Examination Outline Cross-Reference	Level	RO
205000 Shutdown Cooling System (RHR Shutdown Cooling Mode) **Knowledge of electrical power supplies to the following: K2.02 Motor operated valves**	**Tier #**	2
	Group #	1
	K/A #	205000 K2.02
	Rating	2.5
	Rev / Date	1 / 11-01-11

Question 1

The supply breaker which directly connects electrical power to the motor (for motive force) on RHR Shutdown Cooling Outboard Suction Valve, E12-F008 is located at…

A. 15AA

B. 15BA3

C. 15B31

D. 15P31

Answer: C

Explanation:

See the RHR SOI, Attachment IIIE, page 1 of 5 (Electrical Lineup).

'A' is wrong, but plausible unless the Applicant recalls that bus 15AA supplies 4160 VAC to the pump motor instead.

'B' is wrong, but plausible unless the Applicant recalls that 15BA3 is the 480 VAC LCC (not shown explicitly on this SOI's Attachment IIIE) that feeds power to the 480 VAC MCC 15B31.

'C' is correct. This MCC directly connects power to the motor (for moving the valve).

'D' is wrong, but plausible unless the Applicant recalls that this is a 120 VAC source (supplying the motor heater), where the MOV itself is a 480 VAC component.

Technical References:

04-1-01-E12-1, RHR SOI

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-E1200, Objective 7.3			
Question Source:	Bank #		
(note changes; attach parent)	Modified Bank #		
	New	X	
Question History:	Last NRC Exam	No	
Question Cognitive Level:	Memory/Fundamental	X	
	Comprehensive/Analysis		
10CFR Part 55 Content:	55.41(b)(8)		
	55.43(b)		

Examination Outline Cross-Reference	Level	RO
215003 Intermediate Range Monitor (IRM) System **Knowledge of INTERMEDIATE RANGE MONITOR (IRM) SYSTEM design feature(s) and/or interlocks which provide for the following: K4.04 Varying system sensitivity levels using range switches**	**Tier #**	2
	Group #	1
	K/A #	215003 K4.04
	Rating	2.9
	Rev / Date	0 / 10-14-11

Question 2

A plant startup is in progress with reactor power indicating on the IRMs.

IRM 'D' is indicating 80 on Range 4 when it is inadvertently placed on Range 6.

Which of the following will result from this action?

A. Half-scram RPS 'A'

B. Half-scram RPS 'B'

C. Rod withdrawal block

D. DOWN pushbutton illuminates, only

Answer: D

Explanation:

This action results in an indication of 8 on Range 6.

'A' is wrong because the IRM has been ranged UP (not DOWN), resulting in a downscale reading on that IRM. The choice is plausible to the Applicant who believes there are IRM Downscale RPS scram functions and associates IRM 'D' with RPS Trip System 'A'.

'B' is wrong because the IRM has been ranged UP (not DOWN), resulting in a downscale reading on that IRM. The choice is plausible to the Applicant who believes there are IRM Downscale RPS scram functions and associates IRM 'D' with RPS Trip System 'B'.

'C' is wrong because the resulting Range 6 indication is still above the IRM Downscale rod withdrawal block setpoint of 5/125 of scale. The choice is plausible to the Applicant who fails to recall the 5/125 scale setpoint.

'D' is correct because the resulting Range 6 indication is below the IRM Range Down setpoint of 15/125 scale.

Technical References: Vendor Manual 460000971, Nuclear Monitoring Systems, Volume IV, Part 2			
References to be provided to applicants during exam: None			
Learning Objective: GLP-OPS-C5102, Objective 7.2			
Question Source:	Bank # GGNS-OPS-07108		X
(note changes; attach parent)	Modified Bank #		
	New		
Question History:	Last NRC Exam		No
Question Cognitive Level:	Memory/Fundamental		
	Comprehensive/Analysis		X
10CFR Part 55 Content:	55.41(b)(7)		
	55.43(b)		

Examination Outline Cross-Reference	Level	RO
259002 Reactor Water Level Control System **Ability to monitor automatic operations of the REACTOR WATER LEVEL CONTROL SYSTEM including: A3.06 Reactor water level setpoint setdown following a reactor scram**	**Tier #**	2
	Group #	1
	K/A #	259002 A3.06
	Rating	3.0
	Rev / Date	0 / 10-14-11

Question 3

The plant is operating at rated power when a total loss of feedwater causes a reactor scram on low reactor water level.

Assuming reactor water level never rises above Level 3, what will be the setpoint on the FW MASTER LEVEL Controller 7 seconds after the scram?

A. 54.0”

B. 36.0”

C. 18.0”

D. 12.4”

Answer: A

Explanation:

The Low level (Level 3) scram setpoint is 11.4”. DFCS will initiate Setpoint Setdown also at 11.4”. Once initiated, the FW MASTER LEVEL Controler changes to 54.0” for <u>either</u> 10 seconds <u>or</u> until 12.4” is reached, at which time the setpoint changes to 18.0”.

The stem explicitly states to assume that level never rises above Level 3 (i.e., 11.4”); as such, the FW MASTER LEVEL Controller is still at 54.0” because we’re only 7 seconds (rather than 10 seconds) into the event.

For this reason, choice ‘A’ is correct.

‘C’ is wrong for the reason ‘A’ is correct. It’s plausible because it does represent a possible Setpoint Setdown value.

‘D’ is wrong for the reason ‘A’ is correct. It’s plausible because this value does play a role in the Setpoint Setdown logic.

‘B’ is wrong for the reason ‘A’ is correct. In need of a 4th answer choice, this value has been

chosen as a plausible one because it does represent the usual controller pre-scram value.			
Technical References: 04-1-01-N21-1, Feedwater SOI			
References to be provided to applicants during exam: None			
Learning Objective: GLP-OPS-C3401, Objective 12			
Question Source:	Bank # GGNS-LORQT-06088	X	
(note changes; attach parent)	Modified Bank #		
	New		
Question History:	Last NRC Exam	No	
Question Cognitive Level:	Memory/Fundamental	X	
	Comprehensive/Analysis		
10CFR Part 55 Content:	55.41(b)(7)		
	55.43(b)		

Examination Outline Cross-Reference	Level	RO
295001 Partial or Complete Loss of Forced Core Flow Circulation **2.4.11 Knowledge of abnormal condition procedures.**	**Tier #**	1
	Group #	1
	K/A #	295001 2.4.11
	Rating	4.0
	Rev / Date	1 / 11-01-11

Question 4

Following a trip of Recirc Pump 'A' from 75% power, operators are inserting CRAM rods to exit the Restricted Region.

While in the Restricted Region, which of the following would require an immediate manual scram per the Reduction in Recirc System Flowrate ONEP?

A. Oscillations of LPRM readings with peak-to-peak swings of 50 watts/cm^2 at a 2-second period.

B. Oscillations of APRM readings with peak-to-peak swings of 8% power at a 6-second period.

C. Power and/or noise level oscillations with a period of 10 to 15 seconds.

D. PBDS Ch A Hi Hi Decay Ratio alarm, alone (i.e., without any other PBDS Ch A indication).

Answer: A

Explanation:

See the "Reduction in Recirc System Flowrate" ONEP step 4.9 and its NOTE. These criteria define the onset of Thermal Hydraulic Instability (THI). Per ONEP step 3.1.1, an immediate manual scram is required if THI is observed while operating in the Restricted Region.

'A' is correct per the THI value specified in step 4.9.3 and the period specified in the NOTE.

'B' is wrong because it is below the value specified in step 4.9.2 and longer than the period specified in the NOTE. Plausibility speaks for itself.

'C' is wrong because the 10 – 15 second period far in excess of the period specified in the NOTE for power and/or noise levels. Plausibility speaks for itself.

'D' is wrong per step 4.9.5 which requires the alarm be confirmed against accompanying hi or rising LPRM counts for a PBDS channel. Plausibility speaks for itself.

Technical References: 05-1-02-III-3, Reduction in Recirc System Flowrate ONEP			
References to be provided to applicants during exam: None			
Learning Objective: GLP-OPS-ONEP, Objectives 1 and 2			
Question Source:	Bank # GGNS-LORQT-06407	X	
(note changes; attach parent)	Modified Bank #		
	New		
Question History:	Last NRC Exam	No	
Question Cognitive Level:	Memory/Fundamental	X	
	Comprehensive/Analysis		
10CFR Part 55 Content:	55.41(b)(10)		
	55.43(b)		

Examination Outline Cross-Reference	Level	RO
219000 RHR/LPCI: Torus/Suppression Pool Cooling Mode	**Tier #**	2
2.4.4 Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures.	**Group #**	2
	K/A #	219000 2.4.4
	Rating	4.5
	Rev / Date	0 / 10-14-11

Question 5

The plant is operating at rated power with the following:

- RCIC is operating for its normal quarterly pump surveillance
- RHR 'A' is operating in Suppression Pool Cooling to support the RCIC pump run

If for some reason the RCIC pump run is prolonged and suppression pool temperature continues to rise, at what temperature will EP-3, Containment Control, direct operators to also place RHR 'B' in Suppression Pool Cooling?

A. 90°F

B. 95°F

C. 105°F

D. 110°F

Answer: B

Explanation:

EP-3 entry for SP Temperature is at 95°F, at which time, step SPT-2 directs operators to "operate all available SP cooling" (i.e., in this case, place the additional RHR 'B' in service).

For this reason, only choice 'B' is correct.

'C' is wrong for the reason 'B' is correct. It is plausible because this is the Supp Pool Temperature LCO when operating at >1% RTP and adding heat to the pool for testing.

'D' is wrong for the reason 'B' is correct. It is plausible because this is the Supp Pool Temperature LCO requiring an immediate manual scram.

'A' is wrong for the reason 'B' is correct. This is the only answer choice that does not represent a "real" value related to this topic. It's been determined to been a discriminating distracter, however, based on this Exam Author's experience…much more so than would be

the only remaining “real” value of 120°F (again, an LCO value requiring depressurization).		
Technical References: EP-3, Containment Control		
References to be provided to applicants during exam: None		
Learning Objective: GLP-OPS-EP3, Objective 5		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b)(10)	
	55.43(b)	

Examination Outline Cross-Reference	Level	RO
290002 Reactor Vessel Internals **Knowledge of the physical connections and/or cause effect relationships between REACTOR VESSEL INTERNALS and the following: K1.09 LPCI**	**Tier #**	2
	Group #	2
	K/A #	290002 K1.09
	Rating	3.2
	Rev / Date	0 / 10-14-11

Question 6

Which of the following systems can be aligned to inject to the RPV both inside <u>and</u> outside the core shroud?

A. RHR/LPCI

B. Feedwater

C. RCIC

D. CRD

Answer: A

Explanation:

'A' is correct. RHR/LPCI injects inside the shroud via the LPCI injection valves, E12-F042A(B, C), but outside the shroud when aligned to inject via the RHR Shutdown Cooling Return valves to feedwater, E12-F053A(B).

'B' is wrong. Feedwater is strictly an outside the shroud injection path via the B21-F065A(B) feedwater inlet valves. The choice is plausible because Feedwater is one of the "Preferred ATWS Systems" used in EP-2A, ATWS RPV Control.

'C' is wrong. RCIC is also only an outside the shroud injection pathway. It is plausible for the same reason as choice 'B'.

'D' is wrong. CRD water enters below plenum outside shroud. It is plausible for the same reason as choice 'B'.

Technical References:

EP-2A, ATWS RPV Control

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-B1300, Objective 5.3			
Question Source:	Bank #		
(note changes; attach parent)	Modified Bank #		
	New	X	
Question History:	Last NRC Exam	No	
Question Cognitive Level:	Memory/Fundamental		
	Comprehensive/Analysis	X	
10CFR Part 55 Content:	55.41(b)(3) & (8)		
	55.43(b)		

Examination Outline Cross-Reference	Level	RO
262002 Uninterruptable Power Supply (A.C./D.C.) **Knowledge of the physical connections and/or cause effect relationships between UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.) and the following: K1.01 Feedwater level control**	**Tier #**	2
	Group #	1
	K/A #	262002 K1.01
	Rating	2.8
	Rev / Date	0 / 10-14-11

Question 7

The plant is operating at rated power.

The output breaker of inverter 1Y99 trips and its static switch fails to transfer

Which of the following describes the response of the Reactor Feed Pumps (RFPs) and Digital Feed Control System (DFCS)?

A. RFPs shift to manual and lock up at present speed; DFCS shifts to Master Manual at present signal.

B. RFPs continue to operate as before; DFCS shifts to Master Manual at present signal.

C. RFPs continue to operate as before; DFCS continues to operate as before.

D. RFPs shift to manual and lock up at present speed; DFCS continues to operate as before.

Answer: C

Explanation:

DFCS and RFP INFI-90 control systems have 3 power sources (see N21 SOI, Attachment III): 1Y99 inverter output, 1Y99 alternate power, and a BOP backup power supply. When the inverter static switch failed to auto-transfer, the BOP backup supply was there to ensure continuity of power. Thus, there was no impact on either the RFPs or the DFCS system (other than a P680 alarm.

For this reason, only choice ‘C’ is correct.

‘A’, ‘B’, ‘D’ are wrong for the reason that ‘C’ is correct. They plausible because they each represent another potential RFP/DFCS response were those components to actually see an interruption in INFI-90 control power.

Technical References:

04-1-01-N21-1, N21 SOI E-0035, Reactor Feedwater Control Cabinet Power Supplies drawing			
References to be provided to applicants during exam: None			
Learning Objective: GLP-OPS-C3400, Objective 14			
Question Source:	Bank # GGNS-OPS-01940		X
(note changes; attach parent)	Modified Bank #		
	New		
Question History:	Last NRC Exam		No
Question Cognitive Level:	Memory/Fundamental		
	Comprehensive/Analysis		X
10CFR Part 55 Content:	55.41(b)(4) & (7)		
	55.43(b)		

Examination Outline Cross-Reference	Level	RO
295019 Partial or Complete Loss of Instrument Air **Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR: AA1.01 Backup air supply**	**Tier #**	1
	Group #	1
	K/A #	295019 AA1.01
	Rating	3.5
	Rev / Date	0 / 10-14-11

Question 8

The plant is operating at rated power with the following Plant Air Compressor (PAC) configuration:

- PAC ‘A’ running
- PAC ‘B’ shutdown (idle)
- PAC ‘C’ in Standby

A lockout occurs on ST-21.

No operator action has yet been taken to re-energize the buses affected by the lockout.

30 seconds has elapsed since the lockout.

Which of the following describes what control room operators could do to recover a PAC?

A. Verify SSW ‘B’ has automatically aligned for cooling the PACs at P870; verify PAC ‘C’ has automatically started at P854.

B. Manually align SSW ‘B’ to cool the PACs at P870; manually start PAC ‘A’ at P870.

C. Manually align SSW ‘B’ to cool the PACs at P870; manually start PAC ‘C’ at P854.

D. Verify SSW ‘B’ has automatically aligned for cooling the PACs at P870; verify PAC ‘A’ has automatically re-started at P870.

Answer: B

Explanation:

Although PAC ‘A’ was running before the ST-21 lockout, this PAC (the only ESF-powered of the 3 PACs) is considered the “backup air supply” after the greatly simplified final GGNS modifications of the P51 Plant Air System completed during RF17. Thus, the KA match.

Loss of ST-21 takes away Bus 16AB (tripping the running PAC ‘A’) and Bus 11HD (taking away the MCC 11B31 control power needed to auto-start the Standby PAC ‘C’). Thus, we’re left with no PACs running. Without the 11B31 control power, PAC ‘C’ cannot be manually started either. Of course, the Div 2 DG has automatically re-powered Bus 16AB, as well as restored the needed control power for PAC ‘A’ (i.e., MCC 16B42 is restored via LSS re-sequencing). So, operators can manually start PAC ‘A’. However, because the ST-21 lockout gave only a BUV signal, and not a LOP signal, to Bus 16AB, SSW ‘B’ did not auto-align to cool the PACs. However, operators could manually align SSW ‘B’ to cool PAC ‘A’.

For these reasons, only choice ‘B’ is correct.

‘A’, ‘C’, ‘D’ are wrong for the reason ‘B” is correct. They are plausible because they each represent another potential system response/required operator action should the Applicant fail to recall the described power distribution and impact of a BUV on Bus 16AB.

Technical References:

04-1-01-P51-1, Plant Air SOI
04-1-01-R21-1, Load Shedding & Sequencing System SOI

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-P5100, Objective 8

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b)(4)	
	55.43(b)	

Examination Outline Cross-Reference	Level	RO
295002 Loss of Main Condenser Vacuum **Knowledge of the interrelations between LOSS OF MAIN CONDENSER VACUUM and the following: AK2.07 Offgas system**	**Tier #**	1
	Group #	2
	K/A #	295002 AK2.07
	Rating	3.1
	Rev / Date	0 / 10-14-11

Question 9

The plant is operating at rated power with the following:

- F045 TREAT/AUTO/BYPASS switch at P845 is in its normal position
- Offgas Post-treatment Rad Monitor 'B' is in an INOP trip condition

Offgas Post-treatment 'A' Rad Monitor fails hard upscale.

What is the automatic system/plant response?

A. Alarms at P601, only.

B. Offgas System switches over to the BYPASS mode.

C. Offgas System switches over to the TREAT mode.

D. Main condenser vacuum begins to slowly degrade.

Answer: D

Explanation:

Per IOI-1, the AUTO position for the F045 switch is not used; rather the switch is already in its TREAT position at rated power (the Applicant is expected to recognize this). There is no function that automatically switches the system over to BYPASS from having been in the TREAT mode (unlike there would be in the opposite direction). Stem conditions depict a 2-out-of-2 Post-treatment rad monitor response (both INOP conditions in this case), sufficient to auto-close the N64-F060 valve (Offgas Exhaust to Radwaste Ventilation). Closure of this F060 valve effectively bottles up the Offgas system exhaust flowpath, causing the start of a slow degradation of main condenser vacuum. This degradation has been demonstrated to be very slow (even at rated power conditions) here at GGNS.

For this reason, only choice 'D' is correct.

'A' is wrong for the reason 'D' is correct. It is plausible because it represents a system response where no automatic action occurs.

‘B’ is wrong for the reason ‘D’ is correct. It is plausible to the weaker Applicant who believes this system has an automatic switchover to the BYPASS mode.

‘C’ is wrong for the reason ‘D’ is correct. It is plausible to the weaker Applicant who cannot recall the normal switch position but who does understand that only makes sense that the system should now be in the TREAT mode of operation given the stem conditions.

Technical References:

ARI Window 04-1-02-1H13-P601-19A-C8
03-1-01-1 (IOI-1), Cold Shutdown to Generator Carrying Minimum Load

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-N6465, Objective 12

Question Source:	Bank # GGNS-OPS-03194	X
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b)(4)	
	55.43(b)	

Examination Outline Cross-Reference	Level	RO
295033 High Secondary Containment Area Radiation Levels **Ability to determine and/or interpret the following as they apply to HIGH SECONDARY CONTAINMENT AREA RADIATION LEVELS: EA2.01 Area radiation levels**	**Tier #**	1
	Group #	2
	K/A #	295033 EA2.01
	Rating	3.8
	Rev / Date	0 / 10-14-11

Question 10

Refer to the next sheet that shows part of Table 10 of EP-4, Auxiliary Building Control.

Operators have entered EP-4.

A system that cannot be isolated from the RPV is discharging outside the primary CTMT.

Which of the following situations requires that an Emergency Depressurization be performed?

A. RHR-B EQUIP AREA TEMP at 200°F <u>with</u> RHR ROOM B at 2 x 10^5 mr/hr

B. RCIC EQUIP AREA TEMP at 220°F <u>with</u> RCIC ROOM at 9 x 10^4 mr/hr

C. RHR ROOM A at 1 x 10^5 mr/hr <u>with</u> RCIC ROOM at 8.5 x 10^4 mr/hr

D. RHR-A EQUIP AREA TEMP at 240°F <u>with</u> RCIC EQUIP AREA TEMP at 200°F

Table 10
Aux Building Area Parameters

Area	Operating Limit	Max Safe Value
TEMPERATURE		
MSL PIPE TUNNEL TEMP	185°F (P601-19A/18A-A3/A4)	250°F (E31-N604A,B,C,D,E,F)
RHR-A EQUIP AREA TEMP	165°F (P601-20A-B1)	225°F (E31-N608A, N610A)
RHR-B EQUIP AREA TEMP	165°F (P601-20A-B1)	225°F (E31-N608B, N610B)
RCIC EQUIP AREA TEMP	185°F (P601-21A-G3)	212°F (E31-N602A/B)
RWCU-PUMP ROOM 1 TEMP	170°F (P680-11A-A1)	NA
RWCU-PUMP ROOM 2 TEMP	170°F (P680-11A-A2)	NA
HVAC EXHAUST RADIATION LEVEL		
AUX BLDG FHA VENT EXHAUST	3.6 mr/hr (P601-19A-B9/C9)	NA
AUX BLDG FHA POOL EXHAUST	30 mr/hr (P601-19A-B10/C10)	NA
RADIATION LEVEL		
RHR ROOM A	10^2 mr/hr (P844-1A-D4)	8×10^4 mr/hr
RHR ROOM B	10^2 mr/hr (P844-1A-D4)	8×10^4 mr/hr
RHR HX A HATCH	10^2 mr/hr (P844-1A-C4)	8×10^4 mr/hr
RHR HX B HATCH	10^2 mr/hr (P844-1A-C4)	8×10^4 mr/hr
RCIC ROOM	10^2 mr/hr (P844-1A-D4)	8×10^4 mr/hr
MAIN STEAM LINE RAD MONITOR	Set Point Log (P601-19A-D4)	8×10^4 mr/hr
SGTS FLTR TRN	2.5 mr/hr (P844-1A-C5)	8×10^2 mr/hr

Answer: C			
Explanation: Per EP-4, step 10, an ED is required only when 2 or "max safe values" are reached for a single given parameter (i.e., 2 temps, or 2 water levels, or 2 rad levels…not combinations among these parameters). 'A' and 'B' are wrong because they each suggest inter-parameter combinations. Their plausibility is based on the fact that they still require an analysis of Table 10 in order to draw a proper conclusion. 'C' is correct because both of these rad levels are above their "max safe values". 'D' is wrong because the RCIC EQUIP AREA TEMP is still below its "max safe value". Its plausibility is based on the fact that it still requires an analysis of Table 10 in order to draw a proper conclusion.			
Technical References: EP-4, Aux Building Control			
References to be provided to applicants during exam: None			
Learning Objective: GLP-OPS-EP4, Objective 3			

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b)(10)	
	55.43(b)	

Examination Outline Cross-Reference	Level	RO
211000 Standby Liquid Control System **2.1.31 Ability to locate control room switches, controls, and indications, and to determine that they correctly reflect the desired plant lineup.**	**Tier #**	2
	Group #	1
	K/A #	21100 2.1.31
	Rating	4.6
	Rev / Date	0 / 10-14-11

Question 11

Which of the following identifies two indications, at P601, associated with a successful initiation of SLC 'A' from the control room?

A. SQUIB VLV READY light extinguished
SLC PMP A/B DISCH PRESS approximately 100 psig greater than reactor pressure

B. SQUIB VLV READY light lit
SQUIB A LOSCONT/PWR LOSS status light (postage stamp) lit

C. STOR TK OUTL VLV C41-F001A red light lit (green light extinguished)
SLC SYS A OOSVC annunciator not in alarm

D. STOR TK OUTL VLV C41-F001A red light extinguished (green light lit)
SLC PMP A/B DISCH PRESS approximately 100 psig greater than reactor pressure

Answer: A

Explanation:

First, a NOTE concerning the KA match. We opted to keep this KA (rather than reject and replace) and justify the match as follows: All of the SLC control room indications are at one small location on panel P601; therefore, there is no way to construct a test question so as to have the Applicant demonstrate his ability to "locate" anything. Therefore, this test item does not attempt to do so. However, this test item does quite effectively examine the Applicants' "ability to determine that they correctly reflect the desired plant lineup". It is because this Exam Team believes this is an important concept to examine that we've chose to keep this KA and submit this test item.

A successfully fired squib is indicated by the SQUIB VLV READY light extinguished plus the SLC SYS A OOSVC annunciator in alarm plus the SQUIB A LOSCONT/PWRLOSS status light lit. Pump can't start unless its suction valve is open (i.e., STOR TK VLV C41-F001A red light lit). Once running, it tracks at ~100 psig above reactor pressure.

For all of these reasons, only answer choice 'A' is correct.

'B' is wrong because it suggests the SQUIB VLV READY light should still be lit. Plausibility

should be obvious from the above discussion. 'C' is wrong because it suggests the SLC SYS A OOSVC annunciator should not be in alarm. Plausibility should be obvious from the above discussion. 'D' is wrong because it suggests the STOR TK OUTL VLV C41-F001A should be closed. Plausibility should be obvious from the above discussion.			
Technical References: 04-1-01-C41-1, SLC System SOI			
References to be provided to applicants during exam: None			
Learning Objective: GLP-OPS-C4100, Objective 9			
Question Source:	Bank # GGNS-OPS-08982	X	
(note changes; attach parent)	Modified Bank #		
	New		
Question History:	Last NRC Exam	No	
Question Cognitive Level:	Memory/Fundamental		
	Comprehensive/Analysis	X	
10CFR Part 55 Content:	55.41(b)(6) & (7)		
	55.43(b)		

Examination Outline Cross-Reference	Level	RO
201005 Rod Control and Information System (RCIS) **Knowledge of the operational implications of the following concepts as they apply to ROD CONTROL AND INFORMATION SYSTEM (RCIS): K5.09 High power setpoints BWR-6**	**Tier #**	2
	Group #	2
	K/A #	201005 K5.09
	Rating	3.5
	Rev / Date	1 / 11-01-11

Question 12

A power ascension is in progress with reactor power at 28%.

Operators have not yet discovered that the Bypass Valves are about 10% open.

If not corrected, how does this condition impact RC&IS notch withdrawal restraints as power ascension continues towards rated power?

Notch withdrawal restraints are currently…

A. less conservative than required and will remain so all the way to rated power.

B. per design but will become less conservative than required at some higher power.

C. per design but will become more conservative than required at some higher power.

D. more conservative than required and will remain so all the way to rated power.

Answer: B

Explanation:

RC&IS gets its reactor power signal from HP Turbine 1st Stage pressure. Stem conditions indicate 28% power (understood to be indicated APRM), but with some steam portion being diverted away from the HP Turbine and into the Condenser via the 10% open Bypass Valves. Therefore, at 28% APRM power, the corresponding 1st Stage pressure signal (i.e., reactor power signal) sent to RC&IS is erroneously low. RC&IS, therefore, "thinks" reactor power is lower than it actually is (but, in fact, it's not). Assume (for this question) the Bypass Valves were already 10% open as power rose above the LPSP (26% power), RC&IS invoked the RWL 4-notch restrants later than it should have in that case. So what. By now, power is already up to 28%, so "notch withdrawal restraints are currently per design" (i.e., the RWL 4-notch restraints in effect). However, as power continues to rise, the HPSP (62%) will actually be reached before the HP Turbine 1st Stage pressure sees it. Therefore, again, RC&IS will be late to invoke the new 2-notch restraint of the RWL. This means that actual power will be something greater than 62% for some amount of time, during which RWL continues to permit

4-notch withdrawals...this, now, is a situation that is "less conservative than required". For these reasons, 'B' is correct. ‘A’, ‘C’, ‘D’ are wrong. They are plausible because they require a firm comprehension of the relationships already described in order to eliminate them as potential answers.			
Technical References: Bases for Tech Spec 3.3.2.1 Tech Spec Table TR 3.3.2.1-1			
References to be provided to applicants during exam: None			
Learning Objective: GLP-OPS-C1102, Objective 14.3			
Question Source:	Bank # GGNS-OPS-08687	X	
(note changes; attach parent)	Modified Bank #		
	New		
Question History:	Last NRC Exam	No	
Question Cognitive Level:	Memory/Fundamental		
	Comprehensive/Analysis	X	
10CFR Part 55 Content:	55.41(b)(5) & (7)		
	55.43(b)		

Examination Outline Cross-Reference	Level	RO
226001 RHR/LPCI: Containment Spray System Mode **Ability to manually operate and/or monitor in the control room: A4.12 Containment/drywell pressure**	**Tier #**	2
	Group #	2
	K/A #	226001 A4.12
	Rating	3.8
	Rev / Date	0 / 10-14-11

Question 13

Operators have manually initiated CTMT Sprays per EP-3, Containment Control.

At which control room panel can the operator monitor CTMT pressure as it lowers, and when does EP-3 direct operators to terminate CTMT Sprays?

A. P601
Before CTMT pressure drops to 0 psig

B. P601
When CTMT pressure drops to 0 psig

C. P870
Before CTMT pressure drops to 0 psig

D. P870
When CTMT pressure drops to 0 psig

Answer: C

Explanation:

There is no CTMT pressure indication of any type (meter or recorder) at P601 (chosen as an answer choice because this is the panel from which CTMT Sprays are operated). Only at P870 are there two Divisional post-accident indicators for CTMT/DRWL PRESSURE (1M71-R601 A & B). Steps CNT-5 and PCP-4 of EP-3 direct operators to terminate sprays "before CTMT pressure drops to 0 psig."

For these reasons, only choice 'C' is correct.

'A', 'B', 'D' are wrong for the reason 'C' is correct. They are plausible because each choice either suggests the existence of a CTMT pressure indicator at the P601 panel (the same panel where the operator would be actually operating CTMT sprays) and/or suggest that EP-3 tells them to secure the sprays at a later time ("when", vice "before").

Technical References: EP-3, Containment Control		
References to be provided to applicants during exam: None		
Learning Objective: GLP-OPS-M7101, Objective 5.1		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b)(7) & (10)	
	55.43(b)	

Examination Outline Cross-Reference	Level	RO
295004 Partial or Complete Loss of D.C. Power **Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF D.C. POWER and the following: AK2.02 Batteries**	**Tier #**	1
	Group #	1
	K/A #	295004 AK2.02
	Rating	3.0
	Rev / Date	1 / 11-01-11

Question 14

Should something occur that causes a Divisional battery bank to have to carry the loads on that bus, rather than its charger, then the "coping time" for that battery becomes a concern.

How long is the coping time for the Div 1 (or Div 2) battery bank?

A. 4 hours

B. 6 hours

C. 8 hours

D. 12 hours

Answer: A

Explanation:

Div 1 and 2 battery banks have a coping time of 4 hours. Div 3 is only 2 hours.

For this reason, only choice 'A' is correct.

'B', 'C' and 'D' are wrong. No such coping times exist. Their plausibility is dependent upon the Applicant even remembering the 4 hour coping time; if he doesn't, then a 6, 8, or 12 hour or choice is just as discriminating.

Technical References:

UFSAR, Sections 8.3.2.1.6.2 and 8.3.2.1.7.5

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-L1100, Objective 5.2

Question Source:	Bank #	

(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b)(8)	
	55.43(b)	

Examination Outline Cross-Reference	Level	RO
295025 High Reactor Pressure **Ability to determine and/or interpret the following as they apply to HIGH REACTOR PRESSURE:** **EA2.03 Suppression pool temperature**	**Tier #**	1
	Group #	1
	K/A #	295025 EA2.03
	Rating	3.9
	Rev / Date	0 / 10-14-11

Question 15

A high reactor pressure condition…

A. will result in having to emergency depressurize at a lower suppression pool temperature than would a low reactor pressure condition, when HCTL is considered.

B. will result in having to emergency depressurize at a higher suppression pool temperature than would a low reactor pressure condition, when HCTL is considered.

C. leads to entering the "Possible Boiling" region at lower drywell/CTMT temperatures than would a low reactor pressure condition, when the RPV Saturation Temperature curve is considered.

D. allows for a lower injection rate that should keep core debris inside the RPV than would a low reactor pressure condition, when the MDRIR curve is considered.

Answer: A

Explanation:

A review of HCTL (EP-1, Figure 1) clearly shows that for a given suppression pool level, the higher the reactor pressure the lower the suppression pool temperature that will cause entry into the UNSAFE region, requiring an emergency depressurization. For this reason, 'A' is correct and 'B' is wrong (its plausibility speaks for itself).

A review of the RPV Saturation Temperature curve (EP-1 CAUTIONS) clearly shows that higher reactor pressure conditions allow for a greater margin (not a lesser margin) to entry into the "possible boiling" region. For this reason, 'C' is wrong (its plausibility speaks for itself).

A review of the MDRIR curve (EP-1, Figure 6), clearly shows that the higher the reactor pressure the greater the required injection rate in order to remain in the SAFE region of the curve. For this reason, 'D' is wrong (its plausibility speaks for itself).

Technical References:

EP-1 CAUTIONS

EP-1 Figure 1, HCTL EP-1 Figure 6, MDRIR			
References to be provided to applicants during exam: None			
Learning Objective: GLP-OPS-EP02, Objective 7			
Question Source:	Bank #		
(note changes; attach parent)	Modified Bank #		
	New	X	
Question History:	Last NRC Exam	No	
Question Cognitive Level:	Memory/Fundamental		
	Comprehensive/Analysis	X	
10CFR Part 55 Content:	55.41(b)(10)		
	55.43(b)		

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: EK3.03 Lowering reactor water level**	**Tier #**	1
	Group #	1
	K/A #	295037 EK3.03
	Rating	4.1
	Rev / Date	2 / 11-17-11

Question 16

The crew has just entered EP-2A, ATWS RPV Control.

With the MSIVs still open and turbine bypass valves available, the operators have begun to intentionally lower reactor water level.

For the above conditions, per the EP Technical Bases, which of the following identifies two reasons why the operators are intentionally lowering reactor water level?

A. Reduce reactor power
Minimize energy input into primary CTMT

B. Reduce core inlet subcooling
Minimize energy input into primary CTMT

C. Reduce reactor power
Reduce core inlet subcooling

D. Reduce core inlet subcooling
Preemptively reduce the static driving head of water level should a LOCA occur

Answer: C

Explanation:

See EP Tech Bases (02-S-01-40), Attachment V, pages 17-28. This section discusses the two reasons for intentionally lowering level to the various level control bands, depending which particular EP-2A step is being implemented at the time. The two are: 1) reduction of reactor power, and 2) the reduction or core inlet subcooling (to minimize the risk of potential core instabilities that would otherwise occur due to operating at high power/flow ratios with a reduced recirc flow. Thus, answer choice ‘C’ is correct.

Because the stem indicates the MSIVs are open and TBVs are available, the Applicant is expected to conclude (thus, the Higher Cognitive classification of this test item) that no SRVs are open; therefore, there is no heat being added to the suppression pool. For this reason, choices ‘A’ and ‘B’ are wrong (but plausible for the reason just discussed).

Choice 'D' is wrong because such a reason as "preemptively reduce the static driving head of water level should a LOCA occur" is found nowhere in the EP Technical Bases.			
Technical References: EP Technical Bases (02-S-01-40)			
References to be provided to applicants during exam: None			
Learning Objective: GLP-OPS-EP02, Objective 2			
Question Source:	Bank #		
(note changes; attach parent)	Modified Bank #		
	New	X	
Question History:	Last NRC Exam	No	
Question Cognitive Level:	Memory/Fundamental		
	Comprehensive/Analysis	X	
10CFR Part 55 Content:	55.41(b)(10)		
	55.43(b)		

Examination Outline Cross-Reference	Level	RO
203000 RHR/LPCI: Injection Mode (Plant Specific) **Knowledge of the effect that a loss or malfunction of the RHR/LPCI: INJECTION MODE (PLANT SPECIFIC) will have on following: K3.03 Automatic depressurization logic**	**Tier #**	2
	Group #	1
	K/A #	203000 K3.03
	Rating	4.2
	Rev / Date	1 / 11-01-11

Question 17

ADS has <u>not</u> been Inhibited.

Drywell pressure has been above 2.0 psig for 3 minutes.

Reactor water level has been at -155" for 2 minutes.

All of LPCI is manually overridden off; LPCS is tagged out.

Then, a fault occurs on ESF-21.

The ESF buses are re-energized by their DGs.

What is the status of the ADS Valves?

A. Open

B. Closed but will open in about 105 seconds

C. Closed but will open in about 9.2 minutes

D. Closed

Answer: A

Explanation:

The ESF-21 loss takes power away from the ESF 4.16KV buses 16AB and 17AC. When this happens the manual overrides drop out for LPCI 'B' and LPCI 'C'. Thus, when the Div 2 DG restores power to the 16AB bus, with the pre-existing LPCI initiation signal (as indicated in the stem conditions), these LPCI pumps will re-start. Only a single Division of ADS logic is required to actuate in order to open the ADS valves. In this case, the Div 2 related LPCI pumps are available to service that logic. Since the stem indicates that reactor water level has been below -155" for 2 minutes, and DW pressure has been above 2.0 psig for 3 minutes, we are already well beyond the 105-second ADS logic requirement. Thus, as soon as the LPCI pumps restart and develop a discharge pressure, the ADS go open.

For these reasons, 'A' is correct and 'B' and 'D' are wrong (but plausible for reasons described above).

'C' is wrong. Its plausibility is based on the fact that the high drywell pressure portion of the logic is bypassed if reactor water level has remained below -150.3" for more than 9.2 minutes.

Technical References:

04-1-01-B21-1, Nuclear Boiler System SOI
E-1161-005, ADS Relay Logic
E-1181-44, RHR Pump B Control Schematic Diagram

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-E2202, Objective 25

Question Source:	Bank # GGNS-OPS-07538A	X
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b)(7)	
	55.43(b)	

Examination Outline Cross-Reference	Level	RO
217000 Reactor Core Isolation Cooling System (RCIC) **Ability to monitor automatic operations of the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) including: A3.01 Valve operation**	**Tier #**	2
	Group #	1
	K/A #	217000 A3.01
	Rating	3.5
	Rev / Date	1 / 11-01-11

Question 18

The plant is operating at rated power when a total loss of feedwater occurs.

RCIC has automatically initiated.

Reactor water level is now -50" and rising.

Which of the following identifies an <u>expected</u> RCIC valve indication at P601?

A. E51-F046, RCIC WTR TO TURB LUBE OIL CLR – Green light ON, Red light OFF

B. E51-F095, RCIC STM SPLY BYPASS VLV – Green light OFF, Red light ON

C. E51-F019, RCIC MIN FLO TO SUPP POOL – Green light ON, Red light OFF

D. E51-F054, RCIC STM SPLY DR TRAP BYP – Green light OFF, Red light ON

Answer: C

Explanation:

'A' is wrong. This choice suggests that F046 should indicate closed. If it were, oil temperature to the turbine would rise, resulting in a RCIC turbine trip. Plausible because this valve does have P601 position indication.

'B' is wrong. This choice suggests that F095 should still indicate open. In fact, F096 auto-closed 6 seconds after system initiation. Plausible because this valve does have P601 position indication.

'C' is correct. With RCIC already injecting to recover level, F019 should already be in the closed position due to high flow. Plausible because this valve does have P601 position indication.

'D' is wrong. F054 should have already auto-closed as soon as the RCIC Steam Supply (E51-F045) auto-opened. Plausible because this valve does have P601 position indication.

Technical References: 04-1-01-E51-1, RCIC SOI			
References to be provided to applicants during exam: None			
Learning Objective: GLP-OPS-E5100, Objective 9.3			
Question Source:	Bank #		
(note changes; attach parent)	Modified Bank #		
	New	X	
Question History:	Last NRC Exam	No	
Question Cognitive Level:	Memory/Fundamental		
	Comprehensive/Analysis	X	
10CFR Part 55 Content:	55.41(b)(7)		
	55.43(b)		

Examination Outline Cross-Reference	Level	RO
212000 Reactor Protection System **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 #	212000 K6.01
	Rating	3.6
	Rev / Date	1 / 11-02-11

Question 19

The plant is operating at rated power when an ST-21 lockout occurs.

30 seconds has elapsed since the lockout.

What is the status of the power available lights above the RPS Bus Power Transfer handswitches on control room backpanel P610?

A. RPS A Normal Feed – OFF   Alternate Feed – ON

RPS B Normal Feed – ON   Alternate Feed – OFF

B. RPS A Normal Feed – ON   Alternate Feed – OFF

RPS B Normal Feed – ON   Alternate Feed – OFF

C. RPS A Normal Feed – ON   Alternate Feed – OFF

RPS B Normal Feed – OFF   Alternate Feed – ON

D. RPS A Normal Feed – ON   Alternate Feed – ON

RPS B Normal Feed – OFF   Alternate Feed – OFF

Answer: D

Explanation:

With an ST-21 lockout comes a loss of Bus 13R and ESF Xfmr 21. With the loss of 13R comes the loss of BOP Xfmr 12A and, consequently, Bus 14AE. The loss of ESF-21 de-energizes Bus 16AB. Since Bus 14AE was powering RPS MG Set ‘B’ (the NORMAL Feed) via MCC 14B22, and power has not been restored to that MCC, that light is now OFF. Even though the Div 2 DG

has re-powered Bus 16AB (i.e., 30 seconds has elapsed), and with it MCC 16B42 (i.e., Div 2 LSS has re-sequenced the MCC back on), that Alternate Feed light in nonetheless OFF. The reason is that the Alternate Source RPS 'B' EPA Breakers both tripped open on Undervoltage/Underfrequency on the initial power loss. Thus, the power is still not available to the 'B' RPS Bus until an operator is dispatched to manually reset the EPA relays in the MG Set Room. Therefore, both lights (Normal and Alternate are) OFF for RPS B; both lights are still ON for RPS 'A' (because there has been <u>no</u> power loss to either Bus 15AA or Bus 13AD). For the above reasons, only choice 'D' is correct. 'A', 'B', 'C' are wrong for reasons described above. They are plausible based on the Applicant's need to recall and comprehend the power distribution relationship with RPS and the effect that a loss of that power has on the EPA trip relay status.			
Technical References: 04-1-01-C71-1, Attachment III E-0001, Main One Line (electrical distribution)			
References to be provided to applicants during exam: None			
Learning Objective: GLP-OPS-C7100, Objective 11			
Question Source:	Bank # GGNS-OPS-00395	X	
(note changes; attach parent)	Modified Bank #		
	New		
Question History:	Last NRC Exam	No	
Question Cognitive Level:	Memory/Fundamental		
	Comprehensive/Analysis	X	
10CFR Part 55 Content:	55.41(b)(4) & (6) & (7)		
	55.43(b)		

Examination Outline Cross-Reference	Level	RO
295005 Main Turbine Generator Trip	**Tier #**	1
	Group #	1
Ability to operate and/or monitor the following as they apply to MAIN TURBINE GENERATOR TRIP: AA1.05 Reactor/turbine pressure regulating system	**K/A #**	295005 AA1.05
	Rating	3.6
	Rev / Date	1 / 11-01-11

Question 20

If the main generator trips but the main turbine fails to automatically trip, the "Turbine and Generator Trips" ONEP directs the operator to manually trip the turbine then "bias the turbine valves closed."

How does the operator "bias the turbine valves closed"?

A. Run the SP DEMAND indication to Zero (0) rpm using the SP DEMAND LOWER pushbutton.

B. Run the TURBINE STEAM PRESSURE DEMAND indication fully upscale using the PRESS REF RAISE pushbutton.

C. Run the MHC START DEV POS indication to Zero (0) rpm using the MHC START DVC LOWER pushbutton.

D. Run the LOAD DEMAND indication fully downscale using the LOAD DEMAND LOWER pushbutton.

Answer: C

Explanation:

See the referenced ONEP subsequent action step 3.1.2. Only choice 'C' is correct. The MHC (Manual Hydraulic Control) subsystem is used to reset the turbines hydraulic control and protective trip devices. Taking the MHC START DEVICE to zero rpm resets the startup fluid to the Turbine Control Valve TCV) actuators; control of the TCVs with this device is always able to trump EHC's control of the TCVs.

'A' is wrong. SP DEMAND is used for turbine startup, acceleration to rated speed, generator synch, and initial loading to 175 MWe, only. Its plausibility is based on the Applicant's need to recall when/how SP DEMAND is used.

'B' is wrong. PRESSURE REFERENCE raise fully upscale (1050 psig) will keep the Bypass Valves closed but will not fully close the Turbine Control Valves in this case. Its plausibility is based on the Applicant's need to understand the limits of the Initial Pressure Controller (IPC) portion of EHC.

'D' is wrong. The Load Controller portion of EHC doesn't function without a feedback signal in the form of actual main generator output; in this case, the main generator has tripped, causing MWe output to drop below 157 MWe which will automatically turn-off the Load Controller. Its plausibility is based on the Applicant's need to understand when/how the Load Controller functions with respect to generator load.

Technical References:

05-1-02-I-2, Turbine and Generator Trips ONEP
04-1-01-N32-2, Turbine Generator Control SOI

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-N3202, Objective 8

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b)(4) & (10)	
	55.43(b)	

Examination Outline Cross-Reference	Level	RO
295030 Low Suppression Pool Water Level **Knowledge of the interrelations between LOW SUPPRESSION POOL WATER LEVEL and the following: EK2.02 RCIC**	**Tier #**	1
	Group #	1
	K/A #	295030 EK2.02
	Rating	3.7
	Rev / Date	0 / 10-14-11

Question 21

The RCIC SOI Precautions/Limitations specify a suppression pool level below which the RCIC Pump should not be operated (except in an emergency) in order to ensure proper NPSH for the pump.

What is that specified level?

A. 18.34 feet

B. 14.5 feet

C. 14.25 feet

D. 10.5 feet

Answer: B

Explanation:

'A' is wrong. This is the EP-3 (Containment Control) entry for SP low level and is plausible for this reason.

'B' is correct per the RCIC SOI (04-1-01-E51-1), P/L 3.4.

'C' is wrong. Per the EP Technical Bases (02-S-01-40), Attachment III, page 9 of 11, CAUTION #2 discussion, this is the level below which SP temperature elements will be exposed; it is plausible for this reason.

'D' is wrong. Per EP-2 (RPV Control), this is the lowest SP level at which an emergency depressurization may be performed via the ADS/SRVs; it is plausible for this reason.

Technical References:

04-1-01-E51-1, RCIC SOI
02-S-01-40, EP Technical Bases

EP-3, Containment Control EP-2, RPV Control			
References to be provided to applicants during exam: None			
Learning Objective: GLP-OPS-E5100, Objective 13.1			
Question Source:	Bank #		
(note changes; attach parent)	Modified Bank #		
	New	X	
Question History:	Last NRC Exam	No	
Question Cognitive Level:	Memory/Fundamental	X	
	Comprehensive/Analysis		
10CFR Part 55 Content:	55.41(b)(8) & (10)		
	55.43(b)		

Examination Outline Cross-Reference	Level	RO
215004 Source Range Monitor (SRM) System **Knowledge of the effect that a loss or malfunction of the following will have on the SOURCE RANGE MONITOR (SRM) SYSTEM: K6.05 Trip units**	**Tier #**	2
	Group #	1
	K/A #	215004 K6.05
	Rating	2.6
	Rev / Date	1 / 11-02-11

Question 22

A reactor startup is in progress following a mid-cycle scram from rated power.

SRM/IRM overlap has been checked and SRM count rate is being maintained between 10^2 and 10^4 cps.

An internal short causes the SRM 'B' Upscale Trip unit to de-energize.

How does SRM Channel 'B' respond with respect to RC&IS and RPS?

A. Generates a rod block, only.

B. Generates a rod block and causes a full scram.

C. Causes a full scram, only.

D. Does not generate a rod block and does not cause a full scram.

Answer: A

Explanation:

Stem conditions defining this reactor startup as following a mid-cycle scram ensures there is no doubt about the fact that all RPS shorting links are in fact installed. [Note – GGNS never removes its shorting links anyway, even for a startup following a refueling outage.] As such, when this trip de-energizes (i.e., its "normal" fail-safe mode of operation, which the Applicant is expected to recognize), the trip unit generates an "upscale" flux trip signal (i.e., above the 2 x 10^5 cps scram setpoint). However, with the RPS shorting links installed (an RPS system configuration), no full scram occurs.

For this reason, choices 'B' and 'C' are wrong, but plausible to the Applicant who fails to recall that GGNS never pulls shorting links.

However, the rod block setpoint is at 1 x 10^5 cps; that setpoint has been exceeded. For this reason, choice 'A' is correct and choice 'D' is wrong (its plausibility speaks for itself).

Technical References: GLP-OPS-C5101, SRM lesson plan		
References to be provided to applicants during exam: None		
Learning Objective: GLP-OPS-C5101, Objective 3.10		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b)(7)	
	55.43(b)	

Examination Outline Cross-Reference	Level	RO
300000 Instrument Air System (IAS) **Knowledge of the connections and / or cause effect relationships between INSTRUMENT AIR SYSTEM and the following: K1.04 Cooling water to compressor**	**Tier #**	2
	Group #	1
	K/A #	300000 K1.04
	Rating	2.8
	Rev / Date	0 / 10-14-11

Question 23

The plant is operating at rated power.

TBCW Return From Air Compressor, P43-F289, fails closed due to a solenoid failure.

Which of the following describes the response of the Plant Air Compressors (PACs)?

A. Running PAC immediately trips on low cooling water flow; the low cooling flow prevents the Standby PAC from auto-starting.

B. Running PAC remains operating for some amount of time while its temperatures rise; Standby PAC <u>will</u> auto-start when the running PAC trips on high temperatures.

C. Running PAC remains operating for some amount of time while its temperatures rise; Standby PAC will <u>not</u> auto-start when the running PAC trips on high temperatures.

D. Running PAC immediately trips on low cooling water flow; Standby PAC auto-starts and remains operating.

Answer: B

Explanation:

P43-F289 is the TBCW Return for all 3 PACS; therefore its failing CLOSED stops TBCW flow through all PACs. The PACs do <u>not</u> have a “low cooling water flow trip” designed in their control system; they do however have several high temperature trips. Which of these will result in tripping the PAC the earliest after a loss of cooling flow?...hard to say. Nonetheless, the running PAC will not immediately trip; rather, it will continue to run for some amount of time while its temperatures rise. Because the Standby PAC has not been running during this time of no cooling water flow, it is not pre-disposed to having any high temperatures; therefore, it will auto-start when the running PAC finally trips on high temperatures (i.e., one of the immediate auto-start signals for a Standby PAC is a trip of the running PAC).

For all of these reasons, only choice ‘B’ is correct.

‘A’, ‘C’, ‘D’ are wrong for reasons described above. They are plausible because they represent either possible PAC responses were these machines to have low-flow trips, or suggest that a machine that has not yet been running would fail to auto-start due to a pre-existing high temp condition (forcing the Applicant to reason that one cannot yet possibly exist).

A NOTE concerning the KA match – The configuration of the Plant Air / Instrument Air Systems at GGNS is such that to discuss the cause-effect relationship between “cooling water to the compressors” and the resulting “status of the PACs” (as in the context of this test item) is in fact the same as to discuss the cause-effect relationship between “cooling water to the compressors” and the resulting “status of the Instrument Air System”.

Technical References:

Alarm Window 04-1-02-1H13-P870-5A-E3
Alarm Window 04-S-02-SH13-P854-1A-A4

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-P5100, Objective 15

Question Source:	Bank # GGNS-OPS-08927	X
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b)(4)	
	55.43(b)	

Examination Outline Cross-Reference	Level	RO
295024 High Drywell Pressure **Knowledge of the interrelations between HIGH DRYWELL PRESSURE and the following: EK2.02 HPCS**	**Tier #**	1
	Group #	1
	K/A #	295024 EK2.02
	Rating	3.7
	Rev / Date	0 / 10-14-11

Question 24

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, the Rover stops the HPCS Pump using its handswitch.

Which of the following describes the restart capability of the HPCS Pump?

A. Pump will automatically restart if reactor water level drops below -41.6”.

B. Pump will automatically restart if drywell pressure drops below 1.39 psig then rises above it again.

C. Depressing the HPCS INIT RESET pushbutton right now will return automatic start capability to the pump.

D. Arming and depressing the HPCS MAN INIT pushbutton right now will immediately restart the pump.

Answer: C

Explanation:

‘A’ is wrong. With the existing Div 3 LOCA signal (> 1.39 psig DW pressure) and the pump’s handswitch having been placed to STOP, the pump is overridden OFF; therefore it will not auto-restart on another low level signal. Plausibility speaks for itself.

‘B’ is wrong. Even if the only remaining Div 3 LOCA signal were to clear, the initiation logic would still have to be reset using the HPCS INIT RESET pushbutton in order to restore the logic to service and allow the pump to automatically respond to another high DW pressure signal. Plausibility speaks for itself.

‘C’ is correct. Once the low water level condition has cleared (i.e., above -41.6”), the initiation logic can be reset, even with an existing high DW pressure signal, simply by depressing the

HPCS INIT RESET pushbutton. Once the logic is reset, the pump is able to automatically respond to another initiation signal. ‘D’ is wrong. See E-1183-023 (HPCS initiation logic). The automatic initiation signal already energized the K9 seal-in relay. Depressing the MAN INIT pushbutton would only try to energize this same already energized K9 relay…nothing would be accomplished. Plausibility speaks for itself.			
Technical References: E-1183-023, HPCS Relay Logic			
References to be provided to applicants during exam: None			
Learning Objective: GLP-OPS-E2201, Objective 12			
Question Source:	Bank # GGNS-OPS-03286		X
(note changes; attach parent)	Modified Bank #		
	New		
Question History:	Last NRC Exam		No
Question Cognitive Level:	Memory/Fundamental		
	Comprehensive/Analysis		X
10CFR Part 55 Content:	55.41(b)(7)		
	55.43(b)		

Examination Outline Cross-Reference	Level	RO
295006 SCRAM **2.4.2 Knowledge of system set points, interlocks and automatic actions associated with EOP entry conditions.**	**Tier #**	1
	Group #	1
	K/A #	295006 2.4.2
	Rating	4.5
	Rev / Date	0 / 10-14-11

Question 25

Operators have entered EP-2, RPV Control, on low reactor water level.

Reactor water level drops to -35" before recovering to its normal band.

Which of the following describes a current system status?

A. ADS Initiation Timers are timing out.

B. Drywell purge compressors are running.

C. RWCU is isolated.

D. Recirc pumps are running in slow speed.

Answer: D

Explanation:

'A' is wrong. Although the +11.4" Narrow Range (EP-2 entry condition) does provide the confirmatory level signal to ADS, none of the other logic portions necessary to start the ADS timers have been satisfied. The choice is plausible because the EP-2 entry value is involved, nonetheless, in the ADS logic.

'B' is wrong. Drywell purge compressors auto-start at 1.39 psig DW pressure, not on Level 3. The choice is plausible to the Applicant who mistakenly applies a Level 3 condition to the purge compressors.

'C' is wrong. RWCU auto-isolates at -41.6", not on Level 3. The choice is plausible to the Applicant who fails to recall the RPV level at which RWCU isolates.

'D' is correct. Recirc pumps auto-downshift from fast to slow speed at Level 3 (+11.4"),

Technical References:

05-1-02-III-5, Automatic Isolations ONEP (RWCU Group 8 Isolations list) 04-1-01-E61-1, Combustible Gas Control SOI 04-1-01-B21-1, Nuclear Boiler System SOI 04-1-01-B33-1, Reactor Recirc System SOI			
References to be provided to applicants during exam: None			
Learning Objective: GLP-OPS-B2101, Objective 8.1			
Question Source:	Bank #		
(note changes; attach parent)	Modified Bank #		
	New	X	
Question History:	Last NRC Exam	No	
Question Cognitive Level:	Memory/Fundamental	X	
	Comprehensive/Analysis		
10CFR Part 55 Content:	55.41(b)(7) & (10)		
	55.43(b)		

Examination Outline Cross-Reference	Level	RO
261000 Standby Gas Treatment System **Knowledge of the effect that a loss or malfunction of the STANDBY GAS TREATMENT SYSTEM will have on the following: K3.05 Secondary containment radiation/ contamination levels**	**Tier #**	2
	Group #	1
	K/A #	261000 K3.05
	Rating	3.2
	Rev / Date	0 / 10-14-11

Question 26

Both SGTS trains have automatically initiated on valid high radiation levels detected by all channels of Fuel Handling Area Ventilation Exhaust.

<u>Two minutes</u> later, SGTS operation has stabilized but Enclosure Building Pressure is <u>actually</u> reading steady at -0.22" w.c., rather than the expected -0.32" w.c., on recorders at P870.

Secondary Containment is <u>not</u> breached.

How does this difference between the "actual" and "expected" pressure impact the airborne radiation levels in Secondary Containment?

A. Radiation levels are <u>higher</u> than they would otherwise be, but only in the Enclosure Building.

B. Radiation levels are <u>higher</u> than they would otherwise be, throughout all of Secondary Containment.

C. Radiation levels are <u>lower</u> than they would otherwise be, but only in the Enclosure Building.

D. Radiation levels are <u>lower</u> than they would otherwise be, throughout all of Secondary Containment.

Answer: B

Explanation:

Only choice 'B' is correct. Reduction of airborne radiation levels within the Auxiliary Bldg during SGTS operation is accomplished by a combination of dilution and filtration. The contaminated boundary air volume is mixed with clean air in-leakage resulting from the negative building-to-atmospheric (outside) d/p. With there being a lower than design d/p (i.e., -0.22" versus -0.32") indicated in the stem conditions, there is less air in-leakage (from outside) being provided as part of the "dilution"…therefore, for a given high radiation level problem, the result will be that radiation levels are higher than they would otherwise be throughout Secondary CTMT (i.e., both the Enclosure Building and Auxiliary Building).

‘A’ is wrong because it suggests that only a portion of the Secondary CTMT would be impacted by this “lack of sufficient dilution” problem. It’s plausible to the Applicant who cannot recall the Secondary CTMT boundary.

‘C’ and ‘D’ are wrong for the reason that ‘B’ is correct. They are plausible to the Applicants who fail to comprehend the concept of there being a relationship between building d/p and an amount of air in-leakage that contributes to keeping airborne radiation levels as low as possible.

Technical References:

UFSAR Section 6.5.3.2 (Secondary Containment), especially pages 6.5-12 thru 6.5-15

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-T4801, Objective 3

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b)(8)	
	55.43(b)	

Examination Outline Cross-Reference	Level	RO
264000 Emergency Generators (Diesel/Jet) **Knowledge of the operational implications of the following concepts as they apply to EMERGENCY GENERATORS (DIESEL/JET): K5.05 Paralleling A.C. power sources**	**Tier #**	2
	Group #	1
	K/A #	264000 K5.05
	Rating	3.4
	Rev / Date	0 / 10-14-11

Question 27

Per the Precautions & Limitations of the Standby Diesel Generators (P75) SOI, which of the following is a potential consequence of running two DGs in parallel with a single ESF Transformer at the same time?

A. Inability of either DG to maintain a constant load (KW)

B. Overloading of the associated ESF Transformer

C. An overcurrent condition on one or both of the associated ESF 4.16 KV buses

D. Inability of one or both DG output breakers to automatically trip should a LOCA signal be received.

Answer: A

Explanation:

See the Div 1(2) DG SOI (04-1-01-P75-1). The context of this question is P/L 3.17. That P/L states: "Do NOT parallel two DGs to the same power supply. DG voltage regulators AND governors do NOT function properly with two Diesels in parallel." This exam author has chosen to frame this question in a Higher Cognitive way, requiring the Applicant to draw a conclusion based on his comprehension of a poorly functioning DG governor.

'A' is correct. With the machine paralleled to the offsite grid, any fluctuations in DG speed due to a poorly performing governor would be seen as commensurate fluctuations in the real load (KW) on that machine at the time.

'B' and 'C' are wrong. Their plausibility is premised on the Applicant not recalling, at all, this P/L, thereby not being to draw the proper conclusion of choice 'A'. In that case, each of these two choices provide sufficient distraction in that they propose a concern of potentially overloading transformers/buses. They are both wrong simply because there is no basis for either of these conclusions to be drawn from poorly functioning governors and/or voltage regulators.

'D' is wrong. The basis for its plausibility is much the same as for 'B' and 'C', except that in this case, the Applicant is especially distracted because he knows how important it is that the output breaker auto-trip on a LOCA signal. Again, there is no technical basis for concluding that

the breaker response is affected by the concerns of this P/L.			
Technical References: 04-1-01-P75-1, SDG System SOI			
References to be provided to applicants during exam: None			
Learning Objective: GLP-OPS-P7500, Objective 19.1			
Question Source:	Bank #		
(note changes; attach parent)	Modified Bank #		
	New	X	
Question History:	Last NRC Exam	No	
Question Cognitive Level:	Memory/Fundamental		
	Comprehensive/Analysis	X	
10CFR Part 55 Content:	55.41(b)(7) & (10)		
	55.43(b)		

Examination Outline Cross-Reference	Level	RO
202001 Recirculation System **Knowledge of the effect that a loss or malfunction of the following will have on the RECIRCULATION SYSTEM : K6.01 Jet pumps**	**Tier #**	2
	Group #	2
	K/A #	202001 K6.01
	Rating	3.5
	Rev / Date	0 / 10-14-11

Question 28

The plant is operating at rated power when the rams head for Jet Pumps 3 and 4 is completely ejected.

Which of the following is an expected change in a P680 indication resulting from this event?

A. RECIRC PUMP A DRIVING FLOW rises.

B. LOOP A JP TOTAL FLO rises.

C. LOOP B JP TOTAL FLO lowers.

D. RECIRC PUMP B DRIVING FLOW lowers.

Answer: A

Explanation:

When a rams head is ejected, there is less resistance across the core; for this reason, the drive flow in both the affected and unaffected loops rises. For this reason, 'A' is correct and 'D' is wrong (but plausible to the Applicant who fails to comprehend this design principle).

'B' is wrong. Jet Pumps 3 and 4 belong to Loop 'A'. Their loss means two less JPs contributing to that loop's total JP flow; the loop's total measured JP flow goes down, not up. Plausibility is based on the Applicant's need to comprehend this design principle and the assignment of these particular JPs to that drive loop.

'C' is wrong. A decreased core plate d/p also results from this rams head failure, allowing for a greater measured flow through unaffected ('B') loop for JP flow. Plausibility is based on the Applicant's need to comprehend this design principle.

NOTE – all of these parameter changes (before versus after the ejection) have been validated on the GGNS simulator and against the JP Anomalies ONEP for accuracy.

Technical References:

05-1-02-III-6, Jet Pump Anomalies ONEP			
References to be provided to applicants during exam: None			
Learning Objective: GLP-OPS-B3300, Objective 50			
Question Source:	Bank #		
(note changes; attach parent)	Modified Bank #		
	New		X
Question History:	Last NRC Exam		No
Question Cognitive Level:	Memory/Fundamental		
	Comprehensive/Analysis		X
10CFR Part 55 Content:	55.41(b)(2) & (3)		
	55.43(b)		

Examination Outline Cross-Reference	Level	RO
204000 Reactor Water Cleanup System **Ability to predict and/or monitor changes in parameters associated with operating the REACTOR WATER CLEANUP SYSTEM controls including: A1.07 RWCU drain flow**	**Tier #**	2
	Group #	2
	K/A #	204000 A1.07
	Rating	2.9
	Rev / Date	0 / 10-14-11

Question 29

During a plant startup operators are controlling reactor water level with CRD makeup and RWCU blowdown.

Currently:

- RWCU Blowdown Valve (G33-F033) is 20% open
- RWCU Blowdown Orifice Bypass Valve (G33-F031) is closed
- All other RWCU (G33) system valves are in their normal Pre-Pump Mode lineup

Operators decide to open G33-F031.

(1) How will this action change the blowdown flow rate?

(2) Per the G33 SOI, what else can operators do with RWCU Regen HX Return Valve (G33-F042) to change the blowdown flow rate in the same direction?

A. (1) Lowers the blowdown flow rate.

(2) Throttle open G33-F042.

B. (1) Lowers the blowdown flow rate.

(2) Throttle closed G33-F042.

C. (1) Raises the blowdown flow rate.

(2) Throttle open G33-F042.

D. (1) Raises the blowdown flow rate.

(2) Throttle closed G33-F042.

Answer: D

Explanation:

G33 SOI Section 5.1 directs all RWCU blowdown activities. Step 5.1.2a(5) allows operators to open G33-F031 (Orifice Bypass) should increased blowdown flow be desired. Step 5.1.2a(6) also allows the throttling closed of G33-F042 (the normally-open RWCU Regen HX Return) to help divert that much more system flow towards the blowdown line. [System lesson plan GFIG-OPS-G3336, Figure 1 sufficiently shows G33-F042 in its normal Pre-Pump Mode lineup.]

For these reasons, only choice 'D' is correct.

'A', 'B', 'C' are wrong for the reasons already described. They are plausible because they each represent a possible way to either change the blowdown rate or to change the position of F042.

Technical References:

04-1-01-G33-1, RWCU SOI

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-G3336, Objective 10

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b)(3) & (10)	
	55.43(b)	

Examination Outline Cross-Reference	Level	RO
216000 Nuclear Boiler Instrumentation **Knowledge of the effect that a loss or malfunction of the NUCLEAR BOILER Instrumentation will have on following: K3.24 Vessel level monitoring**	**Tier #**	2
	Group #	2
	K/A #	216000 K3.24
	Rating	3.9
	Rev / Date	0 / 10-14-11

Question 30

The reference leg for condensing pot B21-D004A is leaking.

Which of the following is a consequence of this condition?

A. Upset Range level will indicate lower than actual level.

B. Shutdown Range level will indicate higher than actual level.

C. One Wide Range channel will indicate lower than actual level.

D. One Narrow Range channel will indicate higher than actual level.

Answer: D

Explanation:

See GFIG-OPS-B2101, Figure 1. The Upset Range and Shutdown Range instruments use the B21-D002 condensing pot. For this reason, choices ‘A’ and ‘B’ are wrong, but plausible unless the Applicant recalls this fact.

The B21-D004A thru D pots service the four Narrow Range and Wide Range instrument reference legs. A leaking reference leg results in a lowering of the normal d/p that would exist between the reference leg (high pressure side of the d/p cell) and variable leg (low pressure side of the cell). The d/p cell only knows a delta-pressure…i.e., that lowering of d/p is usually due to a rise in actual level in the RPV. Thus, the result of this leak is that the affected channel will indicate higher than actual level (i.e., actual level hasn’t changed).

For these reasons, choice ‘D’ is correct and ‘C’ is wrong (but plausible as discussed above).

Technical References:

P&ID M-1077B, Nuclear Boiler System

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-B2101, Objective 4.2		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b)(3) & (7)	
	55.43(b)	

Examination Outline Cross-Reference	Level	RO
2.1.1 Knowledge of conduct of operations requirements.	**Tier #**	3
	Group #	
	K/A #	2.1.1
	Rating	3.8
	Rev / Date	0 / 10-14-11

Question 31

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. Manually opening ADS Valves (two at a time) at P601

C. Initiating and overriding HPCS at P601

D. Up-shifting Reactor Recirc Pumps at P680

Answer: C

Explanation:

The Acceptable Two-Handed Operations for GGNS are listed in Attachment 9.3, Section 3 of EN-OP-115.

'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.

'B' 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.

'C' is correct. It is listed on the Addendum and the method for overriding does in fact require two-handed operation (see 02-S-01-27, Operations Philosophy, Section 6.2.8.b).

'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.

Technical References:

EN-OP-115, Conduct of Operations
02-S-01-27, Operations Philosophy

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-PROC, Objective 8.10

Question Source:	Bank # GGNS-LORQT-06382a	X
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b)(10)	
	55.43(b)	

Examination Outline Cross-Reference	Level	RO
2.2.43 Knowledge of the process used to track inoperable alarms.	**Tier #**	3
	Group #	
	K/A #	2.2.43
	Rating	3.0
	Rev / Date	0 / 10-14-11

Question 32

Which of the following labeling methods is used to identify problem annunciators (i.e., those that are not fully functional) in the control room?

A. Square dot in the corner of the window

B. Red tape across the window

C. Black triangle in the corner of the window

D. Yellow tape across the window

Answer: B

Explanation:

'A' is wrong, but is plausible because a square dot does indicate an annunciator that has re-flash capability.

'B' is correct. Per 02-S-01-25, section 6.2 and Attachment IV, red tape is used to label problem annunciators.

'C' is wrong, but is plausible because a black triangle is used to indicate that there are isolations associated with the annunciator.

'D' is wrong. Yellow tape is not used at all; however, the suggested yellow color is as plausible as is red to the Applicant who cannot recall.

Technical References:

02-S-01-25, Deficient Equipment Identification, Section 6.2 and Attachment IV

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-PROC, Objective 33.4

Question Source:	Bank # Held in NRC Exam Room Bank	X
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b)(10)	
	55.43(b)	

Examination Outline Cross-Reference	Level	RO
2.3.4 Knowledge of radiation exposure limits under normal or emergency conditions.	**Tier #**	3
	Group #	
	K/A #	2.3.4
	Rating	3.2
	Rev / Date	0 / 10-14-11

Question 33

Per procedure 10-S-01-17, Emergency Personnel Exposure Control, the Emergency Director may authorize a worker to receive up to a maximum of ____ Rem (TEDE) for the purpose of protecting a valuable piece of plant equipment.

A. 5

B. 10

C. 20

D. 25

Answer: B

Explanation:

See 10-S-01-17, Section 6.1, Table 1. Only choice 'B' is correct.

'A' is wrong but is plausible because the value of 5 Rem is the lower end of the band for this action suggested in the stem.

'C' is wrong but is plausible to the Applicant who cannot remember these particular dose rate limits.

'D' is wrong but is plausible because ">25 Rem" is the limit for "life saving or for protecting large populations".

Technical References:

10-S-01-17, Emergency Personnel Exposure Control

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-PROC, Objective 49.2

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Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b)(10)	
	55.43(b)	

Examination Outline Cross-Reference	Level	RO
2.4.6 Knowledge of EOP mitigation strategies.	**Tier #**	3
	Group #	
	K/A #	2.4.6
	Rating	3.7
	Rev / Date	0 / 10-14-11

Question 34

Which of the following strategies might be employed while operating per the steps of EP-2A, ATWS RPV Control?

A. With no high pressure injection source available, waiting for level to drop to -204" before Emergency Depressurizing.

B. Rapidly depressurizing the RPV with the bypass valves in anticipation of having to Emergency Depressurize with ADS Valves.

C. While Emergency Depressurizing, waiting to reach a specific reactor pressure before intentionally re-injecting to the RPV.

D. Waiting until HCTL is <u>above</u> an UNSAFE limit before Emergency Depressurizing so long as both SLC pumps are already injecting.

Answer: C

Explanation:

'A' is wrong. This choice suggests the "Steam Cooling" strategy found in EP-2 (RPV Control) but <u>not</u> in EP-2A; thus its plausibility.

'B' is wrong. This choice suggests the "anticipatory blowdown" permitted by the Pressure Leg of EP-2 (Step P-1), but <u>not</u> permitted by the equivalent Step P-1 of EP-2A; thus its plausibility.
.
'C' is correct. This is the strategy of EP-2A, Steps L-12 & L-13, along with their Table 6 (Minimum Steam Cooling Pressure).

'D' is wrong. This choice suggests a strategy that has only face plausibility to the weaker Applicant; it does not exist as an EP-2 or 2A strategy.

Technical References:

EP-2, RPV Control
EP-2A, ATWS RPV Control

References to be provided to applicants during exam: None		
Learning Objective: GLP-OPS-EP02A, Objective 3		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b)(10)	
	55.43(b)	

Examination Outline Cross-Reference	Level	RO
295003 Partial or Complete Loss of A.C. Power **Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER: AA1.04 D.C. electrical distribution system**	**Tier #**	1
	Group #	1
	K/A #	295003 AA1.04
	Rating	3.6
	Rev / Date	0 / 10-14-11

Question 35

The plant is operating at rated power.

Power is suddenly 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 by 5 to 10 volts.

C. Lowers by 60 to 65 volts.

D. Goes to zero volts.

Answer: B

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).

For these reasons only choice ‘B’ is correct.

‘A’ 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.
Choices ‘C’ and ‘D’ are wrong for the reason ‘B’ is correct. They are plausible to the weaker Applicant who cannot recall basic DC battery/charger operating principles (not an uncommon problem in the experience of this Exam Author).

Technical References: 04-1-01-L11-1, Plant DC SOI			
References to be provided to applicants during exam: None			
Learning Objective: GLP-OPS-L1100, Objective 19			
Question Source:	Bank # GGNS-OPS-07461	X	
(note changes; attach parent)	Modified Bank #		
	New		
Question History:	Last NRC Exam	No	
Question Cognitive Level:	Memory/Fundamental		
	Comprehensive/Analysis	X	
10CFR Part 55 Content:	55.41(b)(7)		
	55.43(b)		

Examination Outline Cross-Reference	Level	RO
295016 Control Room Abandonment **2.1.25 Ability to interpret reference materials, such as graphs, curves, tables, etc.**	**Tier #**	1
	Group #	1
	K/A #	295016 2.1.25
	Rating	3.9
	Rev / Date	0 / 10-14-11

Question 36

Use your provided references to answer this question.

The plant is being controlled from the Remote Shutdown Panels.

Reactor pressure is 400 psig.

Indicated reactor water level at P150 is -35”.

What is actual reactor water level?

A. -25”

B. -30”

C. -45”

D. -50”

Answer: C

Explanation:

Refer to the Remote Shutdown ONEP (05-1-02-II-1), Attachment II. Plotting the indicated level of -35” on the 400 psig reactor pressure line, we find that it intersects with an actual level of -45”.

Only choice ‘C’ is correct.

‘A’, ‘B’, ‘D’ are wrong, but each is plausible because it represents a possible actual water level should an Applicant mis-apply/mis-read this ONEP figure.

Technical References:

05-1-02-II-1, Shutdown from the Remote Shutdown Panel

References to be provided to applicants during exam: 05-1-02-II-1, Remote Shutdown ONEP (entire)			
Learning Objective: GLP-OPS-ONEP, Objective 2			
Question Source:	Bank # GGNS-OPS-05433a		X
(note changes; attach parent)	Modified Bank #		
	New		
Question History:	Last NRC Exam		No
Question Cognitive Level:	Memory/Fundamental		
	Comprehensive/Analysis		X
10CFR Part 55 Content:	55.41(b)(10)		
	55.43(b)		

Examination Outline Cross-Reference	Level	RO
295035 Secondary Containment High Differential Pressure **2.1.28 Knowledge of the purpose and function of major system components and controls.**	**Tier #**	1
	Group #	2
	K/A #	295035 2.1.28
	Rating	4.1
	Rev / Date	1 / 11-01-11

Question 37

With respect to Fuel Handling Area Pressure Control at control room panel P842:

- PRESSURE CONTROL SELECT switch is in AUTO
- Fuel Handling Area Pressure Controller (R600) setpoint is “25”
- Chan ‘A’ differential pressure is reading -0.30” w.c.
- Chan ‘B’ differential pressure is reading -0.23” w.c.

If desired, how would operators establish a higher steady-state d/p than what currently exists?

A. Change the R600 Controller setpoint to “30”.

B. Change the R600 Controller setpoint to “20”.

C. Place the PRESSURE CONTROL SELECT switch in CHAN A.

D. Place the PRESSURE CONTROL SELECT switch in CHAN B.

Answer: A

Explanation:

With the PRESSURE CONTROL SELECT switch in AUTO, both of the same d/p transmitter signals that feed the two-pen recorders are routed to a “high value selector”. This selector passes only the signal representing the highest “absolute pressure” (i.e., the least negative d/p) on to the R600 Controller as the controller’s feedback signal. In this case, that signal value is currently -0.23” wc coming from Channel ‘B’. Nonetheless, it is still the R600 controller (in AUTO) with a setpoint of “25” (representing a d/p of -0.25” wc) that is in control of the pressure control damper T42-F021.

‘A’ is correct. For the reasons already discussed, only by changing the R600 setpoint to a higher (i.e., more negative) value (“30” versus “25”) can a higher (more negative) d/p be established.
‘B’ is wrong. Changing the setpoint to “20” would establish a lower (less negative) d/p. Its plausibility speaks for itself.

‘C’ and ‘D’ are wrong. So long as the R600 Controller is in AUTO it always remains in control of the system. All that is accomplished by repositioning the PRESSURE CONTROL SELECTOR switch to either Chan A or Chan B is to hand-pick that signal to be the particular feedback signal to the R600 Controller…rather than letting the “high value selector” function work (as it otherwise would with the switch in the AUTO position). Plausibility of these choices is based on the Applicant’s need to understand this design.

Technical References:

04-1-01-T42-1, Fuel Handling Area Ventilation System SOI

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-T4200, Objective 10a

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b)(7)	
	55.43(b)	

Examination Outline Cross-Reference	Level	RO
203000 RHR/LPCI: Injection Mode	**Tier #**	2
	Group #	1
Ability to predict and/or monitor changes in parameters associated with operating the RHR/LPCI: INJECTION MODE (PLANT SPECIFIC) controls including: A1.03 System flow	**K/A #**	203000 A1.03
	Rating	3.8
	Rev / Date	1 / 11-02-11

Question 38

A LOCA is in progress.

Reactor pressure is 800 psig when operators open 8 ADS Valves for Emergency Depressurization.

All low pressure ECCS systems are lined up for injection.

Operators can expect to see positive indication of RHR/LPCI injection flow into the RPV as **early** as when reactor pressure reaches approximately ______ psig.

A. 630

B. 490

C. 476

D. 285

Answer: D

Explanation:

'A' is wrong but is plausible because this pressure is the earliest at which we can expect to see injection with the Condensate Booster Pumps.

'B' is wrong but is plausible because this pressure is the earliest at which we can expect to see injection with the LPCS Pump.

'C' is wrong but is plausible because this is the auto-opening pressure permissive setpoint for the LPCI injection valve (E12-F042).

'D' is correct. This is the "earliest" pressure at which we'll see positive indication of flow on P601, as identified in the RHR SOI section for "Manual Startup of RHR in LPCI Mode".

Technical References: 04-1-01-E12-1, RHR SOI, Section 5.4.1			
References to be provided to applicants during exam: None			
Learning Objective: GLP-OPS-E1200, Objective 20			
Question Source:	Bank #		
(note changes; attach parent)	Modified Bank #		
	New	X	
Question History:	Last NRC Exam	No	
Question Cognitive Level:	Memory/Fundamental	X	
	Comprehensive/Analysis		
10CFR Part 55 Content:	55.41(b)(8)		
	55.43(b)		

Examination Outline Cross-Reference	Level	RO
295008 High Reactor Water Level **Knowledge of the reasons for the following responses as they apply to HIGH REACTOR WATER LEVEL: AK3.07 HPCS isolation**	**Tier #**	1
	Group #	2
	K/A #	295008 AK3.07
	Rating	3.2
	Rev / Date	0 / 10-14-11

Question 39

The HPCS injection valve E22-F004 isolates on Reactor Water Level 8…

A. to prevent exceeding cooldown rates as water level approaches the vessel flange.

B. to prevent overflow into the main steam lines.

C. to ensure MCPR is maintained within limits.

D. to delay the HPCS suction swap from the CST to the Suppression Pool.

Answer: B

Explanation:

Answer is taken directly from Bases for Tech Spec 3.3.5.1 (ECCS – Operating), Table 3.3.5.1-1, Function 3.c. Only choice ‘B’ is correct.

‘A’ is wrong but is plausible based on a Note found in the “Refueling” Integrated Operating Procedure (03-1-01-5) that states that whenever water level reaches the flange, a rise in flange cooldown rate is expected.

‘C’ is wrong. Its plausibility is taken from the fact that the MCPR limits for the basis for the Level 8 scram function.

‘D’ is wrong. By needlessly feeding the RPV beyond Level 8, the HPCS suction swap will occur sooner; the CST is the preferred suction source. This is the basis for plausibility.

Technical References:

Tech Spec 3.3.5.1, Table 3.3.5.1-1, Function 3.c

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-E2201, Objective 5.5

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b)(7)	
	55.43(b)	

Examination Outline Cross-Reference	Level	RO
295020 Inadvertent Containment Isolation **Knowledge of the operational implications of the following concepts as they apply to INADVERTENT CONTAINMENT ISOLATION: AK1.02 Power/reactivity control**	**Tier #**	1
	Group #	2
	K/A #	295020 AK1.02
	Rating	3.5
	Rev / Date	0 / 10-14-11

Question 40

The plant is operating at rated power.

A singular failure results in losing electrical power only to the following valve:

- P53-F001, INSTRUMENT AIR SUPPLY HDR TO CTMT

Power to the valve cannot be immediately restored.

Which of the following describes the operational impact of this failure, and why?

A. Operators will have to insert a manual scram either in response to, or to preclude, multiple control rod drifts.

B. Operators will have to insert a manual scram in response to the MSIVs beginning to drift closed.

C. Plant may remain at rated power because this valve is an MOV; it will remain as is.

D. Plant may remain at rated power because this valve is a gagged open AOV; it will remain as is.

Answer: A

Explanation:

P53-F001 is a fully-functional (not gagged open), solenoid-operated AOV, normally-open, delivering Instrument Air to major CTMT systems: the Scram Air Header and the MSIVs. Taking electrical power away from the solenoid fails the AOV closed. The operators will soon afterwards experience multiple control rod drifts as Scram Air Header pressure bleeds off, forcing them to insert a manual scram per the requirements of the CRD Malfunctions ONEP (05-1-02-IV-1), Immediate Operator Action 2.3.1.

‘A’ is correct for the reasons already discussed.

‘B’ is wrong. Although the MSIVs will eventually begin to drift closed, this won’t happen until long after the operators will have had to respond to the rod drifts; it will not have been the reason (i.e., the "why") for inserting a manual scram. This choice's plausibility speaks for itself.

'C' is wrong. As stated, this valve is a fail-closed on loss of power AOV, not one of the CTMT isolation MOVs that would otherwise remain as is. This choice's plausibility speaks for itself.

'D' is wrong. As stated, this valve is a fail-closed on loss of power AOV, not one of the several gagged-open Secondary CTMT isolation AOVs (for example: P53-F026A, Instrument Air Supply to Aux Bldg, which is upstream of the P53-F001 valve) that would remain as is on a loss of power. This choice's plausibility speaks for itself.

Technical References:

05-1-02-IV-1, CRD Malfunctions ONEP
05-1-02-III-5, Automatic Isolations ONEP (see section 3.10)

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-P5100, Objective 42

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b)(7) & (10)	
	55.43(b)	

Examination Outline Cross-Reference	Level	RO
209001 Low Pressure Core Spray System **Knowledge of electrical power supplies to the following: K2.03 Initiation logic**	**Tier #**	2
	Group #	1
	K/A #	209001 K2.03
	Rating	2.9
	Rev / Date	0 / 10-14-11

Question 41

Which of the following is the direct source of electrical power to the LPCS initiation logic?

A. 1Y87

B. 1Y96

C. 11DA

D. 11DB

Answer: C

Explanation:

'A' and 'B' are wrong. These are Div 1 and 3 ESF inverters, respectively. LPCS initiation is DC-powered logic. These choices are plausible to the Applicant who confuses ESF instrumentation loop power (ESF inverter powered) with logic power (DC).

'C' is correct. ESF Div 1 DC Bus 11DA directly powers the LPCS initiation logic (see LPCS SOI, Electrical Lineup Attachment III, page 2 of 2, DC breaker 72-11A18 on Bus 11DA distribution panel 1DA1; also see LPCS Relay Logic drawings E-1182, sheets 023& 026).

'D' is wrong. This is the Div 2 ESF DC Bus. This choice is plausible to the Applicant who fails to recall that LPCS is a strictly Div 1 associated system.

Technical References:

E-1182-023 & 026, LPCS Electrical Drawings
04-1-01-E21-1, LPCS SOI

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-E2100, Objective 9.2

Question Source:	Bank #	

(note changes; attach parent)	Modified Bank #		
	New	X	
Question History:	Last NRC Exam	No	
Question Cognitive Level:	Memory/Fundamental	X	
	Comprehensive/Analysis		
10CFR Part 55 Content:	55.41(b)(8)		
	55.43(b)		

Examination Outline Cross-Reference	Level	RO
215005 Average Power Range Monitor/Local Power Range Monitor System **Ability to manually operate and/or monitor in the control room: A4.06 Verification of proper functioning/ operability**	**Tier #**	2
	Group #	1
	K/A #	215005 A4.06
	Rating	3.6
	Rev / Date	0 / 10-14-11

Question 42

The plant is operating at rated power.

If LPRM 26-27B fails downscale, its associated APRM ‘C’ indicated power will ________.

If LPRM 26-27B is then bypassed, its associated APRM ‘C’ indicated power will ________.

(Assume no APRM gain adjustments are performed.)

A. rise
   lower

B. lower
   rise

C. rise
   remain the same

D. lower
   remain the same

Answer: B

Explanation:

If an LPRM being averaged fails downscale, the average (APRM) reading will lower. When the LPRM is then bypassed, its input is kicked out of the lowered average, resulting in new rise in the APRM average.

For these reasons, only choice ‘B’ is correct.

‘A’, ‘C’, ‘D’ are wrong for the reason ‘B’ is correct. They are plausible to the Applicant who does not fully comprehend the APRM/LPRM relationship and/or the APRM averaging circuit operating principle and/or its related indications.

Technical References: 04-1-01-C51-1, Neutron Monitoring SOI 17-S-02-40, Bypassing and Unbypassing LPRMs			
References to be provided to applicants during exam: None			
Learning Objective: GLP-OPS-C5104, Objective 11.1			
Question Source:	Bank #GGNS-OPS-08868a	X	
(note changes; attach parent)	Modified Bank #		
	New		
Question History:	Last NRC Exam	No	
Question Cognitive Level:	Memory/Fundamental		
	Comprehensive/Analysis	X	
10CFR Part 55 Content:	55.41(b)(7)		
	55.43(b)		

Examination Outline Cross-Reference	Level	RO
218000 Automatic Depressurization System **Knowledge of AUTOMATIC DEPRESSURIZATION SYSTEM design feature(s) and/or interlocks which provide for the following: K4.02 Allows manual initiation of ADS logic**	**Tier #**	2
	Group #	1
	K/A #	218000 K4.02
	Rating	3.8
	Rev / Date	0 / 10-14-11

Question 43

Which of the following is required in order to manually initiate ADS using its MAN INIT pushbuttons at control room panel P601?

A. High drywell pressure 1.39 psig or higher

B. Reactor water level -150.3” or lower (wide range)

C. Reactor water level -150.3” or lower (wide range) <u>with</u> +11.4” or lower (narrow range)

D. Corresponding low pressure ECCS pump running at sufficient discharge pressure

Answer: D

Explanation:

Unlike the automatic initiation logic, the manual initiation requires <u>only</u> that the corresponding (Div 1 or Div 2) low pressure ECCS pump be running at sufficient discharge pressure (see E-1161-005, ADS Relay Logic). For this reason, ‘D’ is correct.

‘A’, ‘B’, ‘C’ are wrong but are plausible because they all represent ADS logic input signals/signal combinations.

Technical References:

E-1161-005, ADS Relay Logic

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-E2202, Objective 15

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No

Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b)(7)	
	55.43(b)	

Examination Outline Cross-Reference	Level	RO
239001 Main and Reheat Steam System **Knowledge of MAIN AND REHEAT STEAM SYSTEM design feature(s) and/or interlocks which provide for the following: K4.08 Removal of non condensable gases from reactor head area**	**Tier #**	2
	Group #	2
	K/A #	239001 K4.08
	Rating	2.5
	Rev / Date	1 / 11-02-11

Question 44

Which of the following describes the RPV head vent valve arrangement?

A. Two valves that connect to the drywell equipment drain sump, one that connects to the ‘A’ main steam line

B. One valve that connects to the drywell equipment drain sump, one that connects to the ‘A’ main steam line, and one that connects to the ‘D’ main steam line

C. Two valves that connect to the drywell floor drain sump, one that connects to the ‘A’ main steam line

D. One valve that connects to the drywell floor drain sump, one that connects to the ‘A’ main steam line, and one that connects to the ‘D’ main steam line

Answer: A

Explanation:

B21-F001 and F002 connect the RPV head to the drywell equipment (not floor) drain sump. B21-F005 connects the head to the ‘A’ MSL (see P&ID M-1077A)

For this reason, only choice ‘A’ is correct.

‘B’, ‘C’, ‘D’ are wrong for the reason ‘A’ is correct. They are plausible because they are all possible valve/MSL arrangements.

Technical References:

M-1077A, Nuclear Boiler System P&ID

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-B1300, Objective 2.1

Question Source:	Bank #	

(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b)(3)	
	55.43(b)	

Examination Outline Cross-Reference	Level	RO
245000 Main Turbine Generator and Auxiliary Systems **Knowledge of the effect that a loss or malfunction of the MAIN TURBINE GENERATOR AND AUXILIARY SYSTEMS will have on following:** **K3.05 Reactor feedwater pump**	**Tier #**	2
	Group #	2
	K/A #	245000 K3.05
	Rating	2.7
	Rev / Date	0 / 10-14-11

Question 45

The plant is operating at rated power when the main turbine trips.

Per the hardcard, one RFPT has now been placed in SPEED AUTO and reactor water level control is in Automatic on the Startup Level Control Valve.

Which of the following describes the current steam supply to the one operating RFPT?

A. Main steam into the RFPT's <u>high</u> pressure steam chest

B. MSR reheat steam into the RFPT's <u>high</u> pressure steam chest

C. Main steam into the RFPT's <u>low</u> pressure steam chest

D. MSR reheat steam into the RFPT's <u>low</u> pressure steam chest

Answer: A

Explanation:

With the main turbine operating above ~35%, the RFPT governor is designed such that it recognizes there is sufficient MSR reheat steam to run the RFPTs by admitting steam through the LP Stop Valves and into the <u>LP</u> steam chest. With the turbine off-line, post-scram, sufficient MSR reheat steam pressure no longer exists; instead, the RFPT relies on main steam. Per the N21 SOI hardcard, control room operators place the reactor on Automatic Startup Level Control after having established a sufficient RFPT discharge pressure; they do this by raising RFPT governor position to ~60% open. This is the point where we're past the "cracking point" for the main steam poppet of the governor valve which now allows main steam into the HP steam chest.

For the reasons above, only 'A' is correct.

'B','C', 'D' are wrong for the reason 'A' is correct. Their plausibility speaks for itself.

Technical References:

04-1-01-N21-1, Feedwater SOI			
References to be provided to applicants during exam: None			
Learning Objective: GLP-OPS-N2100, Objective 8			
Question Source:	Bank #		
(note changes; attach parent)	Modified Bank #		
	New	X	
Question History:	Last NRC Exam	No	
Question Cognitive Level:	Memory/Fundamental	X	
	Comprehensive/Analysis		
10CFR Part 55 Content:	55.41(b)(4)		
	55.43(b)		

Examination Outline Cross-Reference	Level	RO
272000 Radiation Monitoring System **Knowledge of RADIATION MONITORING System design feature(s) and/or interlocks which provide for the following: K4.03 Fail safe tripping of process radiation monitoring logic during conditions of instrument failure**	**Tier #**	2
	Group #	2
	K/A #	272000 K4.03
	Rating	3.6
	Rev / Date	1 / 11-02-11

Question 46

Which of the following will cause an automatic isolation of Secondary Containment?

A. Fuel Handling Area Exhaust rad monitor ‘A’ at 4.0 mR/hr with
Fuel Handling Area Exhaust rad monitor ‘D’ INOP trip

B. Fuel Pool Sweep Exhaust rad monitor ‘A’ at 35 mR/hr with
Fuel Pool Sweep Exhaust rad monitor ‘C’ at 40 mR/hr

C. Fuel Handling Area Exhaust rad monitor ‘B’ at 5.0 mR/hr with
Fuel Handling Area Exhaust rad monitor ‘D’ at 3.0 mR/hr

D. Fuel Pool Sweep Exhaust rad monitor ‘B’ at 25 mR/hr with
Fuel Pool Sweep Exhaust rad monitor ‘C’ INOP trip

Answer: A

Explanation:

Whether it be the Fuel Handling Area or the Fuel Pool Sweep Exhaust Vents, the only rad monitor combination that satisfies the required logic is the channels ‘A’ + ‘D’ combination, or channels ‘B’ + ‘C’.

‘A’ is correct. The Fuel Handling Area channel ‘A’ rad monitor is above the trip setpoint of 3.6 mR/hr and the channel ‘D’ rad monitor is providing the other half of the required logic with a valid INOP trip.

‘B’ is wrong because it suggests a wrong channel combination.

‘C’ is wrong because it shows the wrong coincidence combination (B + D) and because the Fuel Handling Area ‘D’ rad monitor being below the trip setpoint (for Secondary CTMT isolation) of 3.6 mR/hr. It is plausible because it represents two possible channel combinations.

‘D’ is wrong because although a proper channel combination exists, FPS channel ‘B’ is below the 30 mR/hr setpoint.

Technical References: ARI Windows 04-1-02-1H13-P601-19A-B9 & B10		
References to be provided to applicants during exam: None		
Learning Objective: GLP-OPS-T4801, Objective 8.6		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b)(7) & (11)	
	55.43(b)	

Examination Outline Cross-Reference	Level	RO
2.1.44 Knowledge of RO duties in the control room during fuel handling such as responding to alarms from the fuel handling area, communication with the fuel storage facility, systems operated from the control room in support of fueling operations, and supporting instrumentation.	**Tier #**	3
	Group #	
	K/A #	2.1.44
	Rating	3.9
	Rev / Date	0 / 10-14-11

Question 47

Per 01-S-06-50, Control of Fuel Services Operations, personnel performing ____________ shall be in constant communication with (specifically) the _____________ who shall monitor for ______________.

A. fuel movements in or out of the spent fuel pool
ACRO
Fuel Handling Area rising radiation levels

B. fuel movements in or out of the spent fuel pool
CRS
Fuel Handling Area rising radiation levels

C. core alterations
ACRO
inadvertent criticality

D. core alterations
STA
inadvertent criticality

Answer: C

Explanation:

See 01-S-06-50, section 6.6.8. The only correct answer choice 'C' is taken directly from this section.

Choices 'A', 'B' and 'D' are wrong. Neither of the statements suggested by these choices are found anywhere in the subject procedure, nor within any other fuel handling-related site-specific or fleet procedures. They are plausible, however, either because they represent possible fuel handling-related activities and/or personnel who logically might be involved in such activities.

Technical References:

01-S-06-50, Control of Fuel Services Operations			
References to be provided to applicants during exam: None			
Learning Objective: GLP-OPS-PROC, Objective 18.3			
Question Source:	Bank #		
(note changes; attach parent)	Modified Bank #		
	New	X	
Question History:	Last NRC Exam	No	
Question Cognitive Level:	Memory/Fundamental	X	
	Comprehensive/Analysis		
10CFR Part 55 Content:	55.41(b)(10)		
	55.43(b)		

Examination Outline Cross-Reference	Level	RO
2.2.35 Ability to determine Technical Specification Mode of Operation.	**Tier #**	3
	Group #	
	K/A #	2.2.35
	Rating	3.6
	Rev / Date	0 / 10-14-11

Question 48

Average reactor coolant temperature is 220 °F.

What is/are the possible MODE(s) of operation that the plant could be operating in at this temperature?

Only the following MODE(s):

A. 1, 2, or 3

B. 2 or 3

C. 2

D. 3

Answer: B

Explanation:

See TS Table 1.1-1, Modes of Operation. By definition, the plant would be in MODE 3 if the Mode Switch were in Shutdown and temperature were >200 F. Per IOI-1, during a plant pressurization, the plant remains in MODE 2 (MS in Startup) while the plant heats up and pressurizes to near rated pressure, after which the MS is transferred to RUN (MODE 1). By the time MODE 1 is reached, reactor coolant temperature is well above 200 F.

For these reasons, only choice ‘B’ is correct.

‘A’, ‘C’, ‘D’ are wrong for the reason ‘B’ is correct. They are plausible because they each represent a possible MODE and the need for the Applicant to firmly grasp/apply MODE definitions in order to eliminate them as potential answers.

Technical References:

Tech Spec Table 1.1-1, MODES 03-1-01-1, Cold Shutdown to Generator Carrying Minimum Load			
References to be provided to applicants during exam: None			
Learning Objective: GLP-OPS-TS001, Objective 5			
Question Source:	Bank # Held in NRC Exam Room Bank	X	
(note changes; attach parent)	Modified Bank #		
	New		
Question History:	Last NRC Exam	No	
Question Cognitive Level:	Memory/Fundamental		
	Comprehensive/Analysis	X	
10CFR Part 55 Content:	55.41(b)(5)		
	55.43(b)		

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 #	
	K/A #	2.3.13
	Rating	3.4
	Rev / Date	1 / 10-24-11

Question 49

During a refueling outage, operators are moving fuel between the reactor cavity and the spent fuel pool.

An irradiated fuel assembly is dropped and damaged on the CTMT elevation 208'.

Rising airborne activity has been confirmed on CTMT elevation 208'.

Radiation levels have <u>not</u> risen to the point of any automatic isolations or system initiations.

Per the "High Radiation During Fuel Handling" ONEP, which of the following describes <u>required</u> operator actions?

A. Start an available SGTS train but leave Containment Cooling in its Normal Cooling Mode.

B. Start an available SGTS train and place Containment Cooling in its Containment Cleanup Mode.

C. Start both SGTS trains but leave Containment Cooling in its Normal Cooling Mode.

D. Start both SGTS trains and place Containment Cooling in its Low Volume Purge Mode.

Answer: B

Explanation:

The ONEP (05-1-02-II-8), section 3.4, directs us to place Containment Cooling in Cleanup Mode (i.e., a second Recirc charcoal filter is placed in service). Section 3.1.2 directs us to start an available SGTS train; starting both trains is <u>not</u> <u>required</u>. For these reasons, choice 'B' is correct.

'A' and 'C' are wrong for the reason 'B' is correct.

'D' is wrong for the reason 'B' is correct. Plausible because the Containment Cooling System does have a Low Volume Purge Mode, as well.

Technical References: 05-1-02-II-8, High Radiation During Fuel Handling ONEP			
References to be provided to applicants during exam: None			
Learning Objective: GLP-OPS-ONEP, Objective 2.0			
Question Source:	Bank #		
(note changes; attach parent)	Modified Bank #		
	New	X	
Question History:	Last NRC Exam	No	
Question Cognitive Level:	Memory/Fundamental	X	
	Comprehensive/Analysis		
10CFR Part 55 Content:	55.41(b)(8) & (10)		
	55.43(b)		

Examination Outline Cross-Reference	Level	RO
295018 Partial or Complete Loss of Component Cooling Water	**Tier #**	1
	Group #	1
2.4.11 Knowledge of abnormal condition procedures.	**K/A #**	295018 2.4.11
	Rating	4.0
	Rev / Date	2 / 11-17-11

Question 50

The plant is operating at rated power when the following occur:

- Three radial well pumps have tripped
- CCW temperature exceeds 100°F due to the loss of PSW
- At the same time, CCW system temperature is now unable to keep reactor recirc pump seal cavity temperatures below their recorder alarm setpoints

What is the required operator action?

A. Immediately insert a manual scram.

B. Immediately insert a manual scram and manually trip the reactor recirc pumps.

C. Immediately reduce core flow to 67 Mlbm/hr.

D. Immediately reduce core flow to 67 Mlbm/hr, wait 10 minutes then insert a manual scram.

Answer: B

Explanation:

The trip of three radial well pumps is a "Partial loss of PSW" (per that ONEP), section 3.1, allowing time for attempted recovery of pumps and methodical monitoring of CCW and TBCW system temperatures. 3.1.3.a says to declare it a "complete" loss of PSW when CCW temperature cannot be restored and maintained at or below 100°F, at which point, step 2.1 has us reduce core flow to 67 Mlbm/hr. However, the 3rd bullet in the stem tells us that we're already above the recirc pump alarm setpoints. Per the Loss of CCW ONEP, section 3.1.4.b, we're directed to manually scram and manually trip the recirc pumps. Thus, answer choice 'B' is correct.

'A' is wrong because it suggests that we need only scram the reactor; it fails to trip the recirc pumps.

'C' is wrong because although it is the action in response to the loss of PSW ONEP (once the 100°F CCW temperature is reached) that action is moot next to the higher-priority action of immediately scramming and tripping the recirc pumps per the loss of CCW ONEP.

'D' is wrong. This action is taken from the Loss of PSW ONEP, which suggests that a quick reduction in core flow to 67 Mlbm/hr will allow about 10 minutes before having to insert a manual scram.			
Technical References: 05-1-02-V-1, Loss of CCW ONEP 05-1-02-V-2, Loss of TBCW ONEP 05-1-02-V-11, Loss of PSW ONEP			
References to be provided to applicants during exam: None			
Learning Objective: GLP-OPS-ONEP, Objective 1			
Question Source:	Bank #		
(note changes; attach parent)	Modified Bank #		
	New	X	
Question History:	Last NRC Exam	No	
Question Cognitive Level:	Memory/Fundamental	X	
	Comprehensive/Analysis		
10CFR Part 55 Content:	55.41(b)(10)		
	55.43(b)		

Examination Outline Cross-Reference	Level	RO
286000 Fire Protection System **Ability to (a) predict the impacts of the following on the FIRE PROTECTION SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: A2.08 Failure to actuate when required**	**Tier #**	2
	Group #	2
	K/A #	286000 A2.08
	Rating	3.2
	Rev / Date	0 / 10-14-11

Question 51

A fire has started in the Div 2 ESF switchgear room.

To manually initiate the CARDOX system an operator has used the local pushbutton station in an effort to release CO2 into the room.

40 seconds later CO2 flow into the room has not started.

Which of the following describes the operation of this portion of the CARDOX system?

A. CO2 flow into the room should have already started.

The operator can flood the room with CO2 by rotating the arm to OPEN then back to CLOSE at the local Electro Manual Station; no other operator assistance is required.

B. CO2 flow into the room should have already started.

The operator can flood the room with CO2 by rotating the arm to OPEN then back to CLOSE at the local Electro Manual Station; operator assistance is required at the 10-ton CO2 Storage Unit.

C. CO2 should begin to flow into the room in another 10 seconds.

If not, the operator can flood the room with CO2 by rotating the arm to OPEN then back to CLOSE at the local Electro Manual Station; no other operator assistance is required.

D. CO2 should begin to flow into the room in another 10 seconds.

If not, the operator can flood the room with CO2 by rotating the arm to OPEN then back to CLOSE at the local Electro Manual Station; operator assistance is required at the 10-ton CO2 Storage Unit.

Answer: B

Explanation:

There are two local stations: a break glass pushbutton station, and a break glass rotate arm (Electro Manual) station. Use of the pushbutton station amounts to a forced automatic CO2 initiation after a 30-second time delay. Therefore, 40 seconds after using the pushbutton CO2 into the room should have already started. The use of the Electro Manual Station's rotate arm to OPEN then back to CLOSE is directed by the CARDOX system SOI (04-1-01-P64-3), section 5.2. This evolution does require the assistance of another operator at the 10-ton CO2 Storage Unit who must first flood the CO2 header with CO2 by removing the pin from the Master Valve Manual SVF373 Release Lever and placing that lever to the OPEN position.

Based on the above, choice 'B' is correct and 'A' is wrong (but plausible based on the need for the Applicant to recall the need for an operator to flood the header at the 10-ton station.)

'C' and 'D' are wrong for the reason 'B' is correct. They are plausible to the Applicant who recalls that there is some amount of time delay, but who confuses the actual 30-second delay with the fact that the ESF room vent exhaust fans automatically trip off in response to a fire detection signal in the room and then auto-restart 50 seconds later (see lesson plan GLP-OPS-Z7700, page 11 of 19).

Technical References:

04-1-01-P64-3, Fire Protection Cardox System

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-P6400, Objective 8

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b)(4) & (10)	
	55.43(b)	

Examination Outline Cross-Reference	Level	RO
295014 Inadvertent Reactivity Addition **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 AA1.06
	Rating	3.3
	Rev / Date	0 / 10-14-11

Question 52

A plant power ascension is in progress with current reactor power at 20%.

A feedwater heating problem causes an inadvertent reactor power excursion.

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) 30%
   (2) PRESS REF LOWER

B. (1) 30%
   (2) LOAD DEMAND RAISE

C. (1) 35%
   (2) PRESS REF LOWER

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

Answer: D

Explanation:

Choices ‘A’ and ‘C’ are wrong. Lowering the Pressure Reference setpoint will result in a throttling open 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 Pressure Reference setpoint.

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 closed of the BPVs.

IOI-2, P/L 2.15 prohibits control rod withdrawal above 35% with open Bypass Valves.

For these reasons, answer choice 'D' is correct.

'B' is wrong because the power level is 35% not 30% as suggested in this answer choice. Plausibility of this 30% value is based on the Applicant's recall that we upshift recirc pumps from slow to fast speed when we've reached 30 to 32% power during the power ascension.

Technical References:

03-1-01-2, Power Operations.

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-N3202, Objective 13

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b)(7) & (10)	
	55.43(b)	

Examination Outline Cross-Reference	Level	RO
295021 Loss of Shutdown Cooling **Knowledge of the interrelations between LOSS OF SHUTDOWN COOLING and the following: AK2.01 Reactor water temperature**	**Tier #**	1
	Group #	1
	K/A #	295021 AK2.01
	Rating	3.6
	Rev / Date	1 / 11-01-11

Question 53

A refueling outage is in progress with RHR Shutdown Cooling (SDC) in operation.

RHR SDC is lost and no alternate means of decay heat removal is available.

Of the following situations, which would result in the **SHORTEST** amount of time for reactor vessel water temperature to reach 200°F?

If this loss of decay heat removal event were to happen…

A. prior to any core alterations and with the reactor cavity flooded up.

B. after core alterations have been completed and with the reactor cavity flooded up.

C. prior to any core alterations and with the reactor cavity drained down.

D. after core alterations have been completed and with the reactor cavity drained down.

Answer: C

Explanation:

Water temperature will increase the fastest for the situation where the core contains the greatest amount of irradiated fuel (i.e., pre-fuel shuffle) and is covered by the least amount of water (i.e., drained down, rather than flooded up).

Based on this, ‘C’ is correct and ‘A’, ‘B’, ‘D’ are wrong but plausible because they each represent a real situation during which a loss of decay heat removal might occur and are accounted for on the “Time To Reach 200°F” Figures of the “Inadequate Decay Heat Removal” ONEP (05-1-02-III-1).

Technical References: 05-1-02-III-1, Inadequate Decay Heat Removal ONEP			
References to be provided to applicants during exam: None			
Learning Objective: GLP-OPS-ONEP, Objective 2			
Question Source:	Bank #		
(note changes; attach parent)	Modified Bank #		
	New	X	
Question History:	Last NRC Exam	No	
Question Cognitive Level:	Memory/Fundamental		
	Comprehensive/Analysis	X	
10CFR Part 55 Content:	55.41(b)(10) & (14)		
	55.43(b)		

Examination Outline Cross-Reference	Level	RO
295031 Reactor Low Water Level **Ability to operate and/or monitor the following as they apply to REACTOR LOW WATER LEVEL: EA1.10 Control rod drive**	**Tier #**	1
	Group #	1
	K/A #	295031 EA1.10
	Rating	3.6
	Rev / Date	0 / 10-14-11

Question 54

While operating in EP-2 (RPV Control), operators have completed lining up CRD for injection per the hardcard.

Which of the following describes that CRD system lineup?

A. One pump running
Flow Control Valve (F002A(B)) fully open
Pressure Control Valve (F003) fully closed

B. Two pumps running
Flow Control Valve (F002A(B)) fully open
Pressure Control Valve (F003) fully open

C. One pump running
Flow Control Valve (F002A(B)) fully closed
Pressure Control Valve (F003) fully open

D. Two pumps running
Flow Control Valve (F002A(B)) fully open
Pressure Control Valve (F003) fully closed

Answer: B

Explanation:

See the CRD system SOI (04-1-01-C11-1), hardcard (Attachment VIII).

In EP-2 (non-ATWS), the Applicant should recognize that CRD will be “maximized for flow”, requiring two pumps running, a fully open flow control valve (F002A or B), and a fully open drive water pressure control valve.

For this reason, only choice ‘B’ is correct.

‘A’, ‘C’, and ‘D’ are wrong; their plausibility is based on possible lineups for these system components. ‘D’, especially, is the lineup for “maximizing CRD for pressure” if operating in EP-

2A (ATWS RPV Control).

Technical References:

EP-2, RPV Control
04-1-01-C11-1, CRD System SOI

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-C111A, Objective 9

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b)(7) & (10)	
	55.43(b)	

Examination Outline Cross-Reference	Level	RO
295038 High Off-Site Release Rate **Knowledge of the interrelations between HIGH OFF-SITE RELEASE RATE and the following: EK2.05 †Site emergency plan**	**Tier #**	1
	Group #	1
	K/A #	295038 EK2.05
	Rating	3.7
	Rev / Date	0 / 10-14-11

Question 55

Which of the following identifies the emergency classification level specified as the EP-4 (Auxiliary Building Control) entry condition for "offsite radiation release"?

A. Unusual Event

B. Alert

C. Site Area Emergency

D. General Emergency

Answer: B

Explanation:

See EP-4 entry conditions. Only choice 'B' is correct. All other choices are wrong but plausible simply because they are the other three possible emergency action levels.

Technical References:

EP-4, Auxiliary Building Control

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-EP4, Objective 5

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	

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

Examination Outline Cross-Reference	Level	RO
600000 Plant Fire On Site **2.4.49 Ability to perform without reference to procedures those actions that require immediate operation of system components and controls.**	**Tier #**	1
	Group #	1
	K/A #	600000 2.4.49
	Rating	4.6
	Rev / Date	0 / 10-14-11

Question 56

A fire has started in Div 2 Diesel Generator Room.

The fire brigade is responding.

Per 10-S-03-2, Response To Fires, which of the following DG Room Outside Air Fans is/are required to be started?

A. Div 1, only

B. Div 3, only

C. Div 1 and Div 3, only

D. Div 2 and Div 3, only

Answer: C

Explanation:

See 10-S-03-2, section 6.2.3 (Control Room Operator Actions), specifically, step 6.2.3.g, which directs the operator to start the ventilation systems (understood to be the O/A fans) in the other rooms (in this case, Div 1 and Div 3 rooms).

For this reason, choice 'C' is correct and the other choices are wrong but plausible simply because they represent potential rooms for which the vent systems must be running unless the Applicant recalls the specific direction given in this procedure.

Technical References:

10-S-03-2, Response To Fires

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-PROC, Objective 58.4

Question Source:	Bank # Held in NRC Exam Room Bank	X
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b)(10)	
	55.43(b)	

Examination Outline Cross-Reference	Level	RO
700000 Generator Voltage and Electric Grid Disturbances **Ability to determine and/or interpret the following as they apply to GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES: AA2.01 Operating point on the generator capability curve**	**Tier #**	1
	Group #	1
	K/A #	7000000 AA2.01
	Rating	3.5
	Rev / Date	0 / 10-14-11

Question 57

Refer the figure on the next sheet to answer this question.

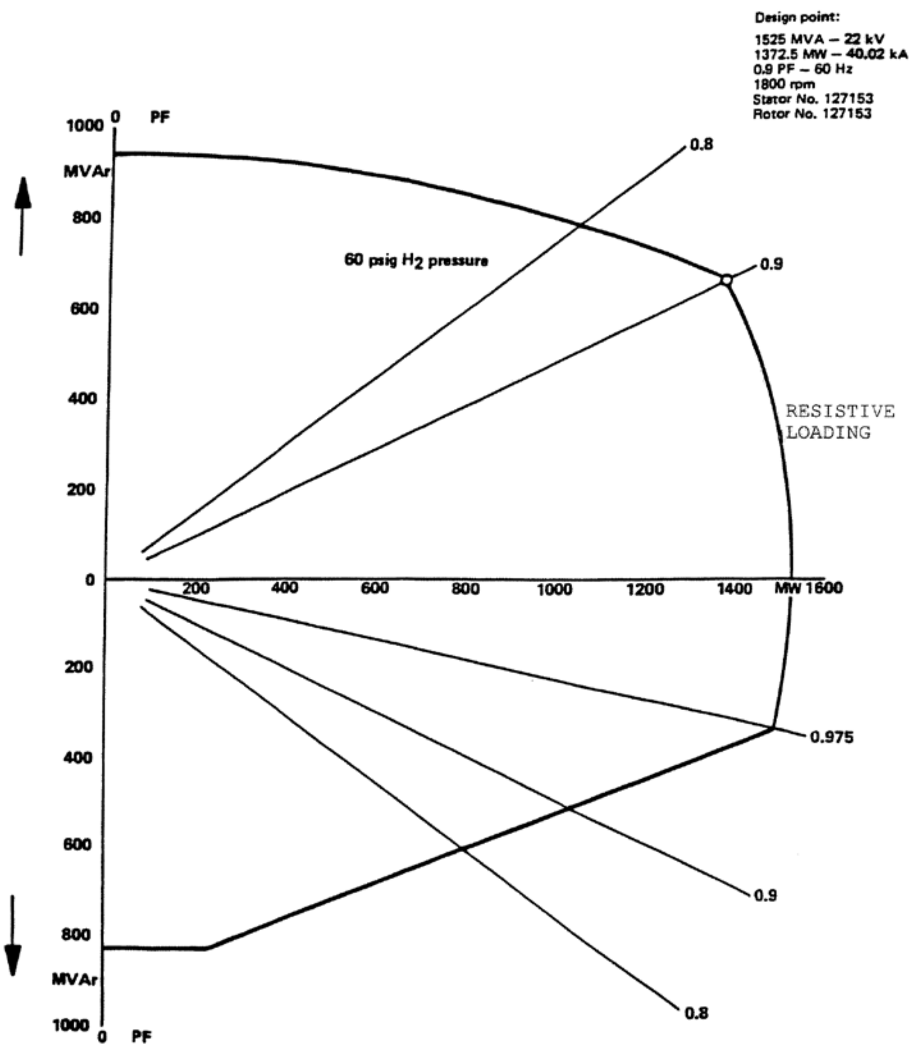
Which of the following is a main generator configuration that poses an operational **RISK** to either grid stability and/or protection of the generator itself?

A. Carrying 800 MWe while operating at a 0.9 pf.

B. Carrying any amount of real load (MWe) with the generator under-excited.

C. Carrying 1000 MWe and +50 MVARs with a steady-state hydrogen gas pressure of 58 psig.

D. Carrying 600 MWe and -100 MVARs with a single hydrogen cooler section out-of-service.



Design point:
1525 MVA – 22 kV
1372.5 MW – 40.02 kA
0.9 PF – 60 Hz
1800 rpm
Stator No. 127153
Rotor No. 127153
1000
0
PF
MVAr
800
600
400
200
0
60 psig H_2 pressure
0.8
0.9
RESISTIVE
LOADING
200
400
600
800
1000
1200
1400
MW 1600
200
400
600
800
MVAr
1000
0
PF
0.975
0.9
0.8

Answer: A			

Explanation:

The figure is the GGNS generator capacity curve ("spider curve") found in its Integrated Operating Procedure 03-1-01-2, Power Operation.

'A' is correct. The suggested MWe-versus-pf operating point clearly shows an MVAR load in excess of +250 MVARs. This violates the IOI-2 Precaution/Limitation 2.4 "emergency operation" limit. In fact, the plotted point is actually about 300 MVARs which exceeds +272 MVARs, the value at which the generator's "Reverse Power Relay" may not recognize a reverse power condition (as described in this same P/L).

'B' is wrong. The Applicant is expected to interpret "under-excited" as simply meaning to operate with –MVARs. There are no restrictions to operating under-excited (except not to exceed -170 MVARs, as stated in P/L 2.4); nor does under-excitation "at any real load" in itself represent a risk. Plausibility of this choice is based simply on the need for the Applicant to consider the concept of "under-excitation" and related it back to his recall of P/L 2.4.

'C' is wrong. This is the more difficult distracter because GGNS's generator capacity does not provide the typical "spider curve" family of curves (i.e., several curves for differing hydrogen gas pressures). This is because GGNS always attempts to maintain 60 psig (see the operator actions for alarm response instruction 04-1-02-1H22-P148-2A-B4, "H2 PRESSURE LOW"). However, that ARI shows that not until a pressure of 54 psig would we consider unloading the generator for a possible generator shutdown.

'D' is wrong. This choices suggests a MWe-versus-MVAR operating point that yields a total reactive load (MVA) that is still well below the "90% of rated MVA" that we can operate at with a single hydrogen cooler section out-of-service. Plausibility of choice is based on the Applicant being able to determine that this operating point must be well within the limits for a single cooler OOS.

Technical References:

03-1-01-2, Power Operations
04-1-02-1H22-P148-2A-B4, H2 PRESSURE LOW alarm response instruction
04-1-01-N40-1, Main Generator and Auxiliaries SOI

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-N4151, Objective 14

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

	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b)(10)	
	55.43(b)	

Examination Outline Cross-Reference	Level	RO
295028 High Drywell Temperature **Knowledge of the operational implications of the following concepts as they apply to HIGH DRYWELL TEMPERATURE: EK1.01 Reactor water level measurement**	**Tier #**	1
	Group #	1
	K/A #	295028 EK1.01
	Rating	3.5
	Rev / Date	0 / 10-14-11

Question 58

Refer to the EP-1 CAUTION 1 on the next sheet to answer this question.

A LOCA is in progress with the following:

- Wide Range level is -10"
- Fuel Zone level is -25"
- Upset Range level is 5"
- Shutdown Range level is 10"
- 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 instruments **is/are** usable?

A. Fuel Zone Range, only

B. Fuel Zone Range and Wide Range, only

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

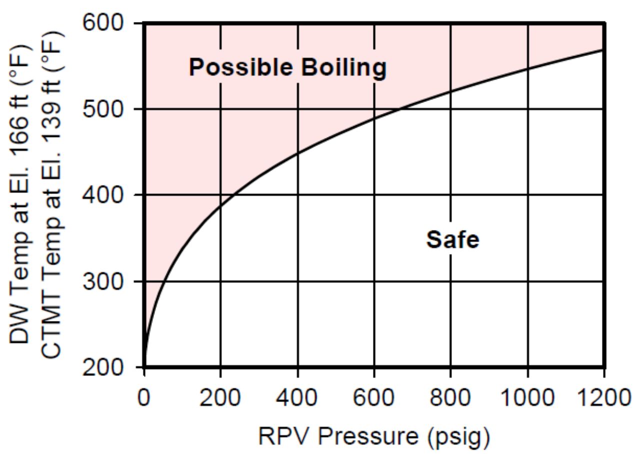
D. Fuel Zone Range, Wide Range, Upset Range, and Shutdown Range

CAUTIONS

1 RPV level indications are affected by instrument run temperatures and RPV pressure:

1. If the temperature near any RPV level instrument leg is in the **Possible Boiling** zone of the **RPVST** (Fig 2), the instrument may be unreliable due to boiling in the run.

Fig 2: RPV Saturation Temperature (RPVST)



2. Do <u>not</u> use any RPV level instrument that exceeds <u>both</u> limits in the following table:

Level Instrument	Indicated Level		Reference Leg Temperature
Wide Range	< –131 in.	and	> 143°F at CTMT El. 139 ft.
Upset Range	< 159 in.	and	> 195°F at DW El. 166 ft.
Shutdown Range	< 139 in.	and	> 66°F at DW El. 166 ft.

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. 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.

For these reasons (i.e., only the Fuel Zone and Wide Ranges are usable), choice 'B' is correct.

Choices 'A', 'C', 'D' are wrong but are plausible based on the Applicant's need to apply Caution 1 as already described.

Technical References:

EP-1, CAUTION 1

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-EP02, Objective 8

Question Source:	Bank # GGNS-OPS-05326a	X
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b)(10)	
	55.43(b)	

Examination Outline Cross-Reference	Level	RO
295023 Refueling Accidents **Knowledge of the reasons for the following responses as they apply to REFUELING ACCIDENTS: AK3.02 Interlocks associated with fuel handling equipment**	**Tier #**	1
	Group #	1
	K/A #	295023 AK3.02
	Rating	3.4
	Rev / Date	0 / 10-14-11

Question 59

For fuel handling operations, interlocks that prevent adding excessive reactivity to the core that could lead to a refueling accident are the…

A. fuel handling platform bridge and trolley interlocks.

B. refueling platform bridge and main hoist interlocks.

C. fuel handling platform main hoist interlocks.

D. refueling platform frame-mounted and monorail-mounted auxiliary hoist interlocks.

Answer: B

Explanation:

'A' is wrong. The fuel handling platform is used exclusively at the spent fuel pool, not at the core. The choice is plausible to the Applicant who fails to recall this fact.

'B' is correct. The refueling platform bridge (Reverse Travel #1 & #2) and Main Hoist interlocks provide such excess reactivity addition protection, as described in the GLP-RF-F1101 lesson plan.

'C' is wrong for the same reason as 'A'. It is plausible for the same reason.

'D' is wrong. Although these interlocks are on the refueling bridge, they have nothing to do with protection against adding excess reactivity; rather, they protect against hoist over-loading and hoist over-travel. Plausibility is similar to that of choices 'A' and 'C'.

Technical References:

04-1-01-F11-1, Refueling Platform SOI

References to be provided to applicants during exam: None

Learning Objective: GLP-RF-RF1101, Objective 25		
Question Source:	Bank # Held in NRC Exam Room	X
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	**YES (2009 NRC)**
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b)(6)	
	55.43(b)	

Examination Outline Cross-Reference	Level	RO
295026 Suppression Pool High Water Temperature **2.2.39 Knowledge of less than or equal to one hour Technical Specification action statements for systems.**	**Tier #**	1
	Group #	1
	K/A #	295026 2.2.39
	Rating	3.9
	Rev / Date	0 / 10-14-11

Question 60

The plant is operating at rated power with a suppression pool average temperature of 85°F.

A single SRV sticks open, remains open, and suppression pool temperature begins to rise.

At 95°F, operators enter Tech Spec 3.6.2.1, Suppression Pool Average Temperature, and take the required action.

What is that LCO required action?

A. Verify suppression pool temperature is ≤ 110°F within 1 hour.

B. Restore suppression pool temperature to ≤ 95°F within 1 hour.

C. Reduce thermal power to ≤ 1% RTP within 1 hour.

D. Immediately place the mode switch in shutdown.

Answer: A

Explanation:

See LCO 3.6.2.1, ACTION A.1

'A' is correct per ACTION A.1.

'B' is wrong. Per ACTION A.2, we have 24 hours (not 1 hour) to restore to ≤ 95°F. Plausibility is based on this possible ACTION choice.

'C' is wrong. This choice is a hybrid of ACTION B.1 and the consistency of a "1-hour" action statement "from memory" for RO Applicants; plausibility is linked to it discriminatory ability.

'D' is wrong. Per ACTION D.1, this is the required action for a 110°F temperature, not 95°F. Plausibility is based on this possible ACTION choice.

Technical References: Tech Spec LCO 3.6.2.1, Suppression Pool Average Temperature			
References to be provided to applicants during exam: None			
Learning Objective: GLP-OPS-TS001, Objective 39			
Question Source:	Bank #		
(note changes; attach parent)	Modified Bank #		
	New	X	
Question History:	Last NRC Exam	No	
Question Cognitive Level:	Memory/Fundamental	X	
	Comprehensive/Analysis		
10CFR Part 55 Content:	55.41(b)(10)		
	55.43(b)		

Examination Outline Cross-Reference	Level	RO
295022 Loss of CRD Pumps **Ability to operate and/or monitor the following as they apply to LOSS OF CRD PUMPS: AA1.01 CRD hydraulic system**	**Tier #**	1
	Group #	2
	K/A #	295022 AA1.01
	Rating	3.1
	Rev / Date	0 / 10-14-11

Question 61

The plant is operating at rated power when the running CRD pump trips.

Which of the following describes the P601 indications associated with the in-service CRD flow control valve (F002A), and its controller, that result from the pump trip?

	F002A Green light	**F002A Red light**	**Controller Output**
A.	ON	ON	70%
B.	ON	OFF	0%
C.	OFF	ON	0%
D.	OFF	ON	100%

Answer: D

Explanation:

Prior to the pump trip, controller output is ~70% keeping the in-service F002A throttled to maintain about 60 gpm cooling water header flow. With F002A throttled, it has a dual (Green & Red lights both ON) status at P601. After the pump trips system flow goes to zero, the flow controller senses this, maximizes its output signal (i.e., 100%), and fully opens the in-service F002A in a fruitless attempt to restore system flow. With F002A fully open, its Green light goes OFF, leaving only its Red light ON.

For these reasons, only choice ‘D’ is correct.

‘A’ is wrong (but plausible because it represents a possible indication option). This choice suggests the valve remains as is after the pump trips.

‘B’ is wrong (again plausible for the same reason). This choice suggests the controller fully closes the valve.

‘C’ is wrong (again plausible for the same reason). This choice suggests a relationship of the valve going fully open with a 0% controller output (i.e., reverse acting).			
Technical References: GLP-OPS-C111A, CRD System			
References to be provided to applicants during exam: None			
Learning Objective: GLP-OPS-C111A, Objective 8			
Question Source:	Bank #		
(note changes; attach parent)	Modified Bank #		
	New	X	
Question History:	Last NRC Exam	No	
Question Cognitive Level:	Memory/Fundamental		
	Comprehensive/Analysis	X	
10CFR Part 55 Content:	55.41(b)(7)		
	55.43(b)		

Examination Outline Cross-Reference	Level	RO
209002 High Pressure Core Spray System (HPCS) **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: A2.05 D.C. electrical failure: BWR-5,6**	**Tier #**	2
	Group #	1
	K/A #	209002 A2.05
	Rating	2.8
	Rev / Date	0 / 10-14-11

Question 62

With an ATWS and LOCA in progress, HPCS has been manually overridden at P601.

LOCA initiation signals still exist in all three divisions.

Then, the following occur at P601:

- HPCS SYS OOSVC annunciator seals-in
- HPCS LOGIC POWER FAILURE amber light (postage stamp) is lit

(1) What RO action, if any, is required <u>immediately</u> after receiving these alarms?

(2) What RO action will be required should HPCS injection to the RPV become necessary per EP-2A?

A. (1) Close the HPCS injection valve and stop the HPCS pump.
(2) Manually align the system for injection using the HPCS SOI.

B. (1) No action is required.
(2) Manually align the system for injection using the HPCS SOI.

C. (1) No action is required.
(2) Start the HPCS pump and open the HPCS injection valve.

D. (1) Close the HPCS injection valve and stop the HPCS pump.
(2) Start the HPCS pump and open the HPCS injection valve.

Answer: C

Explanation:

The stem condition alarms show that the DC-powered HPCS initiation logic power has been lost. The Part (1) operator action suggested by answer choices ‘A’ and ‘D’ are plausible to the Applicant who is confused into believing that a logic power loss causes the pre-existing “manual override” to drop out, resulting in an auto-start of the HPCS pump and auto-opening of its injection valve…historically, this is a fairly common initial RO/SRO trainee misconception. This is not the way the system is designed, however; therefore, choices ‘A’ and ‘D’ are wrong.

The specific answer to the second part of the question is found in the alarm response instruction (ARI) 04-1-02-1H13-P601-16A-H5, HPCS SYS OOSVC. That ARI identifies the accompanying HPCS LOGIC POWER FAILURE amber light and reminds operators that a HPCS initiation signal cannot be generated (manually or automatically). Therefore, in order to begin injecting with HPCS the operator simply needs to start the pump and open the injection valve.

For these reasons, choice ‘C’ is correct and choice ‘B’ is wrong (its plausibility is inherent in the above discussions).

Technical References:

04-1-02-1H13-P601-16A-H5, HPCS SYS OOSVC alarm response instruction
04-1-01-E22-1, HPCS System SOI
E-1183-023, HPCS System Relay Logic schematic

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-E2201, Objective 13.2

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b)(7) & (10)	
	55.43(b)	

Examination Outline Cross-Reference	Level	RO
215005 Average Power Range Monitor/Local Power Range Monitor System **Knowledge of the physical connections and/or cause effect relationships between AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM and the following: K1.16 Flow converter/comparator network**	**Tier #**	2
	Group #	1
	K/A #	215005 K1.16
	Rating	3.3
	Rev / Date	0 / 10-14-11

Question 63

The plant is operating at rated power.

An I&C technician places the APRM FLOW UNIT 'A' Unit Mode switch to TEST.

Which of the following occurs?

A. Control rod block
Half-scram generated by INOP from APRM 'A' only

B. Control rod block
Half-scram generated by INOP trips from both APRM 'A' and APRM 'E'

C. No control rod block
Half-scram generated by INOP from APRM 'A' only

D. No control rod block
Half-scram generated by INOP trips from both APRM 'A' and APRM 'E'

Answer: B

Explanation:

When an APRM flow unit is taken to TEST, an INOP Trip signal is generated due to a "flow unit switch out of operate" condition. The APRM Flow Unit 'A' feeds both APRM 'A' and APRM 'E'. Because it does, both of those APRMs generate INOP Trips as a result, thus producing the RPS half-scram, while also sending a control rod block request to RC&IS.

For these reasons, choice 'B' is correct.

'A', 'C', 'D' are wrong as described above, but are plausible because they each represent a possible result given the above description.

Technical References: 04-1-01-C51-1, Neutron Monitoring System SOI			
References to be provided to applicants during exam: None			
Learning Objective: GLP-OPS-C5104, Objective 9.1			
Question Source:	Bank # GGNS-OPS-08843		X
(note changes; attach parent)	Modified Bank #		
	New		
Question History:	Last NRC Exam		No
Question Cognitive Level:	Memory/Fundamental		
	Comprehensive/Analysis		X
10CFR Part 55 Content:	55.41(b)(7)		
	55.43(b)		

Examination Outline Cross-Reference	Level	RO
217000 Reactor Core Isolation Cooling System (RCIC) Ability to (a) predict the impacts of the following on the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC); 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 vacuum pump	**Tier #**	2
	Group #	1
	K/A #	217000 A2.09
	Rating	2.9
	Rev / Date	0 / 10-14-11

Question 64

A small Feedwater line leak in the drywell causes operators to insert a manual scram due to rising drywell pressure.

As the scram occurs, both RFPTs trip, causing RCIC to automatically initiate.

Then, all ECCS systems automatically initiate on high drywell pressure.

Operators secure Condensate and Feedwater and isolate the leak.

Reactor water level has been restored to its normal band.

All <u>ECCS</u> initiation logics have been <u>reset</u>.

What operator action, if any, is required to return the RCIC gland seal compressor to its normal standby configuration?

A. No operator action is required.

B. Open the gland seal compressor supply breaker then re-close it.

C. Depress the RCIC INIT RESET pushbutton.

D. Momentarily take the RCIC GL SEAL COMPR handswitch to STOP then release it.

Answer: D

Explanation:

Normal standby configuration for the gland seal compressor is to have its “auto-start” feature armed. This is done per the system SOI, step 4.1.2.f, where the RO simply momentarily places the compressor handswitch to STOP and releases it back to its spring-return to neutral position. This allows the compressor to auto-start on a reactor water Level 2 automatic RCIC initiation (or manual pushbutton initiation from P601), so long as <u>no</u> Div 1 ECCS LOCA initiation signal is present at that time. Once running, the compressor will automatically trip if a Div 1 ECCS LOCA initiation signal is received. Once tripped, it cannot be manually re-started until an operator resets the Div 1 ECCS LOCA initiation logic. And even after that, in order to “return it to its normal standby configuration” (i.e., auto-start feature enabled), the operator must again momentarily place its handswitch to STOP.

‘A’ is wrong. The Applicant would choose this as the answer if he remembers that the Div 1 LOCA signal tripped the compressor off but doesn’t remember the need to manually re-arm the auto-start feature.

‘B’ is wrong for the reason ‘D’ is correct. Plausibility is based on the fact that this Exam Author’s experience has been that, often, when it comes to these lower level (and lesser trained-on) components, the uncertain Applicant can recall concepts, but not specifics. This choice suggests the need to momentarily open then re-close the component’s supply breaker (a concept that sounds familiar to that Applicant), rather than momentarily cycling the handswitch (the unfamiliar specific the Applicant needed to recall).

‘C’ is wrong for the reason ‘D’ is correct. This distracter is aimed at the Applicant who has no recollection at all about the handswitch feature. Given that, this would be the most logical choice for him to default to in the experience of this Author.

‘D’ is correct. See the RCIC SOI, step 4.1.2.f for this specific operator action.

Technical References:

04-1-01-E51-1, RCIC System SOI

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-E5100, Objective 8.16

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b)(7) & (10)	
	55.43(b)	

Examination Outline Cross-Reference	Level	RO
218000 Automatic Depressurization System **Knowledge of the physical connections and/or cause-effect relationships between AUTOMATIC DEPRESSURIZATION SYSTEM and the following: K1.05 Remote shutdown system**	**Tier #**	2
	Group #	1
	K/A #	218000 K1.05
	Rating	3.9
	Rev / Date	0 / 10-14-11

Question 65

The plant is operating at rated power.

A fire occurs at the Offgas control panel P845 in the control room.

The fire is extinguished but, due to the smoke, the control room is evacuated.

In route to the Remote Shutdown Panel, a LOCA occurs.

Drywell pressure is 1.45 psig.

Reactor water level is -155”.

Reactor pressure is 910 psig and lowering.

Which of the following identifies the number of SRVs that indicate open at the Remote Shutdown panel two minutes after the above conditions?

A. 0

B. 2

C. 4

D. 6

Answer: B

Explanation:

Stem conditions indicate that the automatic initiation logic for ADS has been satisfied…≤ -150.3” water level WITH ≥ 1.39 DW pressure WITH 105 seconds elapsed WITH low pressure ECCS pump running (assumed to have started on the Div 1 (2) LOCA (level) signal).

Of the 20 total SRVs, only the 6 that are also Lo-Lo Set SRVs have controls/indications at the

Remote Shutdown Panel (RSP). Two of those 6 are also ADS valves. Therefore, upon arriving at the RSP "two minutes after", operators will see a total of 2 SRVs indicating OPEN due to the ADS auto-initiation.

For these reasons, choice 'B' is correct.

'A' is wrong but plausible to the Applicant who fails to remember that two of the 6 Lo-Lo Set SRVs at the RSP are also ADS valves.

'C' is wrong but plausible to the Applicant who seems to recall that only two of the 6 are not also ADS valves.

'D' is wrong but plausible to the Applicant who seems to recall that all 6 Lo-Lo Set SRVs are also ADS valves.

Technical References:

04-1-01-B21-1, Nuclear Boiler system SOI

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-E2202, Objective 15

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b)(7)	
	55.43(b)	

Examination Outline Cross-Reference	Level	RO
223002 Primary Containment Isolation System/Nuclear Steam Supply Shut-Off Ability to predict and/or monitor changes in parameters associated with operating the PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF controls including: A1.02 Valve closures	**Tier #**	2
	Group #	1
	K/A #	223002 A1.02
	Rating	3.7
	Rev / Date	0 / 10-14-11

Question 66

The plant is operating at rated power when both RFPTs trip (reason unknown).

Reactor water level drops to -50" before operators restore it to normal with HPCS and RCIC.

Without operator action, which of the following identifies an expected change in plant/system status as a result of level dropping that low?

A. Drywell temperatures are rising.

B. CRD system flow is zero.

C. FPCC filter-demin temperatures are rising.

D. Enclosure Building d/p is zero.

Answer: A

Explanation:

See the Automatic Isolations ONEP (05-1-02-III-5), Automatic Isolation Checklists for each of the 10 Groups. Only Groups 6, 7, and 8 isolate at Level 2 (-41.6"). A Group 6 isolation takes Chilled Water (P72) away from the drywell. Without operator action to re-open these isolation valves 30 seconds after the isolation signal (an interlock), DW temperatures will begin to rise. Therefore, choice 'A' is correct.

'B' is wrong. CRD pumps trip on an ESF (Div 1 (2)) load shed (i.e., at -150.3"). This choice is plausible to the Applicant who fails to recall this fact.

'C' is wrong. See the ONEP. A Group 6 isolation also stops Fuel Pool Cooling & Cleanup (G41) flow, tripping those pumps. The result will be lowering filter-demin temperatures in that system, not rising temperatures. Choice is plausible for the Applicant who confuses this relationship of system flow-versus-f/d temperatures.

'D' is wrong. Level 2 is one of the auto-initiation signals for Standby Gas Treatment (SGTS). Once in service, SGTS takes over the control of Enclosure Building d/p to maintain it at the usual

negative pressure of ~-0.30” w.c. Choice is plausible for the Applicant who fails to recall SGTS auto-initiation signals.			
Technical References: 05-1-02-III-5, Automatic Isolations ONEP GLP-OPS-T4801, SGTS lesson plan GLP-OPS-C111A, CRD system lesson plan			
References to be provided to applicants during exam: None			
Learning Objective: GLP-OPS-M7101, Objective 27			
Question Source:	Bank #		
(note changes; attach parent)	Modified Bank #		
	New	X	
Question History:	Last NRC Exam	No	
Question Cognitive Level:	Memory/Fundamental		
	Comprehensive/Analysis	X	
10CFR Part 55 Content:	55.41(b)(7)		
	55.43(b)		

Examination Outline Cross-Reference	Level	RO
239002 Relief/Safety Valves **Knowledge of the effect that a loss or malfunction of the RELIEF/SAFETY VALVES will have on following: K3.01 Reactor pressure control**	**Tier #**	2
	Group #	1
	K/A #	239002 K3.01
	Rating	3.9
	Rev / Date	0 / 10-14-11

Question 67

Operators are implementing the EPs.

There is a need to manually control reactor pressure with SRVs but 11DA power to the 'A' solenoids for all of the SRVs has been lost.

At which panel can operators manually control reactor pressure with SRVs?

A. Remote Shutdown Panel P150

B. Upper Control Room Panel P628

C. Control Room Panel P601

D. Control Room Panel P631

Answer: D

Explanation:

See lesson plan GLP-OPS-E2202, pages 34-35.

'A' is wrong. RSP panel P150 is the Div 1 ('A' solenoid) panel where all of the 6 Lo-Lo Set SRVs can be controlled; no 'B' solenoids are there (they are at RSP P151). Plausible because it is an SRV control/indication related panel.

'B' is wrong. P628 has SRV indications, only; no control switches. Plausible because it is an SRV control/indication related panel.

'C' is wrong. Control Room Panel P601 would be where operators normally control with SRVs; however, it uses the 'A' solenoids, exclusively. Plausible because it is an SRV control/indication related panel.

'D' is correct. Control Room (backpanel) P631 has controls and indications for all 20 SRVs using the 'B' solenoids, exclusively.

Technical References: GLP-OPS-E2202, ADS system lesson plan 04-1-01-B21-1, Nuclear Boiler System SOI			
References to be provided to applicants during exam: None			
Learning Objective: GLP-OPS-E2202, Objective 10			
Question Source:	Bank #		
(note changes; attach parent)	Modified Bank #		
	New	X	
Question History:	Last NRC Exam	No	
Question Cognitive Level:	Memory/Fundamental		
	Comprehensive/Analysis	X	
10CFR Part 55 Content:	55.41(b)(7)		
	55.43(b)		

Examination Outline Cross-Reference	Level	RO
263000 D.C. Electrical Distribution **K3.03 Systems with D.C. components (i.e. valves, motors, solenoids, etc.)**	**Tier #**	2
	Group #	1
	K/A #	263000 K3.03
	Rating	3.4
	Rev / Date	0 / 10-14-11

Question 68

The 11DK breaker that powers the ARI system has tripped open (breaker problem) and will not re-close.

How is the ARI system capability impacted and why?

A. Can still function to depressurize the scram air header because only one Channel is disabled.

B. Can still function to depressurize the scram air header because only one vent path is disabled.

C. Cannot function to depressurize the scram air header because all 3 vents paths are disabled.

D. Cannot function to depressurize the scram air header because 6 of the ARI valve solenoids are disabled.

Answer: C

Explanation:

Bus 11DE powers the logic for ARI Channel 1 and the solenoids for ARI valves F162B, F162D, and F164A. Bus 11DK powers the logic for ARI Channel 2 and the solenoids for ARI valves F162A, F162C, and F160. The normally-closed vent paths are thru: F162A & B; F162C & D; F164A & F160. Given these 4 answer choices, the only one that fits is choice ‘C’. In other words…without 11DK power, we cannot energize the solenoids for one of the ARI valves that we need to open in each of the 3 possible vent paths; i.e., none of the vent paths can be opened and so the header cannot be depressurized by ARI. Note, these are single-solenoid, not dual-solenoid valves.

For these reasons, ‘C’ is correct and ‘A’, ‘B’, ‘D’ are wrong (but their plausibility is based on the Applicant’s need to firmly understand both the DC power interrelationship within the ARI system and the ARI vent path arrangement).

Technical References:

GLP-OPS-C111A, CRD system lesson plan
GFIG-OPS-C111A, CRD system lesson plan figures…Figures 9 & 10 for ARI arrangement
P&ID M-1081A, CRD Hydraulic
E-6066, Sheet 006, ATWS ARI Valves

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-C111A, Objective 5.5

Question Source:	Bank # GGNS-OPS-08767	X
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b)(6) & (7)	
	55.43(b)	

Examination Outline Cross-Reference	Level	RO
300000 Instrument Air System (IAS) **Knowledge of electrical power supplies to the following: K2.01 Instrument air compressor**	**Tier #**	2
	Group #	1
	K/A #	300000 K2.01
	Rating	2.8
	Rev / Date	0 / 10-14-11

Question 69

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:

All three Plant Air Compressors (PACs) are 4.16 KV machines. Bus 13AD powers PAC ‘C’, Bus 14AE powers PAC ‘B’, and Bus 16AB powers PAC ‘A’.

For this reason, choice ‘A’ is correct and ‘B’, ‘C’, ‘D’ are wrong (but plausible based on the above discussion; notwithstanding choice ‘C’ which is a necessary placekeeper distracter; i.e., GGNS only has 3 machines, but we need 4 answer choices.)

Technical References:

04-1-01-P51-1, Plant Air System SOI, Attachment III Electrical Lineup (Page 2 of 2)

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-P5100, Objective 8

Question Source:	Bank # GGNS-OPS-08926	X
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	No

Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b)(4)	
	55.43(b)	

Examination Outline Cross-Reference	Level	RO
400000 Component Cooling Water System (CCWS) **Knowledge of the effect that a loss or malfunction of the following will have on the CCWS: K6.07 Breakers, relays, and disconnects**	**Tier #**	2
	Group #	1
	K/A #	400000 K6.07
	Rating	2.7
	Rev / Date	0 / 10-14-11

Question 70

The plant is operating at rated power with the following:

- CCW pumps 'A' and 'C' running
- CCW pump 'B' in Standby

ST-21 trips and locks out.

Which of the following identifies the CCW pump(s) that is/are running?

A. 'A' and 'B' only

B. 'B' only

C. 'C' only

D. 'A' and 'C' only

Answer: C

Explanation:

A loss of Service Transformer ST-21 results in losing power to the 11HD bus, which powers CCW pump 'A' through an associated MCC), and in losing power to the 16AB bus, which powers CCW pump 'B' through an MCC. Therefore, the running 'A' pump trips and the Standby 'B' pump has no power with which to auto-start. The running 'C' pump is unaffected by the ST-21 lockout because it relies on the other Service Transformer ST-11 to get its power to the 12HE bus and an associated MCC.

For these reasons, choice 'C' is correct and the other choices are wrong (their plausibility is based on the Applicant's need to recall the AC power distribution to these pumps).

[NOTE – Concerning the KA match. Obviously, a transformer lockout amounts to tripping all of that transformer's nearest "breakers"…thus, the KA match has been satisfied.]

Technical References: 04-1-01-P42-1, CCW System SOI, Attachment III Electrical Lineup (page 3 of 4)			
References to be provided to applicants during exam: None			
Learning Objective: GLP-OPS-P4200, Objective 5			
Question Source:	Bank # GGNS-OPS-01439	X	
(note changes; attach parent)	Modified Bank #		
	New		
Question History:	Last NRC Exam	No	
Question Cognitive Level:	Memory/Fundamental		
	Comprehensive/Analysis	X	
10CFR Part 55 Content:	55.41(b)(7)		
	55.43(b)		

Examination Outline Cross-Reference	Level	RO
241000 Reactor/Turbine Pressure Regulating System **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 A3.08
	Rating	3.8
	Rev / Date	0 / 10-14-11

Question 71

Currently:

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

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

What EHC related indications should the operator expect to see at P680?

A. Turbine Control Valves <u>close</u> as necessary to control reactor pressure; Bypass Control Valves remain as is.

B. Bypass Control Valves <u>close</u> as necessary to control reactor pressure; Turbine Control Valves remain as is.

C. Turbine Control Valves <u>open</u> as necessary to control reactor pressure; Bypass Control Valves remain as is.

D. Bypass Control Valves <u>open</u> as necessary to control reactor pressure; Turbine 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 950 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…making choice ‘B’ correct.

‘A’ is wrong for the reason ‘B’ is correct. It is plausible to the Applicant who believes the IPC would first pass its demand signal to the TCVs rather than to the BPVs.

‘C’ and ‘D’ are wrong for the reason ‘B’ is correct. They are plausible to the Applicant who believes that IPC attempts to maintain HP turbine 1st stage pressure rather than reactor pressure (not an uncommon misconception among some LOT students in the experience of this Exam Author).

Technical References:

GLP-OPS-N3202, EHC Control lesson plan

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-N3202, Objective 9

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b)(4) & (5)	
	55.43(b)	

Examination Outline Cross-Reference	Level	RO
262001 A.C. Electrical Distribution **Knowledge of A.C. ELECTRICAL DISTRIBUTION design feature(s) and/or interlocks which provide for the following: K4.01 Bus lockouts**	**Tier #**	2
	Group #	1
	K/A #	262001 K4.01
	Rating	3.0
	Rev / Date	0 / 10-14-11

Question 72

The plant is operating at rated power.

Which of the following will directly trip breakers that result in de-energizing Bus 11HD?

A lockout of transformer…

A. BOP-11A

B. BOP-12A

C. BOP-13

D. BOP-14

Answer: B

Explanation:

See the E-0001, Main One Line (GGNS Electrical Distribution) for the following explanation.

Bus 11HD is normally powered from Bus 13R via BOP-12B Xfmr. Since BOP-12A Xfmr is also on Bus 13R, a lockout of that Xfmr locks out BOP-12B as well…meaning that the feeder from 12B to 11HD gets a direct trip. Therefore, choice 'B' is correct.

There is no connection at all between BOP-11A (on Bus 12R) and Bus 11HD, nor is there for BOP-14 (also on Bus 12R). Likewise, there is no way for a lockout of BOP-13 (on Bus 11R) being able to interrupt power to Bus 13R and thus to Bus 11HD. For these reasons, choices 'A', 'C', 'D' are wrong (but are all plausible unless the Applicant has a solid recall of the electrical distribution arrangement, as well as a firm understanding of how lockout protection works).

Technical References:

E-0001, Main One Line Diagram
04-1-01-R21-11, BOP Bus 11HD SOI

References to be provided to applicants during exam: None			
Learning Objective: GLP-OPS-R2700, Objective 24			
Question Source:	Bank # GGNS-OPS-08752b	X	
(note changes; attach parent)	Modified Bank #		
	New		
Question History:	Last NRC Exam	No	
Question Cognitive Level:	Memory/Fundamental		
	Comprehensive/Analysis	X	
10CFR Part 55 Content:	55.41(b)(8)		
	55.43(b)		

Examination Outline Cross-Reference	Level	RO
2.4.29 Knowledge of the emergency plan.	**Tier #**	3
	Group #	
	K/A #	2.4.29
	Rating	3.1
	Rev / Date	0 / 10-14-11

Question 73

Per the GGNS Emergency Plan, in no case shall the NRC be notified later than ________ after the declaration of a(n) ________________.

A. 2 hours
Unusual Event

B. 90 minutes
Alert

C. 1 hour
Site Area Emergency

D. 30 minutes
General Emergency

Answer: C

Explanation:

See procedure 10-S-01-1 (Activation of the Emergency Plan), step 6.1.6.d NOTE. The limit is 1-hour regardless of the event level.

For this reason, choice 'C' is correct.

'A', 'B', 'D' are wrong for the reason 'C' is correct. Their plausibility is based on: 1) the Applicant's need to recall 1-hour requirement, and 2) not be distracted into believing that the NRC needn't be notified until an event as escalated to some more threatening level of action (e.g., SAE or GE).

Technical References:

10-S-01-1, Activation of the Emergency Plan

References to be provided to applicants during exam: None

Learning Objective: GLP-EP-EPT25, Objective 10		
Question Source:	Bank # Held in NRC Exam Room	X
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b)(10)	
	55.43(b)	

Examination Outline Cross-Reference	Level	RO
2.1.2 Knowledge of operator responsibilities during all modes of plant operation.	**Tier #**	3
	Group #	
	K/A #	2.1.2
	Rating	4.1
	Rev / Date	0 / 10-14-11

Question 74

The plant is in MODE 2 with a steady-state reactor power of 17%.

Per GGNS and/or Fleet procedures…

A. the Shift Manager should now direct operators to take the Mode Switch to RUN.

B. any on-shift SRO should direct an immediate power reduction.

C. the ACRO should immediately reduce power using the pull sheet.

D. the ACRO should immediately insert a manual scram.

Answer: D

Explanation:

MODE 2 requires that the Mode Switch be in the STARTUP position. In that position, the APRM high neutron flux automatic scram setpoint is 15%. The only resolution to this plant condition is to correct the situation by placing the Mode Switch in SHUTDOWN…i.e., immediately insert a manual scram. Per EN-OP-115 (Conduct of Operations), section 5.2[1], the RO is required to take this action on his own without the need for SRO direction.

For these reasons, choice ‘D’ is correct and ‘A’, ‘B’, ‘C’ are wrong (but plausible for reasons described above).

Technical References:

EN-OP-115, Conduct of Operations

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-PROC, Objective 1.3

Question Source:	Bank #	

(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b)(10)	
	55.43(b)	

Examination Outline Cross-Reference	Level	RO
2.2.12 Knowledge of surveillance procedures.	**Tier #**	3
	Group #	
	K/A #	2.2.12
	Rating	3.7
	Rev / Date	0 / 10-14-11

Question 75

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-02-3 (Author's Guide), section 5.1 definitions for each of these 4 symbols.

'A' is correct as defined in the question stem.

'B' is wrong. This symbol is used for steps requiring initials or data to be recorded. Plausibility speaks for itself.

'C' is wrong. This symbol denotes inter-procedural Tech Spec Logic System Functional Test overlap points. Plausibility speaks for itself.

'D' is wrong. This symbol denotes items associated with Inservice Inspection Acceptance Criteria.

Technical References:

01-S-02-3, Author's Guide

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-PROC, Objective 11			
Question Source:	Bank #		
(note changes; attach parent)	Modified Bank #		
	New	X	
Question History:	Last NRC Exam	No	
Question Cognitive Level:	Memory/Fundamental	X	
	Comprehensive/Analysis		
10CFR Part 55 Content:	55.41(b)(10)		
	55.43(b)		

Examination Outline Cross-Reference	Level	SRO
295001 Partial or Complete Loss of Forced Core Flow Circulation **Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION: AA2.05 Jet pump operability**	**Tier #**	1
	Group #	1
	K/A #	295001 AA2.05
	Rating	3.4
	Rev / Date	1 / 11-7-11

Question 76

The plant is operating at rated power.

The CRS reviews the latest Jet Pump Operability Daily surveillance and discovers the following:

- Drive flows in each recirc loop versus their flow control valve positions differ by 9% from established patterns
- Total recirc loop drive flow versus total core flow differs by 12% from established patterns
- Jet Pump #7 indicated flow differs by 11% from established patterns

What is the status of the completed surveillance (SAT/UNSAT) and <u>why</u>, and what is the Tech Spec Required Action, if any?

A. The surveillance is SAT because two of the three stated criteria are within their limits; there is <u>no</u> Tech Spec Required Action.

B. The surveillance is SAT because all three of the stated criteria are within their limits; there is <u>no</u> Tech Spec Required Action.

C. The surveillance is UNSAT because only one of the three stated criteria is within its limit; immediately enter Tech Spec LCO 3.0.3.

D. The surveillance is UNSAT because only one of the three stated criteria is within its limit; be in MODE 3 within 12 hours.

Answer: D

Explanation:

See Tech LCO SR 3.4.3.1. So long as at least two of the three SR 3.4.3.1 criteria (a, b, c) are met, the jet pump remains OPERABLE.

The completed surveillance is UNSAT because of the second and third criteria, both of which are beyond their 10% limits. This requires entry in LCO 3.4.3 and the Required Action of Condition A.1, which is to place the plant in MODE 3 within 12 hours.

For these reasons only choice ‘D’ is correct. Plausibility of the ‘A’, ‘B’, ‘C’ is evident.			
Technical References: Tech Spec LCO 3.4.3 and SR 3.4.3.1			
References to be provided to applicants during exam: None			
Learning Objective: GLP-OPS-TS001, Objective 14			
Question Source:	Bank #		
(note changes; attach parent)	Modified Bank #		
	New	X	
Question History:	Last NRC Exam	No	
Question Cognitive Level:	Memory/Fundamental	X	
	Comprehensive/Analysis		
10CFR Part 55 Content:	55.41(b)		
	55.43(b)(2)		

Examination Outline Cross-Reference	Level	SRO
295003 Partial or Complete Loss of A.C. Power **2.4.41 Knowledge of the emergency action level thresholds and classifications.**	**Tier #**	1
	Group #	1
	K/A #	295003 2.4.41
	Rating	4.6
	Rev / Date	0 / 10-14-11

Question 77

Use your provided references to answer this question.

From rated power a normal plant shutdown has begun with all three Emergency Diesel Generators inoperable and unavailable.

The following occurs:

- Tornado results in a loss of all offsite power
- RCIC automatically initiates and restores reactor water level to a band of -30” to +30”
- 45 minutes after the initial power loss, offsite power is restored to buses 15AA and 17AC

Bus 16AB remains de-energized.

What is the emergency classification for this event?

A. Unusual Event

B. Alert

C. Site Area Emergency

D. General Emergency

Answer: C

Explanation:

Per EAL SS1, choice ‘C’ is correct. Stem conditions are consistent with the SS1 declaration criteria: Loss of all offsite power and loss of all onsite AC power (i.e., no EDGs available) to Div 1, 2, & 3 ESF buses (i.e., 15AA, 16AB, 17AC) for >15 minutes.

Because RCIC restored level and because power was restored to at least one ESF bus (actual to two buses) within 4 hours, EAL SG1 does not apply, making answer choice ‘D’ plausible, but wrong.

The plausibility of choices ‘A’ and ‘B’ are based on the SRO Applicant wrongly applying EALs SU1 or SA1 for the loss of power that occurred. SU1 is wrong because it considers a loss of power to only the Div 1 and Div 2 buses. SA1 is wrong. Although it considers a loss of power to the point of a Station Blackout (or one failure short of a Blackout), to remain at the Alert level, the condition must last no more than 15 minutes.			
Technical References: 10-S-01-1, Section EPP 01-02			
References to be provided to applicants during exam: EAL Emergency Classification Flowcharts 10-S-01-1, Activation of the Emergency Plan, Attachment II (Bases for Emergency Classifications), only pages 91-113 (covers all “System Malfunctions”)			
Learning Objective: GLP-EP-EPTS6, Objective 1			
Question Source:	Bank # GGNS-LORQT-06395	X	
(note changes; attach parent)	Modified Bank #		
	New		
Question History:	Last NRC Exam	No	
Question Cognitive Level:	Memory/Fundamental		
	Comprehensive/Analysis	X	
10CFR Part 55 Content:	55.41(b)(10)		
	55.43(b)(5)		

Examination Outline Cross-Reference	Level	SRO
295021 Loss of Shutdown Cooling **2.2.25 Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits.**	**Tier #**	1
	Group #	1
	K/A #	295021 2.2.25
	Rating	4.2
	Rev / Date	1 / 11-03-11

Question 78

Use your provided references to answer this question.

Following a core reload, reactor cavity water level has been lowered to allow for removal of the main steam line plugs.

RHR 'A' is operating in Shutdown Cooling (SDC) when it trips due to a motor electrical fault.

Operators enter the "Inadequate Decay Heat Removal" ONEP

The CRS declares RHR 'A' inoperable enters the Tech Spec LCO 3.9.9, RHR – Low Water Level.

Which of the following combinations would permit the CRS to then **exit** that LCO?

A. RHR 'B' and is operating in SDC mode but is not OPERABLE in that mode.
   RWCU is available and has been verified capable of handling the decay heat

B. RHR 'B' is OPERABLE in SDC mode but the Div 2 DG is not OPERABLE.
   ADHRS is OPERABLE and is operating for decay heat removal.

C. RHR 'B' is OPERABLE in SDC mode and Div 2 DG is OPERABLE.
   ADHRS is OPERABLE and is operating for decay heat removal.

D. RHR 'B' is OPERABLE in SDC mode and Div 2 DG is OPERABLE.
   RWCU has been verified capable, and is operating for decay heat removal

Answer: C

Explanation:

To "exit" LCO 3.9.9, we must have "Two decay heat removal subsystems OPERABLE and one decay heat removal subsystem in operation." Without the Applicant's knowledge of the Bases for this LCO, however, this LCO statement can be mis-leading. See the Bases discussion on Bases page B 3.9-30. There we find that in fact that the expectation is that both RHR Shutdown Cooling (SDC) subsystems are OPERABLE…however, it allows ADHRS to be substituted for one of them. That same Bases discussion states that the DG has to also be

OPERABLE for at least one of the two SDC subsystems.

‘A’ is wrong. This is because we have no OPERABLE RHR Shutdown Cooling subsystems in this answer choice (i.e., RHR ‘A’ is broken; RHR ‘B’ has for whatever reason not been declared OPERABLE. Plausibility speaks for itself.

‘B’ is wrong. Div 2 DG needs to be OPERABLE in order to support RHR ‘B’ being OPERABLE in SDC mode.

‘C’ is correct. This combination does satisfy the LCO Bases as already described.

‘D’ is wrong. This combination forces us to remain in the LCO because RWCU can only be used as an “alternate method of decay heat removal” within the context of LCO CONDITION 3.9.9.A. In other words, RWCU does not qualify as a “decay heat removal subsystem”; therefore, this answer choice combination presents us with only a single OPERABLE decay heat removal subsystem, rather than the two required by the LCO.

Technical References:

Tech Spec LCO 3.9.9 (RHR – Low Water Level)
Tech Spec LCO 3.9.9 Bases

References to be provided to applicants during exam:

LCO 3.9.9 in its entirety **without the Bases**

Learning Objective: GLP-OPS-E1200, Objective 22

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b)	
	55.43(b)(2)	

Examination Outline Cross-Reference	Level	SRO
295025 High Reactor Pressure **2.4.6 Knowledge of EOP mitigation strategies.**	**Tier #**	1
	Group #	1
	K/A #	295025 2.4.6
	Rating	4.7
	Rev / Date	2 / 11-17-11

Question 79

Operators have just entered EP-2, RPV Control, as directed from EP-4, Auxiliary Building Control.

All control rods are fully inserted.

Reactor water level is being maintained between -30" and +30" with Condensate and Feedwater

Reactor pressure is 700 psig.

RCIC Room radiation level is quickly approaching its "Max Safe" value..

The CRS should **NEXT** direct the operators to…

A. depressurize the reactor to the main condenser.

B. inhibit ADS.

C. open 8 ADS valves.

D. manually override HPCS.

Answer: A

Explanation:

It is only from step 9 of EP-4 that we are directed to enter EP-2, and this is only the result of step 8 recognizing that we're approaching a max safe value for either an area temperature, water level, or radiation level. Once in EP-2, we invoke the allowance of its Pressure Leg to perform an anticipatory blowdown of reactor pressure to the main condenser in advance of having to do an Emergency Depressurization via SRVs to the Suppression Pool…which would be necessary per step 10 of EP-4 when we reach two "max safe" values. Thus, choice 'A' is correct.

'B' is wrong. It would be plausible if we either in EP-2A for an ATWS; we are not (all rods are fully inserted), or if we were having to enter the "Alternate Level Control Leg of EP-2 because we weren't effectively maintaining level…we are (-30" to +30").

'C' is wrong. It would be plausible to the Applicant who forgets the anticipatory blowdown allowance of the EP-2 Pressure Leg.

'D' is wrong. It would be plausible to the Applicant who forgets that's we manually manually override HPCS only upon entry into EP-2A.

Technical References:

EP-2, RPV Control
EP-4, Auxiliary Bldg Control
EP-2A, ATWS RPV Cntrol

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-EP02, Objective 3

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b)	
	55.43(b)(5)	

Examination Outline Cross-Reference	Level	SRO
295038 High Off-Site Release Rate **Ability to determine and/or interpret the following as they apply to HIGH OFF-SITE RELEASE RATE: EA2.03 †Radiation levels**	**Tier #**	1
	Group #	1
	K/A #	295038 EA2.03
	Rating	4.3
	Rev / Date	0 / 10-14-11

Question 80

Use your provided references to answer this question.

A General Emergency has been declared.

Radioactive release from the Containment Vent System has begun and is expected to be in progress for 90 minutes.

Weather conditions are partly cloudy with no rain expected in the forecast.

Offsite and onsite monitoring team data has been obtained.

Dose commitments at the Site Boundary are: 2000 mrem TEDE and 4000 mrem Thyroid

Dose commitments at 5 miles are: 800 mrem TEDE and 2000 Thyroid CEDE.

Which of the following would be the Protective Action Recommendation for these conditions?

A. Shelter 5 mile EPZ

B. Shelter 10 mile EPZ

C. Evacuate 2 miles all sectors
   Evacuate 5 miles downwind sectors
   Shelter remainder of 10 mile EPZ
   Consider use of KI

D. Evacuate 2 miles all sectors
   Evacuate 10 miles downwind sectors
   Shelter remainder of 10 mile EPZ
   Consider use of KI

Answer: C

Explanation:

See GGNS procedure 10-S-01-12, Radiological Assessment and PARs, section 6.2.1.

All answer choices are plausible by virtue their being taken directly from the section 6.2.1 PAR criteria for either “Standard”, “Extended”, or the two conditions for “Shelter” PARs.

The stem does not indicate that any conditions exist where evacuation would be impossible or otherwise dangerous. Also, the stem does indicate that CTMT venting is expected to be greater than 1 hour and has already begun. For these reasons, the “Shelter” PARs are not appropriate, making choices ‘A’ and ‘B’ wrong.

The dose commitments stated in the stem show that we are still below the levels for an “Extended” PAR at 5 miles, making choice ‘D’ wrong.

Therefore, only the “Standard” PAR applies, making choice ‘C’ correct.

Technical References:

10-S-01-12, Radiological Assessment and PARs, <u>entire Section 6.2, only</u>

References to be provided to applicants during exam: None

Learning Objective: GLP-EP-EPTS6, Objective 2

Question Source:	Bank # GGNS-LORQT-06071	X
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b)	
	55.43(b)(4)	

Examination Outline Cross-Reference	Level	SRO
600000 Plant Fire On Site **Ability to determine and interpret the following as they apply to PLANT FIRE ON SITE: AA2.17 Systems that may be affected by the fire**	**Tier #**	1
	Group #	1
	K/A #	600000 AA2.17
	Rating	3.6
	Rev / Date	0 / 10-14-11

Question 81

Use your provided references to answer this question.

The plant is in MODE 4.

RHR ‘B’ is operating in Shutdown Cooling (SDC) when a fire starts in fire zone 1A215.

Which of the following describes a potential concern and why?

A. Inability to isolate the SDC common suction because one of its isolation valves is physically located in this fire zone.

B. Inability to isolate the SDC common suction because cabling to one of its isolation valves runs through this fire zone.

C. Loss of reactor water inventory to the suppression pool due to “hot short” inadvertent opening of RHR ‘B’ pump min flow valve.

D. Unplanned MODE change when SDC heat sink is lost due to “hot short” inadvertent isolation of SSW flow to RHR ‘B’ HX.

Answer: B

Explanation:

[First, with respect to this being an SRO ONLY test item: Per the GGNS procedure for “Response To Fires” (10-S-03-2), Section 6.2.4 (“Shift Manager Action”)…step 6.2.4c states that it his responsibility to “use Attachment III to help in assessing fire damage potential to SAFE SHUTDOWN equipment in the fire zone.” And in fact that is the way we train/qualify GGNS licensed operators.]

See Attachment III, page 1 of 2, General Note #2, where we find that in Attachment IV a ‘D’ beside each component in the fire zone tables means that component is physically located in that fire zone. However, a ‘C’ means that a cable or circuit for that component runs through that fire zone.

See Attachment IV, pages 53-55, which lists the components affected by a fire in fire zone

1A215. A review of this list reveals that this is strictly a Div 1 impacting fire zone (i.e., all of these are 'A' subsystem/loop related components); therefore, answer choices 'C' and 'D' are wrong, although plausible by virtue of the fact that the stem has RHR 'B' operating in SDC. Until the SRO Applicant carefully reviews the Attachment IV fire zone 1A215 table, he has no way of eliminating these two answer choices. If RHR 'A' were operating in SDC, rather than 'B', either of them would be a "potential concern".

The 1A215 list includes the SDC common suction isolation valve Q1E12F008-A, which has a 'C' designator to the right-side of it. As already described above, the 'C' denotes that a cable runs through the fire zone, making choice 'B' correct and choice 'A' wrong, but plausible unless the Applicant has taken the time to review the General Notes of Attachment III.

Technical References:

10-S-03-2, Response To fires

References to be provided to applicants during exam:

10-S-03-2 (Response To Fires), the following portions only: Attachment III (both pages); Attachment IV pages 40 through 59

Learning Objective: GLP-OPS-PROC, Objective 58.8

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b)	
	55.43(b)(5)	

Examination Outline Cross-Reference	Level	SRO
295036 Secondary Containment High Sump/Area Water Level **2.4.6 Knowledge of EOP mitigation strategies.**	**Tier #**	1
	Group #	2
	K/A #	295036 2.4.6
	Rating	4.7
	Rev / Date	0 / 10-14-11

Question 82

Following a seismic event the plant remains operating at rated power.

Then, the following occur:

- RHR A PMP RM FLOODED seals-in at P870
- LPCS PMP RM FLOODED seals-in at P870
- RHR RM A SMP LVL HI-HI seals-in at P680
- LPCS RM SMP LVL HI-HI seals-in at P680
- Suppression pool level is 18.7 feet and lowering

The Rover closes the suppression pool suction valves for both RHR ‘A’ and LPCS.

Suppression pool level stabilizes at 18.4 feet.

Per EP-4, Auxiliary Building Control, the CRS should direct operators to…

A. continue to operate sump pumps as necessary to restore normal room water levels; no additional action is yet required.

B. continue to operate sump pumps as necessary to restore normal room water levels, and prepare to commence a normal plant shutdown using the IOIs.

C. manually scram the reactor, enter EP-2, and closely monitor suppression pool level for entry into EP-3.

D. manually scram the reactor, enter EP-2, and perform an Emergency Depressurization.

Answer: B

Explanation:

The SRO Applicant is expected to recognize that the two ROOM FLOODED alarms indicate that room water levels are above the “max safe” values of EP-4, Table 10, requiring only a normal plant shutdown per Step 7 of that EP while continuing efforts to pump the water out of the rooms. For this reason, choice ‘B’ is correct.

‘A’ is wrong. This would be the action to take per Step 7 if we had only a single RM FLOODED alarm (i.e., only one “max safe”). Based on the above discussion, its plausibility speaks for itself. ‘C’ is wrong. This would be a variation of the action to take if the SRO Applicant mistook the stem scenario for being a “primary system discharging outside primary CTMT” (per Step 6) and therefore inappropriately proceeded to Steps 8 and 9. Again, its plausibility speaks for itself. ‘D’ is wrong. The SRO Applicant who has convinced himself that proceeding to Steps 8 and 9 was the right thing to do will choose this answer (derived from step 10), recognizing there are in fact two RM FLOODED alarms. Again, its plausibility speaks for itself.			
Technical References: EP-4, Auxiliary Building Control ARI Windows 04-1-02-1H13-P870-2A-E1 & F1 ARI Windows 04-1-02-1H13-P680-8A1-A2 & A4			
References to be provided to applicants during exam: None			
Learning Objective: GLP-OPS-EP4, Objective 3			
Question Source:	Bank #		
(note changes; attach parent)	Modified Bank #		
	New	X	
Question History:	Last NRC Exam	No	
Question Cognitive Level:	Memory/Fundamental		
	Comprehensive/Analysis	X	
10CFR Part 55 Content:	55.41(b)		
	55.43(b)(5)		

Examination Outline Cross-Reference	Level	SRO
209001 Low Pressure Core Spray System **2.2.37 Ability to determine operability and/or availability of safety related equipment.**	**Tier #**	2
	Group #	1
	K/A #	209001 2.2.37
	Rating	4.6
	Rev / Date	0 / 10-14-11

Question 83

A Condition Report (CR) has been initiated to document housekeeping deficiencies in the LPCS Pump Room.

A large, unsecured tool box has been found stored in this room.

The tool box is adjacent to LPCS TEST RTN TO SUPP POOL VLV, E21-F012.

The tool box has not yet been removed.

What Operability Code should be assigned to the initial screening of this CR in PCRS?

A. NOT REQUIRED

B. EQUIPMENT FUNCTIONAL

C. INOPERABLE

D. EQUIPMENT NON-FUNCTIONAL

Answer: C

Explanation:

This item tests the SRO Applicant's ability to apply "big picture" concepts of the "Operability Determination Process" (fleet procedure EN-OP-104) from memory.

Given the 4 answer choices, the Applicant should recall that the "Functional" ("Nonfunctional") concept applies exclusively to non-Tech SSCs. Thus, he should eliminate answer choices 'B' and 'D', which are nonetheless plausible because they are in fact part of the EN-OP-104 process and are possible initial screening codes (see section 3.0[19]).

'A' is wrong. The NOT REQUIRED code is applied only where an SSC is not within the scope of either an Operability Determination or of a Functionality Assessment. Because this is a possible code, it is a plausible answer choice.

The unsecured tool box potentially threatens the "safety function" of the E21-F012 valve in the

event of a DBE earthquake. See the following EN-OP-104 sections for some discussions about Degraded/Nonconforming Conditions and more of this: 3.0[6]; 3.0[16]; 5.4[3] thru [5]; 5.4[6]; 5.4[7]; Attachment 9.1. As such, section 5.4[7] suggests the system is INOPERABLE. For these reasons, only choice 'C' is correct.			
Technical References: EN-OP-104, Operability Determination Process			
References to be provided to applicants during exam: None			
Learning Objective: GLP-OPS-PROC, Objective 42.3			
Question Source:	Bank #		
(note changes; attach parent)	Modified Bank #		
	New		X
Question History:	Last NRC Exam		No
Question Cognitive Level:	Memory/Fundamental		
	Comprehensive/Analysis		X
10CFR Part 55 Content:	55.41(b)		
	55.43(b)(5)		

Examination Outline Cross-Reference	Level	SRO
211000 Standby Liquid Control System **2.2.40 Ability to apply Technical Specifications for a system.**	**Tier #**	2
	Group #	1
	K/A #	211000 2.2.40
	Rating	4.7
	Rev / Date	0 / 10-14-11

Question 84

The plant is operating at rated power.

Both SLC subsystems are declared INOPERABLE after an engineering evaluation identifies a nonconforming condition common to both squib valves.

What is the Completion Time allowed by LCO 3.1.7, SLC System for restoring one of these SLC subsystems to an OPERABLE status?

A. 7 days

B. 72 hours

C. 12 hours

D. 8 hours

Answer: D

Explanation:

See LCO 3.1.7, ACTION C.1 Completion Time…it's 8 hours, making choice 'D' correct.

'A' is wrong, but plausible as this is the Completion Time for ACTION A.1 (only one SLC subsystem inoperable).

'B' is wrong, but plausible as this is the Completion Time for ACTION B.1 (out-of-spec boron concentration).

'C' is wrong, but plausible as this is the Completion Time for ACTION D.1 (to be in MODE 3 when the other ACTIONs have not been met).

Technical References:

Tech Spec LCO 3.1.7, SLC System

References to be provided to applicants during exam: None			
Learning Objective: GLP-OPS-TS001, Objective 40			
Question Source:	Bank #		
(note changes; attach parent)	Modified Bank #		
	New	X	
Question History:	Last NRC Exam	No	
Question Cognitive Level:	Memory/Fundamental	X	
	Comprehensive/Analysis		
10CFR Part 55 Content:	55.41(b)		
	55.43(b)(2)		

Examination Outline Cross-Reference	Level	SRO
261000 Standby Gas Treatment System **Ability to (a) predict the impacts of the following on the STANDBY GAS TREATMENT SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: A2.11 High containment pressure**	**Tier #**	2
	Group #	1
	K/A #	261000 A2.11
	Rating	3.3
	Rev / Date	0 / 10-14-11

Question 85

A seismic event has resulted in the following:

- Small-break LOCA in the drywell
- Auto-initiation of RCIC and HPCS
- RCIC RM FLOODED alarm due to suction line penetration leak at suppression pool wall
- EP-4 entry on Area Water Level, <u>only</u>
- EP-2 and EP-3 entries

Drywell pressure has reached 2.3 psig and has stabilized.

Drywell temperature has reached 127°F and has stabilized.

Reactor water level is being controlled in its <u>normal</u> band with RCIC.

Suppression pool level is 18.7 feet and slowly lowering.

At this point, operators are **REQUIRED** to…

A. Make up to the suppression pool using P11, if available.

B. Place <u>all</u> available drywell cooling in service.

C. Expand the reactor water level control band to -30” to +30”.

D. Defeat the high drywell pressure signal using Att. 31, then secure SGTS and return normal ventilation to service.

Answer: D

Explanation:

‘A’ is wrong. See EP-3 entry condition for Low SP Level (18.34’). We’re still well above this level. Not until level drops below 18.34’ would the CRS proceed to EP-3 step SPL-6 and begin normal SP makeup. Plausibility of this choice speaks for itself given the stem conditions.

‘B’ is wrong. See EP-3 entry condition for DW temperature (135°F). Only when we reach 135°F would the CRS place all available DW cooling in operation per EP-3 step DWT-2. Plausibility of this choice speaks for itself given the stem conditions.

‘C’ is wrong. See GGNS procedure 02-S-01-27 (Operations Philosophy), section 6.6.8d (Level Control Bands table). Given the stem conditions…i.e., we’re successfully controlling in the normal band with a low-flow, non-ECCS pump (RCIC)…there is no reason to apply the Expanded Band. Plausibility of this choice speaks for itself given the stem conditions.

‘D’ is correct. See EP-4, step 1. Stem conditions clearly state the CRS has entered EP-4 on Area Water Level, only. Additionally, the Applicant is expected to recognize from the stem conditions that SGTS has auto-initiated (both the Level 2 and high DW pressure signals existed at some point…the DW pressure signal is still present). Having entered the EP only On Area Water Level validates that no ventilation exhaust high rad conditions exist. Therefore, step 1 directs operators to secure SGTS and return normal Aux Bldg / Fuel Handling Area / Fuel Pool Sweep ventilation systems to service. Before SGTS can be secured, however, EP Attachment 31 will have to be implemented; this defeats the high DW pressure signal that is still present.

NOTE – Per the allowance of NUREG-1021, we’ve opted to match only the second part (part (b)) of this KA in the construct of this SRO-ONLY question. There is no way to employ an SRO-ONLY part (a) “predict” piece into such a system as SGTS for sampled KA.

Technical References:

EP-2, RPV Control
EP-3, Containment Control
EP-4, Auxiliary Building Control
EP-1 Attachment 31 – Aux Bldg Ventilation Restoration with LOCA Signal

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-EP4, Objective 3

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X

10CFR Part 55 Content:	55.41(b)
	55.43(b)(5)

Examination Outline Cross-Reference	Level	SRO
264000 Emergency Generators (Diesel/Jet) **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 #**	2
	Group #	1
	K/A #	264000 2.4.30
	Rating	4.1
	Rev / Date	0 / 10-14-11

Question 86

Per 01-S-06-5 (Reportable Events or Conditions), which of the following is an NRC 8-Hour Notification Requirement?

A. Safety Limit Violation

B. Tech Spec deviation authorized per 10CFR50.54(x)

C. Automatic start of Div 1 DG on a valid BUV condition

D. Initiation of a shutdown required by Tech Specs

Answer: C

Explanation:

See 01-S-06-5, Attachment III.

‘A’ and ‘B’ are wrong, but plausible because these are 1-hour requirements.

‘D’ is wrong, but plausible because this is a 4-hour requirement.

‘C’ is correct. This is an 8-hour requirement.

Technical References:

01-S-06-5, Reportable Events or Conditions

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-PROC, Objective 52.8

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No

Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b)	
	55.43(b)(5)	

Examination Outline Cross-Reference	Level	SRO
201003 Control Rod and Drive Mechanism **2.1.25 Ability to interpret reference materials, such as graphs, curves, tables, etc.**	**Tier #**	2
	Group #	2
	K/A #	201003 2.1.25
	Rating	4.2
	Rev / Date	0 / 10-14-11

Question 87

Use your provided references to answer this question.

The table below shows the scram time test results for two control rods, tested at a reactor pressure of 1000 psig.

Notch Position	Rod 28-33 Scram Time (sec)	Rod 32-41 Scram Time (sec)
43	0.28	0.30
29	0.80	0.81
13	1.44	1.48

What should the CRS conclude from his review of this data?

A. Rod 28-33 is satisfactory; rod 32-41 is “slow”.

B. Both rods are satisfactory.

C. Rod 28-33 is “slow”; rod 32-41 is satisfactory.

D. Both rods are slow.

Answer: A

Explanation:

See Tech Spec Table 3.1.4-1 (Control Rod Scram Times), specifically footnote (c) which directs the SRO to use linear interpolation of the table’s scram time acceptance criteria for reactor pressures that are in between 950 and 1050 psig. Therefore, a 1000 psig test would yield the following times for acceptance criteria: Notch 43 = 0.305 sec; Notch 29 = 0.81 sec; Notch 13 = 1.465 sec.

A review of the data for rod 28-33 shows its scrams times to be less than the interpolated 1000 psig acceptance criteria times at all three notch positions. This makes answer choices ‘C’ and ‘D’ wrong.

A review of the data for rod 32-41 shows its times to be satisfactory for notch positions 43 and

29, but unsatisfactory (i.e., “slower”) than the interpolated 1.465 sec time for notch 13. This makes choice ‘A’ correct and choice ‘B’ wrong.

The plausibility of each distracter speaks for itself based on the above discussions.

Technical References:

Tech Spec LCO 3.1.4 – Control Rod Scram Times, Table 3.1.4-1

References to be provided to applicants during exam:

LCO 3.1.4 in its entirety, including Table 3.1.4-1

Learning Objective: GLP-OPS-C111B, Objective 12

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b)	
	55.43(b)(2)	

Examination Outline Cross-Reference	Level	SRO
202001 Recirculation System **2.4.11 Knowledge of abnormal condition procedures.**	**Tier #**	2
	Group #	2
	K/A #	202001 2.4.11
	Rating	4.2
	Rev / Date	1 / 11-02-11

Question 88

The plant is operating at 67 Mlbm/hr core flow in the Monitored Region after operators take action to stabilize a degrading main condenser vacuum.

Vacuum has stabilized at 25” Hg Vac.

Both PBDS channels are OPERABLE.

Based on this the CRS should…

A. now enter the Reduction in Recirc System Flow Rate ONEP and assign a THI watch with no concurrent duties.

B. now enter the Reduction in Recirc System Flow Rate ONEP and assign a THI watch with concurrent duties allowed.

C. assign a Reactivity SRO the responsibility of within 15 minutes initiating action to exit the Monitored Region by cramming rods.

D. assign a Reactivity SRO the responsibility of within 15 minutes initiating action to exit the Monitored Region by lowering core flow.

Answer: B

Explanation:

See the Reduction in Recirc System Flow Rate ONEP (05-1-02-III-3).

Answer choice ‘B’ comes specifically from the Subsequent Action 3.0 NOTE on page 3, where we find we can simply assign a THI watch with concurrent duties after the plant has stabilized from an unintentional entry into the Monitored Region (MR).

‘A’ is wrong for the reason ‘B’ is correct.

‘C’ and ‘D’ are wrong because the stem conditions clearly state both PBDS channels are OPERABLE; therefore, there is no reason to expedite an exit from the MR within 15 minutes, whether it be by rods or core flow. See Step 3.4 for the plausibility of these answer choices.

Technical References: 05-1-02-III-3, Reduction in Recirc System Flow Rate ONEP			
References to be provided to applicants during exam: None			
Learning Objective: GLP-OPS-ONEP, Objective 2			
Question Source:	Bank #		
(note changes; attach parent)	Modified Bank #		
	New	X	
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory/Fundamental		
	Comprehensive/Analysis	X	
10CFR Part 55 Content:	55.41(b)		
	55.43(b)(5)		

Examination Outline Cross-Reference	Level	SRO
290003 Control Room HVAC **2.2.40 Ability to apply Technical Specifications for a system.**	**Tier #**	2
	Group #	2
	K/A #	290003 2.2.40
	Rating	4.7
	Rev / Date	0 / 10-14-11

Question 89

The plant is in MODE 3 when <u>one</u> control room AC subsystem is declared INOPERABLE.

What is the Completion Time allowed by LCO 3.7.4, Control Room Air Conditioning (AC) System, for restoring this subsystem to an OPERABLE status?

A. 12 hours

B. 36 hours

C. 7 days

D. 30 days

Answer: D

Explanation:

See the Completion Times associated with the ACTIONs of this LCO.

'A' and 'B' are wrong but plausible because these are the two successive Completion Times for ACTIONs C.1 & C.2 (in the event of failure to meet the other Completion Times).

''C' is wrong but plausible because this is the Completion Time for ACTION B.2 (when two subsystems are INOPERABLE).

'D' is correct. This is the Completion Time for a single INOPERABLE subsystem per ACTION A.1

Technical References:

Tech Spec LCO 3.7.4 – Control Room AC System

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-TS001, Objective 40

Question Source:	Bank #		
(note changes; attach parent)	Modified Bank #		
	New	X	
Question History:	Last NRC Exam	No	
Question Cognitive Level:	Memory/Fundamental	X	
	Comprehensive/Analysis		
10CFR Part 55 Content:	55.41(b)		
	55.43(b)(2)		

Examination Outline Cross-Reference	Level	SRO
295037 SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown Ability to determine and/or interpret the following as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN: EA2.01 Reactor power	**Tier #**	1
	Group #	1
	K/A #	295037 EA2.01
	Rating	4.3
	Rev / Date	0 / 10-14-11

Question 90

An ATWS and LOCA have begun.

The crew has just entered EP-2A and the following conditions exist:

- ARI/RPT has been initiated
- Reactor power is 5%
- MSIVs are closed
- Reactor water level is -50” and slowly lowering
- RPV injection has not yet been terminated
- CTMT pressure is 3.0 psig and slowly rising
- Suppression pool level is 19.0 feet
- Suppression pool temperature is 100°F and slowly rising

The CRS should direct operators to…

A. Terminate injection; stop lowering level when power drops below 4% and establish a band between there and -191”.

B. Maintain injection and establish a level band of -70” to -130”.

C. Inject with both SLC pumps.

D. Initiate CTMT sprays.

Answer: C

Explanation:

See EP-2A.

‘A’ is wrong but plausible because this is the action directed from the top-most row of EP-2A step L-5 if SP temperature were above 110°F. L-5 sends us to L-8 where level is lowered to bring power below 4% and then a band is established between there and -191”.

‘B’ is wrong but plausible because this is the step L-6 action if power were already below 4% when step L-5 were reached…it wasn’t.

‘C’ is correct. The Power leg steps Q-2 and Q-4 clearly show the need to start both SLC pumps where power is above 4%.

‘D’ is wrong. This choice brings EP-3 (CTMT Control) into play, given the stem conditions identifying SP level and CTMT pressure. The Applicant is expected to recognize that PSP (Pressure Suppression Pressure) is of no concern with these two parameter values. Thus, the EP-3 need to spray CTMT for PSP would not be appropriate. Plausibility speaks for itself.

Technical References:

EP-2A, ATWS RPV Control
EP-3, CTMT Control

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-EP02A, Objective 3

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b)	
	55.43(b)(5)	

Examination Outline Cross-Reference	Level	SRO
212000 Reactor Protection System **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: A2.21 †Failure of individual relays to reposition: Plant-Specific**	**Tier #**	2
	Group #	1
	K/A #	212000 A2.21
	Rating	3.9
	Rev / Date	0 / 10-14-11

Question 91

The plant is operating at rated power when the following occurs:

- An <u>actual</u> under-voltage condition occurs on the output of RPS 'B' MG Set
- <u>Both</u> of the under-voltage relays that should have sensed the condition failed to do so
- As a result, RPS 'B' MG Set is still powering RPS 'B' Bus
- All loads normally energized from RPS 'B' Bus are still energized

Which of the following describes the potential impact of sustained operation with this under-voltage condition (per the Tech Spec Bases), <u>and</u> describes the Tech Spec required action for the failure of the under-voltage relays?

A. Loss of scram pilot valve function due to solenoid chatter.
Remove RPS 'B' MG Set from service within 1 hour.

B. Premature dropout of scram pilot valve solenoids.
Remove RPS 'B' MG Set from service within 1 hour.

C. Loss of scram pilot valve function due to solenoid chatter.
Remove RPS 'B' MG Set from service within 72 hours.

D. Premature dropout of scram pilot valve solenoids.
Remove RPS 'B' MG Set from service within 72 hours.

Answer: A

Explanation:

See Tech Spec 3.3.8.2 and the "Background" portion of its Bases.

Applicant is expected to know that the stem condition (2^{nd} bullet) statement "Both of the…" means that <u>both</u> of the EPAs for the RPS 'B' MG Set (i.e., the in-service RPS power supply) have malfunctioned, making both EPAs inoperable.

Per Tech Spec 3.3.8.2, Action B.1, the MG Set must be removed from service within 1 hour.

See the Bases "Background" discussion (4th paragraph from the top) for the consequence of operating for an extended period with this low voltage condition. For these reasons, 'A' is correct. 'B' is incorrect because the Background discussion does not imply anything about excessive current beyond the solenoid design current. 'C' and 'D' are incorrect because they imply that Tech Spec 3.3.8.2, Action A.1 is applicable. It would apply if only one of the EPAs for the MG Set were inoperable.			
Technical References: Tech Spec 3.3.8.2 and Bases			
References to be provided to applicants during exam: None			
Learning Objective: GLP-OPS-C7100, Objective 17			
Question Source:	Bank #		
(note changes; attach parent)	Modified Bank #		
	New	X	
Question History:	Last NRC Exam	No	
Question Cognitive Level:	Memory/Fundamental		
	Comprehensive/Analysis	X	
10CFR Part 55 Content:	55.41(b)		
	55.43(b)(2)		

Examination Outline Cross-Reference	Level	SRO
500000 High Containment Hydrogen Concentration **2.4.6 Knowledge of EOP mitigation strategies.**	**Tier #**	1
	Group #	2
	K/A #	500000 2.4.6
	Rating	4.7
	Rev / Date	2 / 11-17-11

Question 92

A LOCA is in progress with the following:

- All control rods are fully inserted
- Emergency Depressurization has been performed
- Operators are able to keep reactor water level between -210” and -195” with HPCS injecting at 7,100 gpm
- CTMT hydrogen concentration is reading 3% and slowly rising
- Drywell hydrogen concentration is reading 2.8% and slowly rising
- Drywell radiation levels are reading 100 R/hr

The CRS should…

A. Immediately exit the EPs and enter the SAPs because level is still below -191”.

B. Immediately exit the EPs and enter the SAPs because of the current drywell radiation levels.

C. Immediately exit the EPs and enter the SAPs because of the CTMT hydrogen concentration.

D. Wait until drywell hydrogen concentration is at least 2.9% then exit the EPs and enter the SAPs.

Answer: C

Explanation:

‘A’ is wrong. This choice refers to steps L-16 and L-17 of EP-2A (ATWS RPV Control), where we are directed to exit the EPs and enter the SAPs if we fail to restore level above -191” after having Emergency Depressurized and are now re-injecting. Stem conditions clearly indicated all rods are fully inserted…we have no ATWS and are not in EP-2A.

‘B’ is wrong. This choice is taken from EP-5 (RPV Flooding), where we find another EP exit/SAP entry criteria in the form of drywell radiation level…200 R/hr…a symptom of core damage having occurred. First. In this scenario, we are not in EP-5, and secondly, the indicated drywell rad level is only 100 R/hr.

‘C’ is correct. The second override step of the Hydrogen Leg of EP-3 (CTMT Control) directs us to immediately exit the EPs and enter the SAPs when either CTMT hydrogen or DW hydrogen reaches 2.9%.

‘D’ is wrong for the reason ‘C’ is correct.

Technical References:

EP-3, Containment Control

References to be provided to applicants during exam: None

Learning Objective: GLP-EP-EPTS26, Objective 15.2

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b)	
	55.43(b)(5)	

Examination Outline Cross-Reference	Level	SRO
295029 High Suppression Pool Water Level **2.2.40 Ability to apply Technical Specifications for a system. |**	**Tier #**	1
	Group #	2
	K/A #	295029 2.2.40
	Rating	4.7
	Rev / Date	0 / 10-14-11

Question 93

The plant is operating at rated power with suppression pool water level just below the Tech Spec LCO upper limit.

Due to a delay in collecting pump vibration data during a RCIC Pump performance surveillance, suppression pool water level rises 3” above the Tech Spec LCO upper limit.

What is the Tech Spec Required Action?

A. Restore suppression pool level to within its limits within 2 hours; otherwise be in MODE 3 within 12 hours thereafter.

B. Restore suppression pool level to within its limits within 12 hours; otherwise be in MODE 3 within 12 hours thereafter.

C. Restore suppression pool level to within its limits within 2 hours; otherwise be in MODE 3 within 36 hours thereafter.

D. Restore suppression pool level to within its limits within 12 hours; otherwise be in MODE 3 within 36 hours thereafter.

Answer: A

Explanation:

See LCO 3.6.2.2 – Suppression Pool Water Level.

‘A’ is correct per Conditions A and B of the LCO.

‘B’ is wrong because Condition A is a 2-hour action, not a 12-hour.

‘C’ is wrong because Condition B is a 12-hour action, not a 36-hour.

‘D’ is wrong because both the Action Times for both Conditions (A and B) are wrong.

Plausibility of each answer choice is based on using actual action times found in the LCO.

Technical References: Tech Spec LCO 3.6.2.2 – Suppression Pool Water Level			
References to be provided to applicants during exam: None			
Learning Objective:			
Question Source:	Bank #		
(note changes; attach parent)	Modified Bank #		
	New	X	
Question History:	Last NRC Exam	No	
Question Cognitive Level:	Memory/Fundamental	X	
	Comprehensive/Analysis		
10CFR Part 55 Content:	55.41(b)		
	55.43(b)(2)		

Examination Outline Cross-Reference	Level	SRO
2.1.36 Knowledge of procedures and limitations involved in core alterations.	**Tier #**	3
	Group #	
	K/A #	2.1.36
	Rating	4.1
	Rev / Date	1 / 11-03-11

Question 94

Per 01-S-06-50 (Control of Fuel Services Operations), which of the following activities requires that the Refueling Supervisor In Charge (RSIC) first obtain Refuel SRO authorization?

A. An LPRM is to be removed from the core.

B. A spent control rod is to be lowered into the Spent Fuel Pool.

C. A new (non-irradiated) fuel assembly is to be lowered into the core.

D. A control rod is to be lowered into a core cell where there are no fuel assemblies.

Answer: C

Explanation:

Per 01-S-06-50. Section 5.1 defines CORE ALTERATIONS. Attachment III shows the activities for which the RSIC must obtain Refuel SRO authorization. These are CORE ALTERATIONS, only.

'A', 'B', and 'D' are all wrong because these are not Core Alterations; they are plausible because they require the Applicant to recognize them as such.

'C' is correct because it is a Core Alteration.

Technical References:

01-S-06-50, Control of Fuel Services Operations

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-PROC, Objective 18.2

Question Source:	Bank # (Held in NRC Exam Room Bank	X
(note changes; attach parent)	Modified Bank #	

	New	
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b)	
	55.43(b)(6)	

Examination Outline Cross-Reference	Level	SRO
2.2.23 Ability to track Technical Specification limiting conditions for operations.	**Tier #**	3
	Group #	
	K/A #	2.2.23
	Rating	4.6
	Rev / Date	0 / 10-14-11

Question 95

Procedure 02-S-01-17 is used for "Control of Limiting Conditions for Operation".

Per that procedure, which of the following requires the SRO to initiate (in eSOMS) an LCO Tracking Record, specifically of the type called "Technical Specification (TS) LCOTR"?

A. An inoperable system/component is not required to be operable under current plant conditions but would be required if plant mode or conditions changed.

B. Tracking the entry date/time of an LCO associated with Fire Protection systems or components.

C. During a refueling outage, a component is declared inoperable but that component is not required to be operable under current plant conditions.

D. A component required to be operable under current plant conditions has been made inoperable due to maintenance activities.

Answer: D

Explanation:

'A' is wrong per procedure section 5.1.2. This would be the specific type called: (PTS) LCOTR…Potential Tech Spec LCO Tracking Record. Plausibility speaks for itself.

'B' is wrong per procedure section 5.1.5. This would be the specific type called: Fire Protection TRM (FTR) LCOTR. Plausibility speaks for itself.

'C' is wrong per procedure section 5.1.4. This would be the specific type called: (OPT) LCOTR…Outage Potential Tech Spec LCO Tracking Record. Plausibility speaks for itself.

'D' is correct per procedure section 5.1.1.

Technical References:

02-S-01-17, Control of Limiting Conditions for Operation

References to be provided to applicants during exam: None			
Learning Objective: GLP-OPS-TSLCO, Objective 3			
Question Source:	Bank #		
(note changes; attach parent)	Modified Bank #		
	New	X	
Question History:	Last NRC Exam	No	
Question Cognitive Level:	Memory/Fundamental	X	
	Comprehensive/Analysis		
10CFR Part 55 Content:	55.41(b)		
	55.43(b)(5)		

Examination Outline Cross-Reference	Level	SRO
2.4.38 Ability to take actions called for in the facility emergency plan, including supporting or acting as emergency coordinator if required.	**Tier #**	3
	Group #	
	K/A #	2.4.38
	Rating	4.4
	Rev / Date	1 / 11-03-11

Question 96

Per 10-S-01-1 (Activation of the Emergency Plan), there is only one type of emergency where the Emergency Director is directed to **not activate** any of the onsite or offsite Emergency Response Facilities (even though an ALERT or higher EAL may have been declared).

What is that type of emergency?

A. Armed attack against the plant

B. Tornado resulting in Visible Damage within the Protected Area

C. Validation that an Operating Basis Earthquake has caused major damage to plant vital structures

D. Validated notification of an airborne attack threat

Answer: A

Explanation:

See 10-S-01-1, section 6.1.7. This section prescribes the ED's responsibilities unique to an "armed attack against the plant". Section 6.1.7.f specifically directs the ED to "not activate…any of the ERFs". For this reason, choice 'A' is correct.

'B', 'C' are wrong for the reason 'A' is correct, but are plausible because they each represent types of emergencies that might indicate the need to potentially shelter personnel (i.e., be concerned for safety).

'D' is wrong for the reason 'A' is correct. This is the strongest distracter in that it may lead to a conclusion of an armed attack on the plant as well. However, per the EALs (i.e., HA1a), an "airborne attack threat" is simply an airliner less than 30 minutes away from the plant.

Technical References:

10-S-01-1, Activation of the Emergency Plan
Emergency Action Levels (Flowcharts)

References to be provided to applicants during exam: None			
Learning Objective: GLP-EP-EPTS6, Objective 8			
Question Source:	Bank #		
(note changes; attach parent)	Modified Bank #		
	New	X	
Question History:	Last NRC Exam	No	
Question Cognitive Level:	Memory/Fundamental	X	
	Comprehensive/Analysis		
10CFR Part 55 Content:	55.41(b)		
	55.43(b)(5)		

Examination Outline Cross-Reference	Level	SRO
2.1.1 Knowledge of conduct of operations requirements.	**Tier #**	3
	Group #	
	K/A #	2.1.1
	Rating	4.2
	Rev / Date	0 / 10-14-11

Question 97

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[7]e. Only choice 'B' is correct.

'A', 'C', 'D' are wrong for the reason 'B' is correct. Each is as plausible as is the correct answer to the Applicant who cannot recall the 10 minute requirement.

Technical References:

EN-OP-115, Conduct of Operations

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-PROC, Objective 1.0

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X

	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b)(10)	
	55.43(b)	

Examination Outline Cross-Reference	Level	SRO
2.2.18 Knowledge of the process for managing maintenance activities during shutdown operations, such as risk assessments, work prioritization, etc.	**Tier #**	3
	Group #	
	K/A #	2.2.18
	Rating	3.9
	Rev / Date	1 / 11-03-11

Question 98

Use your provided references to answer this question.

The plant is in MODE 5.

HPCS operability is not required at this time.

The following is the current status of AC electrical power sources:

- Baxter-Wilson 500 KV feed is OPERABLE and feeding all Div 1 ESF buses electrical distribution subsystems
- Div 1 DG is not OPERABLE but is AVAILABLE, capable of supplying all Div 1 ESF buses and electrical distribution subsystems
- There are no other AC electrical power sources (offsite feeds or DGs) OPERABLE or AVAILABLE currently

There are no unusually complex evolutions in progress, or any plant conditions which themselves represent an unusually high risk.

What is the risk level color assigned to the "AC Power Control" Safety Function?

A. Green

B. Yellow

C. Orange

D. Red

Answer: B

Explanation:

See the SOPP Section V Tables. Mode 5 falls into SOPP "Shutdown Condition 2". For SOPP risk assessment purposes, OPERABILITY is not the relevant criterion, rather AVAILABLITY (as defined in SOPP Section, page 6) is the dominant risk assessment factor. A review of page 22 shows the "AC Power Control" risk assessment criteria for Shutdown Condition 2 (Mode

5). What matters is found in the rectangular boxes on the right of the page. The top-most box identifies offsite power sources, where we find a risk color of Yellow because we have only a single offsite feeder available (Baxter-Wilson in this case). The bottom-most box identifies the diesel generators, where again we find a risk color of Yellow because only the Div 1 DG available. Thus, the risk color assigned to the AC Power Control Safety Function, currently, is YELLOW…making choice 'B' correct.

'A', 'C', 'D' are wrong for the reason 'B' is correct. They are plausible because they each represent potential risk colors dependent upon other combinations of available power sources.

Technical References:

Shutdown Operations Protection Plan (SOPP)

References to be provided to applicants during exam:

Shutdown Operations Protection Plan (SOPP), in its entirety

Learning Objective: GLP-OPS-PROC, Objective 29.10

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b)	
	55.43(b)(5)	

Examination Outline Cross-Reference	Level	SRO
2.4.16 Knowledge of EOP implementation hierarchy and coordination with other support procedures or guidelines such as, operating procedures, abnormal operating procedures, and severe accident management guidelines.	**Tier #**	3
	Group #	
	K/A #	2.4.16
	Rating	4.4
	Rev / Date	1 / 11-03-11

Question 99

Operators have entered EP-3 (Containment Control) due to a complete loss of CTMT Cooling.

CTMT temperature has reached 183°F when operators manually initiate CTMT Sprays.

RHR ‘A’ begins to successfully spray but RHR ‘B’ does not.

Electricians report a tripped breaker at 16B31 for RHR ‘B’ CTMT Spray Valve (E12-F028B).

They report no apparent damage to either the breaker or the bus and recommend one attempt to re-close the breaker.

Meantime, CTMT temperature has reached 186°F and is slowly rising.

The CRS should…

A. immediately enter the Emergency Depressurization leg of EP-2.

B. direct the electrician to re-close the E12-F028B breaker one time in an effort to restore CTMT Spray ‘B’; if it appears that both loops of Sprays will be effective in restoring temperature below 185°F, the Emergency Depressurization will not be required.

C. dispatch operators to manually open E12-F028B before deciding if an Emergency Depressurization is necessary.

D. Immediately exit the EPs and enter the SAPs.

Answer: B

Explanation:

See EP-3, Containment Control, CTMT Temperature Leg. Step CNT-3 tells us to spray before reaching 185°F, while step CNT-6 tells us to ensure spray according to the CSIPL curve requirements (i.e., ensuring that it’s in fact safe to spray at all). Step CNT-6 is also the step that directs us to Emergency Depressurize if we cannot “restore and maintain” CTMT temperature below the 185°F limit with sprays.

See Operations Philosophy procedure 02-S-01-27, section 6.2.6. This section gives us the allowance of avoiding an unnecessary Emergency Depressurization when we have confidence (based on a qualified judgment) that we have (or will soon have) sufficient systems capability to restore and maintain a given EP parameter (CTMT temperature in this case) below a limit. Therefore, given this stem's scenario, it is prudent to let the breaker be closed, see if CTMT Spray 'B' will begin to spray, and see if the combined CTMT Spray 'A' and 'B' will reverse the rising temperature trend…i.e., "restore and maintain" has been achieved. For this reason, answer choice 'B' is correct.

'A' is wrong for the reason 'B' is correct.

'C' is wrong because the E12-F028B valve is physically located in CTMT where temperature has already reached 186°F, presenting an extreme personal safety hazard.

'D' is wrong because there is no transition point from EPs to SAPs based on CTMT temperature.

Technical References:

EP-3, Containment Control
02-S-01-27, Operations Philosophy

References to be provided to applicants during exam: None

Learning Objective:

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b)	
	55.43(b)(5)	

Examination Outline Cross-Reference	Level	SRO
2.3.11 Ability to control radiation releases.	**Tier #**	3
	Group #	
	K/A #	2.3.11
	Rating	4.3
	Rev / Date	1 / 11-03-11

Question 100

Operators are in EP-3 (CTMT Control) attempting to control a rising CTMT pressure.

The CRS has determined it is now necessary to vent the CTMT in order to keep CTMT pressure below 22.4 psig.

The EOF is fully operational.

Just before opening the 20" vents, the CRS/Shift Manager is <u>required</u> to <u>directly</u> notify the ____________ that an unfiltered release of radioactivity will occur.

A. TSC Coordinator

B. Emergency Plant Manager

C. Radiological Assessment Coordinator

D. Emergency Director

Answer: D

Explanation:

See EP-1, Attachment 13 (page 3 of 10), step 2.7, where direction is given to notify the Emergency Director, not the Emergency Plant Manager (the EPM is in the TSC).

For this reason, 'D' is correct and 'B' is wrong.

'A' is wrong for the reason 'D' is correct, but plausible because the TSC Coordinator is the second key player in the TSC next to the EPM.

'C' is wrong for the reason 'D' is correct, but is plausible because of this person's vital role in the release rate itself, and because he is a key assistant to the Emergency Director in the EOF.

Technical References:

EP-1, Attachment 13 – Containment Venting/Defeating Containment Vent Path Isolation Interlocks

References to be provided to applicants during exam: None			
Learning Objective: GLP-OPS-EP, Objective			
Question Source:	Bank #		
(note changes; attach parent)	Modified Bank #		
	New		X
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory/Fundamental		X
	Comprehensive/Analysis		
10CFR Part 55 Content:	55.41(b)		
	55.43(b)(5)		

GGNS LOT 2011 NRC INITIAL LICENSED OPERATOR WRITTEN EXAMINATION

RO EXAM

ANSWER KEY

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

GGNS LOT 2011 NRC INITIAL LICENSED OPERATOR WRITTEN EXAMINATION

SRO EXAM

ANSWER KEY

76	D
77	C
78	C
79	A
80	C
81	B
82	B
83	C
84	D
85	D
86	C
87	A
88	B
89	D
90	C
91	A
92	C
93	A
94	C
95	D
96	A
97	B
98	B
99	B
100	D