Examination Outline Cross Reference	Level	RO
203000 RHR/LPCI: Injection Mode 2.4.31 Knowledge of annunciator alarms, indications, or response procedures.	Tier	2
	Group #	1
	K/A	203000
	Rating	4.2
	Revision	3

Revision Statement:

Rev 1 & 2: Editorial changes in stem and answers

Rev 3: Revised question to reflect single action (not actions), editorial changes.

Question: 29

A LOCA has occurred.

CRS directs an Emergency Depressurization on Reactor water level.

Following are alarms received on 1H13-P601:

- LPCS/LPCI A INJ VLV RPV PRESS LO
- LPCI B/C INJ VLV RPV PRESS LO

Which of the following indicates the **setpoint** and action associated with these alarms?

A. ≤ 450 psig Reactor pressure; Verify immediate RHR injection flow into the vessel.

B. ≤ 476 psig Reactor pressure; Verify immediate RHR injection flow into the vessel.

C. ≤ 450 psig Reactor pressure; Verify RHR injection valves auto open.

D. ≤ 476 psig Reactor pressure; Verify RHR injection valves auto open.

Answer: D

Explanation:

Per 04-1-02-1H13-P601-17A-C3 and 21A-F7:

- Possible Causes - RPV pressure ≤ 476 psig
- Automatic Actions - If a valid LOCA signal is present, valves E12-F042A, B, & C will auto open

However, RHR pumps will not inject at > 250 psig, but the injection valves will auto open.

Distracters:

A is wrong, but plausible. 450 psig is the setpoint for manual opening of the E12-F042A, B, & C, no alarm is associated with this pressure for all systems except for RHR ‘C’. This pressure is also from the RHR piping, not Reactor pressure. No flow will enter the RPV due to discharge head of the pump is lower than this pressure. Injection will begin around 250 psig.

B. is wrong, but plausible. This pressure will cause the injection valve to open, however no flow will enter the RPV due to discharge head of the pump is lower than this pressure. Injection will begin around 250 psig.

C is wrong, but plausible. 450 psig is the setpoint for manual opening of the E12 F042A, B, & C, no alarm is associated with this pressure for ‘A’ and ‘B’.

K/A Match

This question requires the applicant to have the knowledge of the LPCI/RHR system alarms and indications.

Technical References:

04-1-02-1H13-P601-17A-C3, Rev. 39

04-1-02-1H13-P601-21A-F7, Rev. 38

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-E1200, OBJ. 8.9 & 15

Question Source:	**Bank #**	
(note changes and 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	

PRA Applicability:
RHR is #3 on list of System Importance to CDF.

Examination Outline Cross Reference	Level	RO
205000 Shutdown Cooling A4 Ability to manually operate and/or monitor in the control room: A4.01 SDC/RHR pumps	Tier	2
	Group #	1
	K/A	205000
	Rating	3.7
	Revision	2
Revision Statement: **Rev 1: Editorial changes in stem and answers** **Rev 2: Changed format of question to match other (1) (2) questions.**		

Question: 30

Plant in Mode 3, cooldown in progress.

Crew is starting RHR 'A' in **Shutdown Cooling**.

When RHR 'A' pump is started, discharge flow transmitter fails to zero gpm.

(1) Which of the following describes the result of the failed component?

And

(2) What is the action required?

A. (1) RHR 'A' Min Flow valve E12-F064A will auto open in 8 seconds.
(2) Trip RHR 'A' pump and isolate suction flow path from the RPV.

B. (1) RHR 'A' pump will trip in 8 seconds.
(2) Open discharge flowpath fully.

C. (1) RHR 'A' Min Flow valve E12-F064A will auto open in 8 seconds.
(2) Open discharge flowpath fully.

D. (1) RHR 'A' pump will trip in 8 seconds.
(2) Isolate suction flow path from the RPV.

Answer:	**A**
Explanation: Per 04-1-01-E12-2, Shutdown Cooling and Alternate Decay Heat Removal Operation SOI, Caution, If greater than 1154 gpm is not established within 8 seconds of pump start, Then F064A, RHR A MIN FLO TO SUPP POOL automatically opens, establishing flow path from Reactor to Suppression Pool.	

Knowing that with the flow indication failed low, the min flow valve will open and maintain flowpath from the RPV to the suppression pool.

Tripping the pump should be the first action, however, isolating the suction flow path is required because even if the pump is secured, flow will continue due to differential pressure between the RPV and the suppression pool.

Distracters:

B is wrong, but plausible. Pump will not auto trip and the manual actions should be stop draining the RPV.

C is wrong, but plausible. First part is correct however, manual actions should be stop draining the RPV.

D is wrong, but plausible. Pump will not auto trip.

K/A Match

This question requires the applicant to have the knowledge of the ability to operate or monitor the RHR shutdown cooling system.

Technical References:

04-1-01-E12-2, Rev. 123, step 4.1.2c (18) CAUTION.

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-E1200, OBJ. 8.2 & 14

Question Source:	**Bank #**	
(note changes and 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	

PRA Applicability:

RHR is #3 on list of System Importance to CDF.

Examination Outline Cross Reference	Level	RO
209001 LPCS K5 Knowledge of the operational implications of the following concepts as they apply to LOW PRESSURE CORE SPRAY SYSTEM : K5.01 Indications of pump cavitation	Tier	2
	Group #	1
	K/A	209001
	Rating	2.6
	Revision	1
Revision Statement: **Editorial changes in stem and answers** **Re-Ordered answers short to long**		

Question: 31

With LPCS injecting to the reactor, what indications are available in the Control Room that a valve disk/stem separation has occurred on E21-F001, LPCS Suction from the Suppression Pool?

A. LPCS SYS OOSVC annunciator is in alarm.

B. E21-R600, LPCS PMP DISCH FLO indicator, is oscillating.

C. E21-R600, LPCS PMP DISCH FLO indicator, is full upscale.

D. LPCS/LPCI A INJ VLV RPV PRESS LO annunciator is in alarm.

Answer: **B**

Explanation:

A valve disk/stem separation will cause the disk to drift shut. This will cause pump cavitation. The result will be observed by oscillating discharge pressure and flow. Discharge pressure will not continuously be 0 gpm since water will be forced out of the pump then drain back into the pump to be forced out again. This results in pressure and flow oscillations. The MOV will not overload in this condition or lose power.

Distracters:

A is wrong, but plausible. No reason for the out of service alarm to come for this event.

C is wrong, but plausible. as stated in the explanation

D is wrong, but plausible. RPV Press Lo annunciator comes in when RPV pressure is below 476 psig and should already be in.

K/A Match

This question requires the applicant to have the knowledge of indication of ECCS pump cavation.

Technical References: GLP-OPS-COM02, Pumps, Rev 1 GSMS-RO-EP047, Simulator Lesson Plan, Rev. 9 GLP-OPS-E2100, Low Pressure Core Spray Lesson Plan, Rev. 13
Handouts to be provided to the Applicants during exam: NONE
Learning Objective: GLP-OPS-COM02, OBJ. 3

Question Source:	**Bank # 65**	X
(note changes and attach parent)	**Modified Bank #**	
	New	
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.41(b)(14)	
Level of Difficulty:	2	
PRA Applicability:		

Examination Outline Cross Reference	Level	RO
209001 LPCS A3 Ability to monitor automatic operations of the LOW PRESSURE CORE SPRAY SYSTEM including: A3.06 Lights and alarms	Tier	2
	Group #	1
	K/A	209001
	Rating	3.6
	Revision	1
Revision Statement: **Editorial changes in stem and answers** **Removed third part (#3) of stem and third part of all answer.**		

Question: 32

A LOCA has occurred.

Division 1 ECCS has failed to initiate due to Drywell pressure trip units in GROSS FAIL.

RO manually initiates Division 1 ECCS by arming and depressing LPCS/RHR A MAN INIT pushbutton.

What is the status of the following LPCS indications?

1. White status light above LPCS/RHR A INIT RESET pushbutton
2. Alarm LPCS SYS ACTUATED

A. 1 – OFF
2 – ON

B. 1 – OFF
2 – OFF

C. 1 – ON
2 – OFF

D. 1 – ON
2 – ON

Answer: D
Explanation: When MANUALLY initiated the following alarms are received:

- LPCS SYS ACTUATED

LPCS SYS ACTUATED is received due to an initiation signal either auto or manual

White status light above the LPCS/RHR A INIT RESET pushbutton is ON, regardless of start signal (Auto or arm and depress pushbutton man initiate)

Distracters:

A is wrong – White status light above the LPCS/RHR A INIT RESET pushbutton will illuminate anytime an auto start or MAN initiate pushbutton is depressed. Plausible if the applicant believes that an arm and depress MAN initiate will not illuminate the white light.

B is wrong - White status light above the LPCS/RHR A INIT RESET pushbutton and LPCS SYS ACTUATED alarm will illuminate anytime an auto start or MAN initiate pushbutton is depressed. Plausible if the applicant believes that an arm and depress MAN initiate will not illuminate the white light or the Actuated alarm.

C is wrong – LPCS SYS ACTUATED alarm will illuminate anytime an auto start or MAN initiate pushbutton is depressed. Plausible if the applicant believes that an arm and depress MAN initiate will not illuminate the Actuated alarm.

K/A Match

This question requires the applicant to have the knowledge of verifying the indications of proper ECCS initiation even with failed auto start logic.

Technical References:

04-1-02-1H13-P601-21A-B8, Rev. 33, LPCS SYS ACTUATED

04-1-02-1H13-P601-21A-E7, Rev. 110, DRWL PRESS HI

04-1-02-1H13-P601-21A-H8, Rev. 141, LPCS SYS OOSVC

GLP-OPS-E2100, Rev. 13, Low Pressure Core Spray System Lesson Plan

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-E2100, OBJ. 9 & 12

Question Source:	**Bank #**	
(note changes and 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	
PRA Applicability:		

Examination Outline Cross Reference	Level	RO
209002 HPCS A2 Ability to (a) predict the impacts of the following on the HIGH PRESSURE CORE SPRAY SYSTEM (HPCS) ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: A2.13 Low condensate storage tank level BWR-5,6	**Tier**	2
	Group #	1
	K/A	209001
	Rating	3.6
	Revision	2

Revision Statement:

Rev. 1 Editorial changes in stem and answers

Rev.2 Changed the order of #2 question.
Deleted reference to HPCS in #2 question

Question: 33

HPCS is injecting due to a loss of feedwater.

CST LVL LO alarm occurs on 1H13-P601.

(1) Which of the following describes the effects on the HPCS system?

(2) IAW 04-1-01-E22-1, High Pressure Core Spray System, which valve must be operated first to restore CST suction after CST level is restored?

A. (1) Will continue to inject and suction will auto swap
(2) E22-F015, HPCS PMP SUCT FR SUPP POOL

B. (1) Will trip and then suction will auto swap
(2) E22-F001, HPCS PMP SUCT FR CST

C. (1) Will continue to inject and suction will auto swap
(2) E22-F001, HPCS PMP SUCT FR CST

D. (1) Will trip and then suction will auto swap
(2) E22-F015, HPCS PMP SUCT FR SUPP POOL

Answer:	A
Explanation:	
When HPCS is operating and a low CST level is received, an auto suction swap will occur without any stoppage of HPCS injection.	

Per HPCS SOI, 04-1-01-E22-1, section 5.4, after CST level is restored the E22-F015 (Supp Pool suction) must be closed first due to interlock on the E22-F001 (CST suction). The F001 will not open if the F015 is full open. Procedure states to close E22-F015 first, when E22-F015 has left FULL OPEN position, Then open E22-F001.

Distracters:

B and D are wrong, but plausible. The HPCS pump will not trip like the RHR pump would if no suction path is available.

C is wrong, but plausible. The E22-F001 can not be opened first it must see the Supp Pool suction valve "NOT FULL OPEN" before it will close.

K/A Match

This question requires the applicant to have the ability to predict the actions on a CST level low and using procedures correct the actions to restore.

Technical References:

04-1-02-1H13-P601-16A-C4, Rev. 151, CST LVL LO

04-1-01-E22-1, Rev. 123, High Pressure Core Spray System SOI, section 5.4

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-E2200, OBJ. 9

Question Source:	**Bank #**	
(note changes and 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	

PRA Applicability:

HPCS is #16 on list of System Importance to CDF.

Examination Outline Cross Reference	Level	RO
211000 SLC K1 Knowledge of the physical connections and/or cause-effect relationships between STANDBY LIQUID CONTROL SYSTEM and the following: K1.09 Core spray system: Plant-Specific	Tier	2
	Group #	1
	K/A	209001
	Rating	3.2
	Revision	0
Revision Statement:		

Question: 34

The Standby Liquid Control system injects into which of the following system's discharge piping for RPV injection?

A. HPCS

B. LPCS

C. RCIC

D. RHR C

Answer: A
Explanation: The SLC system uses the HPCS injection line between the testable check and the manual isolation valve.
Distracters: B is wrong – LPCS is not used but plausible due to it's the other spray system, if an applicant only knows that an ECCS spray system is used. C is wrong – RCIC system is not used but plausible due to the RCIC system can be used during an ATWS per EP Attachment 28 to inject Boron. D is wrong – RHR C is not used but plausible due to the ADHR system uses RHR C to return to the vessel.
K/A Match This question requires the applicant to have the knowledge of system interrelationships between SLC and HPCS.
Technical References:

04-1-01-E22-1, Rev. 123,High Pressure Core Spray System SOI

04-1-01-C41-1, Rev. 123, Standby Liquid Control System SOI

GLP-OPS-C4100, Rev. 15, Standby Liquid Control System Lesson Plan

M1082, Rev. 35, High Pressure Core Spray System P&ID

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-E2200, OBJ. 13.9

Question Source:	**Bank #**	
(note changes and 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	

PRA Applicability:

HPCS system is #16 on System Importance to CDF.

Examination Outline Cross Reference	Level	RO
211000 SLC A1 Ability to predict and/or monitor changes in parameters associated with operating the STANDBY LIQUID CONTROL SYSTEM controls including: A1.04 Valve operations	Tier	2
	Group #	1
	K/A	209001
	Rating	3.6
	Revision	2

Revision Statement:

Rev 1: Changed stem to fill in the blank with 2 parts
Changed answers to 2X2 format
Changed 10CFR to 6

Rev 2: changed LOD to 2

Question: 35

Following an initiation of SLC, both squib valves fail to open.

SLC storage tank level is ____(1)______ and SLC pump discharge pressure indicates _____(2)______.

A. (1) constant
(2) 1100 to 1200 psig

B. (1) constant
(2) 1500 to 1700 psig

C. (1) decreasing
(2) 1100 to 1200 psig

D. (1) decreasing
(2) 1500 to 1700 psig

Answer: B

Explanation:

When the SLC system is initiated and all components perform their intended function the discharge pressure on the P601 panel will indicate about 200 psig above reactor pressure for two pumps and about 100 psig above reactor pressure for one pump or approximately 1200 psig. If the squib valves fail (last valve in the system prior to the reactor) discharge pressure will go very high and both pumps will go on min flow/discharge relief valve flow. Discharge pressure will indicate about 1700 psig, which is about 700 psig above reactor pressure and also the relief setpoint for the system relief valve. No fluid will be lost so tank level will remain constant. Per 04-1-01-C41-1, SLC SOI, attachment VI, Verification of SLC Injection it states to verity observe SBLC pump discharge pressure exceeds reactor pressure, SBLC tank level lowering and nuclear instrumentation lowering. The applicant must be able to understand the system lineup and what is required for proper system response.

Distracters:

A is wrong – Plausible because if both squib valves failed tank level would remain the same due to pump min flow returns to the suction and the discharge relief returns to the suction and no movement of fluid will occur from the tank. SLC discharge pressure would only be 1200 psig if all worked as designed.

C is wrong – Tank level will not go down, but plausible if one squib valve fired.

D is wrong – Tank level will not go down, but plausible if one squib valve fired. Discharge pressure is too high for correct operation of the system. With no flow path discharge pressure would be at maximum for a positive displacement pump.

K/A Match

This question requires the applicant to have the ability to determine/predict if the SLC system is performing as designed and to recognize the indications of a failed valve.

Technical References:

04-1-01-C41-1, Rev. 123, Standby Liquid Control System SOI

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-C4100, OBJ. 12

Question Source:	**Bank #**	
(note changes and 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:	2	
PRA Applicability:		

Examination Outline Cross Reference	Level	RO
212000 RPS K6 Knowledge of the effect that a loss or malfunction of the following will have on the REACTOR PROTECTION SYSTEM : K6.01 A.C. electrical distribution	Tier	2
	Group #	1
	K/A	209001
	Rating	3.6
	Revision	2

Revision Statement:

Rev 1: Editorial changes in stem and answers
Removed reference to EPA breakers and third part of answers

Rev 2: Removed why the transformer tripped.

Question: 36

Plant operating at rated power.

Service Transformer 21 trips.

Which of the following describes status of power supplies to the RPS Motor Generators?

A. RPS A M/G power available
RPS B M/G power unavailable

B. RPS A M/G power unavailable
RPS B M/G power available

C. RPS A M/G power unavailable
RPS B M/G power unavailable

D. RPS A M/G power available
RPS B M/G power available

Answer: **A**

Explanation:

A loss of ST-21 will cause a loss of 14AE when aligned to preferred source, which, in turn supplies power to 14B22 MCC then to RPS M/G ‘B’.

RPS Motor Gen. ‘A’ remains unaffected due to being powered from ST-11, via 13B22 MCC

Distracters: B is wrong because RPS A M/G did not lose power, it is powered from ST-11 and B is de-energized. C is wrong because RPS A M/G did not lose power, it is powered from ST-11 D is wrong because RPS B M/G is de-energized. All distracters are plausible is the applicant does not remember normal power supply of RPS motor generators.		
K/A Match This question requires the applicant to have the knowledge of normal power supply to the RPS M/G sets and what result a loss of Service Transformer 21 would have on them.		
Technical References: 04-1-01-C71-1, Rev. 36, Reactor Protection System SOI 04-1-01-R21-13, Rev. 32, BOP Bus 13AD 04-1-01-R21-14, Rev. 26, BOP Bus 14AE		
Handouts to be provided to the Applicants during exam: NONE		
Learning Objective: GLP-OPS-7100, OBJ. 5.1 & 11.4		
Question Source:	**Bank # 880**	2008 NRC Q# 19
(note changes and attach parent)	**Modified Bank #**	
	New	
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(7)	
Level of Difficulty:	3	
PRA Applicability:		
RPS is #5 on list of System Importance to CDF.		

Examination Outline Cross Reference	Level	RO
215003 IRM K5 Knowledge of the operational implications of the following concepts as they apply to INTERMEDIATE RANGE MONITOR (IRM) SYSTEM : K5.03 Changing detector position	Tier	2
	Group #	1
	K/A	215003
	Rating	3.0
	Revision	1
Revision Statement: **Changed to Memory/Fundamental**		

Question: 37

A reactor scram has occurred.

ATC is performing subsequent actions of 05-1-02-I-1, Reactor Scram ONEP.

Scram has been reset.

Per step 3.7.2 of this procedure, which IRMs are inserted and why?

A. Only one IRM detector at a time
   To ensure that all areas of the core are monitored

B. Only one IRM detector at a time
   Prevent inadvertent scram signal

C. Only one division of IRM detectors at a time
   Prevent inadvertent scram signal

D. Only one division of IRM detectors at a time
   To ensure that all areas of the core are monitored

Answer: C
Explanation: 05-1-02-I-1, Reactor Scram ONEP states, “Insert IRMs as follows: If Reactor scram signal has not been reset then insert all IRMs. If Reactor scram signal has been reset then insert one RPS division of IRMs, when the first RPS division of IRMs are fully inserted then insert the second RPS division of IRMs.” 04-1-01-C51-1 P&L 3.7 and 4.2.2 CAUTION states: “Due to possible converter failures which May cause a half scram.”
Distracters:

A is wrong, but plausible. per step 3.7.2 of Reactor Scram ONEP one division is inserted not one at a time. The second part is also incorrect because its to prevent a scram not to ensure all areas of the core are monitored. B is wrong, but plausible. per step 3.7.2 of Reactor Scram ONEP one division is inserted not one at a time. D is wrong, but plausible. The second part is incorrect because it's to prevent a scram not to ensure all areas of the core are monitored. .
K/A Match This question requires the applicant to have the knowledge of operational implications for moving IRM detectors.
Technical References: 05-1-02-I-1, Rev 130, Reactor Scram ONEP 04-1-01-C51-1, Rev 29, Neutron Monitoring SOI
Handouts to be provided to the Applicants during exam: NONE
Learning Objective: GLP-OPS-C5100, GLP-OPS-IOI

Question Source:	**Bank #**	5/2017 NRC Q# 36
(note changes and 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	
PRA Applicability:		

Examination Outline Cross Reference	Level	RO
215004 Source Range Monitor Knowledge of SOURCE RANGE MONITOR (SRM) SYSTEM design feature(s) and/or interlocks which provide for the following: K4.01 Rod withdrawal blocks	Tier	2
	Group #	1
	K/A	215004
	Rating	3.7
	Revision	1
Revision Statement: **Changed answer B from "anytime" TO "in all conditions"** **Defined SRM/IRM overlap in Explanation** **Changed LOD to 3**		

Question: 38

Which of the following will cause a Control Rod Withdrawal Block?

A. Any SRM indicating > 2.5 X 10^5 with all IRMs on range 9

B. Any SRM Channel Mode switch out of OPERATE in all conditions

C. Any SRM period meter indicating < 50 seconds with all IRMs on range 3 or below

D. Any SRM indicating 100 cps and detector not full in during IRM/SRM overlap verification

Answer: D

Explanation:

SRM Low Flux and Detector Position - A rod block signal is generated if the SRM count rate is < 106 cps and the detector is not fully inserted.

This trip indicates that the SRM detector has been excessively withdrawn.

The SRM Low Flux and Detector Position rod block is bypassed under any of the following conditions:

- IRM Range Switches are $\geq$ range 3
- Reactor Mode Switch is in RUN
- SRM joystick is in BYPASS
- Detector fully inserted

IRM/SRM overlap occurs at range 1 of IRMs during a startup.

Distracters:

A is wrong, but plausible. being above range 8 on IRMs this rod block is bypassed.

B is wrong, but plausible. When the mode switch is placed in RUN all blocks are bypassed from the

SRMs.

C is wrong, but plausible. There are no rod blocks from the period indication, only administrative requirement to insert rods to move period to > 50 seconds.

K/A Match

This question requires the applicant to have the knowledge actions of a SRM Rod block.

Technical References:

GLP-OPS-C5101, Rev 8, Source Range Monitoring (SRM) System Lesson Plan

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-C5101, Objective 7.3

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.41(b)(2)	
Level of Difficulty:	3	
PRA Applicability:		

Examination Outline Cross Reference	Level	RO
215005 APRM / LPRM K3 Knowledge of the effect that a loss or malfunction of the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM will have on following: K3.05 Reactor power indication	Tier	2
	Group #	1
	K/A	215005
	Rating	3.8
	Revision	1
Revision Statement: **Editorial changes in stem**		

Question: 39

The plant is at rated thermal power.

The LPRM Chassis for APRM Channel 2 has malfunctioned causing all 22 LPRMs to send a zero output signal to APRM.

What is the current power indication for APRM Channel 2?

Approximately:

A. 100%

B. 75%

C. 50%

D. 0%

Answer: C
Explanation: The LPRM chassis supports the APRM channel by gathering half the LPRM detector signals for the APRM channel and managing some of the channel communication between the APRM and PCI channels. For each APRM channel 1 through 4, an averaging circuit produces an output signal that is the average of all the valid LPRM inputs for that specific APRM channel. Each APRM Chassis can only receive inputs from 22 LPRMs. Therefore, an LPRM Chassis is used to receive inputs from the other 22 LPRMs assigned to that APRM channel, providing a total of 44 LPRM inputs to each APRM Channel.

Half of the inputs have now gone to zero therefore the average is reduced by 50% and the APRM power indication is approximately 50%.

Distracters:

A is wrong, Power indication will lower not, plausible if the applicant believes that LPRMs are auto bypassed if out of tolerance.

B is wrong, Power will lower due to half of the inputs have gone to zero, this given failure will not cause an auto bypass of LPRM inputs, plausible if the applicant believes that the inputs will be auto bypassed.

D is wrong, Power will lower by approximately 50% not to 0%, plausible if the applicant does not remember the total number of LPRMs feeding an ARPM.

K/A Match

This question requires the applicant to have the knowledge of how power is determined by the APRM/LPRM system.

Technical References:

GLP-OPS-C5104, Rev 9, Average Power Range Monitoring (APRM) System Lesson Plan

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-C5104, Objective 3.1 & 11.1

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(2)	
Level of Difficulty:	3	
PRA Applicability:		

Examination Outline Cross Reference	Level	RO
217000 RCIC Knowledge of electrical power supplies to the following: K2.02 RCIC initiation signals (logic)	Tier	2
	Group #	1
	K/A	217000
	Rating	2.8
	Revision	1

Revision Statement:

Changed LOD to 3

Question: 40

Which of the following provides power to the RCIC initiation logic?

A. 120 VAC UPS Div 1

B. 120 VAC ESF Div 1

C. 125 VDC 11DA

D. 125 VDC 11DB

Answer: **C**

Explanation:

Per GLP-OPS-E5100 Page 45 of 66 “The RCIC System Initiation Logic is powered from 125 VDC Bus 11DA.”

Distracters:

A is wrong, Plausible due to this system does provide isolation logic power just not RCIC.

B is wrong, Plausible due to being ESF powered AC.

D is wrong, Plausible because this is Div 2 DC power and it does supply power to Div 2 isolation logic for RCIC.

K/A Match

This question requires the applicant to have the knowledge of RCIC initiation logic.

Technical References:

GLP-OPS-E5100, Rev. 16, Reactor Core Isolation Cooling (RCIC) System Lesson Plan
Handouts to be provided to the Applicants during exam: NONE
Learning Objective: GLP-OPS-E5100, OBJ. 10.5

Question Source:	**Bank #**	
(note changes and 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	
PRA Applicability:		
RCIC is #17 on list of System Importance to CDF.		

Examination Outline Cross Reference	Level	RO
218000 ADS K1 Knowledge of the physical connections and/or cause-effect relationships between AUTOMATIC DEPRESSURIZATION SYSTEM and the following: K1.05 Remote shutdown system: Plant-Specific	Tier	2
	Group #	1
	K/A	218000
	Rating	3.9
	Revision	0
Revision Statement:		

Question: 41

How many **ADS** valves can be operated from the remote shutdown panels?

A. 2

B. 4

C. 6

D. 8

Answer: A

Explanation:

The Remote Shutdown Panels (Div 1 and Div 2) have controls for a total of 6 SRVs, 2 of which are designated as ADS valves.

Distracters:

B is wrong, Plausible because if the applicant believes that its 2 per panel then the answer would be 4, but it's a Div 1 and Div 2 handswitch for each valve.

C is wrong, Plausible due to there is a total of 6 SRVs that can be controlled from the RSD panels.

D is wrong, Plausible due to there is a total of 8 ADS valves out of the 20 SRVs but only 2 can be controlled from the RSD panels.

K/A Match

This question requires the applicant to have the knowledge of physical location of ADS handswitches on the RSD panels.

Technical References:

GLP-OPS-C6100, Rev. 13, Remote Shutdown Panels System Lesson Plan 05-1-02-II-1, Rev. 49, Shutdown From the Remote Shutdown Panel ONEP
Handouts to be provided to the Applicants during exam: NONE
Learning Objective: GLP-OPS-C6100, OBJ. 10.0

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.41(b)(3)	
Level of Difficulty:	2	
PRA Applicability:		
ADS is #6 on list of System Importance to CDF.		

Examination Outline Cross Reference	Level	RO
223002 PCIS/Nuclear Steam Supply Shutoff K3 Knowledge of the effect that a loss or malfunction of the PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF will have on following: K3.10 Reactor water cleanup	Tier	2
	Group #	1
	K/A	223002
	Rating	2.9
	Revision	2

Revision Statement:

Rev 1: Editorial changes in stem and answers
Added noun name of G33-F004

Rev 2: Underlined the word "only" in answer B.

Question: 42

RWCU is operating in the Post-pump mode.

Div 2 NUS temperature switch for RWCU Pump Room malfunctions and indicates full upscale, but the NUS INOP light does **NOT** come on.

What effect will this have on the RWCU system?

A. Group 8 Div 2 valves isolate and both RWCU pumps trip.

B. Group 8 Div 2 valves isolate and only B RWCU pump trips.

C. Group 8 valves isolate (both divisions) and both RWCU pumps trip.

D. ONLY G33-F004, RWCU PMP SUCT CTMT OTBD ISOL, isolates and no RWCU pump trip.

Answer: **A**

Explanation:

When a NUS temperature switch for Group 8 actuates depending on which division the NUS switch belongs to determines which valves will close.

Div 1 NUS activation will close G33-F004, F034, F039, F054, F250, F253

Div 2 NUS activation will close G33-F001, F028, F040, F053, F251, F252

Regardless of which division activates the RWCU system will isolate and cause both RWCU pumps to trip.

Distracters:

B is wrong, Both pumps will trip, plausible if the applicant believes that Div 2 isolation is tied to the B RWCU pump.

C is wrong, Only the Div 2 valves will close, plausible if the applicant doesn't understand the isolation logic for RWCU system.

D is wrong, Div 2 isolation will occur, plausible if the applicant believes that only one valve will close. The G33-F004 is the only valve that will close on a SLC 'A' initiation.

K/A Match

This question requires the applicant to have the knowledge of NS4 containment isolation logic and RWCU.

Technical References:

GLP-OPS-G3336, Rev. 17, Reactor Water Cleanup (RWCU) System Lesson Plan

GLP-OPS-M7100, Rev. 14, Containment And Drywell Instrumentation And Control System Lesson Plan

05-1-02-III-5, Rev. 49, Automatic Isolations ONEP

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-G3300, OBJ. 9.8 & 8.6
GLP-OPS-M7100, OBJ 10

Question Source:	**Bank #**	
(note changes and 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	
PRA Applicability:		

Examination Outline Cross Reference	Level	RO
239002 SRVs K2 Knowledge of electrical power supplies to the following: K2.01 SRV solenoids	Tier	2
	Group #	1
	K/A	239002
	Rating	2.8
	Revision	1
Revision Statement: **Changed answer order**		

Question: 43

What is the power supply to Division 2 solenoids of the SRVs?

A. 11DE

B. 11DK

C. 11DB

D. 11DC

Answer:	**C**
Explanation: SRV solenoids are powered from ESF 125 VDC buses 11DA for 'A' division 1 solenoid and 11DB for 'B' division 2 solenoid.	
Distracters: A is wrong, plausible due to this is 125 VDC bus that supplies power to Division 2 'B' solenoid power to ATWS / ARI valves in the CRD system. B is wrong, plausible due to this is 125 VDC bus that supplies power to Division 1 'A' solenoid power to ATWS / ARI valves in the CRD system. D is wrong, plausible due to 11DC is also an ESF 125 VDC bus.	
K/A Match This question requires the applicant to have the knowledge of power supply for the SRV solenoids.	
Technical References:	

04-1-01-B21-1, Rev. 52, Nuclear Boiler System SOI

E1161 SH 4, Rev. 12

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-E2202, OBJ. 9.1

Question Source:	**Bank # 745**	2013 NRC Q# 43
(note changes and attach parent)	**Modified Bank #**	
	New	
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.41(b)(3)	
Level of Difficulty:	2	

PRA Applicability:

125 VDC ESF is #7 on list of System Importance to CDF.

Examination Outline Cross Reference	Level	RO
259002 Reactor Water Level Control K4 Knowledge of REACTOR WATER LEVEL CONTROL SYSTEM design feature(s) and/or interlocks which provide for the following: K4.12 Manual and automatic control of the system	Tier	2
	Group #	1
	K/A	259002
	Rating	3.5
	Revision	2
Revision Statement: **Rev 1: Editorial changes in stem and answers** **Added “initially” in stem** **Replaced inch symbol with inches** **Rev 2: Added the word ‘Then’ to stem and deleted the word ‘if’ in (2) answers**		

Question: 44

Initially plant is operating at rated thermal power with the following:

- ‘A’ NR level indicates 36 inches
- ‘B’ NR level indicates 34 inches
- ‘C’ NR level indicates 38 inches
- Upset Range level indicates 35 inches

Then ‘A’ NR level instrument rapidly fails downscale (“hard” failure).

(1) What is the response of <u>actual</u> reactor water level?

(2) What would cause the level control system to auto transfer to MANUAL?

A. (1) Lowers then returns to the initial level
(2) All 4 level signals have a hard failure

B. (1) Lowers then returns to the initial level
(2) Upset Range now has a hard failure

C. (1) Rises then stabilizes at a level higher than initial level
(2) Upset Range now has a hard failure

D. (1) Rises then stabilizes at a level higher than initial level
(2) All 4 level signals have a hard failure

Answer: **D**		
Explanation: Since ‘A’ NR is the middle indication of the 3 (prior to the failure), it is controlling. When it fails, Digital Feedwater Control System (DFCS) throws it out and substitutes Upset in its place. Now, among ‘B’ NR, ‘C’ NR, and Upset, the new “middle” indication (with which DFCS will begin to control) is Upset, which is indicating 35” (as compared to the pre-failure indication from ‘A’ NR which was 36”). Since the Master Level Controller setpoint hasn’t changed (it’s still set at 36”), and DFCS now “sees” level at 35” (i.e., an inch lower than “desired”), it will increase RFPT speed to RAISE actual reactor level to the Master Level Controller 36” setpoint. Once at 36”, actual level will remain there. Per GLP-OPS-C3400, Feedwater Level control system, “If all four level channels “hard fail”, the level control stations will Automatically swap to MANUAL.		
Distracters: A is wrong, but plausible. Level doesn’t lower and if Upset fails along with one Narrow Range then the lesser of the two remaining Narrow range channels will be selected. B is wrong, but plausible. if Upset fails along with one Narrow Range then the lesser of the two remaining Narrow range channels will be selected. C is wrong, but plausible. Level doesn’t lower. Plausible if applicant does not remember the actions of a single failed instrument.		
K/A Match This question requires the applicant to have the knowledge of Feedwater control system failures.		
Technical References: GLP-OPS-C3400, Rev. 17, Digital Feedwater Control System Lesson Plan, Page 15 OF 57		
Handouts to be provided to the Applicants during exam: NONE		
Learning Objective: GLP-OPS-C3400, OBJ. 3.6		
Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #205**	X
	New	
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(7)	

Level of Difficulty:	3	
PRA Applicability:		

Examination Outline Cross Reference	Level	RO
261000 SGTS K6 Knowledge of the effect that a loss or malfunction of the following will have on the STANDBY GAS TREATMENT SYSTEM : K6.04 Process radiation monitoring	Tier	2
	Group #	1
	K/A	261000
	Rating	2.9
	Revision	1
Revision Statement: **Editorial changes in stem and answers**		

Question: 45

‘A’ FHA Rad Monitor is INOP with its function switch in STANDBY.

‘D’ FHA Rad Monitor spikes to indicate 4.5 mrem/hr.

Which of the following describes the status of the Standby Gas Treatment system?

A. Both SBGT trains are operating

B. ONLY ‘B’ SBGT train is operating

C. ONLY ‘A’ SBGT train is operating

D. Both SBGT trains have a 1/2 initiation signal

Answer: C
Explanation: Per 04-1-02-1H13-P601-19A-B9 and C9, the INOP part is caused by function switch not in operate and >3.6 mrem/hr is the setpoint for rad levels. With A and D detectors in this condition this will cause an initiation of SBGT A only. B and C detectors would cause an initiation of SBGT B.
Distracters: All distractors are combinations of SBGT trains. Plausible if the applicant does not remember SBGT start logic. ‘D’ would be correct if a combination of Rad monitors A and B or C and D
K/A Match This question requires the applicant to have the knowledge of auto start logic of the SBGT system from a

malfunction with the Radiation monitoring system.

Technical References:

04-1-01-T48-1, Rev. 36, Standby Gas Treatment System SOI

GLP-OPS-T4800, Rev. 10, Standby Gas Treatment System Lesson Plan

04-1-02-1H13-P601-19A-B9, Rev. 102, FH AREA EXH DIV 1, 4 RAD HI-HI/INOP Alarm Response

04-1-02-1H13-P601-19A-C9, Rev. 102, FH AREA EXH DIV 2, 3 RAD HI-HI/INOP Alarm Response

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-T4800, OBJ. 8.7 & 9.6

Question Source:	**Bank # 659**	X
(note changes and 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	
PRA Applicability:		

Examination Outline Cross Reference	Level	RO
262001 AC Electrical Distribution K3 Knowledge of the effect that a loss or malfunction of the A.C. ELECTRICAL DISTRIBUTION will have on following: K3.02 Emergency generators	Tier	2
	Group #	1
	K/A	262001
	Rating	3.8
	Revision	1
Revision Statement: **Editorial changes in stem** **Changed 3000 to 3.0 KV to match indicator**		

Question: 46

A grid transient causes voltage on all ESF 4.16 KV buses to indicate 3.0 KV for 3 seconds, then returns to normal within 2 seconds.

What is the status of the ESF 4.16 KV buses one minute later?

A. Div 1, 2, and 3 are powered from their DGs.

B. Div 1 and 2 are powered from offsite; Div 3 is powered from its DG.

C. Div 1 is powered from offsite; Div 2 and 3 are powered from their DGs.

D. Div 1 and 2 are powered from their DGs; Div 3 is powered from offsite.

Answer: **B**

Explanation:

Div 1 (2) BUV setpoints are 3744 volts for 9 seconds and 2912 volts for 0.5 seconds. Stem indicates that neither condition was met. Therefore, Buses 15AA and 16AB are still on their offsite sources.

Div 3 has a 73% BUV setpoint of 3045 volts (zero (0) second time delay). Clearly this condition was met. The Div 3 BUV relay immediately tripped the 17AC incoming feeders, started the Div 3 DG, and now the bus is powered from the machine.

Distracters:

A is wrong – as stated above Div 3 will get a start signal and assume the bus 17AC. Plausible if all 3 D/Gs have the same start setpoint as Div 1 and 2.

C is wrong - as stated above Div 1 and 2 did not get a start signal. Plausible if all 3 D/Gs have the same start setpoint as div 3.

D is wrong - as stated above Div 1 and 2 did not get a start signal and Div 3 will get a start signal and assume the bus 17AC. Plausible if applicant remembers the starts reversed.

K/A Match

This question requires the applicant to have the knowledge of auto start logic of the Emergency Diesel generators during a malfunction of the AC power grid.

Technical References:

04-1-01-P75-1, Standby Diesel Generator System SOI, Rev. 104

GLP-OPS-P7500, Standby Diesel Generator System Lesson Plan, Rev. 26

04-1-01-P81-1, High Pressure Core Spray Diesel Generator SOI, Rev. 76

GLP-OPS-P8100, High Pressure Core Spray Diesel Generator Lesson Plan, Rev. 18

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-P7500, OBJ. 8, 15.2
GLP-OPS-P8100, OBJ. 8

Question Source:	**Bank # 237**	X
(note changes and attach parent)	**Modified Bank #**	
	New	
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(7)	
Level of Difficulty:	3	

PRA Applicability:

R21 ESF is #8 on list of System Importance to CDF.

Div. 1 & 2 EDGs is #11 on list of System Importance to CDF.

Failure to align alternate power to 4.16KV or 6.9 KV buses is #6 on Operation Action Importance to CDF.

Examination Outline Cross Reference	Level	RO
262001 AC Electrical Distribution K5 Knowledge of the operational implications of the following concepts as they apply to A.C. ELECTRICAL DISTRIBUTION: K5.01 Principle involved with paralleling two A.C. sources	Tier	2
	Group #	1
	K/A	262001
	Rating	3.1
	Revision	4
Revision Statement: **Rev 1 Per NRC: New question** **Rev 2 Editorial changes in stem and answers, Changed 10CFR to 8** **Rev 3 Changed answer B** **Rev 4: Changed LOK from F to H and LOD from 2 to 3**		

Question: 47

An operator is monitoring Div 2 Emergency Diesel Generator paralleled with Bus 16AB for a monthly surveillance test.

Operator inadvertently takes the UNIT PRL CONT DG 12 handswitch to RESET.

Which of the following describes the operational implications of the switch manipulation?

A. Diesel Generator trips

B. Generator MVARS rises

C. Generator output breaker opens

D. Generator voltage regulator trips to manual

Answer: C
Explanation: 04-1-01-P75-1, Standby Diesel Generator System, Step 3.18; **IF** Diesel Generator is in parallel, placing UNIT PRL CONT DG 11 [12] handswitch to RESET opens D/G breaker.

The unit parallel switch (HS-M604A/B) is a three-position switch with switch positions RESET, OFF, and PAR. The switch spring-returns to the OFF position from the RESET or PAR positions.

The RESET position is used when the generator is not in parallel. Placing the switch in RESET when the generator is in parallel causes the generator output breaker to trip.

The OFF position is the normal position. The PAR position is used when the generator is in parallel with the grid. In the PAR position, droop circuits are placed in the voltage regulator circuitry to improve stability when the generator is in parallel with the grid.

The purpose of the droop circuitry is to allow the generator to operate in parallel with the grid. The circuit is designed to allow load sharing, while preventing the generator from assuming the entire grid load.

Distracters:

A is wrong – The D/G will not trip, only a Generator breaker trip will occur. Plausible if the applicant determines an engine trip will occur.

B is wrong. – MVARS will actually fall due to output breaker trip. Plausible if the applicant doesn't remember the actions during switch manipulation.

D is wrong – A Generator breaker trip will occur not voltage regulator trip to man. Plausible due to there is a control switch that will place the voltage regulator in the MANUAL mode but not automatically.

K/A Match

This question requires the applicant to have the knowledge of principle involved with paralleling two A.C. sources including DROOP operation of the Generator.

Technical References:

04-1-01-P75-1, Standby Diesel Generator System SOI, Rev. 104

GLP-OPS-P7500, Standby Diesel Generator System Lesson Plan, Rev. 26

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-P7500, OBJ. 17 & 19

Question Source:	**Bank #**	
(note changes and 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:	2	
PRA Applicability:		
R21 ESF is #8 on list of System Importance to CDF. Div. 1 & 2 EDGs is #11 on list of System Importance to CDF.		

Examination Outline Cross Reference	Level	RO
262002 UPS (AC/DC) A4 Ability to manually operate and/or monitor in the control room: A4.01 Transfer from alternative source to preferred source	Tier	2
	Group #	1
	K/A	262002
	Rating	2.8
	Revision	3

Revision Statement:

Rev 1: Changed answers from short to long

Rev 2: Reworded answer C

Rev 3: Editorial correction in answer C

Question: 48

The following annunciator was received and sealed in:

- STATIC INVRTR 1Y97 TROUBLE

A local operator reports that 1Y97 indicates Alternate Source Supplying Load.

Which of the following describes when this annunciator should clear?

After Normal source is restored and ____________________.

A. local alarm acknowledge pushbutton is depressed.

B. static transfer switch auto transfers to Normal source.

C. INVERTER TO LOAD pushbutton is depressed.

D. manual transfer switch is transferred to Normal source to load.

Answer: **B**

Explanation:

1Y97 is a BOP inverter, therefore, when a loss of normal supply occurs the alarm is received. After normal supply is restored the BOP inverters will auto transfer back to the normal supply which will then clear the alarm.

Distracters:

A is wrong, this has no effect on the control room alarm, plausible due to other local panels will clear trouble alarms when acknowledged locally.

C is wrong, but plausible, this action is required for ESF inverters not BOP.

D is wrong, but plausible, the swap took place on an auto transfer, the manual switch was not used, if it had been manually transferred then this statement would be true.

K/A Match

This question requires the applicant to have the knowledge of indications of auto transfer and return to normal for BOP inverters.

Technical References:

04-1-01-L62-1, Static Inverters System SOI, Rev. 49

GLP-OPS-L6200, Static Inverter System Lesson Plan, Rev. 14

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-L6200, OBJ. 4.2, 5.1, 8

Question Source:	**Bank # 750**	2013 NRC Exam Q# 48
(note changes and attach parent)	**Modified Bank #**	
	New	
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(7)	
Level of Difficulty:	3	
PRA Applicability:		

Examination Outline Cross Reference	Level	RO
263000 DC Electrical Distribution Ability to monitor automatic operations of the D.C. ELECTRICAL DISTRIBUTION including: A3.01 Meters, dials, recorders, alarms, and indicating lights	Tier	2
	Group #	1
	K/A	263000
	Rating	3.2
	Revision	0
Revision Statement:		

Question: 49

Power is lost to Div 3 battery charger 1C4 when its feeder breaker from MCC 17B01 trips open (breaker internal fault).

Which of the following identifies the initial response of 11DC bus voltage as indicated on control room panel P601?

A. Remains constant.

B. Lowers to zero volts.

C. Lowers by 60 to 65 volts.

D. Lowers by 5 to 10 volts.

Answer: D

Explanation:

Unlike the Div 1 and Div 2 batteries which have two load-sharing chargers always connected, the Div 3 battery only has one charger (normally the 1C4 charger) connected at a time. Therefore, when that charger loses its MCC AC power source, it de-energizes and is no longer able to float the bus at the normal 5-10 volts above battery bank terminal voltage. As such, the resulting P601 battery bus indication will drop by 5 to 10 volts (i.e., the bus will now be carried by the battery itself).

Distracters:

B is wrong. This choice represents the response if the same failure were to occur for one of the Div 1 or Div 2 battery chargers (i.e., where the load-sharing charger would continue to float the bus at the normal float voltage); its plausibility should speak for itself in this regard.

A and C are wrong but plausible to the weaker Applicant who cannot recall basic DC battery/charger operating principles.

K/A Match

This question requires the applicant to have the knowledge of basic DC battery/charger operating

principles.
Technical References: 04-1-01-L11-1, Plant DC Systems SOI, Rev. 127 GLP-OPS-L1100, Plant DC System Lesson Plan, Rev. 19
Handouts to be provided to the Applicants during exam: NONE
Learning Objective: GLP-OPS-L1100, OBJ. 7

Question Source:	**Bank # 339**	2011 NRC Exam Q# 35
(note changes and attach parent)	**Modified Bank #**	
	New	
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(7)	
Level of Difficulty:	3	

PRA Applicability:
125 VDC ESF is #7 on list of System Importance to CDF.

Examination Outline Cross Reference	Level	RO
264000 EDGs A2 Ability to (a) predict the impacts of the following on the EMERGENCY GENERATORS (DIESEL/JET) ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: A2.03 Operating unloaded, lightly loaded, and highly loaded	**Tier**	2
	Group #	1
	K/A	264000
	Rating	3.4
	Revision	2
Revision Statement: **Rev 1: Editorial changes in answers** **Rev 2: Changed wording in both questions.** **Changed LOD to 3**		

Question: 50

Div. 1 Diesel Generator was started for a maintenance retest and has been running for more than 5 minutes unloaded.

(1) If the diesel remains running, what could occur when it is loaded?

And

(2) IAW 04-1-01-P75-1, Standby Diesel Generator SOI, what are the requirements for minimum runtime and load?

A. (1) Excessive smoke from exhaust manifold.
(2) Greater than 60 minutes at more than 50% load

B. (1) Oscillations in engine speed
(2) Greater than 30 minutes at more than 60% load

C. (1) Excessive smoke from exhaust manifold
(2) Greater than 30 minutes at more than 60% load

D. (1) Oscillations in engine speed
(2) Greater than 60 minutes at more than 50% load

Answer: A
Explanation: Per 04-1-01-P75-1, Standby Diesel Generator System: 3.22 <u>**IF**</u> Diesel Generators are run for longer than five minutes unloaded, excessive amounts of smoke <u>May</u> be generated from exhaust manifold following loading of Diesel Generators. 3.52 Following an extended engine run of no/light loading, the generator <u>Should</u> be loaded to more than 50 percent load for more than one hour. CARB approved action in response to INPO IER L2-11-46. Reference CR 2011-8269.
Distracters: B is wrong, but plausible. This choice represents the response if Governor oil level is low and oil has to be added, could cause oscillations with speed. Minimum loading time is from Div 3 D/G precaution running unloaded. C is wrong, but plausible. Minimum loading time is from Div 3 D/G precaution running unloaded. D is wrong, but plausible. This choice represents the response if Governor oil level is low and oil has to be added, could cause oscillations with speed.
K/A Match This question requires the applicant to have the knowledge of procedure requirements for emergency diesel generators.
Technical References: 04-1-01-P75-1, Standby Diesel Generator System SOI, Rev. 104 GLP-OPS-P7500, Standby Diesel Generator System Lesson Plan, Rev. 26
Handouts to be provided to the Applicants during exam: NONE
Learning Objective: GLP-OPS-P7500, OBJ. 19

Question Source:	**Bank #**	
(note changes and attach	**Modified Bank #**	

parent)		
	New	X
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.41(b)(10)	
Level of Difficulty:	3	
PRA Applicability:		
Div. 1 & 2 EDGs is #11 on list of System Importance to CDF.		

Examination Outline Cross Reference	Level	RO
264000 EDGs A4 Ability to manually operate and/or monitor in the control room: A4.05 Transfer of emergency generator (with load) to grid	Tier	2
	Group #	1
	K/A	264000
	Rating	3.4
	Revision	2

Revision Statement:

Rev 1: Editorial changes in stem
Changed to Memory Fundamental

Rev 2: Changed wording of stem.

Question: 51

Div 1 D/G has auto started and re-energized 15AA bus.

The CRS has directed you to **parallel an offsite power source** to the Div 1 D/G.

To adjust bus frequency within range of offsite frequency, use _____(1)______ handswitch so that synchroscope is rotating SLOWLY in ___(2)___ direction.

A. (1) DG 11 VR MAN CONT
   (2) FAST

B. (1) DG 11 GOV MAN CONT
   (2) FAST

C. (1) DG 11 VR MAN CONT
   (2) SLOW

D. (1) DG 11 GOV MAN CONT
   (2) SLOW

Answer: **D**

Explanation:

Per 04-1-01-P75-1, Standby Diesel Generator system, step 4.3.2e:

e. **ADJUST** Standby Diesel Generator 11 [12] speed to bring frequency within range of Offsite System frequency by **USING** GOV MAN CONT DG 11 [12] handswitch so that synchroscope indicator is **ROTATING SLOWLY** in the SLOW direction (counterclockwise).

Distracters:

A is wrong, but plausible. This choice represents the handswitch used to adjust voltage or VARS if voltage Regulator is in MAN. The procedure step states in the slow direction not fast, fast direction would be used if the D/G was being paralleled to an offsite source instead of the reverse.

B is wrong, but plausible. The procedure step states in the slow direction not fast, fast direction would be used if the D/G was being paralleled to an offsite source instead of the reverse.

C is wrong, but plausible. This choice represents the handswitch used to adjust voltage or VARS if voltage Regulator is in MAN.

K/A Match

This question requires the applicant to have the knowledge of procedure requirements for emergency diesel generators being paralleled to an offsite source.

Technical References:

04-1-01-P75-1, Standby Diesel Generator System SOI, Rev. 104

GLP-OPS-P7500, Standby Diesel Generator System Lesson Plan, Rev. 26

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-P7500, OBJ. 17

Question Source:	**Bank #**	
(note changes and 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	

PRA Applicability:

Div. 1 & 2 EDGs are #11 on list of System Importance to CDF.

Examination Outline Cross Reference	Level	RO
300000 Instrument Air Knowledge of electrical power supplies to the following: K2.01 Instrument air compressor	Tier	2
	Group #	1
	K/A	300000
	Rating	2.8
	Revision	0
Revision Statement:		

Question: 52

Which of the following is the electrical power supply to the compressor motor for Plant Air Compressor 'C'?

A. 13AD

B. 14AE

C. 15AA

D. 16AB

Answer: A
Explanation: Per 04-1-01-P51-1, Attachment III: 13AD powers PAC 'C'. 14AE...PAC 'B'. 16AB...PAC 'A'.
Distracters: B is wrong, but plausible. Bus 14AE provides power to PAC 'B’. C is wrong. An ESF bus does provide power to a Plant Air compressor, but, not the 15AA bus. 16AB bus provides power to PAC ‘A’. Plausible if the applicant confuses the two ESF buses. D is wrong, but plausible. Bus 16AB provides power to PAC 'A'.
K/A Match This question requires the applicant to have the knowledge of Plant air compressor power supply.
Technical References: 04-1-01-P51-1, Plant Air System SOI, Rev. 16

GLP-OPS-P5100, Plant Air Systems Lesson Plan, Rev. 8		
Handouts to be provided to the Applicants during exam: NONE		
Learning Objective: GLP-OPS-P5100, OBJ. 8		
Question Source:	**Bank # 123**	X
(note changes and attach parent)	**Modified Bank #**	
	New	
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.41(b)(4)	
Level of Difficulty:	2	
PRA Applicability:		
Plant Air is #15 on list of System Importance to CDF.		

Examination Outline Cross Reference	Level	RO
400000 Component Cooling Water 2.1.28 Knowledge of the purpose and function of major system components and controls.	Tier	2
	Group #	1
	K/A	400000
	Rating	4.1
	Revision	0
Revision Statement:		

Question: 53

The purpose of the CCW Temperature Control Valve is to control ____(1)______ system flow on the ______(2)________ of the CCW system heat exchanger.

A. (1) PSW
   (2) Inlet

B. (1) CCW
   (2) Outlet

C. (1) CCW
   (2) Inlet

D. (1) PSW
   (2) Outlet

Answer: D

Explanation:

The CCW system uses the PSW system as a heat sink by controlling PSW flow on the outlet of the CCW system heat exchangers.

Distracters:

A is wrong. CCW Temp Control Valve is located on the outlet of the CCW system heat exchangers. Plausible if the applicant doesn't remember the location of the CCW system TCV.

B is wrong. PSW flow is controlled not CCW. Plausible if the applicant doesn't remember which system is modulated to control temperature.

C is wrong. PSW flow is controlled not CCW and CCW Temp Control Valve is located on the outlet of the CCW system heat exchangers. Plausible if the applicant doesn't remember which system is modulated to control temperature and the location of the CCW system TCV.

K/A Match

This question requires the applicant to have the knowledge of CCW system Temperature Control Valve purpose and function.
Technical References: 04-1-01-P42-1, Component Cooling Water (CCW) SOI, Rev. 53 GLP-OPS-P4200, Component Cooling Water System Lesson Plan, Rev. 15
Handouts to be provided to the Applicants during exam: NONE
Learning Objective: GLP-OPS-P4200, OBJ. 7.5

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.41(b)(4)	
Level of Difficulty:	2	
PRA Applicability:		

Examination Outline Cross Reference	Level	RO
201005 RCIS A2 Ability to (a) predict the impacts of the following on the ROD CONTROL AND INFORMATION SYSTEM (RCIS) ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: A2.04 Withdraw block: BWR-6	**Tier**	2
	Group #	2
	K/A	201005
	Rating	3.2
	Revision	1
Revision Statement: **Changed condition to event in stem**		

Question: 54

A plant Startup is in progress.

Control rods are being withdrawn to achieve criticality.

(1) Which of the following describes an event that will cause RC&IS to stop control rod **withdrawal ONLY**?

And

(2) What procedure should be used to mitigate the event?

A. (1) Withdrawing a control rod one notch past the bank position.
(2) 05-1-02-IV-1, Control Rod/Drive Malfunctions ONEP.

B. (1) Withdrawing a control rod and a reed switch on one channel fails to close.
(2) 05-1-02-IV-1, Control Rod/Drive Malfunctions ONEP.

C. (1) Withdrawing a control rod one notch past the bank position.
(2) 04-1-01-C11-2, Rod Control and Information System.

D. (1) Withdrawing a control rod and a reed switch on one channel fails to close.
(2) 04-1-01-C11-2, Rod Control and Information System.

Answer:	**D**
Explanation: Withdrawing a control rod and a reed switch on one channel fails to close will cause a Control Rod Withdrawal Block due to DATA FAULT. Using section 4.11 of 04-1-01-C11-2, Rod Control and Information System SOI, RC&IS Substitute Data will allow the other channel to "substitute position" and	

reset the control rod block.

Distracters:

A is wrong, but plausible. This action will also cause an INSERT block, the ONEP is not entered due to only one notch past its intended position, the ONEP gives guidance for “more than one notch”.

B is wrong, but plausible. The ONEP is not entered due to only one notch past its intended position, the ONEP gives guidance for “more than one notch”.

C is wrong, but plausible. This action will also cause an INSERT block.

K/A Match

This question requires the applicant to predict what will occur within RC&IS when a Rod Block occurs and what procedure to use to mitigate this event.

Technical References:

04-1-01-C11-2, Rod Control and Information System SOI, Rev. 42

GLP-OPS-C1102, Rod Control and Information System (RC&IS) Lesson Plan, Rev. 10

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-C1102, OBJ. 13.3

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(2)	
Level of Difficulty:	3	
PRA Applicability:		

Examination Outline Cross Reference	Level	RO
202001 Recirculation K1 Knowledge of the physical connections and/or cause-effect relationships between RECIRCULATION SYSTEM and the following: K1.19 Feedwater flow	Tier	2
	Group #	2
	K/A	202001
	Rating	3.2
	Revision	1

Revision Statement:

Used "Recirc" in stem
Changed to Memory Fundamental
Changed to LOD 3

Question: 55

The plant is operating at rated thermal power.

Which of the following describes a condition that will cause the Reactor Recirc pumps to automatically downshift to SLOW speed?

A. Total core flow is reduced to <70 mlbm/hr.

B. One Reactor Feedwater pump suction flow transmitter fails to 0 mlbm/hr.

C. One Feedwater flow signal in the Reactor Level Control System fails to 0 mlbm/hr.

D. Total Feedwater flow signal in the Reactor Level Control System fails to <2.5 mlbm/hr.

Answer: **D**

Explanation:

The cavitation interlock for the Reactor Recirc Pumps is Total Feedwater flow <3.0 mlbm/hr. This signal is generated within the Reactor Level Control System Total Feedwater indication.

Distracters:

A is wrong. This action occurs manually by the operations staff in several occasions and a Reactor Recirc FCV runback will lower core flow even further with no downshift to slow speed. Plausible due to this is a well known core flow that is used during events as a setpoint to lower reactor power.

B is wrong. Recirc FCV runback signal is generated from a Feedpump trip and low RPV water level. The Feedpump trip signal is generated from low suction flow. (GLP-OPS-B3300, PAGE 42), plausible due to this causes an automatic event within the recirc system but not a downshift.

C is wrong. IF one of the two flow signals fail to zero the Feedwater Reactor level control system will cause the level control system to transfer to single element. (GLP-OPS-C3400, PAGE 29). Plausible if the applicant believes the low flow downshift signal is generated from the FLCS.

K/A Match

This question requires the applicant to have the knowledge of the Recirculation system interrelationship with the Feedwater system.

Technical References:

04-1-01-B33-1, Reactor Recirculation System SOI, Rev. 163

GLP-OPS-B3300, Reactor Recirculation System Lesson Plan, Rev. 30

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-B3300, OBJ. 25

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

Examination Outline Cross Reference	Level	RO
204000 RWCU K3 Knowledge of the effect that a loss or malfunction of the REACTOR WATER CLEANUP SYSTEM will have on following: K3.06 Area radiation levels	Tier	2
	Group #	2
	K/A	204000
	Rating	2.6
	Revision	1
Revision Statement: **Changed 10CFR to 12**		

Question: 56

The plant is operating at rated thermal power.

Which of the following RWCU malfunctions will cause area radiation levels in the MSL tunnel to rise FIRST?

A. Both RWCU pumps trip

B. Spurious Group 8 isolation

C. RWCU F/D septum rupture

D. P42-F103, RWCU HX ISOL VLV, inadvertently closes

Answer: **C**

Explanation:

Per 05-1-02-V-12, Condensate/Reactor Water High Conductivity ONEP, Pages 1 & 7.

1.4 Resin May be injected into the Reactor from **EITHER** the RWCU filter - demineralizers (F/D) **OR** the condensate precoat filters **OR** demineralizers. In **EITHER** case the results are the same. The resin intrusion Will cause water conductivity **AND** activity to rise substantially while pH substantially reduces. Depending on the amount **AND** rate of resin injected, these changes May become detectable immediately **OR** only after several hours. The most serious injection Could cause a rapid change in vessel chemistry resulting in a large release of nitrogen-16 into the steam lines. Care Should be exercised to ensure a main steam line radiation alarm is **NOT** the result of a fission product release caused by a fuel element failure. A resin instrusion May cause a slow positive reactivity event for several hours due to decreases in the surface tension of the water followed by negative reactivitry due to accumulation of organic impurities causing increased voiding in core. The overall effect May take several days to return to normal conditions.

Symptoms of a G33 Demineralizer resin break through are normally indicated by an initial rise in Reactor Water conductivity G33 Demineralizer effluent conductivity high alarms **OR** post strainer high dP alarms May also be indicative of a resin break through. A rise in MSL Radiation monitor **AND** Offgas Pre-Treatment Radiation monitor readings May also be observed.

Distracters:

A is wrong, but plausible. This would reduce the radiation around the MST due to no flow in the system.

B is wrong, but plausible. Group 8 is the RWCU system therefore this would reduce the radiation around the MST due to no flow in the system,

D is wrong, but plausible. This would cause the temperature to rise in the RWCU system to the point of isolation, therefore this would reduce the radiation around the MST due to no flow in the system,

K/A Match

This question requires the applicant to have the knowledge of the interrelationship with the RWCU system malfunctions and area radiation levels.

Technical References:

04-1-01-G33-1, Reactor Water Cleanup SOI, Rev. 160

GLP-OPS-G3336, Reactor Water Cleanup (RWCU) System Lesson Plan, Rev. 17

05-1-02-V-12, Condensate/Reactor Water High Conductivity ONEP, Rev. 26

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-G3300, OBJ. 8

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(12)	
Level of Difficulty:	3	
PRA Applicability:		

Examination Outline Cross Reference	Level	RO
219000 RHR/LPCI: Torus/Pool Cooling Mode K5 Knowledge of the operational implications of the following concepts as they apply to RHR/LPCI: TORUS / SUPPRESSION POOL COOLING MODE: K5.04 Heat exchanger operation	**Tier**	2
	Group #	2
	K/A	219000
	Rating	2.9
	Revision	1
Revision Statement: **Changed order of answers from short to long**		

Question: 57

An ATWS has occurred.

Reactor power is approximately 10%.

At time 0 all immediate actions are performed.

The CRS directs both RHR systems to be aligned for Suppression Pool Cooling.

Which of the following indicates when the RHR heat exchangers will provide **<u>continuous</u>** full flow cooling operation?

A. Immediately E12-F048A and B, RHR HX A/B BYP VLV, automatically closes.

B. E12-F048A and B, RHR HX A/B BYP VLV, automatically closes after a 4 minute time delay.

C. Immediately by manually closing E12-F048A and B, RHR HX A/B BYP VLV, using handswitch.

D. Manually closing E12-F048A and B, RHR HX A/B BYP VLV, using handswitch after 4 minute time delay.

Answer: D
Explanation: The applicant must recognize that an ATWS >5% immediate action is to initiate and override low pressure ECCS systems by using the ECCS manual initiate pushbuttons at time 0.

E12-F048A and B, RHR HX A/B BYP VLV, are interlocked open for 4 minutes after a LOCA signal or manual initiation. Then must be manually closed. However, the F048A/B can be closed within the 4 minute time delay but it will return to the full open position. The question states "provide continuous full flow", the only way that can happen is to wait for the 4 minute TD.

Distracters:

A is wrong. E12-F048A/B do not automatically close. The only action with the handswitch for immediate valve response would be with no initiation signal present. Plausible due to the E21-F024A and B, Test Return valves get a close signal on a LPCI initiation.

B is wrong. E12-F048A/B do not automatically close. The time delay will allow the valves to be closed by removing an open signal. Plausible due to the E21-F024A and B, Test Return valves get a close signal on a LPCI initiation.

C is wrong. E12-F048A and B, RHR HX A/B BYP VLV, are given an open signal in the logic for 4 minutes after a LOCA signal or manual initiation. Then must be manually closed. However, the F048 can be closed within the 4 minute time delay but it will return to the full open position. The question states "provide continuous full flow", the only way that can happen is to wait for the 4 minute TD. Plausible if the applicant doesn't remember the 4 minute TD.

K/A Match

This question requires the applicant to have the knowledge of the implications of the interlocks on the RHR heat exchanger bypass valves with a LOCA signal present and suppression pool cooling is required.

Technical References:

GLP-OPS-E1200, Residual Heat Removal (RHR) System Lesson Plan, Rev. 10

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-E1200, OBJ. 8

Question Source:	**Bank #**	
(note changes and 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	

PRA Applicability:
RHR is #3 on list of System Importance to CDF. Failure to start Suppression Pool Cooling is #9 on list of Operator Action Importance to CDF.

Examination Outline Cross Reference	Level	RO
226001 RHR/LPCI: CTMT Spray Mode A1 Ability to predict and/or monitor changes in parameters associated with operating the RHR/LPCI: CONTAINMENT SPRAY SYSTEM MODE controls including: A1.10 Emergency generator loading	**Tier**	2
	Group #	2
	K/A	226001
	Rating	3.0
	Revision	2

Revision Statement:

Rev 1: Defined LOCA in stem
Changed time for load rise to 18 min.
Changed LOD to 4
Changed inch symbol to inches in answer A

Rev 2: Editorial changes to stem

Question: 58

Time in minutes

At time 0: Total loss of offsite power has occurred

- All diesel generators started and energized respective buses

At time 5: Large break LOCA signal occurred due to high drywell pressure

- All Low Pressure ECCS were overridden

At time 18: RO recognized Div 1 and Div 2 D/G load rose by approximately 0.8 MWe each

What caused this indication?

A. Reactor water level lowered to <-150.3 inches.

B. Drywell pressure lowered to <1.23 psig then rose to >1.4 psig.

C. Auto initiation of Containment Spray mode of both RHR systems.

D. MCCs 15B42 and 16B42 were manually re-energized per ONEP.

Answer: C

Explanation:

With ECCS pumps overridden the only signals that will auto restart the RHR pumps is an auto initiation of Containment spray mode.

Each RHR pump load is approximately 803 KWe.

Distracters:

A is wrong, but plausible. Low pressure ECCS initiation is a seal in signal and must be manually reset to reinstate the auto initiation signals, if drywell pressure lowered and then rose to above the initiation setpoint nothing would change.

B is wrong, but plausible. Low pressure ECCS initiation is a seal in signal and must be manually reset to reinstate the auto initiation signals, even if reactor water level lowered to the initiation setpoint nothing would change due to the logic is already sealed in.

D is wrong, but plausible. This action is performed from the loss of AC power ONEP, however, this one MCC doesn't have as much load as one RHR pump.

K/A Match

This question requires the applicant to have the ability to monitor changes in D/G loading when a containment spray initiation ocurrs.

Technical References:

04-1-01-E12-1, Residual Heat Removal System SOI, Rev. 147

GLP-OPS-E1200, Residual Heat Removal (RHR) System Lesson Plan, Rev. 10

05-1-02-I-4, Loss of AC Power ONEP, Rev. 50

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-E1200, OBJ. 10

Question Source:	**Bank #**	
(note changes and 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:	4	
PRA Applicability:		
RHR is #3 on list of System Importance to CDF. Failure to initiate SPC and Containment Spray is #2 on list of Operator Action Importance to CDF.		

Examination Outline Cross Reference	Level	RO
234000 Fuel Handling Equipment A4 Ability to manually operate and/or monitor in the control room: A4.01 Neutron monitoring system	Tier	2
	Group #	2
	K/A	234000
	Rating	3.7
	Revision	2

Revision Statement:

Rev 1: Editorial changes in stem
Changed 10CFR to 1

Rev 2: Removed part of stem

Question: 59

In-vessel fuel movements are in progress.

(1) Which monitors provide primary indication of neutron flux levels for identifying inadvertent criticality?

<u>**And**</u>

(2) With current plant conditions, how many are required by Tech Specs?

A. (1) Intermediate Range Monitors
(2) 2

B. (1) Source Range Monitors
(2) 2

C. (1) Intermediate Range Monitors
(2) 4

D. (1) Source Range Monitors
(2) 4

Answer: **B**

Explanation:

During a refueling outage Reactor power is so low that the Source Range Monitors (SRMs) are used to monitor for inadvertent criticality.

Tech Spec 3.3.1.2.2 requires an operable SRM in core quadrant where core alterations are being performed <u>**AND**</u> in adjacent quadrant.

Distracters:

A is wrong, but plausible. IRMs are used after the reactor is critical and power is > SRM indication.

C is wrong, but plausible. IRMs are used after the reactor is critical and power is > SRM indication. And 4 is the number of SRMs required for startup.

D is wrong, but plausible. 4 is the number of SRMs required for startup.

K/A Match

This question requires the applicant to have the ability to monitor for an inadvertent criticality during fuel handling outage activities and which monitor to use.

Technical References:

03-1-01-5, Refueling IOI, Rev. 137

Tech Specs 3.3.1.2, Amendment No. 120

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-IOI05, OBJ. 2.3

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.41(b)(1)	
Level of Difficulty:	2	
PRA Applicability:		

Examination Outline Cross Reference	**Level**	**RO**
241000 Reactor/Turbine Pressure Regulator A3 Ability to monitor automatic operations of the REACTOR/TURBINE PRESSURE REGULATING SYSTEM including: A3.08 Steam bypass valve operation	**Tier**	2
	Group #	2
	K/A	241000
	Rating	3.8
	Revision	1
Revision Statement: **Editorial changes in stem and answers**		

Question: 60

Currently:

- Reactor power is 18%
- Main generator is on the grid
- Bypass Control Valves are 27% open
- Load reference is OFF

The ACRO inserts a control rod which reduces reactor power to approximately 17%.

What Reactor/Turbine pressure control related indications should the operator expect to see at P680?

A. Bypass Control Valves open;
Turbine Control Valves remain as is

B. Bypass Control Valves close;
Turbine Control Valves remain as is

C. Turbine Control Valves open;
Bypass Control Valves remain as is

D. Turbine Control Valves close;
Bypass Control Valves remain as is

Answer: **B**		
Explanation: At this power level EHC is being controlled by IPC (pressure regulator). Reducing reactor power has the effect of lowering reactor pressure. If the Bypass Valves weren't already open 27%, the Turbine Control Valves would close as needed to control pressure (i.e., to restore pressure back to the normal 935 psig main steam line equalizing header setpoint). However, with the Bypass Valves open, they instead will close as necessary to control pressure. Because the rod insertion reduced power by only 1%, with the Bypass Valves starting at 27% open, they alone will be able to fully restore pressure to IPC setpoint (i.e., the TCVs will not have to move). Thus, only the Bypass Valves need to close as necessary to control pressure, leaving the TCVs as is.		
Distracters: A & C are wrong. Plausible to the Applicant who believes that IPC attempts to maintain HP turbine 1st stage pressure rather than reactor pressure. D is wrong. It is plausible to the Applicant who believes the IPC would first pass its demand signal to the TCVs rather than to the BPVs.		
K/A Match This question requires the applicant to have the ability to monitor automatic operations of the REACTOR/TURBINE PRESSURE REGULATING SYSTEM including bypass valve operation.		
Technical References: GLP-OPS-N3202, Main Turbine EHC Control System Lesson Plan, Rev. 11		
Handouts to be provided to the Applicants during exam: NONE		
Learning Objective: GLP-OPS-N3202, OBJ. 9		
Question Source:	**Bank # 375**	2011 NRC Exam Q# 71
(note changes and 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	

PRA Applicability:

Examination Outline Cross Reference	Level	RO
245000 Main Turbine Gen. / Aux. K6 Knowledge of the effect that a loss or malfunction of the following will have on the MAIN TURBINE GENERATOR AND AUXILIARY SYSTEMS: K6.06 Electrical distribution	Tier	2
	Group #	2
	K/A	245000
	Rating	3.0
	Revision	0
Revision Statement:		

Question: 61

The main turbine has been tripped and is coasting down.

Service Transformer 21 trips.

As Bearing Oil Header pressure drops to 93 psig, which of the AC Auxiliary Oil Pumps will start?

A. 'A' pump ONLY

B. 'B' and 'C' pumps ONLY

C. 'A' and 'B' pumps ONLY

D. 'A', 'B', and 'C' pumps

Answer: A

Explanation:

A trip of Service Transformer 21 causes a loss of 11HD and 14AE. ‘A’ pump is powered from 13AD. ‘B’ and ‘C’ pumps are powered from 14AE. Only ‘A’ pump has power.

Assume no operator action.

A and B start at 95 psig. C starts at 90 psig.

Distracters:

B is wrong. It is plausible to the Applicant who doesn’t remember the power supplies, this would be true if Service Transformer 11 were to trip.

C is wrong. It is plausible to the Applicant who doesn’t remember the power supplies

D is wrong. It is plausible to the Applicant who doesn’t remember the power supplies and if the applicant believes the D/G will restore the power, which is untrue on BOP buses.

K/A Match This question requires the applicant to have the knowledge of a malfunction or loss of AC power will have on T/G auxiliaries.
Technical References: GLP-OPS-N3402, Main Turbine Lube Oil System Lesson Plan, Rev. 8 04-1-01-N34-2, Main Turbine Lube Oil System SOI, Rev. 47
Handouts to be provided to the Applicants during exam: NONE
Learning Objective: GLP-OPS-N3402, OBJ. 4.1, 6.1

Question Source:	**Bank # 1162**	X
(note changes and 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	
PRA Applicability:		

Examination Outline Cross Reference	Level	RO
256000 Reactor Condensate K2 Knowledge of electrical power supplies to the following: K2.01 System pumps	Tier	2
	Group #	2
	K/A	256000
	Rating	2.7
	Revision	0
Revision Statement:		

Question: 62

Electrical bus 14AE trips.

Which Condensate and Condensate Booster pumps are currently **without** power?

A. ‘C’ Condensate Pump
‘B’ & ‘C’ Condensate Booster Pumps

B. ‘B’ & ‘C’ Condensate Pumps
‘B’ & ‘C’ Condensate Booster Pumps

C. ‘C’ Condensate Pump
‘C’ Condensate Booster Pump

D. ‘B’ & ‘C’ Condensate Pumps
‘C’ Condensate Booster Pump

Answer: D
Explanation: Condensate pump ‘B’ and ‘C’ are powered from bus 14AE. Condensate Booster pump ‘C’ is powered from bus 14AE.
Distracters: A is wrong. ‘B’ condensate pump is powered from 14AE and ‘B’ Booster pump is powered from 13AD. B is wrong. ‘B’ Booster pump is powered from 13AD. C is wrong. ‘B’ condensate pump is powered from 14AE. All distracters are plausible due to combinations of BOP bus power.
K/A Match This question requires the applicant to have the knowledge of power supplies for the condensate system

pumps.

Technical References:

GLP-OPS-N1900, Condensate System Lesson Plan, Rev. 12

04-1-01-N19-1, Condensate System SOI, Rev. 72

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-N1900, OBJ. 10.1 & 10.2

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.41(b)(5)	
Level of Difficulty:	2	

PRA Applicability:

Condensate is #10 on list of System Importance to CDF.

Examination Outline Cross Reference	Level	RO
259001 Reactor Feedwater K1 Knowledge of the physical connections and/or cause-effect relationships between REACTOR FEEDWATER SYSTEM and the following: K1.08 Reactor water level control system	Tier	2
	Group #	2
	K/A	259001
	Rating	3.6
	Revision	1
Revision Statement: **Editorial changes in stem and answers**		

Question: 63

The plant is operating at rated conditions.

The following alarm is received:

- RFPT A GOV VLV CONT TROUBLE

ATC recognizes Feedwater Control System for ‘A’ Reactor Feed pump has shifted to EMERGENCY MANUAL.

What describes response of RFPT A speed to a manual RAISE demand?

A. change at 10 rpm/sec for first second then 120 rpm/sec thereafter
Overspeed trip is possible

B. change from 0% to 100% speed in 15 seconds
Overspeed trip is possible

C. change at 10 rpm/sec for first second then 120 rpm/sec thereafter
Feedpump speed is clamped

D. change from 0% to 100% speed in 15 seconds
Feedpump speed is clamped

Answer: **B**

Explanation:

RFPT governor operation in EMERGENCY MANUAL has no speed feedback signal. These modes of operation have direct control of the Governor control valve; therefore, RFPT overspeed is possibility in these modes. In manual, Feedpump speed can be changed from 0% to 100% in 15 seconds.

Distracters:

A is wrong, but plausible. This speed change is normal using SPEED AUTO.

C is wrong, but plausible. This speed change is normal using SPEED AUTO, this is normal speed control with one feedpump running.

D is wrong, but plausible. This is normal speed control with one feedpump running..

K/A Match

This question requires the applicant to have the knowledge of physical connection or cause/effect relationship with DFCS and the Reactor Feedwater pumps control.

Technical References:

GLP-OPS-C3400, Digital Feedwater Control System (DFCS) Lesson Plan, Rev. 17

GLP-OPS-N2100, Feedwater System Lesson Plan, Rev. 17

04-1-01-N21-1, Feedwater System SOI, Rev. 74

04-1-02-1H13-P680-2A-C5, RFPT A GOV VLV CONT TROUBLE ARI, Rev. 215

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-C3400, OBJ. 15 & 16
GLP-OPS-N2100, OBJ. 19

Question Source:	**Bank #**	
(note changes and 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	
PRA Applicability:		

Examination Outline Cross Reference	Level	RO
286000 Fire Protection 2.4.50 Ability to verify system alarm setpoints and operate controls identified in the alarm response manual.	Tier	2
	Group #	2
	K/A	286000
	Rating	4.2
	Revision	1
Revision Statement: **Used abbreviations for pump** **Editorial changes in answers**		

Question: 64

The following alarm is received on P862 panel:

- JKY FIRE PMP PWR FAIL

(1) Which of the following describes the fire water header pressure **setpoint** that the Motor Driven Fire Pump (MDFP) should start?

And

(2) If the Motor Driven does **not** automatically start what action should the RO then perform?

A. (1) 123 psig
(2) Must dispatch local operator to start the MDFP

B. (1) 129 psig
(2) Must dispatch local operator to start the MDFP

C. (1) 123 psig
(2) Manually start the MDFP from Control Room

D. (1) 129 psig
(2) Manually start the MDFP from Control Room

Answer: D
Explanation: The Motor driven pump will auto start at 129 psig header pressure and if a component does not perform it auto function then perform that action, the Motor driven pump has a start pushbutton on the P862 panel.
Distracters:

A is wrong, but plausible. 123 psig for 5 seconds will start the ‘A’ diesel driven fire pump and the Motor driven pump has a start pushbutton on the P862 panel.

B is wrong, but plausible. The Motor driven pump has a start pushbutton on the P862 panel.

C is wrong, but plausible. 123 psig for 5 seconds will start the ‘A’ diesel driven fire pump..

K/A Match

This question requires the applicant to have the ability to use the Alarm response instructions (ARI).

Technical References:

GLP-OPS-P6400, Fire Protection System Lesson Plan, Rev. 13

04-S-01-P64-1, Fire Protection Water System SOI, Rev. 63

EN-OP-120, Operator Fundamentals Program, Rev. 1

04-S-02-SH13-P862-1A-A6, JKY FIRE PMP PWR FAIL ARI, Rev. 14

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-P6400, OBJ. 6

Question Source:	**Bank #**	
(note changes and 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	
PRA Applicability:		

Examination Outline Cross Reference	Level	RO
290003 Control Room HVAC K4 Knowledge of CONTROL ROOM HVAC design feature(s) and/or interlocks which provide for the following: K4.01 System initiations/reconfiguration: Plant-Specific	Tier	2
	Group #	2
	K/A	290003
	Rating	3.1
	Revision	3
Revision Statement: **Rev 1: Corrected consistency from a previous question with times.** **Rev 2: Made question a fill-in-blank for simplicity** **Rev 3: Minor editorial changes**		

Question: 65

(Time in minutes)

Plant operating at rated thermal power when following occurs:

At time 0: All 4 Control Room Vent Radiation Monitors indicate 6 mRem/hr

At time 5: LOCA occurs

At time 10: HPCS and RCIC auto initiate and restore level to normal band

Control Room Standby Fresh Air Ventilation system Fresh Air Inlet valves, Z51-F007 and Z51-F016, receive an isolation signal at time _________ and can be manually opened at time _________ .

A. 0
10

B. 0
20

C. 10
20

D. 10
10

Answer: **A**

Explanation:

On the receipt of a control room isolation signal of -41.6” RPV level or 1.23 psig D/W pressure or >4.5 mrem/hr on Control Room vent rad monitor, the SBFA unit will auto start in the Recirc mode. Fresh air inlet valves F007 and F016 will auto close and be interlocked closed for 10 minutes. Any subsequent initiation signal during the 10 min time delay has no effect.

Distracters:

B is wrong, but plausible. The valves can be re-opened after 10 min time delay. The 30 second time delay to reopen valves is based on the Containment Isolation bypass timer. There is no such 30 second bypass on Control Room Ventilation.

C is wrong, but plausible. The Valves will auto close at time 0 and can be re-opened after 10 min time delay. The 30 second time delay to reopen valves is based on the Containment Isolation bypass timer. There is no such 30 second bypass on Control Room Ventilation.

D is wrong, but plausible. The Valves will auto close at time 0.

K/A Match

This question requires the applicant to have the knowledge of the Control Room HVAC reconfiguration.

Technical References:

04-S-01-Z51-1, Control Room HVAC System SOI, Rev. 57

GLP-OPS-Z5100, Control Room Ventilation System Lesson Plan, Rev. 13

E-0131 SH 3, Rev. 10

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-Z5100, OBJ. 8, 9 & 11

Question Source:	**Bank # 764**	2013 NRC Exam Q# 65
(note changes and 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	
PRA Applicability:		

Examination Outline Cross Reference	Level	RO
2.1.1 Knowledge of conduct of operations requirements	Tier	3
	Group #	N/A
	K/A	2.1.1
	Rating	3.1
	Revision	1

Revision Statement:

Changed answers from short to long
Changed LOD to 3

Question: 66

Per EN-OP-115, Conduct of Operations, which of the following is an acceptable two-handed operation at GGNS?

A. Manually inhibiting ADS at P601

B. Initiating and overriding HPCS at P601

C. Manually opening ADS Valves at P601

D. Up-shifting Reactor Recirc Pumps at P680

Answer: **B**

Explanation:

The Acceptable Two-Handed Operations for GGNS are listed in Attachment 9.3, Section 3 of EN-OP-115. Initiating and overriding ECCS systems is listed on the Addendum and the method for overriding does in fact require two-handed operation (see 02-S-01-43, Transient Mitigation Strategy, Section 6.2.8.b).

Distracters:

A is wrong. There is no reason to manually inhibit both ADS divisions at the same time; nor is it a human performance behavior that conforms to self-checking expectations; the choice is plausible to the Applicant who fails to recall these facts.

C is wrong. There is no reason to open more than one ADS Valve at a time; nor is it a human performance behavior that conforms to self-checking expectations; the choice is plausible to the Applicant who fails to recall these facts.

D is wrong. Attachment 9.3, Section 3 permits two-handed downshifting of the Recirc Pumps, but not up-shifting. The choice is plausible to the Applicant who fails to recall this fact.

K/A Match This question requires the applicant to have the knowledge of conduct of operations requirements.		
Technical References: EN-OP-115, Conduct of Operations, Rev. 19 02-S-01-43, Transient Mitigation Strategy, Rev. 3		
Handouts to be provided to the Applicants during exam: NONE		
Learning Objective: GLP-OPS-PROC, Objective 1.17		
Question Source:	**Bank # 335**	2011 NRC Exam Q# 31
(note changes and 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	
PRA Applicability:		

Examination Outline Cross Reference	Level	RO
2.1.5 Ability to use procedures related to shift staffing, such as minimum crew complement, overtime limitations, etc.	Tier	3
	Group #	N/A
	K/A	2.1.5
	Rating	2.9
	Revision	1
Revision Statement: **Editorial changes in stem**		

Question: 67

Which of the following is a Tech Spec requirement for **MINIMUM Control Room** shift manning?

A. One RO <u>and</u> one SRO during MODE 1, 2, or 3

B. One RO <u>and</u> one SRO when fuel is in reactor

C. Two ROs <u>and</u> one SRO when fuel is in reactor

D. Two ROs <u>and</u> one SRO during MODE 1, 2, or 3

Answer: **A**

Explanation:

Tech Spec Administrative Controls Section 5.2.2, Technical Requirements Manual (TRM) Section 7.0 (including Table 7.2.2-1), and Conduct of Operations procedure, EN-OP-115, Attachment 9.3, Section 1.

Per Tech Spec Administrative Controls Section 5.2.2, TRM Table 7.2.2-1, and the Conduct of Operation Att. 9.3 combination of step 5 and 5a states that MODE 1, 2, or 3, there need be only one RO <u>and</u> one SRO in the control room.

Distracters:

B is wrong, but plausible. “Fuel is in the reactor” (prospectively) even in MODE 5. The Conduct of Operations Att. 9.3 step 5a (allowing just an RO to be in the control room in, for example, MODE 5) is consistent with Tech Spec 5.5.5.2b and TRM Table 7.2.2-1 (allowing just an RO to be present in the control room in MODE 5). Therefore, this choice’s claim that the MINIMUM is having one RO <u>and</u> one SRO in the control room when fuel is in the reactor is wrong.

C is wrong, but plausible. As stated in answer ‘B’ the requirements for fuel in the reactor is only 1 RO in MODE 5.

D is wrong, but plausible. TRM Table 7.2.2-1 requires there to be at least 2 ROs and one SRO for minimum crew manning and does not specify the location (1 RO may rove). Tech Spec 5.2.2b requires one RO to be present in control room.

K/A Match

This question requires the applicant to have the knowledge of conduct of operations requirements.

Technical References:

EN-OP-115, Conduct of Operations, Rev. 19

Tech Specs 5.2.2, Organization, Unit Staff, Amendment No. 183

Tech Specs 7.2.2-1, Administrative Controls

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-PROC, Objective 1.12

Question Source:	**Bank # 552**	2013 NRC Exam Q# 68
(note changes and attach parent)	**Modified Bank #**	
	New	
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.41(b)(10)	
Level of Difficulty:	2	
PRA Applicability:		

Examination Outline Cross Reference	Level	RO
2.1.45 Ability to identify and interpret diverse indications to validate the response of another indication.	Tier	3
	Group #	N/A
	K/A	2.1.45
	Rating	4.3
	Revision	2

Revision Statement:

Rev. 1: Removed first sentence of stem.

Rev. 2: Changed (1) to state "confirm mis-operation"

Question: 68

(1) According to Procedure EN-OP-115, Conduct of Operations, what is the MINIMUM number of independent indications required to be checked to confirm mis-operation of 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**

Explanation:

Procedure EN-OP-115, Conduct of Operations step 5.4 [1] states;

[1] Do not override an automatic initiation of a safety function unless one of the following conditions exists:

- Adequate core cooling is assured by at least two independent indications.
- Mis-operation in automatic mode is confirmed by at least two independent indications.

- Required by procedures.

EN-OP-200, Plant Transient Response Rules, step 5.2 [2] states,

Annunciators are used during events to guide operators and help establish priorities. It is understood that not all annunciator response procedures are consulted during complex casualties and that abnormal and emergency operating procedures direct operator response under these circumstances.

Distracters:

A is wrong. 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.

C is wrong. 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 EN-OP-115, Conduct of Operations step 5.4 [1] 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.

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

K/A Match

This question requires the applicant to have the knowledge of conduct of operations requirements to recognize reliable diverse indications to validate abnormal plant conditions.

Technical References:

02-S-01-43, Transient Mitigation Strategy, Rev. 3, steps 6.1.7 & 6.1.8

EN-OP-115, Conduct of Operations, Rev. 19, steps 5.4 [1] and 4.0 [1]

EN-OP-200, Plant Transient Response Rules, Rev. 3, step 5.1 [2]

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-PROC, Objective 1.5

Question Source:	**Bank #**	
(note changes and 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	
PRA Applicability:		

Examination Outline Cross Reference	Level	RO
2.2.12 Knowledge of surveillance procedures.	Tier	3
	Group #	N/A
	K/A	2.2.12
	Rating	3.7
	Revision	0
Revision Statement:		

Question: 69

In GGNS surveillance procedures, steps marked with a ______ are required to be completed for Technical Specification Acceptance Criteria.

A. $

B. #

C. ✪

D. I

Answer: A
Explanation: See GGNS procedure 01-S-02-10 (Procedure Format and Content), section 5.19 definitions for each of these 4 symbols.
Distracters: B is wrong. This symbol is used for steps requiring initials or data to be recorded. C is wrong. This symbol denotes inter-procedural Tech Spec Logic System Functional Test overlap points. D is wrong. This symbol denotes items associated with Inservice Testing Acceptance Criteria. All distracters are plausible due to all are symbols that are used in surveillance testing procedures.
K/A Match This question requires the applicant to have the knowledge of symbols used in surveillance procedures.
Technical References:

01-S-02-10, Procedure Format and Content, Rev. 4		
Handouts to be provided to the Applicants during exam: NONE		
Learning Objective: GLP-OPS-PROC, Objective 41.9		
Question Source:	**Bank # 371**	2011 NRC Exam Q# 75
(note changes and attach parent)	**Modified Bank #**	
	New	
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.41(b)(10)	
Level of Difficulty:	2	
PRA Applicability:		

Examination Outline Cross Reference	Level	RO
2.2.14 Knowledge of the process for controlling equipment configuration or status.	Tier	3
	Group #	N/A
	K/A	2.2.14
	Rating	3.9
	Revision	2

Revision Statement:

Rev 1: Added noun name of P21 system
Changed LOD to 3

Rev 2: All caps on the word 'only' in answers A and B

Question: 70

During an outage, the decon crew on the refuel floor has requested the use of an additional P21, Makeup Water Treatment System, test connection to supply water to a header for decon purposes.

Which of the following is required for the use of the P21 test connection?

A. CPC tag, ONLY

B. SRO approval, ONLY

C. SRO approval and CPC Sheet

D. SRO approval and prevention of cross contamination

Answer: **D**

Explanation:

Per 02-S-01-37, 6.1.6, P21, P52, and P66 service connection valves ONLY are exempted from the configuration control requirements of this procedure IF exemption is approved by an on-shift SRO. The SRO exempting these requirements Must ENSURE measures are taken to prevent P21, P52 and P66 systems from being contaminated.

Distracters:

All distractors are combinations of component control per 02-S-01-37, and as stated above in the explanation, the listed systems only require SRO approval.

K/A Match

This question requires the applicant to have the knowledge of system alignment requirements.

Technical References:

02-S-01-37, Component Position Control, Rev. 11, step 6.1.6

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-PROC, Objective 34.5

Question Source:	**Bank # 255**	X
(note changes and 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	
PRA Applicability:		

Examination Outline Cross Reference	Level	RO
2.2.43 Knowledge of the process used to track inoperable alarms.	**Tier**	3
	Group #	N/A
	K/A	2.2.43
	Rating	3.0
	Revision	0
Revision Statement:		

Question: 71

One input to a multiple input annunciator has failed and is causing a nuisance alarm.

This single signal to the annunciator has been bypassed.

How should the associated annunciator be identified?

A. No identification is required for the alarm window.

B. Place two vertical lengths of red tape on the alarm window.

C. Place a length of red tape diagonally across the alarm window.

D. Place two lengths of red tape diagonally across the alarm window to form an ‘X’.

Answer: B
Explanation: Per 02-S-01-25, 6.3.5, If an annunciator input is bypassed, the Annunciator window is marked as follows: Two vertical lengths of red tape are applied to distinguish an annunciator with a bypassed input.
Distracters: A is wrong, but plausible. Per 02-S-01-25, 6.3.5, If an annunciator input is bypassed, the Annunciator window is marked as follows: Two vertical lengths of red tape are applied to distinguish an annunciator with a bypassed input C is wrong, but plausible. Annunciators that are **<u>NOT</u>** functioning properly due to faulty alarm circuitry **IDENTIFY** the alarm window that is **<u>NOT</u>** functioning properly by **PLACING** a length of red tape diagonally across the window, D is wrong, but plausible. **<u>IF</u>** Shift Supervision decides to have the annunciator card removed, **<u>THEN</u> DOCUMENT** action on the Annunciator Status Checksheet. A second length of red tape is placed diagonally across the annunciator window to form an 'X' as shown on Attachment IV.

K/A Match This question requires the applicant to have the knowledge of inop alarms.		
Technical References: 02-S-01-25, Deficient Equipment Identification, Rev. 19, step 6.3.5		
Handouts to be provided to the Applicants during exam: NONE		
Learning Objective: GLP-OPS-PROC, Objective 33.4		
Question Source:	**Bank # 612**	2013 NRC Exam Q# 71
(note changes and attach parent)	**Modified Bank #**	
	New	
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.41(b)(10)	
Level of Difficulty:	2	
PRA Applicability:		

Examination Outline Cross Reference	Level	RO
2.3.13 Knowledge of radiological safety procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.	**Tier**	3
	Group #	N/A
	K/A	2.3.13
	Rating	3.4
	Revision	1
Revision Statement: **Unsat, new question.**		

Question: 72

Per EN-RP-101, Access Control For Radiologically Controlled Areas, who is **required** to give the approval for personnel to enter a Locked High Radiation Area (LHRA) with general area dose rates greater than 1.5 Rem/hr?

A. Operations Manager

B. Radiation Protection Manager

C. Radiation Protection Supervisor

D. General Manager, Plant Operations

Answer: B
Explanation: See EN-RP-101, Access Control For Radiologically Controlled Areas, Section 5.5[10], 6th bullet. "Radiation Protection Manager's approval for entry into LHRA's with general area dose rates greater than 1.5 Rem/hr in the actual work area." 'B' is correct
Distracters: Distracters are all as plausible, to the RO Applicant who has never attended much to this procedure and who can only recall small amounts of information about LHRA access described in Administrative Controls Section 5.7 of GGNS Tech Specs and all distracters are listed within EN-RP-101 for different responsibilities or hold a position for decisions on rad control.
K/A Match

This question requires the applicant to have the knowledge of radiological safety procedures to enter a Locked High Radiation Area (LHRA)

Technical References:

EN-RP-101, Access Control For Radiologically Controlled Areas, Rev 13, Section 5.5[10], 6th bullet

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-PROC Obj 50

Question Source:	**Bank #**	5/2017 NRC Exam Q# 73
(note changes and 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	
PRA Applicability:		

Examination Outline Cross Reference	Level	RO
2.3.14 Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities.	**Tier**	3
	Group #	N/A
	K/A	2.3.14
	Rating	3.4
	Revision	1
Revision Statement: **Editorial changes in stem and answers** **Moved part of all answers to stem and made a fill in blank.**		

Question: 73

According to 10 CFR 20, an accessible area with radiation levels of _______ must be posted as a High Radiation Area.

A. 2.5 mrem/hr

B. 50 mrem/hr

C. 150 mrem/hr

D. 500 rads/hr

Answer: C

Explanation:

Definitions found in 10CFR20.1003

Radiation area means an area, accessible to individuals, in which radiation levels could result in an individual receiving a dose equivalent in excess of 0.005 rem (0.05 mSv) in 1 hour at 30 centimeters from the radiation source or from any surface that the radiation penetrates.

150 mrem/hr is greater than the limit of 100 mr/hr for a high radiation area but less than the 500 rads limit for a very high radiation area.

Distracters:

A is wrong, but plausible. 2.5 mrem/hr is less than the limit for a radiation area at 5 mrem/hr, therefore no postings are required.

B is wrong, but plausible. 50 mrem/hr is greater than the limit for a radiation area at 5 mrem/hr, but below the limit for a high radiation area of 100 mrem/hr.

D is wrong, but plausible. 500 rads exceeds the requirement for a very high radiation area.

K/A Match

This question requires the applicant to know what radiation levels will be present in an area posted as High Radiation. The applicant must be able to distinguish between a Radiation Area, High Radiation Area, and a Very High Radiation Area.

Technical References:

10CFR20.1003

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-CFR01 Obj. 6.2

Question Source:	**Bank # 551**	X
(note changes and 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	

PRA Applicability:

Examination Outline Cross Reference	Level	RO
2.4.12 Knowledge of general operating crew responsibilities during emergency operations.	Tier	3
	Group #	N/A
	K/A	2.4.12
	Rating	4.0
	Revision	2

Revision Statement:

Rev 1Changed answers to a 2X2.

Rev 2 All caps on the word "all"
Added ONLY to answers C and D

Question: 74

IAW EN-OP-200, Plant Transient Response Rules, which of the following alarm responses is used by control room personnel during emergency/abnormal conditions?

A. Announce ALL alarms and refer to ALL ARIs

B. Announce ALL alarms, consulting ARIs not required

C. Announce ONLY those alarms that are of significance and refer to ALL ARIs

D. Announce ONLY those alarms that are of significance, consulting ARIs not required

Answer: **D**

Explanation:

EN-OP-200, Plant Transient Response Rules, step 5.2 [2] (b & c), "During emergency /abnormal conditions, alarms are silenced/acknowledged as soon as practical so as not to interfere with the transient response. The announcement of transient alarms during abnormal/emergency operating procedures is not required. In such cases, the operators are expected to announce those alarms that are of significance to the implementation of the applicable abnormal or emergency operating procedure."

"Annunciators are used during events to guide operators and help establish priorities. It is understood that not all annunciator response procedures are consulted during complex casualties and that abnormal and emergency operating procedures direct operator response under these circumstances."

Distracters:

A is wrong, but plausible. Announce only significant alarms and ARIs are not required to be consulted.

B is wrong, but plausible. Announce only significant alarms.

C is wrong, but plausible. ARIs are not required to be consulted**.**

K/A Match

This question requires the applicant to have the knowledge of transient alarm response during abnormal/emergency procedure usage.

Technical References:

EN-OP-115-08, Annunciator Response, Rev. 4

EN-OP-200, Plant Transient Response Rules Rev. 3

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-PROC, OBJ. 1F.1 & 2.1

Question Source:	**Bank #**	
(note changes and 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	
PRA Applicability:		

Examination Outline Cross Reference	Level	RO
2.4.34 Knowledge of RO tasks performed outside the main control room during an emergency and the resultant operational effects.	Tier	3
	Group #	N/A
	K/A	2.4.34
	Rating	4.2
	Revision	3

Revision Statement:

Rev 1: Reduced amount in answers for ease of reading.

Rev 2: Changed the wording of each question and answers

Rev 3: Minor editorial and format changes, LOD to 3

Question: 75

A Reactor Operator is sent to Remote Shutdown Panel (RSDP) and places Transfer Switch for Lockout Transfer Relay (C61-HSS-M150) at 1H22-P152 to the ON position.

(1) This action is required to be performed when the Control Room is abandoned due to a _______________.

And

(2) What RSDP components are electrically separated as a result of this action?

A. (1) Fire or Security Threat
(2) Div 1 ONLY

B. (1) Fire ONLY
(2) Div 1 and Div 2

C. (1) Fire ONLY
(2) Div 1 ONLY

D. (1) Fire or Security Threat
(2) Div 1 and Div 2

Answer:	A

Explanation:

Per 05-1-02-II-1, ONEP

To provide instructions for shutting down the plant from outside the Control Room AND bringing the plant to a Cold Shutdown condition from the Remote Shutdown Panel (RSP). Section 3.1 addresses shutdown from the remote shutdown panel considering a Control Room fire defeating all equipment except Division 1 Safe Shutdown equipment.

Section 3.2 addresses shutdown from the remote shutdown panel considering normal availability of plant equipment with no Control Room fire.

However,

3.2.5 IF Security Threat has occurred, THEN PERFORM the following:

a PERFORM Attachment III to lineup DIV 1 Electrical equipment for the Control Building/Remote Shutdown Panels. (DIV 1 handswitch lineup effects are listed in Attachment XIX.)

Placing any of the Emergency Transfer switches to LOCAL Will cause loss of Control Room indication AND control. The Div 1 SRV Relief function AND Lo-Lo Set function for SRVs on Remote Shutdown panel Will be disabled. REFER to Attachment XIX for effects of handswitch manipulations.

Distracters:

All distracters are wrong, but plausible, because they are combinations of incorrect actions and results.

K/A Match

This question requires the applicant to have the knowledge of actions outside the control room and the results of this action.

Technical References:

05-1-02-II-1, Shutdown From the Remote Shutdown Panel ONEP, Rev. 49

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-ONEP Obj. 23 & 24

Question Source:	**Bank #**	
(note changes and 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	
PRA Applicability:		

NRC INITIAL LICENSED OPERATOR WRITTEN EXAMINATION
OPEN-REFERENCES TABLE OF CONTENTS
RO EXAM

TAB	PROVIDED HANDOUTS
1	05-1-02-III-3, Reduction In Recirculation Flow Rate ONEP (Rev 115), Figure 1, Power/Flow Map for Normal Feedwater Temperature
2	05-1-02-II-1, Shutdown From The Remote Shutdown Panel ONEP (Rev 049), Attachments I and II
3	05-S-01-EP-1, Emergency/Severe Accident Procedure Support Documents (Rev 036), Figures (Figure 1)
4	05-S-01-EP-1, Emergency/Severe Accident Procedure Support Documents (Rev 036), EP/SAP Cautions (Caution 1 Only)

Additional References Provided Outside of This Binder:

- Steam Tables

GGNS LOT 12/2017 **NRC** INITIAL LICENSED OPERATOR WRITTEN EXAMINATION

SRO EXAM

ANSWER KEY

76	C
77	B
78	A
79	D
80	C
81	A
82	B
83	A
84	C
85	D
86	B
87	A
88	D
89	D
90	B
91	A
92	C
93	D
94	B
95	B
96	A
97	B
98	C
99	C
100	B

Examination Outline Cross Reference	Level	SRO
295003 Partial or Complete Loss of AC A2 Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER: AA2.04 System lineups	Tier	1
	Group #	1
	K/A	295003
	Rating	3.7
	Revision	1
Revision Statement: **Re-ordered answers lo to hi** **Added page numbers to Handout material**		

Question: 76

HANDOUT PROVIDED

The plant is operating at rated thermal power.

- Div 1 D/G outage started 10 days ago and was declared INOP at that time
- RHR A was just started in Suppression Pool Cooling
- ESF Transformer 21 tripped on sudden pressure

With current conditions and system lineups, what is the **most limiting** remaining LCO completion time?

A. 12 hours

B. 72 hours

C. 4 days

D. 7 days

Answer:	**C**
Explanation: First the applicant must realize that the current LCO is 3.8.1 action B4, which is a 14 day LCO for one D/G inop and 3.5.1 Condition A was entered with a 7 day LCO due to RHR 'A' in suppression pool cooling.	

Then the applicant must understand what is lost. ESF transformer 21 powers ESF buses 16AB and 17AC, not Div 1 15AA, therefore Div 1 did not lose any power.

Div 2 and 3 Diesel Generators started and energized their respective buses.

There was no loss of any ECCS systems, therefore 3.5.1 is not re-entered.

The current LCO is 10 days into a 14 day completion time, there is 4 days left on the LCO therefore no other LCOs are entered.

Distracters:

A is wrong. This answer is Condition D, One required offsite circuit inop and one required D/G inop. The LCO requires two qualified circuits, GGNS has three qualified circuits and only two of the three is required so a loss of one will not require an entry into Tech Specs.

B is wrong. If the applicant believes that ESF 21 transformer supplies power to Div 1 15AA bus then a loss of LPCS and RHR A will occur, then LCO 3.5.1 condition C would be entered for 2 ECCS system inop. Since Div 1 15AA never lost power this is not true.

D is wrong. If the applicant believes only one ECCS system is lost.

K/A Match

The applicant must first determine the system lineups after a loss of power to one of the 3 ESF buses. Then they must interpret the remaining systems and determine Tech Spec time limits.

Technical References:

Tech Specs 3.5.1, Emergency Core Cooling Systems (ECCS) and Reactor Core Isolation Cooling (RCIC) System, Amendment 175, 201

Tech Specs 3.8.1, Electrical Power Systems, Amendment 175, 201

Handouts to be provided to the Applicants during exam:

Tech Specs 3.5.1, Pages 3.5-1 thru 3.5-3
Tech Specs 3.8.1, Pages 3.8-1 thru 3.8-5

Learning Objective:

GLP-OPS-TS001, OBJ. 35 & 40

Question Source:	**Bank #**	
(note changes and 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 the applicant to determine LCO completion times and to determine system lineups after a loss of power.**		
PRA Applicability:		
RHR is listed as #3 on System Importance to CDF. ESF (R20) is listed as #4 on System Importance to CDF. Div 1 & 2 EDGs is listed as #11 on System Importance to CDF.		

<table>
<tr><td rowspan="1">Examination Outline Cross Reference</td><td>Level</td><td>SRO</td></tr>
<tr><td rowspan="6">295005 Main Turbine Generator Trip

AA2 Ability to determine and/or interpret the following as they apply to MAIN TURBINE GENERATOR TRIP:

AA2.04 Reactor pressure</td><td>Tier</td><td>1</td></tr>
<tr><td>Group #</td><td>1</td></tr>
<tr><td>K/A</td><td>295003</td></tr>
<tr><td>Rating</td><td>3.7</td></tr>
<tr><td>Revision</td><td>2</td></tr>
<tr><td colspan="2"></td></tr>
<tr><td colspan="3">Revision Statement:

Rev 1: Grammatical errors on capitalization & removed periods from answers C and D.
Added clarity to distractor description.
Changed 10 CFR # to 5

Rev 2: Removed ‘psig’ from first number in answers and caps in answers C and D</td></tr>
</table>

Question: 77

Main Turbine Generator trip occurred due to Turbine Thrust Bearing trip.

An operator on Elevation 166’ Turbine Building reported rubbing and abnormal noises coming from the turbine thrust bearing area.

Actions to mitigate the report have been performed.

(1) Which of the following should the CRS direct the RO to use to control reactor pressure in accordance with EP-2?

AND

(2) What pressure band should the CRS direct?

A. (1) SRV’s ONLY
(2) 450 to 600 psig

B. (1) SRV’s ONLY
(2) 800 to 1060 psig

C. (1) Turbine Bypass Valves and Main Steam Line drains
(2) 450 to 600 psig

D. (1) Turbine Bypass Valves and Main Steam Line drains
(2) 800 to 1060 psig

Answer: **B**
Explanation: Per 05-1-02-I-2, Turbine and Generator Trip ONEP step 3.5, If trip was due to TURB THR BRG TRIP, then dispatch an operator to EL 166' Turbine Building to -monitor for evidence of rubbing or other abnormal noises. If observing abnormal noises from either thrust bearing area or turbine casings, then perform the following to allow Turbine to coast down rapidly to full stop immediately break condenser vacuum and prevent turning gear operation. Those steps will cause loss of reactor feed pumps, condenser bypass valves, and steam jet air ejectors. Reactor level must be controlled using RCIC, CRD or ECCS systems. Reactor pressure must be controlled using SRV's. Pressure band should NOT be lowered to 450 to 600 psig due to RCIC and/or ECCS can provide level control.
Distracters: A is wrong, but plausible, pressure band should not be lowered unless level cannot be controlled with RCIC/HPCS. Per 02-S-01-43 step 6.6.6d, page 15, states: Condition, **<u>IF</u>** level can be maintained **<u>AND</u>** controlled using available systems **<u>AND</u>** manual pressure control is required, Normal pressure band 800 to 1060 psig. RCIC and/or HPCS is still available for level control. C is wrong, but plausible, turbine bypass valves and steam line drains are unavailable, bypass valves will close on a condenser vacuum of 12"Hg Vac and the MSIVs will close at 9" Hg Vac isolating the steam. Pressure band should not be lowered unless level cannot be controlled. Per 02-S-01-43 step 6.6.6 states: Condition, **<u>IF</u>** level can be maintained **<u>AND</u>** controlled using available systems **<u>AND</u>** manual pressure control is required, Normal pressure band 800 to 1060 psig. RCIC and/or HPCS is still available for level control. D is wrong, but plausible, turbine bypass valves and steam line drains are unavailable, bypass valves will close on a condenser vacuum of 12"Hg Vac and the MSIVs will close at 9" Hg Vac isolating the steam. Pressure band should not be lowered unless level cannot be controlled.
K/A Match The applicant must first determine correct pressure control strategy for abnormal turbine trip.
Technical References: 05-1-02-I-2, Turbine and Generator Trips ONEP, Rev. 37, step 3.5 02-S-01-43,Transient Mitigation Strategy, Rev. 3, step 6.6.6d, page 15
Handouts to be provided to the Applicants during exam: **NONE**
Learning Objective:

GLP-OPS-ONEP, OBJ. 10 GLP-OPS-EP01, OBJ. 7		
Question Source:	**Bank #**	
(note changes and 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 the applicant to implement the EPs and direct correct pressure control strategy.**		
PRA Applicability:		
Failure to manually depressurize with ADS/SRVs is ranked #1 in Operator Action Importance to CDF.		

Examination Outline Cross Reference	Level	SRO
295006 SCRAM 2.2.25 Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits.	**Tier**	1
	Group #	1
	K/A	295006
	Rating	4.2
	Revision	3
Revision Statement: **Rev 1: Editorial changes** **Changed K/A to 2.2.25** **Rev 2: Editorial changes in stem and answers** **Rev 3: Changed LOK from H to F**		

Question: 78

In Mode 1:

Tech Spec 3.1.5, Control Rod Scram Accumulators, actions require inserting a manual scram within 20 minutes if charging water header pressure is < 1520 psig and two control rod scram accumulators associated with withdrawn control rods are inoperable.

What is the basis for the allowed completion time of 20 minutes for this action?

A. Allow time to start a CRD pump.

B. Allow time to recharge inoperable scram accumulators.

C. Allow time to drain water from inoperable scram accumulator instrument blocks.

D. Allow time to fully insert and disarm control rods associated with inoperable scram accumulators.

Answer: A
Explanation: TS 3.1.5 Action B.1 assumes no CRD pumps are running if charging water header pressure is <1520 psig. The completion time for Action B starts when 2 or more scram accumulators for withdrawn control rods are declared inoperable concurrently with charging water header pressure < 1520 psig. If the

condition cannot be corrected within 20 minutes, Action B.1 is not met and Action D must be entered, which requires placing the Reactor Mode Switch in SHUTDOWN immediately. The **TS Bases** states the 20 minute completion time provided for Action B.1 should be adequate for starting a CRD pump.

Distracters:

B and C are wrong, but plausible because accumulator low pressure and water detection in the accumulator instrument block both cause the CRD accumulator trouble alarm, required operable by TR 3.1.5. But both answers are wrong. Moisture in the accumulator instrument block does not directly require declaring the accumulator inoperable. Low pressure does require individual accumulators to be declared inoperable, but this is addressed by TS 3.1.5 Actions B.2.1 and B.2.2.

D is wrong, but plausible because inserting and disarming a control rod would be necessary if the rod was declared inoperable, which is an alternative to declaring a control rod “slow” when its accumulator is inoperable. However, this is wrong because TS 3.1.5 Action B does not address urgency for inserting/disarming a control rod, only for restoring a CRD pump to operation.

K/A Match

The applicant must first determine the <1 hour time limit and what actions are required and what is the bases for that action.

Technical References:

TS 3.1.5, Control Rod Scram Accumulators, Amendment No. 120

Bases TS 3.1.5 Action B.1, Revision N0. 1

TS 3.1.3, Control Rod Operability, Amendment No. 142

TR 3.1.5

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-TS001, OBJ. 39

Question Source:	**Bank # 692**	2013 NRC Exam Q# 77
(note changes and attach parent)	**Modified Bank #**	
	New	
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.43(b)(2)	
Level of Difficulty:	2	

SRO Only Justification: **This question requires the applicant to have knowledge of Tech Spec bases from memory.**
PRA Applicability:

Examination Outline Cross Reference	Level	SRO
295021 Loss of Shutdown Cooling AA2 Ability to determine and/or interpret the following as they apply to LOSS OF SHUTDOWN COOLING AA2.03 Reactor water level	Tier	1
	Group #	1
	K/A	295021
	Rating	3.5
	Revision	2

Revision Statement:

Rev1: Editorial changes, changed considered to "available", moved second to third bullet.
Added sections of SOPP to given to the applicants.
Changed LOD to 4
Changed 10 CFR # to 5

Rev 2: Corrected Revision number of reference material, SOPP

Question: 79

HANDOUT PROVIDED

Given the following conditions:

- Plant in Mode 4 using 03-1-01-3, Plant Shutdown IOI
- A spurious Reactor Water level of +11.4 inches (Level 3) occurs on NS4, and **cannot** be reset
- RWCU system not available as a decay heat removal system

What is the risk level color now assigned to the "decay heat removal capability" Safety Function?

A. Green

B. Yellow

C. Orange

D. Red

Answer: D
Explanation: The applicant must recognize that with a level 3 signal on NS4 causes a Group 3 (RHR SDC) isolation,

the common suction path for shutdown cooling is isolated and cannot be restored, therefore RHR ‘A’, ‘B’ and ADHR are unavailable for decay heat removal in Mode 4. Level 3 is 11.4” RPV level and will cause a Group 3 (RHR Shutdown Cooling isolation).

See the Shutdown Operations Protection Plan (SOPP) Rev. 20. Stem conditions indicate that the SHUTDOWN CONDITION 1 portion of Section V applies here. See SOPP page 17, where Decay Heat Removal (SDC) section, specifically its right-most box, states Three available would be Green, Two available would be Yellow, One available would be Orange and Zero available would be Red.

Since RWCU has not been demonstrated or calculated then ZERO Decay Heat Removal Systems are available, therefore, correct color is RED.

Distracters:

All answers are plausible since they are the other colors used for shutdown cooling (SDC) condition per SOPP.

K/A Match

The applicant must first interpret RPV level that being below 11.4” causes a Group 3 SDC isolation causing all RHR and ADHR system unusable for decay heat removal.

Technical References:

SOPP, Shutdown Operations Protection Plan, Rev. 20

EN-OU-108, Shutdown Safety Management Program, Rev. 8

Handouts to be provided to the Applicants during exam:

SOPP, Shutdown Operations Protection Plan, Rev. 20, Sections IV & V, Pages 10 thru 33

Learning Objective:

GLP-OPS-PROC, OBJ. 29.7

Question Source:	**Bank #**	
(note changes and 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:

Application of SOPP is an SRO responsibility.
PRA Applicability:
RHR is listed as #3 on System Importance to CDF.

Examination Outline Cross Reference	Level	SRO
295023 Refueling Accidents 2.4.2 Knowledge of system set points, interlocks and automatic actions associated with EOP entry conditions.	**Tier**	1
	Group #	1
	K/A	295023
	Rating	4.6
	Revision	3

Revision Statement:

Rev 1: Per NRC: Changed stem to add "priority" and added second part of first direction. Changed first part to include ONEP only.

Rev 2: Editorial changes to stem
Changed radiation monitors name to EP-4 Table SC-1 description
Changed 10 CFR # to 4

Rev 3: Editorial changes to stem and reworded question (2).

Question: 80

Refueling is in progress using IOI procedure 03-1-01-5, Refueling.

- While using Fuel Handling Bridge, a spent fuel bundle stuck in the full UP position.
- Due to a leak in the FPCCU system, Spent Fuel Pool level lowers to below the skimmers.

RO reported:

- AUX BLDG FHA POOL EXHAUST is 37 mr/hr and rising
- AUX BLDG FHA VENT EXHAUST is 3 mr/hr and rising

(1) Which of the following procedure(s) is/are the highest priority for the CRS to transition?

AND

(2) What action(s) will the CRS direct?

A. (1) EP-4 ONLY
(2) Verify SBGT initiation

B. (1) ONEP High Radiation During Fuel Handling ONLY

(2) Add water to the Spent Fuel Pool

C. (1) EP-4 and ONEP High Radiation During Fuel Handling
(2) Verify SBGT initiation and evacuate Refuel Floor

D. (1) ONEP High Radiation During Fuel Handling and SOI Fuel Pool Cooling and Cleanup ONLY
(2) Evacuate Refuel Floor and add water to Spent Fuel Pool

Answer: **C**

Explanation:

With actions currently being controlled by the Refueling IOI, a report of high radiation on the refuel floor should cause the CRS to enter ONEP High Rad During Fuel Handling:

4.0 SYMPTOMS

4.1 High radiation alarms indicating abnormal high radiological conditions in areas where fuel is being handled.

4.2 Notified by RP of abnormal radiological conditions.

4.3 Local CAM alarm indicating abnormally high radiological conditions.

4.4 Unexplained lowering of Containment **OR** Spent fuel pool water levels.

The applicant must recognize that Fuel Pool Sweep Vent rad has exceeded the entry conditions for EP-4:

Per Table SC-1 AUX BLDG FHA POOL EXHAUST (P601-19A-B10/C10) alarm is received at 30 mr/hr and that meets the Operating limit on table SC-1.

With the given information, the CRS should transition from IOI 03-1-01-5 to EP-4 and ONEP High Rad During Fuel Handling and recognize the automatic actions associated with EP-4 entry by verifying SBGT auto initiation. The CRS will also direct actions to mitigate the events and evacuate the refuel floor from EP-4 and ONEP.

Distracters:

A is wrong. Yes, it is true the CRS will enter EP-4, but the ONEP holds the actions that are required to mitigate the events. The CRS should recognize that both EP and ONEP needs to be entered.

B is wrong. The CRS will enter the ONEP, but, EP-4 must be entered also. Plausible if the applicant does not recognize entry into EP-4.

D is wrong. The CRS will enter the ONEP, but, EP-4 must be entered also. The SOI will also be entered from steps in the ONEP to raise level in the pool. Plausible if the applicant does not recognize entry into EP-4.

K/A Match

The applicant must recognize that the auto actions and setpoints of high rad for fuel floor ventilation have exceeded and EP-4 should be entered.

Technical References:

03-1-01-5, Refueling IOI, Rev. 137

05-1-02-II-8, High Radiation During Fuel Handling ONEP, Rev. 16

05-S-01-EP-4, Auxiliary Building Control EP, Rev. 29

04-1-01-G41-1, Fuel Pool Cooling and Cleanup SOI, Rev. 77

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-ONEP, OBJ. 14

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.43(b)(4)	
Level of Difficulty:	3	

SRO Only Justification:

This question requires the applicant to recognize entry into the High Rad During Fuel Handling ONEP, but, also recognize auto actions of high rad in the ventilation system. This high rad is also an entry condition into EPs. This action will cause a transition from the ONEP to the EP while maintaining use of the ONEP. This action is a CRS decision process and knowledge.

PRA Applicability:

Examination Outline Cross Reference	Level	SRO
295027 High Containment Temperature EA2 Ability to determine and/or interpret the following as they apply to HIGH CONTAINMENT TEMPERATURE (MARK III CONTAINMENT ONLY) EA2.01 Containment temperature: Mark-III	Tier	1
	Group #	1
	K/A	295027
	Rating	3.7
	Revision	2
Revision Statement: **Rev. 1: Changed to "Given the following"** **Editorial changes** **Changed answers to actions** **Rev. 2: Changed reference material to state Caution 1.**		

Question: 81

HANDOUT PROVIDED

Given the following:

- Reactor pressure is 100 psig and lowering slowly
- CTMT pressure is 4.2 psig and stable
- Average CTMT temperature is 190 °F and rising slowly
- DW temperature at elevation 166' is 210 °F and rising slowly
- Suppression Pool level is 17.5 feet and stable
- Suppression Pool temperature is 160 °F and rising slowly
- DW hydrogen is 2.0% and stable

Which of the following actions will the CRS direct IAW the EPs?

A. Initiate CTMT Sprays

B. Open 8 ADS valves

C. Exit the EPs and enter the SAPs

D. Conclude that one or more RPV level instruments may be unreliable

Answer: **A**

Explanation:

Given the combination of CTMT temperature and CTMT pressure, CSIPL Figure 3 clearly shows we're in the "Safe to Initiate" region. Initiate CTMT spray is the appropriate action per the CTMT pressure leg of EP-3.

Distracters:

B is wrong - When comparing supp pool level against CTMT pressure, there is no reason to open 8 ADS valves for PSP (Figure 4). When considering the combination of supp pool level/reactor pressure/supp pool temperature, there is no reason to open 8 ADS valves for HCTL (Figure 1).

C is wrong - We don't leave the EPs and enter the SAPs for hydrogen until it's above 2.9%.

D is wrong - When comparing DW temp at el. 166' against reactor pressure there is no reason to conclude that any level instruments may be unreliable because of possible boiling (RPVST, EP Caution 1 Figure 2).

K/A Match

The applicant must first determine the actions necessary for the elevated Containment temperatures.

Technical References:

05-S-01-EP-1, Emergency/Severe Accident Procedure Support Documents, Rev. 36, Figures and EP-1 EP/SAP Caution 1

Handouts to be provided to the Applicants during exam:

05-S-01-EP-1, Emergency/Severe Accident Procedure Support Documents, Rev. 36, Figures and EP-1 EP/SAP Caution 1

Learning Objective:

GLP-OPS-EP003, OBJ. 22

Question Source:	**Bank # 207**	X
(note changes and attach parent)	**Modified Bank #**	
	New	
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 the applicant to determine EP transition point.**
PRA Applicability:
Failure to initiate SPC and Containment Spray is ranked #2 on Operator Action Importance to CDF.

Examination Outline Cross Reference	Level	SRO
600000 Plant Fire On Site 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.	Tier	1
	Group #	1
	K/A	600000
	Rating	4.1
	Revision	2
Revision Statement: **Rev 1: Added procedure number of handout material.** **Added "s" to answers C and D to make plural.** **Rev 2: Changed reference material attached to show source document.**		

Question: 82

HANDOUT PROVIDED

A fire has been reported in the 15AA switchgear room.

12 minutes later, Fire Brigade leader reports the automatic fire suppression system has extinguished the fire, but there is a lot of damage to the 15AA bus.

Per 01-S-06-5, Reportable Events or Conditions, what is the maximum time allowed to provide notification to the NRC?

Within:

A. 15 minutes

B. 1 hour

C. 4 hours

D. 8 hours

Answer: B
Explanation: Per 10-S-01-1, Activation of the Emergency Plan, HA4, Fire or Explosion affecting the operability of plant safety systems required to establish or maintain safe shutdown. Fire or Explosion resulting in visible damage to any of the structures or areas in Table H2 containing

safety systems or components or Control Room indication of degraded performance of those safety systems.

Therefore an ALERT should be declared due to damage visible to a safety system in Table H2 which is the Control Building.

Per 01-S-06-5, Reportable Event or Conditions, Attachment III, page 4 of 13, I.2; Immediate Notification (within one hour of occurrence) The declaration of **ANY** of the emergency classes specified in the licensee's approved emergency plan. 50.72(a)(1)(i)

Per 10-S-01-6, Notification of Offsite Agencies and Plant On-Call Emergency Personnel, 6.1.1b NOTE states: The NRC shall be notified of the declaration of the Emergency IMMEDIATELY AFTER THE NOTIFICATION OF STATE AND LOCAL AGENCIES and not later than one hour after the Emergency declaration. An open channel shall be maintained until terminated by the NRC.

Distracters:

A is wrong. Plausible due to this would be the time limit for notification of state and local agencies after a declaration of an event.

C and D are wrong. Plausible due to these are the other reportable times within the procedure.

K/A Match

The applicant must first determine that an EAL has been exceeded, HA4, 01-S-06-5, Reportable Events or Conditions, states that the declaration of **ANY** of the emergency classes specified in the licensee's approved emergency plan is an immediate notification (within one hour of occurrence).

Technical References:

01-S-06-5, Reportable Events or Conditions, Rev. 111

10-S-01-1, Activation of the Emergency Plan, Rev. 126

10-S-01-6, Notification of Offsite Agencies and Plant On-Call Emergency Personnel, Rev. 55

Handouts to be provided to the Applicants during exam:

EAL Flow Charts, 10-S-01-1, Activation of the Emergency Plan, EPP-01-02 (Flowchart)

Learning Objective:

GLP-OPS-PROC, OBJ. 22

Question Source:	**Bank #**	
(note changes and 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 the applicant to analyze the information given and determine that an EAL has been exceeded and then be able to recognize that it is an immediate notification to the NRC.**		
PRA Applicability:		

Examination Outline Cross Reference	Level	SRO
295007 High Reactor Pressure AA2 Ability to determine and/or interpret the following as they apply to HIGH REACTOR PRESSURE AA2.02 Reactor power	Tier	1
	Group #	2
	K/A	295007
	Rating	4.1
	Revision	**2**
Revision Statement: **Rev 1: Per NRC: Made changes to answers for clarity, added lesson plan information for LO-LO-Set operation.** **Rev 2: Editorial changes to stem. Changed to LOD 4**		

Question: 83

An ATWS is occurring.

The CRS is directing actions per EP-2A.

Reactor pressure peaked at 1125 psig and a band of 926 - 1073 psig is being maintained.

What is the direction of the CRS to control reactor pressure and why?

A. Take manual control of SRVs to prevent power oscillations

B. Take manual control of SRVs to prevent exceeding HCTL

C. Allow Lo-Lo Set to control SRVs to prevent power oscillations

D. Allow Lo-Lo Set to control SRVs to prevent exceeding HCTL

Answer: **A**
Explanation: With the pressure band given the applicant should determine that Lo-Lo Set logic has initiated. Once initiated, six SRVs are capable of operating in the Low-Low Set mode with the following adjusted opening and closing setpoints: One SRV, F051D, lifts at 1033 psig and blows down to 926 psig. A second SRV, F051B, lifts at 1073 psig and blows down to 936 psig.

Per 02-S-01-43, Transient Mitigation Strategy, step 6.2.6, "During EP-2A (ATWS) cycling of SRVs on Lo-Lo set should be avoided since it could result in power oscillations or tailpipe damage."

SRV cycling is terminated by manually opening any SRVs that are either open or cycling on high pressure. This action establishes direct operator control of the SRVs and RPV pressure and precludes further automatic actuation cycles.

Distracters:

B is wrong. The reason for manual control is to prevent power swings not exceeding HCTL.

C and D are wrong. Lo-Lo Set is to be avoided during an ATWS per 02-S-01-43, Transient Mitigation Strategy, step 6.2.6, **DURING** EP-2A (ATWS), cycling of SRVs on Lo-Lo Set should be avoided since it could result in power oscillations **OR** tailpipe damage.

Cycling of SRVs on Lo-Lo Set is preferred **DURING** EP-2 conditions, **PROVIDED** localized heating of Suppression Pool temperature is maintained less than 210˚F, long term cycling of SRVs on Lo-Lo Set should be avoided.

K/A Match

The SRO applicant must determine that any pressure oscillations will cause power transients therefore the need for pressure control is required. The SRO applicant must determine that Lo-Lo set function of the SRVs has initiated and currently controlling reactor pressure due to the pressure band being maintained. Site procedures and mitigation strategy instructs to prevent power transients. DO NOT allow pressure to be controlled by Lo-Lo set function during an ATWS.

Technical References:

02-S-01-40, EP Technical Bases, Rev. 8, Attachment V, page 43 of 58

02-S-01-43, Transient Mitigation Strategy, Rev. 3, step 6.2.6

GLP-OPS-E2202, Automatic Depressurization System (ADS) Lesson Plan, Rev. 9

05-S-01-EP-2, RPV Control, Rev. 45

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-EP02A, OBJ. 7 & 9

QuestioSource:	**Bank #**	
(note changes and 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 the applicant to determine the current pressure control is an automatic system that should be avoided during an ATWS and direct the RO to take manual control of the SRV and control pressure within band manually.**		
PRA Applicability:		
Failure to manually depressurize with ADS/SRVs is ranked #1 in Operator Action Importance to CDF.		

Examination Outline Cross Reference	Level	SRO
295015 Incomplete SCRAM AA2. Ability to determine and/or interpret the following as they apply to INCOMPLETE SCRAM AA2.02 Control rod position	Tier	1
	Group #	2
	K/A	295015
	Rating	4.2
	Revision	0
Revision Statement:		

Question: 84

An ATWS is occurring.

The CRS is directing actions per EP-2A.

Control Rods are being inserted and Standby Liquid Control is injecting.

Which of the following will allow the CRS to transition from EP-2A to EP-2?

A. RPV level indication is unknown.

B. Standby Liquid Control Tank indicates 0 gallons.

C. All control rods are inserted to or beyond position 02.

D. Reactor power is being monitored on the SRMs, with IRMs and APRMs downscale.

Answer: C
Explanation: EP-2A is entered when one or more control rods are withdrawn past position 02 and it has not been determined that the reactor will remain shutdown under all conditions without boron. If all rods are inserted to at least position 02 or it is determined that the reactor will remain shutdown under all conditions without boron while EP-2A is in use, the condition that required entry no longer exists. Boron injection may then be terminated, if in progress, and RPV water level, RPV pressure, and reactor power controlled in accordance with EP-2, RPV Control Positive confirmation that the reactor will remain shutdown under all conditions is best obtained by verifying that all control rods are inserted to or beyond position 02. Position 02 is the “Maximum Subcritical Banked Withdrawal Position,” defined to be the greatest banked rod position at which the reactor will remain shutdown under all conditions.

Distracters:

A is wrong. Per EP-2A if level becomes unknown, EP-5A will be entered not EP-2.

B is wrong. If SLC is initiated, boron injection is continued until the entire contents of the SLC tank have been injected. If available, both pumps are initially operated to increase the injection rate and shorten the time required to inject Cold Shutdown Boron. Once SLC tank level drops to 2000 gal, injection of Cold Shutdown Boron has been completed, permitting initiation of RPV depressurization in Step P-7. One SLC pump may then be tripped to preclude vortexing at the tank suction, theorized to be possible under some conditions with two pumps in operation. The remaining contents of the SLC tank are injected, using a single SLC pump, to provide the design basis concentration margin. When the SLC tank level drops to 0 gal, the remaining SLC pump is tripped to avoid mechanical damage to the pump and preserve availability of the SLC system should operation again be needed. The CRS can't make the decision on the reactor remaining subcritical under all condition without analytical help.

D is wrong. The CRS will direct to monitor Power during an ATWS event, even though power is very low and on the Source Range instrumentation, the CRS can't ensure subcritical under all condition without analytical help.

K/A Match

The applicant must be able to interpret control rod position to determine if EOPs can be exited or transitioned from one to another.

Technical References:

02-S-01-40, EP Technical Bases, Rev. 8, Attachment V, page 6 of 58
05-S-01-EP2, RPV Control, Rev. 45, EP-2A step 2

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-EP02A, OBJ. 6

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.43(b)(6)	
Level of Difficulty:	3	

SRO Only Justification: **This question requires the SRO applicant to recall the requirements to declare EP-2A exit and entry into another EOP by using a retention step in EP-2A.**
PRA Applicability:

<table>
<tr><td>Examination Outline Cross Reference</td><td>Level</td><td>SRO</td></tr>
<tr><td rowspan="6">500000 High CTMT Hydrogen Conc.

2.4.41 Knowledge of the emergency action level thresholds and classifications.</td><td>Tier</td><td>1</td></tr>
<tr><td>Group #</td><td>2</td></tr>
<tr><td>K/A</td><td>500000</td></tr>
<tr><td>Rating</td><td>4.6</td></tr>
<tr><td>Revision</td><td>2</td></tr>
<tr></tr>
<tr><td colspan="3">Revision Statement:

Rev 1: Added “and stable” to stem.
Added procedure numbers to Handout material.
Added another procedure to Reference material.

Rev 2: Added ‘current’ to stem for conditions.
Removed inch symbol from bullets
Removed trends from bullets</td></tr>
</table>

Question: 85

HANDOUT PROVIDED

An event has occurred with the following conditions:

- Reactor water level is -159 inches
- Drywell pressure is 8.4 psig
- Drywell radiation indicates 3125 R/hr
- Drywell Hydrogen concentration is 7.2%
- Containment pressure is 9.1 psig
- Containment Hydrogen concentration is 8.5%

This event should be classified as:

A. Unusual Event

B. Alert

C. Site Area Event

D. General Emergency

Answer: **D**

Explanation:

Per 10-S-01-1, FG1 Loss of any two barriers **and** loss or potential loss of a third barrier would be a General Emergency.

Drywell Radiation >3000 R/hr is loss of Fuel Clad.
Drywell Pressure >1.39 psig with indications of a reactor coolant leak in the drywell is loss of Reactor Coolant.
Containment H2 concentration in HDOL Unsafe Zone is a potential loss of Primary containment.

Distracters:

All distractors are other classifications in the ERO. Other classifications possible if different combination of loss or potential loss of barriers.

K/A Match

The applicant must be able to recognize entry into an EAL and correctly classify the event.

Technical References:

10-S-01-1, Activation of the Emergency Plan, Rev. 126, FG1
05-S-01-EP-1, EP/SAP FIGURES, Figure 5, HDOL CURVE, Rev 36

Handouts to be provided to the Applicants during exam:

EAL Flow Charts, 10-S-01-1, Activation of the Emergency Plan, EPP-01-02 (Flowchart)
05-S-01-EP-1, EP/SAP FIGURES, Figure 5, HDOL CURVE, Rev 36

Learning Objective:

GLP-EP-EPTS6, OBJ. 1

Question Source:	**Bank #**	
(note changes and 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 the applicant to classify an event using the station ERO procedures.

PRA Applicability:

Examination Outline Cross Reference	Level	SRO
203000 RHR/LPCI: Injection Mode G2.1.23: Ability to perform specific system and integrated plant procedures during all modes of plant operations	Tier	2
	Group #	1
	K/A	203000
	Rating	4.4
	Revision	**3**

Revision Statement:
Per NRC: added bullets to stem, changed wording to answers, added attachment 12 and PSP curve to references.
Per Validation comments added noun name of SOI attachment

Rev 2: Editorial changes to stem. Moved water level from bullets.

Rev 3: Editorial changes to stem and made (2) the same in answers A and C
Removed inch symbol

Question: 86

A LOCA is occurring.

- CRS has entered EP-2 and EP-3
- Containment Sprays have been initiated
- Emergency Depressurization in progress
- Containment parameters are in Unsafe Zone of PSP curve

Now Reactor water level is -195 inches and lowering.

(1) How should CRS direct the alignment of RHR 'A' and 'B' systems?

(2) What procedure will be used for the alignment?

A. (1) Realign RHR A and B systems for LPCI injection.
(2) EP-1 Attachment 12, Defeating RHR SDC Injection Valve Isolation Interlock

B. (1) Realign RHR A and B systems for LPCI injection.
(2) RHR SOI 04-1-01-E12-1, Attachment VII, Containment Spray Shutdown

C. (1) Maintain one RHR system in Containment spray until Safe Zone of PSP curve is entered and realign one RHR for LPCI injection

(2) EP-1 Attachment 12, Defeating RHR SDC Injection Valve Isolation Interlock

D. (1) Maintain one RHR system in Containment spray until Safe Zone of PSP curve is entered and realign one RHR for LPCI injection
(2) RHR SOI 04-1-01-E12-1, Attachment VII, Containment Spray Shutdown

Answer:	**B**

Explanation:

EP-3 step PCP-6

"Use only RHR pumps not required for adequate core cooling by continuous injection."

With level at -195" adequate core cooling is no longer assured, therefore, the 2 RHR systems should be redirected to LPCI injection until reactor level is controlled.

Per 02-S-01-40, Attachment VI, page 24 of 37;

"Core cooling takes precedence over spraying the containment in this step. Only RHR pumps not needed for core cooling may be placed in the containment spray mode. It is permissible, however, to alternate RHR pumps between modes if *continuous* operation in the LPCI mode is not required."

This also requires the SRO to transition and direct use of a SOI hard card while directing steps within EP-3.

Distracters:

A is wrong. First part if correct, RHR would be used in the LPCI mode, however, injection would be through the LPCI injection not SDC injection line using EP-1 Attachment 12. Plausible if the applicant doesn't remember the SOI attachment hard card will stop CTMT spray and inject through LPCI.

C is wrong. Core cooling takes precedence over spraying the containment in this step. All RHR would be used in the LPCI mode using the SOI Hard Card to realign for LPCI injection, not injection through the SDC injection line using EP-1 Attachment 12. Plausible if the applicant doesn't remember the SOI attachment hard card will stop CTMT spray and inject through LPCI.

D is wrong. Core cooling takes precedence over spraying the containment in this step.

K/A Match

The SRO applicant must first determine that Adequate Core Cooling is lost then direct both loops of RHR to be realigned for LPCI injection using specific system and integrated plant procedures.

Technical References:

02-S-01-40, EP Technical Bases, Rev. 8, Attachment VI page 24 of 37

05-S-01-EP-3, Containment Control, Rev. 29

04-1-01-E12-1, Residual Heat Removal System SOI, Rev. 147, Att. VII Hard Card

05-S-01-EP-1, Emergency/Severe Accident Procedure Support Documents, Figures, Rev. 36

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-EP-EP3, OBJ. 9

Question Source:	**Bank #**	
(note changes and 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 the SRO applicant to make the determination that adequate core cooling is no longer assured, and make decisions based on EP-3 actions to direct RO to realign/transition the RHR systems from containment spray to inject through LPCI.

PRA Applicability:

RHR is listed as #3 on System Importance to CDF.

<table>
<tr><td>Examination Outline Cross Reference</td><td>Level</td><td>SRO</td></tr>
<tr><td rowspan="6">212000 RPS

A2 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:

A2.11 Main steamline isolation valve closure</td><td>Tier</td><td>2</td></tr>
<tr><td>Group #</td><td>1</td></tr>
<tr><td>K/A</td><td>212000</td></tr>
<tr><td>Rating</td><td>4.1</td></tr>
<tr><td rowspan="2">Revision</td><td rowspan="2">4</td></tr>
<tr></tr>
<tr><td colspan="3">Revision Statement:
Rev 1: Per NRC: Removed “immediate actions performed”, changed from 2 questions to one and added “why”. Added more information in explanation.
Per validation comments: added “no subsequent actions have been performed.”

Rev 2: changed to 2 questions, with impact first then specific steps from the procedure used. Changed answers to 2X2.

Rev 3: Reworded question #1 the include “reactor scram”
Changed all answers by removing “reactor scram”
Added procedural information for distracter description answer B2 and D2

Rev 4: Removed referenced alarm from stem and changed to LOD 4</td></tr>
</table>

Question: 87

Plant is operating at rated thermal power.

Electrical bus 16AB feeder breaker trips.

- Loss of AC Power ONEP is entered, no subsequent actions have been performed
- Division 2 D/G auto starts and restores bus power

2 minutes later C71-S003A, MG SET A OUTPUT BREAKER, trips.

(1) What is the cause of the resultant reactor scram?

(2) What is the reactor water level band directed by the CRS?

A. (1) MSIV closure
(2) -30 to +50 inches

B. (1) MSIV closure
(2) +11.4 to +53.5 inches

C. (1) Loss of Instrument Air
(2) -30 to +50 inches

D. (1) Loss of Instrument Air
(2) +11.4 to +53.5 inches

Answer: **A**

Explanation:

A loss of power to an ESF bus (15 or 16) will cause a ½ isolation for that division in NS4 logic due to loss of power to the logic and de-energize the 'B' solenoid for all MSIVs. MSIV/DR VLV TRIP INIT, P601-19A-E4 alarm states that a possible cause is Loss of ESF Bus 15AA or 16AB. Subsequent operator action 3.1.8 of Loss of AC Power ONEP, 05-1-02-I-4, states "On 1H13-P601 panel, RESET respective Div 1 (2) NSSSS isolation logic IF possible. This is required to reset alarm and logic for Group 1 MSIV isolation.

RPS electrical system provides power to the MSIV solenoids. Breaker C71-S003A is an EPA breaker that provides power to the RPS bus 'A' from normal power supply. When this breaker trips, all of RPS 'A' will no longer have power. So when RPS 'A' power is lost the other solenoid will no longer have power and all MSIVs will close.

When the MSIVs close, a reactor scram will occur.

IAW 02-S-01-43, Transient Mitigation Strategy, rev.5, 6.6.6 b, with MSIVs closed SRV operation will be required, therefore EP2 will require a level band of -30 to +50 inches.

a. Under some conditions, Reactor water level of greater than +11.4 inches may be obtainable per EP-2, Step L-3.

   HOWEVER attempting to maintain level greater than +11.4 inches may introduce unnecessary challenges to available injection sources. Examples of such situations include:

   - High capacity injection source (e.g., ECCS) is required for level control.
   - Swell **AND** shrink resulting from SRV operation for pressure control **MAY NOT** allow operator to adequately control level between +11.4 inches **AND** +53.5 inches.
   - **DURING** above situations, it is acceptable to widen (expand) level band to nominal band of +50 inches to -30 inches. Purpose of +50 inches to -30 inches band is to prevent challenges to injection systems due to high level **AND** to prevent Level 2 isolations/initiations.

Distracters:

B is wrong, but plausible. Part one is correct. Part two is the normal band that would be used if the MSIVs were open, IAW with EP-2 Level Control, Step L-3, states to maintain level between 11.4 to 53.5 inches, however 02-S-01-43, Transient Mitigation Strategy, 6.6.6B, allows the Expanded band, -30 to +50 inches, to be used if pressure is being controlled by the SRVS.

C is wrong, but plausible. The reactor will not scram on loss of IA and no loss of IA will occur. If the applicant confuses the 16AB with the 15AA bus. A loss of 15AA will cause a loss of IA in the

Containment and eventually a manual reactor scram. Part two is correct.

D is wrong. but plausible. The reactor will not scram on loss of IA and no loss of IA will occur. If the applicant confuses the 16AB with the 15AA bus. A loss of 15AA will cause a loss of IA in the Containment and eventually a manual reactor scram. Part two is the normal band that would be used if the MSIVs were open, IAW with EP-2 Level Control, Step L-3, states to maintain level between 11.4 to 53.5 inches, however 02-S-01-43, Transient Mitigation Strategy, 6.6.6B, allows the Expanded band, -30 to +50 inches, to be used if pressure is being controlled by the SRVS.

K/A Match

The SRO applicant must first determine the outcome of the events that have happened, which in this case would be a reactor scram on MSIV closure, due to 'B' MSIV solenoids de-energizing from loss of power to Div. 2 (B) logic and loss of power to the 'A' MSIV solenoids. The SRO must recognize that a scram from rated conditions will cause entry into EP-2 and should be transitioned to and using 02-S-01-43, Transient Mitigation Strategy determine correct level band.

Technical References:

04-1-02-1H13-P601-19A-E4, MSIV/DR VLV TRIP INIT ARI, Rev. 154

E1187-04, Rev. 23

E1187-06, Rev. 8

ONEP 05-1-02-III-2, Loss of One or Both RPS Buses ONEP, Rev. 26

ONEP 05-1-02-III-5, Automatic Isolations ONEP, Rev. 49

ONEP 05-1-02-I-4, Loss of AC Power, Rev. 50

ONEP 05-1-02-I-1, Reactor Scram, Rev. 130

05-S-01-EP-2, RPV Control, Rev. 45

GLP-OPS-P7500, Standby Diesel Generator System (P75) Lesson Plan, Rev. 26

GLP-OPS-C7100, Reactor Protection System (RPS) Lesson Plan, Rev. 13

GFIG-OPS-C7100, Reactor Protection System (RPS) System Figures, Figure 1A

02-S-01-43, Transient Mitigation Strategy, Rev.5

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-M7101, OBJ. 8.1 & 11

Question Source:	**Bank #**	

(note changes and 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 the SRO applicant to determine the final outcome of sequence of events and enter the appropriate procedure. The SRO must also determine that entry into EP-2 is inevitable and determine correct level band.**		
PRA Applicability:		
ESF (R20) is listed as #4 on System Importance to CDF. RPS is listed as #5 on System Importance to CDF.		

Examination Outline Cross Reference	Level	SRO
218000 ADS A2 Ability to (a) predict the impacts of the following on the AUTOMATIC DEPRESSURIZATION SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: A2.05 Loss of A.C. or D.C. power to ADS valves	Tier	2
	Group #	1
	K/A	218000
	Rating	3.6
	Revision	2
Revision Statement: **Rev 1: New Question** **No Handout** **Changed 10 CFR # to 2** **Rev 2: Changes to question (2)**		

Question: 88

A complete loss of DC battery bus 11DA has occurred.

(1) What effect will this have on ADS/SRV operation?

<u>And</u>

(2) If no changes are made to current plant status, what action will the CRS direct?

A. (1) Handswitches for all ADS valves ONLY in the Main Control Room WILL NOT work
(2) Enter Tech Spec 3.0.3 immediately

B. (1) Handswitches for all ADS valves ONLY in the Main Control Room WILL NOT work
(2) Restore 11DA or be in Mode 3 within 14 hours

C. (1) Handswitches for all SRVs on P601 only WILL NOT work.
(2) Enter Tech Spec 3.0.3 immediately

D. (1) Handswitches for all SRVs on P601 only WILL NOT work.
(2) Restore 11DA or be in Mode 3 within 14 hours

Answer:	**D**
Explanation:	

ADS/SRVs have two solenoids to actuate an automatic opening or to open manually. ESF 125 VDC powers each solenoid, Div.1 DC, 11DA, powers the 'A' solenoid and Div. 2, 11DB, powers the 'B' solenoid. A loss of Div. 1, 11DA, DC power will only cause a loss of 'A' solenoid power for all ADS/SRVs valves. Valve operation will be prohibited by Div 1 ADS, Div 1 Lo-Lo Set and use of handswitch on the P601 panel which uses the Div 1 (A) solenoid.

However, P631, located in the back panels of the main control room, provides operation of SRVs using the Div 2 (B) solenoid powered from Div 2 11DB DC bus. ADS/SRV valve operation will be available from Div. 2 logic and use of Div. 2 solenoid power.

Tech Specs 3.8.4 should be entered in condition 'C'. Required Action is to restore to operable status within 2 hours. If unable then enter condition 'D', Be in MODE 3 in 12 hours.

Distracters:

A is wrong. All valve handswitches on P601 will not work, but handswitches on P631 still work. Plausible if the applicant does not understand where the different solenoids are operated from. Tech Spec 3.0.3 is not entered, but plausible if the applicant decides that multiple ADS valves are INOP Tech Spec 3.5.1 is entered.

B is wrong. All valve handswitches on P601 will not work, but handswitches on P631 still work.

C is wrong. Tech Spec 3.0.3 is not entered, but plausible if the applicant decides that multiple ADS valves are INOP Tech Spec 3.5.1 is entered.

K/A Match

The applicant must first determine the effect of loss of DC on the ADS valves (loss of solenoid power for one of two). Then determine the correct Tech Spec Completion time for the given status.

Technical References:

GLP-OPS-E2202, Automatic Depressurization System (ADS) lesson plan, Rev. 9

Tech Specs 3.8.4 Amendment 120/201

Tech Specs 3.5.1 Amendment 175/201

Handouts to be provided to the Applicants during exam:

None

Learning Objective:

GLP-OP-TS01, OBJ. 9

Question Source:	**Bank #**	
(note changes and 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 the SRO applicant to recognize how a loss of Div 1 DC will affect the ability to use SRVs and determine Tech Spec completion time.

PRA Applicability:

ADS is listed as #6 on System Importance to CDF.

Failure to manually depressurize with ADS/SRVs is ranked #1 on Operator Action Importance to CDF.

Examination Outline Cross Reference	Level	SRO
239002 SRVs G2.2.12 Knowledge of Surveillance procedures	**Tier**	2
	Group #	1
	K/A	239002
	Rating	4.1
	Revision	3
Revision Statement: **Rev 1 Changed answers to match K/A** **No Handout** **Rev 2 Added noun name to the F041F valve instem** **Reworded answers A1 and B1** **Removed 1B21 from answers C1 and D1** **Added plausibility statements for A and B answers** **Rev 3: Editorial changes in stem and answers used "response" in #1**		

Question: 89

During a startup 06-OP-1B21-R-0002, ADS / SRV Valve Operability, is required to be performed on several valves.

The operator performing the surveillance reports that after the handswitch for 1B21-F041F, MSL B SRV (ADS), was taken to the OPEN position there was no indication that the valve opened.

(1) What is the CRS's response?

<u>And</u>

(2) What is the Mode change requirement?

A. (1) Note discrepancy, finish test, enter one LCO for all INOP valves
   (2) Mode 1 is allowed with restrictions

B. (1) Note discrepancy, finish test, enter one LCO for all INOP valves
   (2) Mode 1 is NOT allowed under any conditions

C. (1) Stop performance of surveillance, and declare F041F INOP
   (2) Mode 1 is NOT allowed under any conditions

D. (1) Stop performance of surveillance, and declare F041F INOP
   (2) Mode 1 is allowed with restrictions

Answer: D

Explanation:

Anytime a surveillance is being performed if a component does not meet the Tech Spec Requirements then the test is immediately stopped and Tech Specs are referenced.

First, the applicant must recognize that the B21-F041F is an ADS valve, With the valve not providing positive indication that the valve opened it must be declared INOP due to failure to comply with SR 3.5.1.7.

This would make one ADS valve inop and LCO 3.5.1 Cond. E would be entered.

Tech Spec 3.0.4 b. states:

LCO 3.0.4 When an LCO is not met, entry into a MODE or other specified condition in the Applicability shall only be made:

b. After performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination, of the acceptability of entering the MODE or other specified condition in the Applicability, and establishment of risk management action, if appropriate; exceptions to this Specification are stated in the individual Specifications

Therefore a MODE change to MODE 1 is allowed with restrictions.

Distracters:

A is wrong. Per 01-S-06-2, Surveillance Program Procedure, “5.7.4. IF Tester CANNOT complete Surveillance-WO Task or surveillance requirement CANNOT be met, THEN: 1. STOP work, 2. PLACE it in a safe condition AND 3. NOTIFY Supervisor. Plausible due to this is the method that any SOI system component standby lineup is completed. If a component is found out of lineup then the discrepancies is noted and continue with the lineup and all out of position components are addressed together. Second part is correct.

B is wrong. Per 01-S-06-2, Surveillance Program Procedure, “5.7.4. IF Tester CANNOT complete Surveillance-WO Task or surveillance requirement CANNOT be met, THEN: 1. STOP work, 2. PLACE it in a safe condition AND 3. NOTIFY Supervisor. Plausible due to this is the method that any SOI system component standby lineup is completed. If a component is found out of lineup then the discrepancies is noted and continue with the lineup and all out of position components are addressed together. MODE change can occur with performance of a risk assessment IAW with Tech Specs 3.0.4 b.

C is wrong. First part is correct however, MODE change can occur with performance of a risk assessment IAW with Tech Specs 3.0.4 b.

K/A Match

The applicant must first determine the correct response when a surveillance requirement cannot be met per procedure then determine if a MODE change is allowed.

Technical References:

01-S-06-12, Surveillance Program Procedure, Rev. 112, step 5.7.4.

Tech Specs 3.5.1, ECCS Operating, Condition E, Amendment 201 Tech Specs 3.0.4, LCO Applicability, Amendment 175
Handouts to be provided to the Applicants during exam: NONE
Learning Objective: GLP-OP-PROC, OBJ. 41.8 GLP-OP-TS001, OBJ. 1

Question Source:	**Bank #**	
(note changes and 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 the applicant to**
PRA Applicability:
ADS is listed as #6 on System Importance to CDF.

<table>
<tr><td>Examination Outline Cross Reference</td><td>Level</td><td>SRO</td></tr>
<tr><td rowspan="5">264000 EDGs

A2 Ability to (a) predict the impacts of the following on the EMERGENCY GENERATORS (DIESEL/JET) ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:

A2.09 Loss of A.C. power</td><td>Tier</td><td>2</td></tr>
<tr><td>Group #</td><td>1</td></tr>
<tr><td>K/A</td><td>264000</td></tr>
<tr><td>Rating</td><td>4.1</td></tr>
<tr><td>Revision</td><td>3</td></tr>
<tr><td colspan="3">Revision Statement:
Per NRC: New question

Rev 1: Made answers two parts for A2.

Rev 2: Changed questions to reduce the amount in the answers.

Rev 3: Deleted “entry” in Question (1).</td></tr>
</table>

Question: 90

Plant in Mode 4.

Div 1 and 2 D/Gs are operable.

AC power is lost to Div 1 D/G Fuel Oil Transfer Pump due to fault in its breaker.

(1) What Tech Spec is the CRS required to enter?

and

(2) What Tech Spec LCO tracking report will the CRS initiate IAW 02-S-01-17, Control of Limiting Conditions for Operation?

A. (1) LCO 3.8.2, AC Sources – Shutdown
(2) Active LCO

B. (1) LCO 3.8.2, AC Sources – Shutdown
(2) Potential LCO

C. (1) LCO 3.8.3, Diesel Fuel Oil, Lube Oil, and Starting Air
(2) Active LCO

D. (1) LCO 3.8.3, Diesel Fuel Oil, Lube Oil, and Starting Air
(2) Potential LCO

Answer: **B**

Explanation:

DG11 F/O xfer pump is required OPERABLE during Mode 4 IAW SR 3.8.2.1 and SR 3.8.1.6. With loss of power to the F/O Xfer Pump, DG11 must be declared inoperable due to inability to meet its mission time of 7 days before refueling. In Mode 4, only the DG associated with one electrical division required OPERABLE IAW TS 3.8.8 is required to be OPERABLE. Since, the stem states that both Division 1 and 2 are operable, it is the division for which TS 3.8.8 is credited, and DG12 meets the requirements of TS 3.8.2. Therefore, TS 3.8.2 is met by DG12 and DG11 is not required to be OPERABLE at this time. Since DG11 is not required to be operable in the current mode but is required to be operable in Modes 1,2,and 3 or for different conditions in Modes 4 and 5, A Potential LCO is required IAW 02-S-01-17.

Distracters:

Answer A is plausible because DG11 could be required to be operable in Mode 4, then an active LCO would be required. It is wrong because Div 2 operable, therefore, Div 2 and DG12 satisfy the requirements of TS 3.8.2 and TS 3.8.8. Active LCOs are only necessary when a required LCO is NOT met.

Answer C is plausible because DG11 Fuel Oil Transfer Pump has lost power and TS 3.8.3 specifically refers to the DG Fuel Oil system. It is wrong because TS 3.8.3 only governs fuel oil quantity and quality. The F/O Xfer pump is specifically required for DG operability by TS 3.8.1/3.8.2 per SR 3.8.1.6. It is also wrong for the reason stated for distractor A.

Answer D is plausible for the reason given for distractor C. It is wrong for the first reason stated for distractor C.

K/A Match

The applicant must first determine how a loss of power to the fuel oil pump affects the operability of the DG11, then decide which procedure should be used to mitigate the event that occurred.

Technical References:

Tech Specs:

- 3.8.1, AC Sources – Operating, Amendment 175
- 3.8.2, AC Sources – Shutdown, Amendment 145
- 3.8.3, Diesel Fuel Oil, Lube Oil, and Starting Air, Amendment 149
- 3.8.8, Distribution Systems – Shutdown, Amendment 145

02-S-01-17, Control of Limiting Conditions for Operation, rev.129

Handouts to be provided to the Applicants during exam: NONE		
Learning Objective: GLP-OP-ONEP, OBJ. 2 & 50		
Question Source:	**Bank #**	
(note changes and 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 the SRO applicant to take the given information and decide which procedure should be used to mitigate the events.** **Using Tech Specs determine operability and corrective actions.**		
PRA Applicability:		
Div. 1 & 2 is listed as #11 on System Importance to CDF.		

<table>
<tr><td>Examination Outline Cross Reference</td><td>Level</td><td>SRO</td></tr>
<tr><td rowspan="5">202001 Recirculation

A2 Ability to (a) predict the impacts of the following on the RECIRCULATION SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:

A2.12 Loss of reactor feedwater</td><td>Tier</td><td>2</td></tr>
<tr><td>Group #</td><td>2</td></tr>
<tr><td>K/A</td><td>202001</td></tr>
<tr><td>Rating</td><td>3.8</td></tr>
<tr><td>Revision</td><td>2</td></tr>
<tr><td colspan="3">Revision Statement:

Rev 1: Added information to the stem, made 2 questions to cover the A2 KA.
Corrected the question info to match K/A.
Change to a new question instead of bank.

Rev 2: Editorial changes to stem and answers
Added “Condition A” to question (1).</td></tr>
</table>

Question: 91

Plant is operating at rated thermal power when RFPT ‘A’ trips.

- Recirc FCV ‘A’ runs back fully
- Recirc FCV ‘B’ stops running back at 45% open
- Total Core flow is 72%
- Recirc loop flow mismatch is 3215 gpm

(1) IAW with Tech Spec 3.4.1, Recirculation Loops Operating, Condition A, what is the completion time?

AND

(2) What action will the SRO direct?

A. (1) 2 hours
(2) Balance loop flows IAW 05-1-02-III-3, Reduction in Recirculation System Flow Rate ONEP

B. (1) 12 hours
(2) Balance loop flows IAW 05-1-02-III-3, Reduction in Recirculation System Flow Rate ONEP

C. (1) 2 hours

(2) Immediately take action to be in Mode 3 IAW with Tech Spec 3.4.1

D. (1) 12 hours
(2) Immediately take action to be in Mode 3 IAW with Tech Spec 3.4.1

Answer: **A**

Explanation:

The applicant must recognize that Reactor Recirc loop flow are not within the requirements of SR 3.4.1.1 "Verify recirculation loop jet pump flow mismatch with both recirculation loops in operation is:

a. ≤ 10% of rated core flow when operating at < 70 % core flow

and

b. ≤ 5% of rated core flow when operating at ≥ 70% core flow.

Current core flow is 72%, therefore loop flows are required to be within 2230 (5%) gpm of each other.

When a FCV runback occurs each valve will close to approximately 20% valve position. With one valve at 20% and the other at 45% the flow miss match is 3215 gpm.

LCO 3.4.1, Condition A, "Recirculation loop jet pump flow mismatch not within limits", Action A.1, "Shutdown one recirculation loop", completion time of 2 hours.

IAW with 05-1-02-III-3, Reduction in Recirculation System Flow Rate ONEP step 3.7.

3.7 **IF** Both loops are operating, **THEN** **PERFORM** the following:

3.7.2 At less than 78.7 Mlbm/hr core flow, **BALANCE** loop flows to within 4460 gpm.

3.7.3 At greater than 78.7 Mlbm/hr core flow, **BALANCE** loop flows to within 2230 gpm.

Distracters:

B is wrong, but plausible, 12 hours is the completion time for 3.4.1 condition B.

C is wrong, but plausible, the second part this is condition B with no recirc loops in operation, be in mode 3 within 12 hours.

D is wrong, but plausible, the second part this is condition B with no recirc loops in operation, be in mode 3 within 12 hours.

K/A Match

The applicant must first determine the results of a loss of a Reactor Feed pump during rated conditions (i.e. Reactor Recirc FCV runback). Then, he must recognize that current valve positions would indicate that the flow mismatch between loop is exceeded. The final result is a requirement from Tech Spec 3.4.1.

Technical References:

Tech Specs 3.4.1, Amendment 205 05-1-02-III-3, Reduction in Recirculation System Flow Rate ONEP, Rev. 115		
Handouts to be provided to the Applicants during exam: NONE		
Learning Objective: GLP-OP-TS001, OBJ. 40		
Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.43(b)(2)	
Level of Difficulty:	3	
SRO Only Justification: **This question requires the applicant to determine that Tech Spec 3.4.1 has been exceeded and be able to know the tech spec completion time from memory.**		
PRA Applicability:		

Examination Outline Cross Reference	Level	SRO
234000 Fuel Handling Equipment K4 Knowledge of FUEL HANDLING EQUIPMENT design feature(s) and/or interlocks which provide for the following: K4.02 †Prevention of control rod movement during core alterations	Tier	2
	Group #	2
	K/A	234000
	Rating	4.1
	Revision	3

Revision Statement:

Rev 1: Added refueling platform position to stem. Added IAW Tech Spec SR. Changed answers small to large.

Rev 2: Moved procedure reference to stem and deleted all references to Refuel Platform loaded and over the core in all answers.

Rev 3: Grammatical errors in stem

Question: 92

IAW with SR 3.9.1.1, Refueling Platform is positioned over the core.

(1) Which of the following being loaded would prevent control rod withdrawal movement during core alterations?

(2) How often are refueling equipment interlocks tested in Mode 5?

A. (1) Any Hoist
(2) 7 days

B. (1) Any Hoist
(2) 31 days

C. (1) Main Hoist ONLY
(2) 7 days

D. (1) Main Hoist ONLY
(2) 31 days

Answer: **C**

Explanation:

The applicant must recognize the difference between Main hoist and trolly hoist.

The Main hoist loaded and the Refuel Platform positioned over the Reactor Core will cause a Control Rod withdrawal block while in Refuel.

Per Tech Specs 3.9.1 SR 3.9.1.1 the frequency for testing is 7days.

Distracters:

A is wrong, but plausible. Rod block interlock #1: This interlock indicates a platform condition that transmits a signal to the control room affecting control rod withdrawal. This occurs When a fuel assembly load is on main hoist and platform is over reactor. Only the Main hoist loaded will cause a rod block not the trolley hoist or frame mounted hoist.

B is wrong. Rod block interlock #1: This interlock indicates a platform condition that transmits a signal to the control room affecting control rod withdrawal. This occurs when a fuel assembly load is on main hoist and platform is over reactor. Only the Main hoist loaded will cause a rod block not the trolley hoist or frame mounted hoist, and the surveillance requirement is incorrect but plausible because this is a very frequently used surveillance requirement time for other LCOs.

D is wrong. The surveillance requirement is incorrect but plausible because this is a very frequently used surveillance requirement time for other LCOs.

K/A Match

The applicant must first determine the interlocks for fuel handling equipment and control rod withdrawal and the SRO applicant must be able to remember the surveillance requirement for the interlocks.

Technical References:

Tech Specs 3.9.1, Amendment 138
04-1-01-F11-1, Refueling Platform SOI, Rev. 47, Attachment V, NO. 5

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-RF-F11001, OBJ. 22, 25

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
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 the applicant to recognize the interlocks for the fuel handling equipment and the SRO applicant must be able to know from memory the surveillance requirement for the interlock.**
PRA Applicability:

Examination Outline Cross Reference	Level	SRO
259001 Reactor Feedwater 2.4.8 Knowledge of how abnormal operating procedures are used in conjunction with EOPs.	**Tier**	2
	Group #	2
	K/A	259001
	Rating	4.5
	Revision	4
Revision Statement: **Rev 1: Changed K/A, randomly selected 2.4.8** **Editorial changes in stem** **Rev 2: Reworded question for better clarity** **Rev 3: Editorial changes in answers, removed psig** **Rev 4: Added reference to procedure to stem**		

Question: 93

An ATWS/LOCA has occurred.

- A condenser leak occurred and vacuum is lowering
- EP-2/2A is entered
- Condenser vacuum stabilizes at 15 inches Hg Vac

IAW 02-S-01-43, Transient Mitigation Strategy, which of the following will the CRS direct to maintain reactor water level band?

A. Align Condensate Booster pumps for injection and lower pressure band to 450 to 600 psig

B. Align Condensate Booster pumps for injection and maintain pressure band 800 to 1060 psig

C. Operate Feedpump Turbines below low vacuum trip and lower pressure band to 450 to 600 psig

D. Operate Feedpump Turbines below low vacuum trip and maintain pressure band 800 to 1060 psig

Answer:	D

Explanation:

Per ARI 04-1-02-1H13-P680-2A-A2, RFPT A TRIP, states the trip setpoint for low vacuum is 16"HgVac. Subsequent operator actions state, "If RFPT is tripped on low condenser vacuum and vacuum is above the MSIV low vacuum trip setpoint (9 inches Hg), trip may be bypassed by depressing the VAC TRIP RESET pushbutton on 1H13-P680. Vacuum bypass will remain in until vacuum exceeds 16 inches Hg."

The CRS should recognize that condenser vacuum has stabilized, therefore, high pressure injection system (ie feedwater) is not being challenged. Reactor pressure should NOT be reduced. 02-S-01-43, Transient Mitigation Strategy, step 6.6.6 pressure control table, "**IF** availability of high pressure systems is inadequate **OR** is being challenged **AND** it is desired to lower pressure to within the capacity of available injection systems."

The table also states "**IF** depressurization is required to reduce driving head due to un-isolable leak **AND** adequate reactor inventory exits to prevent exceeding level bands. (For non-ATWS conditions only)." An ATWS currently exist, therefore, this band is allowed in NON-ATWS only.

Nominal Pressure Band of 800 to 1060 psig should be directed by the SRO.

Distracters:

A is wrong, but plausible. Pressure band should be maintained at 800 to 1060 psig, Booster pumps cannot inject at >600 psig.

B is wrong, but plausible. Booster pumps cannot inject at >600 psig.

C is wrong, but plausible. Pressure band should be maintained at 800 to 1060 psig

K/A Match

The applicant must recognize the correct mitigation strategy for a loss of feedwater event such as a loss of condenser vacuum. The SRO applicant should recognize that availability of high pressure systems is adequate and not being challenged. **AND** it is desired not to lower pressure in an ATWS.

Technical References:

ARI 04-1-02-1H13-P680-2A-A2, RFPT A TRIP, Rev. 227
02-S-01-43, Transient Mitigation Strategy, Rev. 4, step 6.6.6

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-EP002, OBJ 2

Question Source:	**Bank #**	
(note changes and attach	**Modified Bank #**	

parent)		
	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 the applicant to take the information given and determine that a ATWS/LOCA is in progress and take the correct procedural mitigation strategy action.**		
PRA Applicability:		

Examination Outline Cross Reference	Level	SRO
2.1.2 Knowledge of operator responsibilities during all modes of plant operation.	Tier	3
	Group #	N/A
	K/A	2.1.2
	Rating	4.4
	Revision	0
Revision Statement:		

Question: 94

Per EN-OP-115, Conduct of Operations, it is a responsibility of the Shift Manager to "remain within ______ minutes of the Control Room and immediately return to the Control Room to provide oversight of activities during accident or abnormal conditions."

A. 5

B. 10

C. 15

D. 20

Answer: B

Explanation:

See EN-OP-115, section 4.0 Responsibilities, [7](e), "Remain within ten minutes of the Control Room and immediately return to the Control Room and provide oversight of activities during accident or abnormal conditions."

Distracters:

All distracters are plausible to the Applicant who cannot recall the 10 minute requirement.

K/A Match

The SRO applicant must recognize the responsibilities of SRO license holder as Shift Manger during all modes of plant operation.

Technical References:

EN-OP-115, Conduct of Operations, Rev. 20

Handouts to be provided to the Applicants during exam: NONE		
Learning Objective: GLP-OPS-PROC, OBJ. 1.0		
Question Source:	**Bank # 405**	2011 NRC Exam Q# 97
(note changes and attach parent)	**Modified Bank #**	
	New	
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 the applicant to know from memory the responsibilities of SRO license holders.**		
PRA Applicability:		

Examination Outline Cross Reference	Level	SRO
2.1.37 Knowledge of procedures, guidelines, or limitations associated with reactivity management.	Tier	3
	Group #	N/A
	K/A	2.1.37
	Rating	4.6
	Revision	1

Revision Statement:

Changed LOD to 3

Question: 95

Power is to be reduced from 100% to 65% for a control rod sequence exchange.

IAW with 02-S-01-27, Operation's Philosophy, who is responsible for reviewing the Reactivity Maneuver Plan and ensuring that the control rod pull sheets are highlighted to emphasize areas of concern?

A. Reactor Engineering

B. Reactivity Management SRO

C. ON-shift Control Room Supervisor

D. ON-shift Operations Shift Manager

Answer: **B**

Explanation:

02-S-01-27, Operation's Philosophy, classifies the described power change as a Type 3 power maneuver (per step 6.9.1.1c), which requires staffing an additional SRO, the Reactivity Management SRO (RMSRO) (per step 6.9.1.2a). Step 6.9.1.2b states the RMSRO is responsible for reviewing the Reactivity Maneuver Plan and ensuring that the control rod pull sheets are highlighted to emphasize areas of concern.

Distracters:

A is wrong - but, is plausible since Reactor Engineers develop the control rod pull sheets and are involved in during the sequence exchange.

C is wrong - but is plausible since this is an on shift SRO who normally has the control room command function and might be considered capable of assuming dedicated reactivity management duties, but the RMSRO must have NO concurrent duties.

D is wrong - but is plausible because Ops Management is present in the control room for management oversight during power reductions for sequence exchanges.

K/A Match

The applicant must recognize the responsibilities of SRO positions for reactivity management.

Technical References:

02-S-01-27, Operation's Philosophy, Rev. 73

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-PROC, OBJ. 4.10

Question Source:	**Bank # 710**	2015 NRC Exam Q# 95
(note changes and attach parent)	**Modified Bank #**	
	New	
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.43(b)(6)	
Level of Difficulty:	3	

SRO Only Justification:

This question requires the applicant to recognize from memory the responsibilities of SRO positions for reactivity management.

PRA Applicability:

Examination Outline Cross Reference	Level	SRO
2.2.7 Knowledge of the process for conducting special or infrequent tests.	Tier	3
	Group #	N/A
	K/A	2.2.7
	Rating	3.6
	Revision	2

Revision Statement:

Rev 1: Changed LOD to 3

Rev 2: Changed wording in answers A and B

Question: 96

During a refueling outage, per EN-OP-116, Infrequently Performed Test or Evolutions, which of the following is required to be controlled as an Identified Infrequently Performed Test or Evolution (IPTE)?

A. Removing Reactor Head

B. Removing Reactor Steam Separator

C. Placing Reactor Mode switch to REFUEL

D. Raising Reactor water level to the Reactor flange

Answer: **A**

Explanation:

EN-OP-116, Infrequently Performed Test or Evolutions, Attachment 9.1, Identified IPTEs, Reactor Head removal is listed as identified and per body of procedure step 5.1.2, If the proposed Test or Evolution is a previously Identified IPTE (See Attachment 9.1, Identified IPTEs), then go to Section 5.2, IPTE Administrative Requirements.

Distracters:

5.1.1.2 All other evolutions involve activities that would not be expected to introduce transients, plant trips, or reduce margins of safety and therefore do not require increased management attention. These evolutions include the following:

- Normal pre-job preparations conducted before frequently performed activities such as monthly Surveillance Testing or Tagout "hanging".
- Activities which are determined not to meet the requirements for an infrequently performed test or evolution, including Work Order and troubleshooting.
- Activities with negligible potential to reduce margins of safety or introduce transients or plant trips.
- Activities controlled by Emergency Procedures, Emergency Plan Procedures, or Abnormal (Off-Normal) Procedures, in which timely response is directed or otherwise required.

B is wrong – due to the above description, this will not introduce any transients, plant trips, or reduce margins of safety.

C is wrong - due to the above description, this will not introduce any transients, plant trips, or reduce margins of safety.

D is wrong - due to the above description, this will not introduce any transients, plant trips, or reduce margins of safety.

All are plausible due to all being action and evolutions performed during a refueling outage.

K/A Match

The applicant must recognize the correct IPTE administrative requirements per procedure.

Technical References:

EN-OP-116, Infrequently Performed Test or Evolutions, Rev. 12

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-PROC, OBJ. 44.1

Question Source:	**Bank #**	
(note changes and 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 the SRO applicant to determine the IPTE administrative requirements.**		
PRA Applicability:		

Examination Outline Cross Reference	Level	SRO
2.2.21 Knowledge of pre- and post-maintenance operability requirements.	Tier	3
	Group #	N/A
	K/A	2.2.21
	Rating	4.1
	Revision	2

Revision Statement:

Rev 1: Editorial changes in stem
Changed answers
Changed to LOD 3

Rev 2: Changed wording of answers.

Question: 97

A small packing leak was discovered on P11-F130, REFUEL WTR XFER PMP SUCT FM SUPP POOL.

Maintenance has tightened the packing on the valve per a Work Order.

A maintenance leak check was performed and the leak was stopped.

What post-maintenance testing is required to be performed by Operations before the Work Order may be closed?

A. Functional stroke of valve IAW P11 SOI.

B. Timed stroke of valve IAW P11 valve operability surveillance test.

C. Functional stroke of valve with local visual observation IAW P11 SOI.

D. Local leak rate test (LLRT) to verify Suppression Pool leakage within allowable limits IAW Engineering LLRT procedure.

Answer: **B**

Explanation:

The candidate is expected to recognize that this is a safety–related primary containment isolation valve in the BOP condensate and refueling water transfer system.

Maintenance activities to tighten the packing of this valve represent a potential to affect the valve stroke time. A timed valve stroke per the appropriate surveillance procedure is required to demonstrate the valve meets the requirements of Tech Specs and the IST program.

Distracters:

A is wrong, but plausible. Although this will demonstrate functionality of the valve post–maintenance, Tech Specs and the IST program impose valve stroke time requirements that must be determined by a timed valve stroke performed IAW the applicable surveillance procedure.

C is wrong, but plausible. Although this will demonstrate functionality of the valve post–maintenance, Tech Specs and the IST program impose valve stroke time requirements that must be determined by a timed valve stroke performed IAW the applicable surveillance procedure and local observation is not required.

D is wrong, but plausible. Tightening of the valve packing does not present the potential to affect how leak–tight the valve is. While the original packing leak did present a concern for challenging limits on allowable Suppression Pool leakage into the Secondary Containment, a Maintenance leak check PMT is adequate to show that the leakage is stopped. A full LLRT of the penetration is not required.

K/A Match

The applicant must recognize the correct post maintenance test requirements.

Technical References:

TS 3.6.1.3, Primary Containment Isolation Valves (PCIVs), Amendment 120

01–S–07–2, Test Control, Rev. 109, steps 2.6.1b, 5.12, 6.2.4b

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-PROC, OBJ. 27.2

Question Source:	**Bank # 556**	2010 NRC Exam Q# 82
(note changes and attach parent)	**Modified Bank #**	
	New	
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.43(b)(5)	
Level of Difficulty:	3	

SRO Only Justification: **This question requires the applicant to first determine this valve is a PCIV within a BOP system. Then determine retest requirements.**
PRA Applicability:

Examination Outline Cross Reference	Level	SRO
2.3.4 Knowledge of radiation exposure limits under normal or emergency conditions.	Tier	3
	Group #	N/A
	K/A	2.3.4
	Rating	3.7
	Revision	2

Revision Statement:

Rev 1: Changed emergency classification to UE
Change explanation and distracters explanation
Changed to LOD 3
Changed from bank to NEW.
Changed 10 CFR # to 4

Rev 2: Editorial changes to stem

Question: 98

A Unusual Event has been declared.

A control room RO needs to be sent into the plant to perform a task. (not to save a life or protect equipment)

RP has estimated the total whole-body dose (TEDE) the RO will receive.

SRO reviewed the Daily Dose Margin Report and determined RO's current dose.

Per 10-S-01-17, Emergency Personnel Exposure Control, against what dose limits does the SRO compare the RO's current dose to ensure the RO has sufficient exposure margin to complete the task?

A. 10CFR20 Federal Dose Limits

B. 10CFR100 Federal Dose Limits

C. GGNS Administrative Dose Limits

D. Authorized Emergency Exposure Dose Limits

Answer: **C**

Explanation:

See 10-S-01-17, sections 6.1.2 and 6.1.3. By definition, the administrative dose limits are automatically suspended and replaced by the 10CFR20 Federal dose limits at the declaration of an Alert emergency or higher.

Only an UE has been declared.

Clearly then, the SRO compares the RO's current dose against the GGNS Administrative Dose Limits.

Distracters:

A is wrong, but plausible. 10-S-01-17, sections 6.1.2 and 6.1.3. By definition, the administrative dose limits are automatically suspended and replaced by the 10CFR20 Federal dose limits at the declaration of an Alert emergency or higher.

B is wrong, but plausible. 10CFR100 do not contain dose limits.

D is wrong, but plausible. 10-S-01-17, sections 6.2.1 and 6.7.1.b. The latter section clearly states that if the limit is likely to be exceeded (i.e., after determining if sufficient exposure margin exists), then obtain an Exposure Extension Authorization per section 6.1. Section 6.1 deals with the limits suggested by Authorized Emergency Exposure Dose Limits.

K/A Match

The applicant must have knowledge of radiation exposure limits under normal or emergency conditions.

Technical References:

10-S-01-17, Emergency Personnel Exposure Control, Rev. 19

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-PROC, OBJ. 17

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.43(b)(4)	
Level of Difficulty:	3	

SRO Only Justification:

This is an SRO-only question for the following reasons: 1) clearly, it relates an SRO-only responsibility in

the said procedure (i.e., to determine sufficient exposure margin); 2) although the actual question being asked is a fairly elementary one (i.e., “What dose limit is to be used?”), this item of knowledge is not at all common knowledge among the non-Radiation Protection Personnel population, most especially among the RO population. Nonetheless, because this “elementary” information is associated with an SRO’s responsibility during a plant emergency situation, it is reasonable to expect an SRO Candidate to know that we suspend the usual GGNS Admin Dose Limits during a Site Area Emergency.
PRA Applicability:

Examination Outline Cross Reference	Level	SRO
2.4.29 Knowledge of the emergency plan.	Tier	3
	Group #	N/A
	K/A	2.4.29
	Rating	4.4
	Revision	1

Revision Statement:

Changed to LOD 3 and minor grammar (tense) change.

Question: 99

A plant event occurs that meets the threshold for an ALERT level of the EALs, but is rapidly concluded (i.e., event no longer exists) before the CRS/SM actually classified the event.

Per 10-S-01-1, Activation of the Emergency Plan, which of the following describes the classification and notification requirements?

A. Classify the event but notify <u>only</u> the NRC.

B. Do <u>not</u> classify the event and notify <u>only</u> the NRC.

C. Classify the event and make <u>all</u> of the normal notifications to offsite agencies, including the NRC.

D. Do <u>not</u> classify the event and do <u>not</u> make any notifications to offsite agencies, including the NRC.

Answer: **C**

Explanation:

10-S-01-1, Activation of the Emergency Plan, Section 6.1.6.a, where we are directed to both classify the rapidly concluded event and make all of the normal offsite agency notifications, including the NRC.

Distracters:

A is wrong - It is plausible to the SRO Applicant who cannot recall the Section 6.1.6 criteria.

B is wrong - Its plausibility comes directly from Section 6.1.6.b.

D is wrong - It is plausible to the SRO Applicant who cannot recall the Section 6.1.6 criteria.

K/A Match The applicant must have knowledge of rapidly changing EAL conditions and how to classify and notify.
Technical References: 10-S-01-1, Activation of the Emergency Plan, Rev. 126
Handouts to be provided to the Applicants during exam: NONE
Learning Objective: GLP-EP-EPTS6, OBJ. 8

Question Source:	**Bank # 31**	2012 NRC Exam Q# 99
(note changes and attach parent)	**Modified Bank #**	
	New	
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.43(b)(5)	
Level of Difficulty:	3	

SRO Only Justification: **This question requires the SRO applicant to understand rapidly changing EAL conditions and how to classify and notify.**
PRA Applicability:

Examination Outline Cross Reference	Level	SRO
2.4.42 Knowledge of emergency response facilities.	Tier	3
	Group #	N/A
	K/A	2.4.42
	Rating	3.8
	Revision	0
Revision Statement:		

Question: 100

Which of the following describes the lowest classification that activation of the Emergency Operations Facility (EOF) is **required**?

A. Unusual Event

B. Alert

C. Site Area Emergency

D. General Emergency

Answer: B

Explanation:

EN-EP-609, Emergency Operations Facility (EOF) Operations, 5.0 [1], "The EOF may be activated at any time, and shall be activated at an Alert, Site Area Emergency, and General Emergency declaration."

10-S-01-1, Activation of the Emergency Plan, Rev. 126, step 6.1.7j (2), "If an Alert or higher classification has been declared, the entire Emergency Organization, all emergency facilities, must be activated."

Distracters:

All Distracters are the other event classifications.

K/A Match

The applicant must have knowledge of the EOF.

Technical References:

EN-EP-609, Emergency Operations Facility (EOF) Operations, Rev. 3

10-S-01-1, Activation of the Emergency Plan, Rev. 126, step 6.1.7j (1) & (2)

Handouts to be provided to the Applicants during exam: NONE		
Learning Objective: GLP-EP-EPTS6, OBJ. 8		
Question Source:	**Bank #**	
(note changes and 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: Only SRO applicants and SRO license holders take the referenced class and have task associated with the ERO and actions of the Shift Manager/SRO.		
PRA Applicability:		

NRC INITIAL LICENSED OPERATOR WRITTEN EXAMINATION
OPEN-REFERENCES TABLE OF CONTENTS
SRO EXAM

TAB	PROVIDED HANDOUTS
1	05-1-02-III-3, Reduction In Recirculation Flow Rate ONEP (Rev 115), Figure 1, Power/Flow Map for Normal Feedwater Temperature
2	05-1-02-II-1, Shutdown From The Remote Shutdown Panel ONEP (Rev 049), Attachments I and II
3	05-S-01-EP-1, Emergency/Severe Accident Procedure Support Documents (Rev 036), Figures (Figures 1-6)
4	05-S-01-EP-1, Emergency/Severe Accident Procedure Support Documents (Rev 036), EP/SAP Cautions (Caution 1 Only)
5	Tech Spec 3.5.1, ECCS-Operating (Surveillance Requirements not included) Tech Spec 3.8.1, AC Sources-Operating (Surveillance Requirements not included)
6	Shutdown Operations Protection Plan (SOPP) (Rev 020), Sections IV & V (pages 10-33)

Additional References Provided Outside of This Binder:

- Emergency Classification Flowcharts: (10-S-01-1, EPP 01-02), dated 11-21-2013
    - Modes 1 through 3 (page 1 of 2)
    - Modes 4 through De-Fuel (page 2 of 2)
- Steam Tables