GGNS LOT 2/2020 **NRC** INITIAL LICENSED OPERATOR WRITTEN EXAMINATION

RO EXAM

ANSWER KEY

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

Examination Outline Cross Reference	Level	RO
295001 Partial or Complete Loss of Forced Core Flow Circulation / 1 & 4 2.2.22 Knowledge of limiting conditions for operations and safety limits**.**	Tier	1
	Group #	1
	K/A	2.2.22
	Rating	4.0
	Revision	1

Revision Statement:

Rev 1, Per Facility Rep, changed answer 'C' to the RPV pressure SL.

Question: 1

Which of the following **VIOLATES** a Tech Spec Safety Limit (SL)?

A. Two Recirc Pumps are running in slow speed with one Flow Control Valve at MIN ED with reactor power at 27%.

B. Core flow has just been reduced to 70 Mlbm/hr per ONEP; the STA recognizes that MCPR is now reading 1.10.

C. MSIV closure scram with reactor pressure rising to 50 psig above the ATWS/ARI initiation setpoint and all SRVs opened.

D. Plant startup with reactor pressure at 585 psig and both Reactor recirc pumps trip. the STA recognizes that MCPR is now reading 1.15.

Answer: **B**

Explanation:

The Safety limits for GGNS:

2.1.1.1 With the reactor steam dome pressure < 685 psig or core flow < 10% rated core flow Thermal power shall be < 21.8% RTP

2.1.1.2 With the reactor steam dome pressure ≥ 685 psig and core flow ≥ 10% rated core flow MCPR shall be ≥ 1.15 for two recirculation loop operation or ≥ 1.15 for single recirculation loop operation.

With core flow at 70 Mlbm/hr it is safe to assume >685 psig reactor pressure and >10% core flow, therefore, MCPR must be > 1.15 and it is 1.10.

'B' is correct

Distracters:

‘A’ is wrong, This is a low flow condition that is normal when the plant is performing a startup and shifting the recirc pumps to fast speed. This also does not meet the criteria in Tech Specs 2.1. Plausible due to a reactor recirc low flow condition.

‘C’ is wrong, The Reactor pressure safety limit is 1325 psig, 50 psig above the ATWS/ARI setpoint, 1126 psig would be only 1176 psig, below the SL.

‘D’ is wrong, First reactor pressure must be >685 psig to make MCPR an issue and MCPR of 1.15 is allowable. If power and flow were at the correct levels.

K/A Match

Knowledge of Tech Spec LCOs and Safety Limits are required to answer this question.

Technical References:

Tech Spec 2.1.2, Adm 203
051-02-III-3, Reduction in Recirculation System Flow Rate ONEP Rev. 118

Handouts to be provided to the Applicants during exam:

None

Learning Objective:

GLP-OPS-B1300, Rev 15 OBJ. 16
GLP-OPS-TS001, Rev 16 OBJ. 3a

Question Source:	**Bank # 133**	2012 NRC Exam
(note changes and attach parent)	**Modified Bank #**	
	New	
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.41(b)(10)	
Level of Difficulty:	2.0	

PRA Applicability:

None.

Examination Outline Cross Reference	Level	RO
295003 Partial or Complete Loss of AC Power AK3 Knowledge of the reasons for the following responses or actions as they apply to partial or complete loss of AC power: AK3.05 Reactor SCRAM	Tier	1
	Group #	1
	K/A	295003 AK3.07
	Rating	3.7
	Revision	1
Revision Statement: **Rev 1, Per Facility Rep, Reworded answer 'C' for clarity.**		

Question: 2

The plant is operating at rated thermal power when the normal feeder breaker to Bus 15AA trips due to a ground fault on the bus itself.

Since Bus 15AA cannot be immediately re-energized, the Loss of AC Power ONEP directs operators to manually scram the reactor.

Which of the following describes a reason for scramming the reactor?

A. Anticipates the automatic scram or control rod drifts as a result of a loss of Instrument Air to Containment.

B. It is a conservative action based on the sustainability of plant operation due to the unavailability of Plant Air Compressor 'A'.

C. It is a conservative action to place the plant in a safe shutdown and cooled down condition before a station blackout event might occur.

D. Anticipates the automatic scram that will occur on high drywell pressure as drywell temperature rises due to the loss of two drywell chillers.

Answer:	**A**
Explanation: P53-F001, Instrument Air Supply Header To CTMT, is an air-operated valve that fails closed on loss of power to its solenoid. This Div 1 isolation valve's solenoid is powered from 15AA. Therefore, the 15AA loss fails F001 closed, cutting off Instrument Air to that header feeding CTMT. The inboard MSIVs and scram air header are loads on that header. MSIVs will begin to drift closed as air pressure lowers in their control units. An auto-scram will result from the MSIV closure. Control Rods will begin to drift in as scram air pressure lowers to the point that scram valves begin to open. Because this question is asking for just "a reason", this single failure mechanism is enough to justify the correct answer. Pre-empting the auto-scram by inserting a manual scram conforms to the "Conservative Decision-Making" requirements of EN-	

OP-115, Conduct of Operations.

Distracters:

'B' is wrong for the reason already discussed. Also, PAC 'A' is powered from Bus 16AB, not 15AA.

'C' is wrong. With a loss of one ESF bus the Candidate may associate this with nearing a station black out but not the reason for this action.

'D' is wrong. This choice is alluding to the two (of 4 total) drywell chillers that are ESF bus powered (the other 2 being BOP bus powered). However, the 2 that are ESF powered are fed from bus 16AB, not from bus 15AA.

K/A Match

Knowledge of reason for reactor scram per ONEP procedure.

Technical References:

05-1-02-I-4 (Loss of AC ONEP) Rev. 053
EN-OP-115, Conduct of Operations, Rev 026

Handouts to be provided to the Applicants during exam:

None

Learning Objective:

GLP-OPS-ONEP OBJ. 6

Question Source:	**Bank # 466**	NRC Exam 2015
(note changes and attach parent)	**Modified Bank #**	
	New	
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.41(b)(7)	
Level of Difficulty:	3.0	

PRA Applicability:

ESF (R20) is listed as #4 on the System Importance to CDF.

"Failure to align alternate power to 4.16 KV or 6.9 KV buses." Is listed as #6 on Operator Action Importance to CDF.

Examination Outline Cross Reference	Level	RO
295004 Partial or Total Loss of DC Power AA1. Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER : AA1.03 A.C. electrical distribution	Tier	1
	Group #	1
	K/A	295004 AA1.03
	Rating	3.4
	Revision	2

Revision Statement:

Rev 1. Per Ops Rep, Changed order of distractors in the description section.

Rev 2, NRC review and comments

Deleted part of last sentence and now reads "ESF XFMR 12 is available to energize bus 15AA."

Changed question part of stem to the following, "Circuit breaker 152-1511, FDR FRM ESF XFMR 12 should..."

Deleted "Will" from answer 'A'

By changing the last sentence of the stem and the question part answer 'C' is now plausible, the student must know the availability of the circuit breakers without DC control.

Question: 3

Bus 15AA is de-energized.

Battery bus 11DA is de-energized.

ESF XFMR 12 is available to energize bus 15AA.

Circuit breaker 152-1511, FDR FRM ESF XFMR 12 should …

A. still have automatic trip capability.

B. close and reopen once at the breaker.

C. not be capable of remote or any local operation.

D. open remotely but must be closed manually, at the breaker.

Answer: B
Explanation:

Any 4160 breaker cannot be controlled remotely without DC, all operations must be done locally. Any breaker that is already closed when the loss of DC occurs can be opened due to the opening springs are compressed by the closing of the breaker, the opening springs are also compressed when the breaker is closed.

If the breaker is already open then the breaker can be closed and reopened once.

Distracters:

'A' is wrong; trip coil is DC powered.

'C' and 'D' are wrong; the breaker will not operate remotely without DC power, but will operate manually at the breaker as described above.

K/A Match

Knowledge of Electrical distribution breakers to an ESF bus with a loss of DC control power.

Technical References:

05-1-02-I-4, Loss of AC ONEP Rev. 053
05-1-02-V-19, Loss of 125VDC, Rev 01
GLP-OPS-L11, Plant DC System Lesson Plan, Rev 20
04-1-01-L11-1, Plant DC System SOI, Rev 129

Handouts to be provided to the Applicants during exam:

None

Learning Objective:

GLP-OPS-L11 OBJ. 10.1

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.41(b)(8)	
Level of Difficulty:	3.0	

PRA Applicability:

Plant DC is listed as #7 on the System Importance to CDF

Examination Outline Cross Reference	Level	RO
295005 Main Turbine Generator Trip AA2. Ability to determine and/or interpret the following as they apply to MAIN TURBINE GENERATOR TRIP: AA2.04 Reactor pressure	Tier	1
	Group #	1
	K/A	295005 AA2.04
	Rating	3.7
	Revision	2

Revision Statement:

Rev 1, Per Facility Rep, changed wording in answer 'D'

Rev 2, NRC Review and comments

Deleted one in stem

Added "and stabilizes" after constant in answer 'A'.

Question: 4

A main turbine / generator trip has occurred at 45%.

Which of the following describes the effect on reactor pressure?

A. held constant and stabilizes at 935 psig.

B. initial rise then controlled at approximately 935 psig.

C. initial rise then controlled between 926 psig and 1033 psig.

D. held constant at 935 psig then lowers to and stabilizes at approximately 850 psig.

Answer: **B**

Explanation:

At 45% power and a turbine trip occurs a Reactor Scram will also occur. Pressure control will be on the Main Bypass valves but initially Reactor pressure will rise then be controlled at 935psig due to the setpoint on Pressure Set is 935 psig.

Distracters:

'A' is wrong; Even though the bypass valves will open, reactor pressure will rise before they open to maintain pressure, plausible due to too much power is being produced not to have a pressure rise. If it were a lower power then a scram would not have occurred and pressure would have remained constant.
.
'C' is wrong, this assumes the MSIVs have closed due to the pressure band given is the open and close setpoints for SRVs on LO-LO set. Plausible due to setpoints on the LO LO set system

‘D’ is wrong; the setpoint can only be manually lowered below 935. Plausible due to the 850 psig number is the lower band number for manual pressure control in the EOPs

K/A Match

Knowledge of reactor pressure changes during a Turbine trip.

Technical References:

05-1-02-I-4, Turbine and Generator Trips Rev. 041
GLP-OPS-N3202, Main Turbine EHC Control System Lesson Plan, Rev 11
03-1-01-1, Cold S/D to Generator Carrying Min Load IOI, Rev 183

Handouts to be provided to the Applicants during exam:

None

Learning Objective:

GLP-OPS-N3202 OBJ. 4.0, 13.0

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(5)	
Level of Difficulty:	2.0	

PRA Applicability:

None.

Examination Outline Cross Reference	Level	RO
295006 SCRAM AK1. Knowledge of the operational implications of the following concepts as they apply to SCRAM : AK1.02 Shutdown margin	**Tier**	1
	Group #	1
	K/A	295006 AK1.02
	Rating	3.4
	Revision	1
Revision Statement: **Rev 1, NRC Review and comments** **Changed "would" to "should" in stem** **Added "without performing a calculation" to end of stem.**		

Question: 5

A reactor scram has occurred.

Per Technical Specifications, which of the following should still ensure that sufficient shutdown margin is maintained and that the reactor will remain subcritical under all conditions without performing a calculation?

A. One center core control rod is at position 48.

B. Two peripheral control rods are at position 48.

C. 50% of the control rods are at position 02 or beyond.

D. Groups 1 through 4 of the control rods are at position 04.

Answer: **A**
Explanation: Per Tech Specs definitions, “SDM shall be the amount of reactivity by which the reactor is subcritical or would be subcritical assuming that: (c) All control rods are fully inserted except for the single control rod of highest reactivity worth, which is assumed to be fully withdrawn. After a scram the crew can determine the reactor shutdown with only one control rod not full in. If more than one is withdrawn then a calculation must be performed.
Distracters: ‘B’ is wrong; Even if it is a peripheral rod only one can be withdrawn. .

‘C’ and ‘D’ are wrong, see the above explanation. Plausible due to ‘C’ is used in EP-2A to determine if the reactor can remain shutdown and ‘D’ is used when determining core density.

K/A Match

Knowledge of concept of shutdown margin after a reactor scram.

Technical References:

Tech Spec Section 1.1 Definitions

Handouts to be provided to the Applicants during exam:

None

Learning Objective:

GLP-OPS-TS001 OBJ. 4.13

Question Source:	**Bank # 721**	2013 NRC (Q5)
(note changes and attach parent)	**Modified Bank #**	
	New	
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.41(b)(6)	
Level of Difficulty:	2.0	

PRA Applicability:

None.

Examination Outline Cross Reference	Level	RO
295016 Control Room Abandonment AK2. Knowledge of the interrelations between CONTROL ROOM ABANDONMENT and the following: AK2.02 Local control stations: Plant-Specific	Tier	1
	Group #	1
	K/A	295016 AK2.02
	Rating	4.0
	Revision	3
Revision Statement: **Rev. 1, Validation, added switch number for transfer switch.** **Rev 2, added Div 2 to answer A** **Rev 3, NRC Review and comments** **Changed "it isolates" to "it should isolate" in stem.** **Removed "Isolates" from all answers.**		

Question: 6

The control room has been abandoned due to a fire in panel P864.

Control of the plant has been established at the Remote Shutdown Panels (RSPs).

At local panel P152 (Area 25A, El. 111'), operators have placed the following switch in the ON position:

- Transfer Switch for Lockout Transfer Relay**, C61-HSS-M150**

When placing this switch in the ON position, it should isolate what equipment from the control room?

A. ALL the Div 2 ESF powered equipment.

B. BOTH Div 1 and 2 ESF powered equipment.

C. ALL the equipment controlled from RSP P150.

D. BOTH equipment controlled from RSPs P150 and P151.

Answer: **C**
Explanation: Per the site electrical drawings, C61-HSS-M150 energizes lockout relays on H22-P152 which electrically isolate components operated at the H22-P150 (Division 1) Remote Shutdown Panel from the Main Control Room. Other selected Division 1 powered/controlled equipment have isolations provided on

Alternate Shutdown Panels and the Diesel Control Panel. Not all Division 1 equipment is affected by Alternate Shutdown /Remote Shutdown Panels and NO Division 2 equipment is affected.
Distracters: 'A' is wrong. Plausible because division 1 equipment only is bypassed. The transfer switch does not effect any Division 2 equipment. 'B' is wrong. Plausible because P150 and P151 are division one and division two equipment only. The transfer switch does not isolate ALL division one and two equipment from control room. 'D' is wrong. Plausible because P150 is isolated but the P151 is not and is manned at the same time.
K/A Match Knowledge of Alternate Shutdown /Remote Shutdown Panels and their interrelations with the Main Control Room.
Technical References: 05-1-02-II-1, Shutdown from Remote Shutdown Panels ONEP sections 1.3; 1.10; Attachments III, IV, and XXI. GLP-OPS-C6100
Handouts to be provided to the Applicants during exam: None
Learning Objective: GLP-OPS-ONEP, objective 54; 55 GLP-OPS-C6100, objective 11

Question Source:	**Bank # 579**	2015 NRC
(note changes and attach parent)	**Modified Bank #**	
	New	
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.41(b)(10)	
Level of Difficulty:	2.0	
PRA Applicability:		

None.

Examination Outline Cross Reference	Level	RO
295018 Partial or Total Loss of CCW AA2. Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENTCOOLING WATER : AA2.02 Cooling water temperature	Tier	1
	Group #	1
	K/A	295018 AA2.02
	Rating	3.1
	Revision	1
Revision Statement: **NRC Review and comments**		

Question: 7

The plant is operating at rated conditions.

CCW Temperature Control Valve (P44-F501) has failed closed due to a controller failure.

Which of the following should auto isolate due to the effects of this condition?

A. Control Rod Drive Pumps

B. Reactor Recirculation Pumps

C. Reactor Water Cleanup System

D. Fuel Pool Cooling and Cleanup System

Answer: C
Explanation: With the Temp control valve failing close the CCW temp will rise causing RWCU temp to rise and cause the G33-F004 to auto close at 140 degrees Non regen outlet temp. Therefore, B is the correct answer.
Distracters: All distracters are plausible due to being loads of the CCW system. ‘A’ and ‘B’ is wrong, but plausible will not auto isolate. ‘D’ is wrong but plausible because it will isolate on low flow not high temp

K/A Match		
Knowledge of effects on system loads due to high temperature.		
Technical References:		
04-1-01-G33-1 Step 3.1		
Handouts to be provided to the Applicants during exam:		
None		
Learning Objective:		
GLP-OPS-G3336 Objective 8.2		
Question Source:	**Bank # 723**	2013 NRC (Q7)
(note changes and attach parent)	**Modified Bank #**	
	New	
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(4)	
Level of Difficulty:	3.0	
PRA Applicability:		
None.		

Examination Outline Cross Reference	Level	RO
295019 Partial or Complete Loss of Instrument Air AK2. Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR and the following: AK2.09 Containment.	**Tier**	1
	Group #	1
	K/A	295018 AK2.09
	Rating	3.3
	Revision	0
Revision Statement: **New Question**		

Question: 8

The plant is operating at rated power when a total loss of instrument air to the Containment occurs.

Which of the following describes the effects on Containment and Drywell cooling water?

	Containment	Drywell
A.	Available	Available
B.	Available	Unavailable
C.	Unavailable	Available
D.	Unavailable	Unavailable

Answer: C

Explanation:

A loss of Instrument Air will cause the Containment isolation valves for Plant Chilled water to auto close, PCW supplies the containment coolers. A loss of PCW will cause the Containment Temperature to rise. This isolation for this system cannot be overridden to allow cooling.

A loss of Instrument Air has no effect on the Drywell Chilled Water System, because a different cooling water system supplies the Drywell coolers. The Drywell Chilled Water System uses MOVs for Ctmt and Drywell isolation valves.

'C' is Correct

Distracters:

'A' is wrong – but plausible, the candidate must recognize the loss of air only affects air operated valves

not MOVs therefore Cooling to the Containment fan is isolated but Drywell cooling is not.

‘B’ is wrong – opposite effects

‘D’ is wrong - but plausible, the candidate must recognize the loss of air only affects air operated valves not MOVs therefore Cooling to the Containment fan is isolated but Drywell cooling is not

K/A Match

Knowledge of effects on the containment with a loss of instrument air.

Technical References:

04-1-01-P71-1, PCW, step 3.6
04-1-01-P72-1, Drywell Chilled Water

Handouts to be provided to the Applicants during exam:

None

Learning Objective:

GLP-OPS-P7100, objective 13.1
GLP-OPS-M5100, Objective 6.5 & 9.3

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(7)	
Level of Difficulty:	3.0	

PRA Applicability:

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

Examination Outline Cross Reference	Level	RO
295021 Loss of Shutdown Cooling AK3. Knowledge of the reasons for the following responses as they apply to LOSS OF SHUTDOWN COOLING : AK3.01 Raising reactor water level.	Tier	1
	Group #	1
	K/A	295021 AK3.01
	Rating	3.3
	Revision	0
Revision Statement:		

Question: 9

The plant is in Mode 3.

Residual Heat Removal (RHR) B is operating in Shutdown Cooling.

Both Reactor Recirc pumps trip and are unable to be restarted.

The primary reason for the contingency to raise reactor water level for this condition is to:

A. raise circulation through the idle recirc loops.

B. establish a flow path from the core to the feedwater annulus.

C. establish a flow path through the SRVs to the Suppression Pool.

D. provide additional coolant mass to the RPV to absorb decay heat.

Answer: **B**
Explanation: With no Recirc pumps operating, raising reactor water level to above 82 inches allows natural circulation from the core up through the steam separator to the feedwater annulus for core cooling.
Distracters: 'A' is wrong but is plausible because a historically common error is to imagine natural circulation to be through recirc loop piping, but it is actually from core to downcomer to core. 'D' is wrong but is plausible because adding water would provide more heat sink, but that is not why the procedure requires raising level.

‘C’ is wrong but is plausible because the procedure does require raising level to the main steam lines to establish flow through the SRVs for Alternate Shutdown Cooling, but this is not warranted since RHR B is already operating in shutdown cooling. Alternate Shutdown Cooling would involve pumping from the suppression pool to the reactor using an ECCS pump and returning to the suppression pool through SRVs to establish a coolant loop.

K/A Match

Knowledge of reasons why reactor water level is raised on loss of shutdown cooling.

Technical References:

05-1-02-III-1, Inadequate Decay Heat Removal ONEP, CAUTION before step 3.4.2

Handouts to be provided to the Applicants during exam:

None

Learning Objective:

GLP-OPS-ONEP, objective 17 & 20

Question Source:	**Bank # 865**	2007 NRC (Q9)
(note changes and attach parent)	**Modified Bank #**	
	New	
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.41(b)(2)	
Level of Difficulty:	3.0	

PRA Applicability:

RHR is listed as #3 on the System Importance to CDF.

Examination Outline Cross Reference	Level	RO
295023 Refueling Accidents 2.4.41 Knowledge of the emergency action level thresholds and classifications.	Tier	1
	Group #	1
	K/A	2.4.41
	Rating	2.9
	Revision	1
Revision Statement: **NRC Review and comments** **Deleted periods at the end of answers C and D**		

Question: 10

Refueling is in progress, a dropped fuel bundle causes rad levels to rise.

The CRS has entered EP-4 due to Offsite Rad Release.

For the CRS to enter EP-4, the Rad Release must be at the level that the declaration of a(n) _________ is expected?

A. Unusual Event

B. Alert

C. Site Area Emergency

D. General Emergency

Answer: **B**
Explanation: Per EP-4 entry conditions "Offsite Rad Release declaration of Alert Expected", so if the CRS has entered EP-4 due to expected or actual offsite rad release then an Alert should be declared. Answer B is correct
Distracters: All distracters are plausible due to being EAL classifications.
K/A Match

Knowledge of EOP entry conditions and EAL classifications and thresholds.		
Technical References: 02-S-01-40 Rev. 09, EP Technical Bases		
Handouts to be provided to the Applicants during exam: None		
Learning Objective: GLP-OPS-EP4, objective 5		
Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.41(b)(10)	
Level of Difficulty:	2.0	
PRA Applicability:		
None.		

Examination Outline Cross Reference	Level	RO
295024 High Drywell Pressure EA1. Ability to operate and/or monitor the following as they apply to HIGH DRYWELL PRESSURE: EA1.05 RPS	Tier	1
	Group #	1
	K/A	295024 EA1.05
	Rating	3.9
	Revision	1
Revision Statement: **NRC Review and comments** **Changed 'A's to An in stem** **Changed wording in question to "actions should coincide with"**		

Question: 11

An RPV leak has occurred

An RO is monitoring automatic actions with rising drywell pressure.

Which of the following actions should coincide with a RPS reactor scram signal?

A. Div 2 LSS actuation

B. HPCS system initiation

C. Div 1 Diesel Generator auto start

D. P53-F001, INST AIR CTMT ISOL, auto close

Answer: D
Explanation: With rising drywell pressure a reactor scram will automatically occur at 1.23 psig Drywell Pressure. A coinciding action will be the automatically closure of P53-F001, INST AIR CTMT ISOL valve. 'D' is correct.
Distracters: All other distractors are wrong but plausible because all auto initiate at 1.39 psig Drywell pressure

K/A Match

Ability to monitor diverse indications to verify actions that have occurred. The Reactor Scram can be verified with an auto closure of Containment Isolation valves.

Technical References:

05-1-02-III-5 Rev 52, Automatic Isolation ONEP
05-1-02-I-1 Rev 132, Reactor Scram ONEP

Handouts to be provided to the Applicants during exam:

None

Learning Objective:

GLP-OPS-C7100 Rev 16, objective 9
GLP-OPS-M7100 Rev 15, objective 8

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.41(b)(7)	
Level of Difficulty:	2.0	

PRA Applicability:

None.

Examination Outline Cross Reference	Level	RO
295025 High Reactor Pressure EK1. Knowledge of the operational implications of the following concepts as they apply to HIGH REACTOR PRESSURE : EK1.03 Safety/relief valve tailpipe temperature/pressure relationships	Tier	1
	Group #	1
	K/A	295025 EK1.03
	Rating	3.6
	Revision	1
Revision Statement: **NRC Review and comments** **Added "should have" before "initially" in (1)** **Changed "is" to "should be" in (2)**		

Dec 2017 NRC Exam Q# 12

Question: 12

HANDOUT PROVIDED

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

(1) How many SRVs should have initially opened?

(2) What should be the peak SRV **tail pipe** temperature?

A. (1) One
(2) 411°F

B. (1) Two
(2) 411°F

C. (1) One
(2) 558°F

D. (1) Two
(2) 558°F

Answer:	**B**

Explanation:

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

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

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

1118 X 25% = 279.5 psia.

Saturated temperature for 279.5 psia is 411°F.

'B' is correct

Distracters:

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

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

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

K/A Match

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

Technical References:

GLP-OPS-E2202, Rev. 10, Automatic Depressurization System (ADS) Lesson Plan pages 21 and 34 of 45

Handouts to be provided to the Applicants during exam:

Steam Tables

Learning Objective:

GLP-OPS-E2202, OBJ 18.0

Question Source:	**Bank # 1350**	**Dec 2017 NRC Exam Q# 12**
(note changes and attach parent)	**Modified Bank #**	

	New	
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(14)	
Level of Difficulty:	4.0	
PRA Applicability:		
None.		

Examination Outline Cross Reference	Level	RO
295026 Suppression Pool High Water Temperature EK3. Knowledge of the reasons for the following responses as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE: EK3.04 †SBLC injection	Tier	1
	Group #	1
	K/A	295026 EK3.04
	Rating	3.7
	Revision	1

Revision Statement:

NRC Review and comments

Changed question (1) to: "At what suppression pool temperature should the CRS call for SBLC injection?"

Question: 13

An ATWS has occurred

Reactor power is 2.5%

(1) At what suppression pool temperature should the CRS call for SBLC injection?

(2) Which of the following describes the reason for SBLC injection?

A. (1) Before 110 F
(2) Maintain Containment temperature and pressure limits

B. (1) Before 110 F
(2) Prevent Large scale power instabilities leading to core damage

C. (1) After 110 F
(2) Maintain Containment temperature and pressure limits

D. (1) After 110 F
(2) Prevent Large scale power instabilities leading to core damage

Answer: **A**

Explanation:

Per 02-S-01-40, EP Technical Bases, Attachment V page 37 of 58 states the following:

"If reactor power is at or below 5%, thermal-hydraulic instabilities are not of significant concern since the core boiling boundary will be relatively high and the core void content will be relatively low. Some

oscillations may still occur, but large-scale instabilities leading to core damage are not expected. Boron injection then need not be injected unless primary containment integrity is jeopardized.”

‘A’ is correct

Distracters:

‘B’ is wrong. At <5% the bases states “large-scale instabilities leading to core damage are not expected.”

‘C’ is wrong. EP-3 states that before suppression pool 110 inject with boron not after, plausible due to the EPs uses before and after setpoints, the student must know its before setpoint.

‘D’ is wrong. At <5% the bases states “large-scale instabilities leading to core damage are not expected.” AND EP-3 states that before suppression pool 110 inject with boron not after, plausible due to the EPs uses before and after setpoints, the student must know its before setpoint.

K/A Match

This question requires the applicant to have the knowledge of the reason for injecting SBLC during an ATWS.

Technical References:

02-S-01-40, EP Technical Bases Rev 9, Attachment V, Page 37 of 58

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-EP02A, OBJ 7.0

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

None.

Examination Outline Cross Reference	Level	RO
295027 High Containment Temperature (Mark III Containment Only) EK1. Knowledge of the operational implications of the following concepts as they apply to HIGH CONTAINMENT TEMPERATURE (MARK III CONTAINMENT ONLY) : EK1.03 Containment integrity: Mark-III	Tier	1
	Group #	1
	K/A	295027 EK1.03
	Rating	3.8
	Revision	0
Revision Statement: **NRC Review and comments** **UNSAT** **NEW QUESTION**		

Question: 14

Per EP-3, Containment Control, BEFORE Containment temperature reaches ___________°F, the containment design temperature, ______________________is required.

A. (1) 185
   (2) Entry into EP-2, Reactor Scram

B. (1) 210
   (2) Entry into EP-2, Reactor Scram

C. (1) 210
   (2) Emergency RPV Depressurization

D. (1) 185
   (2) Emergency RPV Depressurization

Answer: **A**

Explanation:

Per 02-S-01-40, EP Technical Bases, Attachment VI Per EP-3 Bases, page 16 of 37, this is the containment design temperature…i.e., where containment damage or challenge to equipment operability may occur.

If containment temperature continues to increase, a reactor scram is performed through concurrent entry of EP-2. The scram is prescribed for the following reasons:
- The scram reduces the rate of energy production and thus the heat input to the containment.
- If containment temperature continues to increase, emergency RPV depressurization will be required. The reactor should be scrammed before the emergency depressurization is performed. While a high

drywell pressure scram signal may also be present, this may not be the case in all scenarios; conditions requiring entry of EP-3 do not necessarily require entry of EP-2.

The scram is prescribed indirectly, through entry of EP-2, because:

EP-2 coordinates control of RPV water level and pressure following the scram.

EP-2 branches to EP-2A, *ATWS RPV Control*, to provide appropriate guidance if all rods are not inserted by the scram.

An override in the Pressure branch EP-2 allows rapid depressurization through the main turbine bypass valves in anticipation of a possible emergency depressurization.

If containment temperature cannot be restored and maintained below 185°F, the containment design temperature, emergency RPV depressurization is required. This strategy limits the release of energy into the containment, thus minimizing further containment heatup, and ensures that the RPV is depressurized before temperature rises high enough to damage the containment or challenge equipment operability limits. Consistent with the definition of “cannot be restored and maintained” in Section 3.0, a decision that containment temperature cannot be restored and maintained below 185°F can be made may be made before, when, or after temperature actually reaches this value.

‘A’ is correct

Distracters:

‘B’ is wrong, the temperature is 185 not 210. Plausible because per 02-S-01-40, EP Technical Bases, 210 is the design for the suppression pool not the containment

‘C’ is wrong. the temperature is 185 not 210. Plausible because per 02-S-01-40, EP Technical Bases, 210 is the design for the suppression pool not the containment and an emergency depressurization is required if containment temperature can’t be restored and maintained below 185.

‘D’ is wrong. an emergency depressurization is required if containment temperature can’t be restored and maintained below 185.

K/A Match

This question requires the applicant to have the knowledge of the implications of high containment temperature and the effects.

Technical References:

02-S-01-40, EP Technical Bases Rev 9, Attachment VI, Page 18 of 37

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-EP3, OBJ 6.0

Question Source:	**Bank # 930**	NRC 2008 (Q62)
(note changes and attach parent)	**Modified Bank #**	
	New	

Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.41(b)(10)	
Level of Difficulty:	2.0	
PRA Applicability:		
None.		

Examination Outline Cross Reference	Level	RO
295028 High Drywell Temperature EK3. Knowledge of the reasons for the following responses as they apply to (EMERGENCY): EK3.04 Increased drywell cooling	Tier	1
	Group #	1
	K/A	295028 EK3.04
	Rating	3.6
	Revision	0
Revision Statement:		

NRC May 2017 (Q14)

Question: 15

What is the reason for operating all drywell cooling when Drywell Temperature exceeds the **EP-3 entry condition**?

A. To maintain Drywell temperature less than the ADS qualification temperature where a Reactor scram is required

B. To maintain Drywell temperature less than the Drywell design temperature limit where an Emergency Depressurization is required

C. To prevent exceeding the Drywell temperature where the ADS/SRVs are declared INOP

D. To reduce and maintain Drywell temperature less than the Drywell temperature Tech Spec LCO.

Answer: D
Explanation: The Drywell temp. EP-3 entry condition is 135°F, 02-S-01-40, EP Technical Bases, states “As long as drywell temperature remains below 135°F, the higher of the drywell temperature LCO and the maximum normal operating drywell temp., no further operator action need be taken in the DW temp. branch. ‘D’ is correct
Distracters: ‘A’ is wrong – same as B, by maxing drywell cooling you should maintain below the ADS qualification temperature (355°F), an ED is required if this setpoint cannot be maintained. Not just a reactor scram. ‘B’ is wrong – by starting all available drywell cooling could be to prevent exceeding the design temperature limit (330°F), only a scram is required not an ED.

‘C’ is wrong – the ADS qualification temp is 355°F not 135°F

K/A Match

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

Technical References:

02-S-01-40, EP Technical Bases Rev 9, Attachment VI, Page 10 of 37

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-EP3, OBJ 6.0

Question Source:	**Bank # 1263**	**NRC May 2017 (Q14)**
(note changes and attach parent)	**Modified Bank #**	
	New	
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.41(b)(10)	
Level of Difficulty:	2.0	

PRA Applicability:

None.

Examination Outline Cross Reference	Level	RO
295030 Low Suppression Pool Water Level EK2. Knowledge of the interrelations between LOW SUPPRESSION POOL WATER LEVEL and the following: EK2.08 SRV discharge submergence	Tier	1
	Group #	1
	K/A	295030 EK2.08
	Rating	3.5
	Revision	1
Revision Statement: **Rev 1, NRC Review and comments** **Added "the MINIMUM above which is" to stem**		

Question: 16

Which of the following Suppression pool levels is the MINIMUM above which is required to ensure SRV discharge submergence prior to emergency depressurization?

A. 10.5 ft.

B. 14.5 ft.

C. 14.25 ft.

D. 17.5 ft.

Answer: **A**

Explanation:

The suppression pool level to Emergency Depress is 10.5 ft., 02-S-01-40, EP Technical Bases, SRVs may be opened only if suppression pool water level is above 10.5 ft., the bottom of the suppression pool water level indication range. If suppression pool water level were low offscale, the actual water level could be below the top of the SRV discharge devices. Opening the SRVs with the discharge devices exposed would pass steam directly into the containment airspace, bypassing the suppression pool.

'A' is correct

Distracters:

'B' is wrong – The specified suppression pool water level of 14.5 ft. corresponds to an elevation 2 ft. above the top of the horizontal vents.

'C' is wrong – At GGNS, all suppression pool temperature elements are located at or above 14.25 ft.

'D' is wrong – This is the suppression pool level level setpoint for automatic initiation of Suppression Pool Makeup system

K/A Match

This question requires the applicant to have the knowledge of the location of the SRV spargers in relation to Suppression Pool Level.

Technical References:

02-S-01-40, EP Technical Bases Rev 9, Attachment IV, Page 48 of 53

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-EP3, OBJ 6.0

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.41(b)(10)	
Level of Difficulty:	2.0	

PRA Applicability:

None.

Examination Outline Cross Reference	Level	RO
295031 Reactor Low Water Level 2.4.3 Ability to identify post-accident instrumentation.	Tier	1
	Group #	1
	K/A	2950310 2.4.3
	Rating	3.7
	Revision	0
Revision Statement:		

Question: 17

Which of the following is a “Post Accident” Reactor Vessel Water Level instrumentation indication in the Main Control Room?

A. H13-P680 Narrow Range

B. H13-P680 Upset Range

C. H13-P601 Fuel Zone Range

D. H13-P601 Shutdown Range

Answer: C
Explanation: Per Tech Spec 3.3.3.1 Post Accident Monitoring Instrumentation and 06-OP-1C61-M-0001, Remote Shutdown Panel and Accident Monitoring Instrumentation Channel Check, Fuel Zone level instrumentation is considered Post Accident. ‘C’ is correct
Distracters: ‘A’ is wrong – Wide Range level instruments are listed in the Tech Specs but those instruments are on the P601 panel not P680. ‘B’ and ‘D’ are wrong – not listed in Tech Specs. Plausible due to being a range of level instrumentation and Upset is used for Feedwater level control system, Shutdown is used for high water levels.
K/A Match This question requires the applicant to have the knowledge of the location and which level instrument is considered to be post accident.

Technical References: Tech Specs 3.3.3.1 and 06-OP-1C61-M-0001 Rev 107		
Handouts to be provided to the Applicants during exam: NONE		
Learning Objective: GLP-OPS-TS001, OBJ 6.0		
Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.41(b)(10)	
Level of Difficulty:	2.0	
PRA Applicability: None.		

Examination Outline Cross Reference	Level	RO
295037 SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown EA2. Ability to determine and/or interpret the following as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN : EA2.03 SBLC tank level	Tier	1
	Group #	1
	K/A	295037 EA2.03
	Rating	4.3
	Revision	1
Revision Statement: **Rev 1, NRC Review and comments** **Changed is to should and added approximately to stem.**		

Question: 18

An ATWS has occurred.

Standby Liquid Control has been initiated.

Initial SLC storage tank level was 4800 gallons.

The first SLC pump should be secured per EP-2A in approximately __________ minutes?

A. 24 minutes

B. 34 minutes

C. 68 minutes

D. 116.5 minutes

Answer: **B**
Explanation: The SLC pumps are designed for 41.2 gpm and when SLC is initiated both pumps are started, therefore 82.4 gpm is being injected into the vessel. When SLC storage tank reaches 2000 gallons then one pump is to be shutdown per EP-2A. So 4800 gal – 2000 gal = 2800 gals. 2800 gals / 82.4 gpm = 33.98 minutes. ‘B’ is correct

Distracters:

'A' is wrong – Plausible due to if the student thinks that 2000 gal. must be pumped instead of 2000 gal indicated.

'C' is wrong – Plausible due to the student must remember that both pumps will be started, this is the time for one pump running.

'D' is wrong - Plausible due to this would be the time for the entire tank to be pumped using only one pump.

K/A Match

This question requires the applicant to have the knowledge of the SLC pump capacity, use of both pumps and when one pump is to be secured. And the ability to determine how long it will take to pump the SLC tank.

Technical References:

05-S-01-EP-2, RPV Control, Rev 47
02-S-01-40, EP Technical Bases, Rev 9
GLP-OPS-C4100, Standby Liquid Control system lesson plan, Rev 16.

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-C4100, OBJ 6.0 and 7.1

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(1)	
Level of Difficulty:	4.0	

PRA Applicability:

None.

Examination Outline Cross Reference	Level	RO
295038 High Off-Site Release Rate EK2. Knowledge of the interrelations between HIGH OFF-SITE RELEASE RATE and the following: EK2.10 Condenser air removal system.	**Tier**	1
	Group #	1
	K/A	295038 EK2.10
	Rating	3.2
	Revision	1
Revision Statement: **Rev 1, NRC Review and comments** **Changed stem as follows: "Per procedure 03-1-01-1, IOI, Cold Shutdown to Generator Carrying Minimum Load, to prevent high offsite release the Mechanical Vacuum Pumps (MVPs) should be secured prior to exceeding a MAXIMUM of ___(1)___% power and the Steam Jet Air Ejectors (SJAEs) should be placed in service at approximately _______(2)______psig."**		

Question: 19

Per procedure 03-1-01-1, IOI, Cold Shutdown to Generator Carrying Minimum Load, to prevent high offsite release the Mechanical Vacuum Pumps (MVPs) should be secured prior to exceeding a MAXIMUM of ___(1)___% power and the Steam Jet Air Ejectors (SJAEs) should be placed in service at _______(2)______psig Reactor pressure.

A. (1) 5%
   (2) 300 psig

B. (1) 5%
   (2) 500 – 800 psig

C. (1) 10%
   (2) 300 psig

D. (1) 10%
   (2) 500 – 800 psig

Answer: B
Explanation: Per 03-1-01-1, IOI, complete placing the SJAE in service at 500 to 800 psig and P&L 2.10 states, “Mechanical Vacuum Pumps will NOT be operating for Main Condenser evacuation when Reactor power is 5% or greater as indicated on IRMs using the IRM Scale reading v IRM Range chart in 17-S-02-1, Manual Core Heat Balance.” ‘B’ is correct
Distracters:

‘A’ is wrong – Plausible due to this is the required power level but at 300 psig warmup is started for the SJAE not place in service.

‘C’ is wrong – Plausible due to this is the power level at which APRM gains are adjusted at 300 psig warmup is started for the SJAE not place in service.

‘D’ is wrong - Plausible due to this is the power level at which APRM gains are adjusted

K/A Match

This question requires the applicant to have the knowledge of the interrelationship with Condenser Air Removal system and high release rates.

Technical References:

03-1-01-1, IOI, Cold Shutdown to Generator Carrying Minimum Load, Rev. 183

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-I0I01, OBJ 3.15 and 25.8

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.41(b)(12)	
Level of Difficulty:	3.0	

PRA Applicability:

None.

Examination Outline Cross Reference	Level	RO
700000 Generator Voltage and Electric Grid Disturbances AK1. Knowledge of the operational implications of the following concepts as they apply to GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES: AK1.01 Definition of terms: volts, watts, amps, VARs, power factor.	**Tier**	1
	Group #	1
	K/A	700000 AK1.01
	Rating	3.3
	Revision	1
Revision Statement: **NEW QUESTION** **Rev 1, NRC Review and Comments** **Added "Per 05-1-02-I-4, Loss of AC Power ONEP," and changed will to should to (2)**		

Question: 20

Per 05-1-02-I-4, Loss of AC Power ONEP:

(1) Which of the following system voltages meets the voltage definition of a grid disturbance?

(2) Per 05-1-02-I-4, Loss of AC Power ONEP, if ESF bus voltage cannot be maintained, the crew should ____________.

A. (1) 495 Kv
(2) Manually Scram the Reactor and Trip the Generator

B. (1) 495 Kv
(2) Start Emergency Generators and separate from Offsite power

C. (1) 527 Kv
(2) Start Emergency Generators and separate from Offsite power

D. (1) 527 Kv
(2) Manually Scram the Reactor and Trip the Generator

Answer:	C
Explanation: Per ONEP 05-1-02-I-4, Loss of AC Power, Step 3.4;	

Any of the following are possible symptoms of Grid Instability:

- Low system voltage <491Kv
- Hi system voltage >525 Kv
- Low system frequency <58.3 Hz-(Gen Low Frequency Alarm)
- Oscillating system voltage
- Oscillating system frequency

And step 3.4.6;

IF any ESF bus voltage CANNOT be maintained > 3952 volts, grid frequency CANNOT be maintained > 58.3 Hz, OR 500 KV frequency/voltage is erratic, THEN START respective Emergency Diesel Generators, SYNCHRONIZE to respective bus, AND SEPARATE from Offsite power per Attachment IV.

'C' is correct

Distracters:

'A' is wrong – but plausible, the Voltage does not reach the setpoint for instability. Nowhere in this section of the ONEP does it say to scram the reactor and trip the generator.

'B' is wrong – but plausible, the Voltage does not reach the setpoint for instability.

'D' is wrong - but plausible,Nowhere in this section of the ONEP does it say to scram the reactor and trip the generator.

K/A Match

This question requires the applicant to have the knowledge of the implications if the main generator and grid voltage show indications of grid instability.

Technical References:

05-1-02-I-4, Loss of AC Power, Rev. 50, step 3.4.3 CAUTION

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-ONEP, OBJ 6
GLP-OPS-N4151, OBJ 14

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

	Comprehensive / Analysis	
10CFR Part 55 Content:	55.41(b)(10)	
Level of Difficulty:	3.0	
PRA Applicability:		
None.		

Examination Outline Cross Reference	Level	RO
295008 High Reactor Water Level AA1. Ability to operate and/or monitor the following as they apply to HIGH REACTOR WATER LEVEL: AA1.06 HPCS: Plant-Specific.	Tier	1
	Group #	2
	K/A	295008 AA1.06
	Rating	2.8
	Revision	1
Revision Statement: **Rev 1, NRC Review and comments** **Changed will to should in stem**		

Question: 21

An event occurred causing Drywell pressure to rise to 1.6 psig and stabilize there.

Reactor water level lowered to -65 inches.

HPCS has auto initiated

RPV water level is restoring.

(1) At what reactor water level should HPCS stop injection?

And

(2) What is required to restore injection?

A. (1) Level 8 Narrow Range
   (2) Level less than level 8 and depress the HPCS HI LVL RESET pushbutton

B. (1) Level 8 Wide Range
   (2) Level less than level 8 and depress the HPCS HI LVL RESET pushbutton

C. (1) Level 8 Narrow Range
   (2) Level less than level 8 and depress the HPCS INIT RESET pushbutton

D. (1) Level 8 Wide Range
   (2) Level less than level 8 and depress the HPCS INIT RESET pushbutton

Answer: B
Explanation: HPCS will auto initiate at 1.39 psig Drywell pressure or -41.6 inches RPV level. Once initiated injection

will auto stop at level 8 (+53.5 inches) on the Wide range indicators. At that point the white indicating light above the HPCS HI LVL RESET pushbutton will illuminate. With the initiation signal still present level must be below level 8 and the operator must depress the HPCS HI LVL RESET pushbutton to resume injection.

'B' is correct

Distracters:

'A' is wrong – Plausible due to Narrow range level instrumentation also performs actions at level 8 (RCIC system shutdown) and level 9 Feedwater pump trip.

'C' is wrong – Plausible due to Narrow range level instrumentation also performs actions at level 8 (RCIC system shutdown) and level 9 (+56 inches) Feedwater pump trip.

'D' is wrong – Plausible due to the High Level Reset must be depressed first then the initiation button can be used.

K/A Match

This question requires the applicant to have the ability to operate and monitor HPCS injection, when it auto stops and how to resume injection.

Technical References:

04-1-01-E22-1, HPCS SOI, Rev. 127
GLP-OPS-E2201, HPCS Lesson Plan, Rev 13

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-E2201, OBJ 9.4

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

PRA Applicability:

HPCS is listed as #16 on the System Importance to CDF.

Examination Outline Cross Reference	Level	RO
295010 High Drywell Pressure AK1. Knowledge of the operational implications of the following concepts as they apply to HIGH DRYWELL PRESSURE : AK1.03 Temperature increases.	**Tier**	1
	Group #	2
	K/A	295010 AK1.03
	Rating	3.2
	Revision	0
Revision Statement: **Per NRC review (10 free view) SAT**		

Question: 22

A LOCA has caused drywell pressure to reach 2.8 psig.

Drywell temperature is 211°F

A complete loss of offsite power has occurred, including the Port Gibson line.

All DGs have re-powered the ESF buses.

NO Operator actions have been performed.

Which of the following identifies the method of heat removal from the drywell under the current plant conditions?

A. Heat removal from the drywell is only by ambient losses without air circulation.

B. Drywell Coolers and Drywell Chilled Water System 'B' are operating on SSW 'B'.

C. 'A' Drywell Cooler fans are circulating air but the coolers are without cooling water flow.

D. 'B' Drywell Cooler fans are circulating air but the coolers are without cooling water flow.

Answer: A
Explanation: Drywell pressure caused an LSS LOCA Shed and Sequence. The Drywell Coolers are powered from 15B42 and 16B42, which are shed on a LOCA and not automatically re-energized. Operator action is required to restore power to the Drywell Coolers. 15B42 and 16B42 are shed and re-energized automatically on a loss of power, however the LOCA signal will override the loss of power sequence. Additionally, the Drywell Chilled Water supply and return valves will be automatically isolated due to the

Drywell Pressure.

'A' is correct

Distracters:

'B' is wrong – Plausible, procedures does not allow Drywell Chilled Water to be restored by reopening isolation valves without complete system startup when Drywell temperature is > 200°F, SSW 'B' is aligned for LOCA cooling which is not aligned to the DW chilled water system and 'B' Chilled water system is powered from Div 2 16AB which is locked out on a LOCA by LSS.

'C' and 'D' are wrong – Plausible, the Drywell Coolers are powered from 15B42 and 16B42, which are shed on a LOCA and not automatically re-energized

K/A Match

This question requires the applicant to recognize the implications with drywell temperature when drywell pressure is high.

Technical References:

04-1-01-R21-1 Table 1, Rev 106

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-M5100.00 Objective: 13, 14, 19, 21

Question Source:	**Bank # 418**	X Not used on NRC Exam
(note changes and attach parent)	**Modified Bank #**	
	New	
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(7)	
Level of Difficulty:	3.0	

PRA Applicability:

ESF (R20) is listed as #4 on the System Importance to CDF.
Offsite power is listed as #12 on the System Importance to CDF.

Examination Outline Cross Reference	Level	RO
295020 Inadvertent Containment Isolation AK3. Knowledge of the reasons for the following responses as they apply to INADVERTENT CONTAINMENT ISOLATION: AK3.03 Drywell/containment temperature response.	Tier	1
	Group #	2
	K/A	295020 AK3.03
	Rating	3.2
	Revision	2
Revision Statement: **Per Ops Rep, Changed the stem to state "the CRS directs to restore the Aux Building"** **Rev 2, NRC Review and comments,** **Changed question to fill in the blank, now a new question.**		

Question: 23

An inadvertent Containment isolation signal has occurred and will not reset.

The CRS directs to restore the Auxiliary Building per the Automatic Isolations ONEP.

Five minutes after the actions are complete:

______(1)_________ temperature should remain constant due to ventilation cooling water _______(2)_________.

A. (1) Containment
   (2) automatically aligning to SSW.

B. (1) Containment
   (2) being able to be bypassed and restored.

C. (1) Drywell
   (2) being able to be bypassed and restored.

D. (1) Drywell
   (2) automatically aligning to SSW.

Answer:	C
Explanation: After an isolation signal the ONEP allows to restore certain systems that can bypass the isolation signal:	

Instrument Air
Drywell Chilled Water

Therefore, Drywell temperature will remain the same due to a brief interruption with cooling water.

The Containment coolers are cooled by Plant Chilled water system that receives an isolation signal and no ability to bypass the isolation, therefore no cooling water will be supplied to the coolers and Containment temperature will rise.

'C' is correct

Distracters:

'A' is wrong –as stated above CTMT coolers will no longer have cooling and cannot be restored therefore temperature will rise, SSW will not auto align to Ctmt or Drywell coolers, plausible due to SSW will auto align to the ESF switchgear coolers, Control Room AC with a LOP.

'B' is wrong – as stated above CTMT coolers will no longer have cooling and cannot be restored therefore temperature will rise

'D' is wrong – SSW will not auto align to Ctmt or Drywell coolers, plausible due to SSW will auto align to the ESF switchgear coolers, Control Room AC with a LOP

K/A Match

This question requires the applicant to recognize the reasons for a change in parameters within the Drywell and Containment during Containment isolation.

Technical References:

GLP-OPS-P7100, Rev 15
05-1-02-III-5, Automatic Isolation ONEP Rev. 52

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-M5100.00 Objective: 13, 14, 19, 21

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(7)	
Level of Difficulty:	3.0	

PRA Applicability:
None.

Examination Outline Cross Reference	Level	RO
295029 High Suppression Pool Water Level EA2. Ability to determine and/or interpret the following as they apply to HIGH SUPPRESSION POOL WATER LEVEL : EA2.03 Drywell/containment water level	Tier	1
	Group #	2
	K/A	295029 EA2.03
	Rating	3.4
	Revision	1

Revision Statement:

Rev 1, NRC Review and comments:

Changed 'A' to 'An' in stem

Changed answer 'B' to "ECCS initiation and RPV injection from suppression pool"

Added "received" to answer 'C'

Question: 24

A plant operator reports a pipe break in the containment causing water to drain into the Suppression Pool.

Suppression Pool level is currently 19.5 ft and rising.

Due to a logic failure Suppression pool makeup inadvertently initiates.

An RO is monitoring Suppression Pool Water level and reports that level is stable at 24.3ft.

Which of the following describes the reason Suppression Pool Level indication has stabilized?

A. Overflowing the Drywell weir wall.

B. ECCS initiation and RPV injection from suppression pool.

C. Maximum amount of water received from the upper pools.

D. Level has reached the point of transmitters being submerged.

Answer:	**A**
Explanation: The design of the Suppression Pool Makeup system is without considering any other inputs, an	

inadvertent SPMU dump will raise Suppression Pool level to ≈23.8' if pool level was initially at its normal level of 18.5' or raise level approximately 5.3 ft. With suppression pool level at 19.5' plus the 5.3' dumped is 24.8'. the Weir wall to the Drywell stands at 24.3' therefore the level stabilizing is from overflowing of the weir wall.

'A' is correct

Distracters:

'B' is wrong – but plausible, the max indication on the Suppression Pool level indicator is 25.5' wide range.

'C' is wrong – but plausible, the design of the Suppression Pool Makeup system is without considering any other inputs, an inadvertent SPMU dump will raise Suppression Pool level to ≈23.8' if pool level was initially at its normal level of 18.5' or raise level approximately 5.3 ft. With suppression pool level at 19.5' plus the 5.3' dumped is 24.8'. the Weir wall to the Drywell stands at 24.3'.

'D' is wrong – Plausible, the upper pools dump will fill the suppression pool to 5" below the top of the weir wall with suppression pool level at the high level alarm setpoint.

K/A Match

This question requires the applicant to interpret the indications of a high suppression pool level.

Technical References:

02-S-01-40, EP Technical Bases, Rev 009
GLP-OPS-E3000. Rev 13

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-EP3 Objective: 7

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(9)	
Level of Difficulty:	3.0	

PRA Applicability:
None.

Examination Outline Cross Reference	Level	RO
295032 High Secondary Containment Area Temperature 2.4.47 Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material.	Tier	1
	Group #	2
	K/A	295032 2.4.47
	Rating	4.2
	Revision	1

Revision Statement:

Rev 1, NRC Review and comments

In stem (1) will to should, and reworded, In (2) changed will to should, removed reference handout. Student must still be able to differentiate between exceeding the operating limit and the MAX safe limit.

Question: 25

A steam line within the Auxiliary Building Main Steam Tunnel has a leak.

Current MST temperature is 105°F.

MST room temp is rising at 5° F / min

(1) When should automatic actions initiate?

After the MST reaches the MAX SAFE temperature, RCIC room temperature begins to rise:

(2) If the break cannot be isolated, what is the MINIMUM RCIC room temperature that would procedurally require the CRS to direct an Emergency Depressurization?

A. (1) 16 minutes
(2) 212°F

B. (1) 16 minutes
(2) 225°

C. (1) 29 minutes
(2) 212°F

D. (1) 29 minutes
(2) 225°F

Answer:	A
Explanation:	

Per 02-S-01-40, EP Technical Bases, EP-4 Table SC-1, Automatic actions will occur at 185°F in the MST, a Group 1 isolation will occur. At 5°F/ min it will take 16 minutes for the steam tunnel to rise 80°F.

For EP-4 to call for a Emergency Depressurization it requires an unisolable leak and two Max safe values of the same type (i.e. temp). Therefore per Table SC-1 the Max safe for the MST is 250°F and RCIC is 212°F

'A' is correct

Distracters:

'B' is wrong – Plausible, The given temperature is for the RHR rooms not the RCIC room

'C' is wrong – Plausible, 29 minutes is the time to reach the Max Safe value for the MST of 250°F. 185°F is the setpoint for a Group 1 isolation and EP-4 entry.

'D' is wrong – Plausible, 29 minutes is the time to reach the Max Safe value for the MST of 250°F. 185°F is the setpoint for a Group 1 isolation and EP-4 entry. The given temperature is for the RHR rooms not the RCIC room

K/A Match

This question requires the applicant to diagnose and recognize trends of a steam leak using reference material.

Technical References:

02-S-01-40, EP Technical Bases, Rev 009
05-S-01-EP-4, Auxiliary Building Control, Rev. 31

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-EP4 Objective: 5 and 9

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(7)	
Level of Difficulty:	3.0	
PRA Applicability:		

RCIC is listed as #17 on the System Importance to CDF.

Examination Outline Cross Reference	Level	RO
295035 Secondary Containment High Differential Pressure EK2. Knowledge of the interrelations between SECONDARY CONTAINMENT HIGH DIFFERENTIAL PRESSURE and the following: EK2.01 Secondary containment ventilation	Tier	1
	Group #	2
	K/A	295035 EK2.01
	Rating	3.6
	Revision	0
Revision Statement: **NRC Review and comments** **NEW QUESTION**		

Question: 26

Which of the following describes how differential pressure is **controlled** in the Auxiliary Building?

A. Slightly positive by using a pressure control valve on the supply air

B. Slightly positive by using a pressure control valve on the exhaust

C. Slightly negative by using a pressure control valve on the exhaust

D. Slightly negative by using a pressure control valve on the supply air

Answer: D
Explanation: F021 is at the inlet to the supply fans. With all else remaining constant, if F021 opens more, a greater supply air flowrate occurs, which lessens the building d/p. 'D' is correct
Distracters: 'A' is wrong – Aux building is maintained slightly negative not positive 'B' is wrong – Aux building is maintained slightly negative not positive, , F021 is at the inlet to the supply fans 'C' is wrong – F021 is at the inlet to the supply fans

K/A Match This question requires the applicant to have the knowledge of secondary containment ventilation and the interrelations with building D/P.		
Technical References: 02-S-01-40, EP Technical Bases, Rev 009 05-S-01-EP-4, Auxiliary Building Control, Rev. 31		
Handouts to be provided to the Applicants during exam: NONE		
Learning Objective: GLP-OPS-T4200 Objective: 3.1		
Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(7)	
Level of Difficulty:	3.0	
PRA Applicability:		
None.		

Examination Outline Cross Reference	Level	RO
295036 Secondary Containment High Sump/Area Water Level 2.1.30 Ability to locate and operate components, including local controls.	Tier	1
	Group #	2
	K/A	295036 2.1.30
	Rating	4.4
	Revision	1

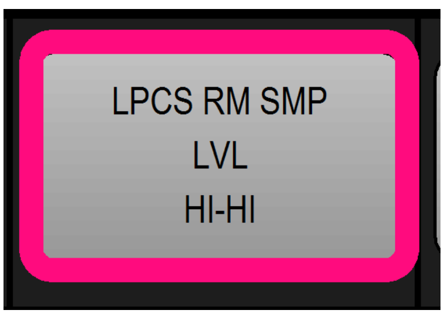
Revision Statement:

Rev 1, NRC Review and comments

Changed from to for in (1) and will to should in (2), deleted "operate all sump pumps and manually start sump pumps"

Question: 27

The plant is at rated conditions when the following alarm is received.



(1) Which of the following describes the crew's response for this alarm?

(2) What location should the actions be performed?

A. (1) Enter EP-4
(2) LPCS room

B. (1) Enter EP-4
(2) Radwaste Control Room

C. (1) Enter ARI P680-8A1-A4
(2) LPCS room

D. (1) Enter ARI P680-8A1-A4
(2) Radwaste Control Room

Answer:	A

Explanation:

The alarm given is outlined in “glowing cerise” color designating that this alarm corresponds to an EP-4 entry condition. EPs have precedence over ARIs, therefore EP-4 should be entered first and actions performed.
The action per EP-4 is to “Operate sump pumps as necessary to restore and maintain area water levels below their operating limits”

LPCS sump pump handswitches are located inside the LPCS pump room.

‘A’ is correct

Distracters:

‘B’ is wrong – Plausible, Radwaste control room contains controls for multiple sump systems and the Auxiliary transfer pump but not room sump pump controls, those are local

‘C’ is wrong – Plausible, EPs have precedence over ARIs

‘D’ is wrong – Plausible, EPs have precedence over ARIs. Radwaste control room contains controls for multiple sump systems and the Auxiliary transfer pump but not room sump pump controls, those are local

K/A Match

This question requires the applicant to have the knowledge of secondary containment high sump water levels and the actions required. Also must have the knowledge of local controls.

Technical References:

02-S-01-40, EP Technical Bases, Rev 009
05-S-01-EP-4, Auxiliary Building Control, Rev. 31
04-1-01-P45-2, Floor Drain Sump System SOI, Rev. 24

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-EP4 Objective: 5 & 9

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(7)	
Level of Difficulty:	3.0	

PRA Applicability:
None.

Examination Outline Cross Reference	Level	RO
203000 RHR/LPCI: Injection Mode (Plant Specific) A2. Ability to (a) predict the impacts of the following on the RHR/LPCI: INJECTION MODE (PLANT SPECIFIC) ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: A2.03 Valve closures	**Tier**	2
	Group #	1
	K/A	203000 A2.03
	Rating	3.2
	Revision	1
Revision Statement: **Rev 1, NRC Review and comments** **Changed will to should in (2), Specified "Wide Range RPV water level"**		

Question: 28

The crew is currently performing actions to mitigate a LOCA.

Current conditions

- Drywell Pressure 1.56 psig
- RPV water level -20 inches and rising by using HPCS and RCIC
- All Low Pressure ECCS Injection valves are OVERRIDDEN CLOSED per EP-2

The LOCA gets worse and level begins to lower.

CRS orders an Emergency Depressurization at -160” Wide Range RPV water level

(1) Which of the following describes the impact on the RHR systems ability to automatically inject into the RPV?

(2) What should be the crew’s action?

A. (1) NOT available to AUTO inject
(2) Manually OPEN injection valves using Handswitch.

B. (1) Will automatically inject at < 476 psig reactor pressure
(2) Verify automatic injection valve opening for all low pressure ECCS systems

C. (1) NOT available to AUTO inject
(2) Depress the RHR A/LPCS and RHR B/RHR C Manual Initiation Pushbuttons

D. (1) Will automatically inject when -150.3 “ Wide Range RPV water level is reached

(2) Verify automatic injection valve opening for all low pressure ECCS systems

Answer: **A**

Explanation:

When Low Pressure ECCS systems are overridden all automatic actions of those components are blocked. By taking the injection valves to overridden closed no auto actions will occur on those valves. Manual opening is required for injection.

‘A’ is correct

Distracters:

‘B’ is wrong – Plausible, normal action without any overriding, however, the student sees another auto initiation signal present.

‘C’ is wrong – Plausible, depressing the auto initiation pushbuttons has no effect on the overridden valves.

‘D’ is wrong – Plausible, if the student is confused about another initiation signal being received then this is plausible.

K/A Match

This question requires the applicant to predict the impact on the LPCI system with a valve closure and using procedures take actions.

Technical References:

04-1-01-E12-1, RHR SOI , Rev. 154

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-E1200 Objective: 8.9

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

10CFR Part 55 Content:	55.41(b)(7)	
Level of Difficulty:	3.0	
PRA Applicability:		
RHR is listed as #3 on the System Importance to CDF.		

Examination Outline Cross Reference	Level	RO
205000 Shutdown Cooling System (RHR Shutdown Cooling Mode) K3. Knowledge of the effect that a loss or malfunction of the SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE) will have on following: K3.01 Reactor pressure	Tier	2
	Group #	1
	K/A	205000 K3.01
	Rating	3.3
	Revision	2
Revision Statement: **Per Ops Rep, Changed stem to state when effect will this have on pressure control.** **Rev 2, NRC Review and comments** **Changed will to should in stem**		

Question: 29

The plant is in Mode 3 with MSIVs closed

RHR B was started a few moments ago in Shutdown Cooling

Reactor pressure is currently 30 psig

A spurious Group 3 isolation has occurred.

Reactor pressure should be controlled and maintained by…

A. SRVs

B. initiating RCIC.

C. the Turbine building Main Steam line Drains

D. the Main Turbine Bypass Valves

Answer: **A**
Explanation: When in Mode 3 decay heat is still high and at 30 psig the Reactor is still very hot. Any loss of cooling will cause reactor pressure to rise until acted upon When shutdown with MSIVs closed the only pressure control is SRVs. ‘A’ is correct

Distracters:

'B' is wrong – Plausible, even though RCIC can be used as level and pressure control, at <60 psig reactor pressure RCIC is isolated and unavailable.

'C' and 'D' are wrong – Plausible, Reactor pressure will rise, steam line drains and bypass valves are unavailable due to MSIVs are closed.

K/A Match

This question requires the applicant to have the knowledge of the effects on reactor pressure during a loss of shut down cooling.

Technical References:

GLP-OPS-E1200 Rev 10
03-1-01-4, Rev 117

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-E1200 Objective: 8.9

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(7)	
Level of Difficulty:	3.0	

PRA Applicability:

RHR is listed as #3 on the System Importance to CDF.

<table>
<tr><td>Examination Outline Cross Reference</td><td>Level</td><td>RO</td></tr>
<tr><td rowspan="5">205000 Shutdown Cooling System (RHR Shutdown Cooling Mode)

K6. Knowledge of the effect that a loss or malfunction of the following will have on the SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE) :

K6.08 RHR service water: Plant-Specific</td><td>Tier</td><td>2</td></tr>
<tr><td>Group #</td><td>1</td></tr>
<tr><td>K/A</td><td>205000 K6.08</td></tr>
<tr><td>Rating</td><td>3.5</td></tr>
<tr><td>Revision</td><td>1</td></tr>
<tr><td colspan="3">Revision Statement:

Rev 1, NRC Review and comments

Changed stem to state "What action (s) should be procedurally performed FIRST?"</td></tr>
</table>

Question: 30

The plant is in Mode 4

RHR A is operating in Shutdown Cooling

Standby Service Water Pump ‘A’ trips.

What action(s) should be procedurally performed FIRST?

A. Raise reactor water level to +82”.

B. Raise reactor water level to between +101” and +129”.

C. Start LPCS and inject to maintain 55 psig in the reactor.

D. Place RHR Pump ‘B’ and SSW pump ‘B’ in Shutdown Cooling.

Answer:	**D**
Explanation: Per 05-1-02-III-1, Inadequate Decay Heat Removal ONEP, “**IF** inservice loop of Shutdown Cooling has been lost, **THEN PLACE** redundant loop in Shutdown Cooling **OR** ADHRS in service per SOI 04-1-01-E12-2.” Just a loss of the SSW system causes a loss of cooling water flow to the RHR heat exchangers, therefore a loss of cooling. ‘D’ is correct	
Distracters:	

‘A’ is wrong – Plausible, Per 05-1-02-III-1, Inadequate Decay Heat Removal ONEP, **IF** forced circulation (no Reactor Recirculation pumps running) has been lost, **THEN RAISE** Reactor water level to 615" actual (this is 82" above instrument zero). This would be the action of both Recirc pumps were lost.

‘B’ is wrong – Plausible, Per 05-1-02-III-1, Inadequate Decay Heat Removal ONEP, **IF** adequate cooling **CANNOT** be maintained **OR** restarted within the estimated time to 200°F, **THEN PERFORM** the following:

> **SLOWLY RAISE** RPV water level using any available injection systems to between +101 inches **AND** +129 inches to establish flow through open SRVs back to Suppression Pool.

‘C’ is wrong - Plausible, Per 05-1-02-III-1, Inadequate Decay Heat Removal ONEP, this step is performed after the above step in ‘B’ is performed:

> **IF** RPV pressure does **NOT** stabilize above 55 psig, **THEN START** additional RHR pump(s) **AND OPEN** its associated Injection Valve (E12-F042A,B,C).
>
> **OR**
>
> **START** LPCS pump **AND OPEN** Injection Valve E21-F005 (1H13-P601) until RPV pressure is above 55 psig.

K/A Match

This question requires the applicant to have the knowledge of the effects during a loss of Service Water for shut down cooling.

Technical References:

05-1-02-III-1, Inadequate Decay Heat Removal ONEP, Rev 46

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-E1200 Objective: 13.3

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(7)	
Level of Difficulty:	3.0	
PRA Applicability:		

RHR is listed as #3 on the System Importance to CDF.

Examination Outline Cross Reference	Level	RO
209001 Low Pressure Core Spray System K5. Knowledge of the operational implications of the following concepts as they apply to LOW PRESSURE CORE SPRAY SYSTEM : K5.05 System venting	Tier	2
	Group #	1
	K/A	209001 K5.05
	Rating	2.5
	Revision	1
Revision Statement: **Rev 1, NRC Review and comments (U)** **Changed question**		

Question: 31

LPCS system is operating in test return to the suppression pool.

Power feeding the 15AA Bus is lost and restored by the emergency DG.

(1) What is the status of the LPCS pump

(2) What is required to restore the system?

A. (1) Running
(2) Restart Jockey pump only

B. (1) NOT Running
(2) Perform system fill and vent

C. (1) Running
(2) Perform system fill and vent

D. (1) NOT running
(2) Restart Jockey pump only

Answer:	B
Explanation: When the 15AA bus is lost, all pump and MOV power is lost. The LPCS pump will trip and all MOVs will fail as is. The Test return valve being open to the suppression pool and the pump not running causes the system to drain to the suppression pool. Per 04-1-01-E21-1, LPCS System Operating Instruction Precaution 2. System damage could occur if LPCS pump is started without discharge line filled and pressurized.	

'B' is correct

Distracters:

'A' is wrong – Plausible, the LPCS pump trips and will NOT restart due to no start signal present, the jockey pump is required to maintain the system pressurized but after system draining a system fill and vent is required.

'C' is wrong – Plausible, the LPCS pump trips and will NOT restart due to no start signal present,.

'D' is wrong - the jockey pump is required to maintain the system pressurized but after system draining a system fill and vent is required.

K/A Match

This question requires the applicant to have the knowledge of the effects during a loss of power to a running ECCS pump in test return mode and the requirements of a system fill and vent.

Technical References:

04-1-01-E21-1, LPCS SOI, Rev 44

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-E2100 Objective: 4.5, 7.0, 11.1

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(7)	
Level of Difficulty:	3.0	

PRA Applicability:

None.

Examination Outline Cross Reference	Level	RO
209001 Low Pressure Core Spray System 2.2.40 Ability to apply Technical Specifications for a system.	Tier	2
	Group #	1
	K/A	209001 2.2.40
	Rating	3.4
	Revision	1
Revision Statement: **Rev 1, NRC Review and comments (U)** **Removed second half of answer C, new answer D. ROs must be able to recognize entry into Tech Specs, answer A is checked on Tech Spec rounds, B is observed during a surveillance and D is a action to place the system in standby but not INOP.**		

Question: 32

Which of the following would require Low Pressure Core Spray (LPCS) system to be declared inoperable?

A. LPCS MOV Test Switch in TEST for 4 hours

B. LPCS flow indication on H13-P601 stuck downscale

C. LPCS Pump Discharge Flow trip unit E21-FIS-N651 failed upscale

D. LPCS system auto start on an inadvertent initiation signal

Answer: C
Explanation: All answers are plausible since they deal with LPCS components. The designated answer is correct since per SOI 04-1-01-E21-1 Section 5.1, step 2, the minimum flow valve must be capable of performing its intended function, otherwise, LPCS must be declared inoperable. With the flow switch failed upscale, min flow valve E21F011 would automatically close, which is not its standby line up, and would not automatically open when required.
Distracters: 'A' is incorrect, because TRM 6.8.2 allows the MOV Test switches to be in test for up to 8 hours before valve operability is affected, ROs should know this due to these switches are checked on the control room tech spec rounds. 'B' and 'D' are incorrect, because these components are not required for LPCS to perform its safety function.
K/A Match

This question requires the applicant to recognize the into Tech Specs.		
Technical References: 04-1-01-E21-1, LPCS SOI, Rev 44 Tech Specs 6.8.2		
Handouts to be provided to the Applicants during exam: NONE		
Learning Objective: GLP-OPS-E2100 Objective: 7.0, 11.1, 13.0		
Question Source:	**Bank # 905**	2007 NRC Exam Q 31
(note changes and attach parent)	**Modified Bank #**	
	New	
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.41(b)(7)	
Level of Difficulty:	3.0	
PRA Applicability:		
None.		

Examination Outline Cross Reference	Level	RO
209002 High Pressure Core Spray System K5. Knowledge of the operational implications of the following concepts as they apply to HIGH PRESSURE CORE SPRAY SYSTEM (HPCS): K5.04 Adequate core cooling: BWR-5,6 .	Tier	2
	Group #	1
	K/A	209002 K5.04
	Rating	3.8
	Revision	1
Revision Statement: **Rev 1, NRC Review and comments (U)** **Changed question to fill in the blank using procedure step in stem. Now a new question instead of bank**		

Question: 33

What are the MINIMUM requirements for flow and RPV level to assure adequate core cooling?

HPCS **OR** LPCS flow above __________ gpm **AND** Reactor water level above ________ inches (Spray Cooling).

A. 7400 gpm; -217”

B. 7000 gpm; -204”

C. 7000 gpm; -217”

D. 7400 gpm; -204”

Answer: C
Explanation: Adequate core cooling is defined as HPCS OR LPCS flow above 7000 gpm and Reactor water level above -217” (spray cooling)
Distracters: ‘A’ is wrong – Flow rate is too high Plausible due to 7400 is rated flow for RHR pumps. ‘B’ is wrong – level is too high -204” is Minimum Zero Injection RPV Water Level. ‘D’ is wrong – Flow rate is too high Plausible due to 7400 is rated flow for RHR pumps, level is too high -204” is Minimum Zero Injection RPV Water Level.

K/A Match
This question requires the applicant to recognize the procedural requirements for adequate core cooling.
Technical References:
04-1-01-E22-1 Rev 121 GLP-OPS-E2200 Rev 10 02-S-01-43 Rev 3
Handouts to be provided to the Applicants during exam:
NONE
Learning Objective:
GLP-OPS-EP01 Objective: 4a

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.41(b)(8)	
Level of Difficulty:	3.0	

PRA Applicability:
None.

Examination Outline Cross Reference	Level	RO
211000 Standby Liquid Control System A1. Ability to predict and/or monitor changes in parameters associated with operating the STANDBY LIQUID CONTROL SYSTEM controls including: A1.06 Flow indication: Plant-Specific	Tier	2
	Group #	1
	K/A	211000 A1.06
	Rating	3.8
	Revision	3
Revision Statement: **Rev 1 - Feedback – added "Per Att. 6."** **Rev 2 - Removed pump running reference**		

Question: 34

An ATWS is in progress at >5% power.

SLC has been initiated.

04-1-01-C41-1, Standby Liquid Control SOI, Attachment 6, Verification of Standby Liquid Control Injection is being performed.

Per Attachment 6, **CHECK** SLC INJECTING INTO the RPV by observing which of the following?

1. Discharge pressure greater than reactor pressure
2. Tank level lowering
3. Squib valves fired
4. Nuclear Instrumentation lowering

A. 2, 3, 4

B. 1, 2, 3

C. 1, 3, 4

D. 1, 2, 4,

Answer: D
Explanation: Per 04-1-01-C41-1, Standby Liquid Control SOI, Attachment 6, Verification of Standby Liquid Control Injection The positive indications that SLC is injecting into the reactor is SLC pump discharge pressure above reactor pressure and tank level lowering and Nuclear Instrumentation lowering.

‘D’ is correct.

Distracters:

‘B’ is wrong – Squib valve firing is not required due to indication may not be reliable. Plausible due to it is listed for checking system initiation but not for injection verification.

‘C’ is wrong – Pump running indication is not required if discharge pressure is above reactor pressure, a pipe break could have occurred and even though the pump is running it is not injecting into the reactor. Plausible due to it is listed for checking system initiation but not for injection verification.

‘D’ is wrong – Pump running indication is not required if discharge pressure is above reactor pressure, a pipe break could have occurred and even though the pump is running it is not injecting into the reactor. Plausible due to it is listed for checking system initiation but not for injection verification and Squib valve firing is not required due to indication may not be reliable. Plausible due to it is listed for checking system initiation but not for injection verification.

K/A Match

This question requires the applicant to have the ability to verify SLC flow is injecting into the reactor from multiple indications.

Technical References:

04-1-01-C41-1 Rev 125

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-C4100 Objective: 12

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.41(b)(8)	
Level of Difficulty:	2.0	

PRA Applicability:

None.

<table>
<tr><td>Examination Outline Cross Reference</td><td>Level</td><td>RO</td></tr>
<tr><td rowspan="5">212000 Reactor Protection System

K3. Knowledge of the effect that a loss or malfunction of the REACTOR PROTECTION SYSTEM will have on following:

K3.02 Primary containment isolation system/nuclear steam supply shut-off: Plant-Specific</td><td>Tier</td><td>2</td></tr>
<tr><td>Group #</td><td>1</td></tr>
<tr><td>K/A</td><td>212000 K3.02</td></tr>
<tr><td>Rating</td><td>3.7</td></tr>
<tr><td>Revision</td><td>1</td></tr>
<tr><td colspan="3">Revision Statement:

Rev 1, NRC Review and comments

Changed will to should in stem</td></tr>
</table>

Question: 35

The plant is operating at rated power.

The incoming feeder breaker to the 16AB bus trips and the Div 2 DG restores 16AB power.

The MSIV/ DR VLV TRIP INIT red alarm is sealed-in.

Which of the following should cause all MSIVs to close?

A. ½ Scram on ‘A’ side

B. Main Steam Tunnel Temp B fail high

C. Main Steam Tunnel delta temp A fail high

D. C71-S003A, MG SET A OUTPUT BREAKER trip

<table>
<tr><td>Answer: D</td></tr>
<tr><td></td></tr>
<tr><td>Explanation:

MSIV solenoids are powered from RPS bus power. Both MSIV solenoids losing power is required for the MSIVs to close.

A loss of 16AB will cause a loss of logic power to the group 1 MSIVs isolation system. This in turn causes a ½ isolation on Div 2 (B) side and loss of power to the Div 2 solenoids for the MSIVs. This is given with the receiving of the MSIV/ DR VLV TRIP INIT red alarm

To close the MSIVs power must be removed from the Div 1 solenoids.

‘D’ is correct. C71-S003A is an EPA breaker but with this breaker tripping will cause all power to be</td></tr>
</table>

removed from the RPS 'A' bus and power to the 'A' MSIV solenoids. Now both solenoids are without power and a full MSIV closure will occur.

Distracters:

'A' is wrong – This action will only cause a 1/2 scram not a MSIV closure, power must be totally removed for the RPS bus to remove power to the B solenoids for the MSIVs. RPS power is only removed from the Scram logic. Plausible due to scram logic and MSIVs solenoids are powered from the RPS bus.

'B' is wrong – With a 'B' temperature in the steam tunnel failing high would cause a Group 1 valve logic to initiate, however this would initiate a Div 2 isolation signal which the Div 2 solenoid is already de-energized from the loss of logic power.

'C' is wrong – Main Steam tunnel delta Temp no longer provides a Group 1 isolation signal.

K/A Match

This question requires the applicant to have the knowledge of the effects on the Containment isolation system on loss of RPS power.

Technical References:

04-1-01-C71-1 Rev 37
04-1-02-1H13-P601-19A-E4, MSIV/ DR VLV TRIP INIT

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-C7100 Objective: 11.6

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(9)	
Level of Difficulty:	3.0	

PRA Applicability:

RPS is listed as #5 on the System Importance to CDF.

Examination Outline Cross Reference	Level	RO
212000 Reactor Protection System A4. Ability to manually operate and/or monitor in the control room: A4.08 Individual system relay status: Plant-Specific	Tier	2
	Group #	1
	K/A	212000 A4.08
	Rating	3.4
	Revision	1
Revision Statement: **Rev 1, NRC Review and comments (U)** **Changed question and answers to determine which status lights are OFF**		

Question: 36

HANDOUT PROVIDED

The plant is currently in a refueling outage and operators are pulling fuses on a clearance for a Reactor Protection System (RPS) logic modification.

- Currently fuse 1C71F27B is removed.
- The Non-licensed operator has just removed fuse 1C71F14C.

Using E1173-019 determine which of the RPS System SCRAM SOL VLV status lights on P680 are OFF.

A. All lights are OFF

B. 1A through 4A and 1B OFF only

C. 1A through 4A only

D. 1B only

Answer: B
Explanation: Ref E1173-019; pulling fuse F27B only de-energizes white indicating light 1B, not the K14B or K14F relays. Pulling fuse F14C, de-energizes K14C and K14G which gives a half scram in channel 'C' which is a Division I channel and all Div 1 lights will be off 1A through 4A 'B' is correct.
Distracters:

‘A’ is wrong – this answer assumes a full scram but by pulling the K14C will cause a half scram only and pulling F27B only deenergizes the light 1B,

‘C’ is wrong – only indicates all div 1 lights not the one div 2 light only

‘D’ is wrong – pulling fuse F27B only de-energizes white indicating light, not the K14B or K14F relays. Pulling fuse F14C, de-energizes K14C and K14G which gives a half scram in channel 'C' which is a Division I channel.

K/A Match

This question requires the applicant to have the ability to determine individual system relay status with given conditions with the RPS system.

Technical References:

E1173-019

Handouts to be provided to the Applicants during exam:

E1173-019

Learning Objective:

GLP-OPS-C7100 21, 5.3, 6.3

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(8)	
Level of Difficulty:	3.0	

PRA Applicability:

RPS is listed as #5 on the System Importance to CDF.

Examination Outline Cross Reference	Level	RO
215003 Intermediate Range Monitor (IRM) System K2. Knowledge of electrical power supplies to the following: K2.01 IRM channels/detectors	Tier	2
	Group #	1
	K/A	215003 K2.01
	Rating	2.5
	Revision	1
Revision Statement: **Rev 1, NRC Review and comments** **Changed 'A' to BOP DC**		

Question: 37

Which of the following is the power supply to the IRM detectors?

A. BOP DC

B. BOP Inverters

C. ESF Inverters

D. ESF MCC AC

Answer: C
Explanation: IRMs are powered from ESF Inverters 1Y87, 1Y88, 1Y95, and 1Y96 'C' is correct.
Distracters: 'A' is wrong – The IRMs are AC power not DC 'B' is wrong – BOP inverters deliver the same power but to non ESF systems 'D' is wrong – ESF power supply is correct but a MCC is 480Vac not 120Vac
K/A Match This question requires the applicant to have the knowledge of the IRM power supplies.
Technical References:

GLP-OPS-C5102, Rev 6, IRM Lesson Plan
04-1-01-C51-1, Neutron Monitoring SOI, Rev 32

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-C5102, Objective 6

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.41(b)(7)	
Level of Difficulty:	2.0	

PRA Applicability:

None.

Examination Outline Cross Reference	Level	RO
215004 Source Range Monitor (SRM) System K4. Knowledge of SOURCE RANGE MONITOR (SRM) SYSTEM design feature(s) and/or interlocks which provide for the following: K4.04 Changing detector position	Tier	2
	Group #	1
	K/A	215004 K4.04
	Rating	2.8
	Revision	1
Revision Statement: **Rev 1, NRC Review and comments (U)** **Changed answers to make more plausible**		

Question: 38

A reactor startup is in progress with the reactor having just gone critical.

- Operators have verified SRM/IRM overlap
- Operators are continuing to withdraw control rods
- SRMs are being withdrawn as necessary to maintain SRM count rate between 1×10^2 and 1×10^5 cps.

The SRM 'E' detector fails and indicates 0 cps output.

Which of the following describes the result?

A. Div 1 half scram only

B. Control Rod Withdrawal Block and Div 1 half scram

C. Control Rod Withdrawal Block only

D. SRM DNSC and RETRACT PERMIT lights only

Answer:	**C**
Explanation: With Overlap confirmed then you must assume that IRMs are between range 1 and 3. If the SRM detector is not full in and indication is <100 cps a rod block is generated. 'C' is correct.	

Distracters:

'A' is wrong – Even though this is considered an INOP, SRMs will not provide scram signals.

'B' is wrong – SRMs will not provide scram signals, only Withdraw blocks

'D' is wrong – The DNSC (downscale) light will be lit however, the retract permit will not.

K/A Match

This question requires the applicant to have the knowledge of the SRM detector wrong position rod block.

Technical References:

GLP-OPS-C5101, Rev 11, SRM Lesson Plan
04-1-01-C51-1, Neutron Monitoring SOI, Rev 32

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-C5101, Objective 8.2

Question Source:	**Bank # 652**	2013 Audit Exam
(note changes and attach parent)	**Modified Bank #**	
	New	
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(7)	
Level of Difficulty:	3.0	

PRA Applicability:

None.

<table>
<tr><td>Examination Outline Cross Reference</td><td>Level</td><td>RO</td></tr>
<tr><td rowspan="6">215005 Average Power Range Monitor/Local Power Range Monitor System

K2. Knowledge of electrical power supplies to the following:

K2 02 APRM channels</td><td>Tier</td><td>2</td></tr>
<tr><td>Group #</td><td>1</td></tr>
<tr><td>K/A</td><td>215005 K2.02</td></tr>
<tr><td>Rating</td><td>2.6</td></tr>
<tr><td>Revision</td><td>1</td></tr>
<tr></tr>
<tr><td colspan="3">Revision Statement:

Rev 1, NRC Review and comments

Changed “is” to should be in question part of stem.</td></tr>
</table>

Question: 39

Which of the following is the power supply to APRM Channel 1 cabinet?

What should be the result if this power is lost?

A. 1Y99
Division 1 half scram

B. 1Y87
Division 1 half scram

C. 1Y87
Control Rod Block only

D. 1Y99
Control Rod Block only

Answer: **B**

Explanation:

The UPS system supplies regulated 120VAC power to the particular APRM channel and Voter. Each channel and voter has individual UPS power supply. 1Y87 feeds Channel 1. A loss of 120VAC UPS will result in a half scram condition for the particular RPS due to the Voter de-energizing.

‘B’ is correct.

Distracters:

‘A’ is wrong – 1Y99 has the same features as an ESF inverter and provides power to important control features however it is considered a BOP inverter and does not supply power to the APRMs

‘C’ is wrong – A loss of cabinet power will de-energize the voter and cause a half scram

‘D’ is wrong – 1Y99 has the same features as an ESF inverter and provides power to important control features however it is considered a BOP inverter and does not supply power to the APRMs and A loss of cabinet power will de-energize the voter and cause a half scram.

K/A Match

This question requires the applicant to have the knowledge of the APRM power supplies and the resulting actions if lost.

Technical References:

GLP-OPS-C5104, Rev 11, APRM Lesson Plan

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-C5101, Objective 7.2, 10, 11.2, 14

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.41(b)(7)	
Level of Difficulty:	3.0	

PRA Applicability:

None.

Examination Outline Cross Reference	Level	RO
215005 Average Power Range Monitor/Local Power Range Monitor System A4. Ability to manually operate and/or monitor in the control room: A4.04 LPRM back panel switches, meters and indicating lights	Tier	2
	Group #	1
	K/A	215005 A4.04
	Rating	3.2
	Revision	3
Revision Statement: **Rev 1 – Added "to initiate a control rod block" to the stem for clarification** **Rev 2 – Added (not bypassed) to stem for clarification**		

Question: 40

APRM Channel 2 has the following LPRM indications on the backpanels:

- 8 LPRMs are bypassed
- 2 additional LPRMs are indicating 0 (not bypassed)

(1) How many more LPRMs must be bypassed to initiate a control rod withdrawal block?

(2) How many LPRMs are required per level for APRM operability?

A. (1) 14
   (2) 2

B. (1) 14
   (2) 3

C. (1) 16
   (2) 2

D. (1) 16
   (2) 3

Answer:	**D**
Explanation: The APRM count and level feature monitors and indicates the number of valid LPRMs assigned to an APRM channel. Each LPRM which is not in BYPASS is counted by the APRM. LPRM outputs which are failed low or high are considered valid inputs until they are bypassed at the respective APRM cabinet. If there are less than 21 valid LPRM inputs or less than 3 LPRMs per detector level to the APRM channel,	

the count circuit will initiate a control rod withdrawal block and an APRM Trouble Annunciator.

'D' is correct.

Distracters:

'A' is wrong – The student must know if the LPRM is reading 0 then it is still counted in the counting circuit. Two LPRMs are indicating 0 but are still counted. Less than 3 LPRMs per detector level to the APRM channel will cause an inop.

'B' is wrong – The student must know if the LPRM is reading 0 then it is still counted in the counting circuit. Two LPRMs are indicating 0 but are still counted

'C' is wrong – Less than 3 LPRMs per detector level to the APRM channel will cause an inop.

K/A Match

This question requires the applicant to have the ability to monitor LPRM indications in the control room to determine status.

Technical References:

GLP-OPS-C5104, Rev 11, APRM Lesson Plan
04-1-01-C51-4, Neutron Monitoring SOI, Rev 32

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-C5104, Objective 7.1

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(7)	
Level of Difficulty:	3.0	

PRA Applicability:

None.

Examination Outline Cross Reference	Level	RO
217000 Reactor Core Isolation Cooling System (RCIC) 2.4.34 Knowledge of RO tasks performed outside the main control room during an emergency and the resultant operational effects.	Tier	2
	Group #	1
	K/A	217000 2.4.34
	Rating	4.2
	Revision	2

Revision Statement:

Rev 1 – Validation, validators recognized that 'A' was also correct, so added "plant operator has been dispatched to the RCIC room" for clarification

Rev 2, NRC Review and comments (U)

Clarified question in the stem, the RCIC system Group 4 isolation signal cannot be reset from the RSP, all other auto actions can be reset or bypassed from outside the control room. Added to the stem "...for continued operation from the remote shutdown panel." Changed answer 'C' to an actual isolation signal that the student must recognize as such.

Question: 41

Events have occurred and the Control Room has been evacuated.

RCIC is initiated and controls are taken from the Remote Shutdown Panel.

Plant operator has been dispatched to the RCIC room.

Which of the following should cause RCIC to be unavailable for continued operation from the Remote Shutdown Panel?

A. Overspeed trip

B. RPV level 8 is reached

C. RCIC Exhaust Diaphragm pressure 15 psig

D. Auto suction swap to Suppression Pool

Answer: **C**

Explanation:

If RCIC is operating from the remote Shutdown Panel and a Group 4 isolation is received then RCIC will no longer be available for level control due to the Group 4 can't be reset from the remote shutdown panel. The correct answer of "RCIC Exhaust Diaphragm pressure of 15 psig" is above the setpoint for a Group 4 isolation signal. The setpoint is 10 psig. The student has to determine that the Group 4 isolation signal was exceeded and that a Group 4 isolation will prevent operation of RCIC from the Remote Shutdown Panel.

'C' is correct.

Distracters:

'A' is wrong – The overspeed trip can be reset locally and RCIC restarted.

'B' is wrong – The F045 will auto close but RCIC can be restarted when level 8 auto resets.

'D' is wrong – A suction swap has not effect on the operation of the RCIC system.

K/A Match

This question requires the applicant to have the Knowledge of RO tasks for RCIC performed outside the main control room during an emergency and the resultant operational effects.

Technical References:

GLP-OPS-E5100, Rev 11, RCIC Lesson Plan
04-1-01-E51-1, RCIC SOI, Rev 142
05-1-02-II-1, Shutdown From the Remote Shutdown Panel ONEP, Rev 51

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-E5100, Objective 11.5

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(7)	
Level of Difficulty:	3.0	

PRA Applicability:

RCIC is listed as #17 on the System Importance to CDF.

Examination Outline Cross Reference	Level	RO
218000 Automatic Depressurization System A1. Ability to predict and/or monitor changes in parameters associated with operating the AUTOMATIC DEPRESSURIZATION SYSTEM controls including: A1.01 ADS valve tail pipe temperatures	Tier	2
	Group #	1
	K/A	218000 A1.01
	Rating	3.4
	Revision	1
Revision Statement: **Rev 1 – Validation, made a 1 / 2 question**		

Question: 42

A loss of all RPV level indication has occurred during a LOCA.

The CRS has entered EP-5 and directed injection into the RPV.

The CRS has directed you to monitor for RPV flooding.

(1) Which of the following describes the tail pipe temperature indication for RPV flooding?

And

(2) Which valves should be monitored?

A. (1) Rising
(2) ADS valves only

B. (1) Lowering
(2) ADS valves only

C. (1) Rising
(2) LO-LO Set SRVs only

D. (1) Lowering
(2) LO-LO Set SRVs only

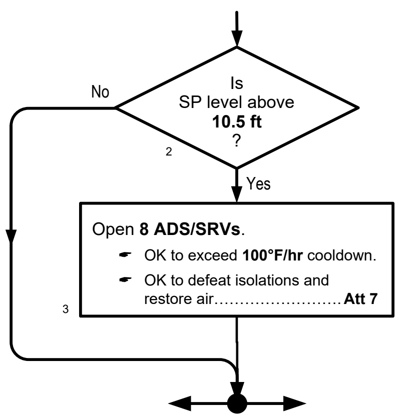
Answer:	B
Explanation: Per EP-5 Flooded RPV Indications for SRV tail pipe is "Lowering SRV tailpipe temperatures"	

Flooded RPV Indications

NOTE: Some combination of the following conditions should be observed as water fills the steam lines and flows through SRVs. No single indication can be relied on in all events.

- Rising RPV pressure
- Lowering SRV tailpipe temperatures
- If injecting from the suppression pool, SP level lowers as the RPV and steam lines are flooded, then stabilizes when the steam lines are full.
- If injecting from outside the CTMT, SP level rising.
- MSL or RCIC high flow trips.
- If MSIVs are open, two phase flow audible near the main steam tunnel, main steam chest, or main turbine valves.

When EP-5 was entered one of the first directions is to Emergency Depressurize the RPV. That direction is to open 8 ADS valves.



Open 8 ADS/SRVs

RPV depressurization can most easily and rapidly be performed by opening SRVs. Those associated with the ADS function are used first, consistent with automatic system functions. The ADS SRVs are generally considered the most reliable and are arranged to provide uniform distribution of the heat load around the suppression pool. If one or more ADS valves cannot be opened, other SRVs are opened until the total number of open SRVs equals the number of SRVs dedicated to ADS function.

'B' is correct

Distracters:

'A' is wrong – Even though water will be exiting the SRV the Water temperature is lower that the steam temp therefore, tailpipe temp will lower not rise, the EP-5 says to open 8 ADS valves not LO-LO set valves.

'C' is wrong – Even though water will be exiting the SRV the Water temperature is lower that the steam temp therefore, tailpipe temp will lower not rise.

‘D’ is wrong – EP-5 says to open 8 ADS valves not LO-LO set valves.

K/A Match

This question requires the applicant to have the ability to monitor tail pipe temperatures during ADS valve operation.

Technical References:

02-S-01-40, EP TECHNICAL Bases, Rev 9

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-EP5, OBJ 7

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(14)	
Level of Difficulty:	3.0	

PRA Applicability:

ADS is listed as #6 on the System Importance to CDF.

<table>
<tr><td>Examination Outline Cross Reference</td><td>Level</td><td>RO</td></tr>
<tr><td rowspan="5">223002 Primary Containment Isolation System/Nuclear Steam Supply Shut-Off

K1. Knowledge of the physical connections and/or cause effect relationships between PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF and the following:

K1.14 Containment drainage system</td><td>Tier</td><td>2</td></tr>
<tr><td>Group #</td><td>1</td></tr>
<tr><td>K/A</td><td>223002 K1.14</td></tr>
<tr><td>Rating</td><td>2.8</td></tr>
<tr><td>Revision</td><td>1</td></tr>
<tr><td colspan="3">Revision Statement:

Rev 1, NRC Review and comments

Changed “is” to “should be” in the stem</td></tr>
</table>

Question: 43

The Plant has experienced a RPV leak inside the Drywell.

Drywell pressure is currently 1.5 psig.

What should be the status of the **Containment** floor and equipment drain systems?

	Floor Drain	Equipment Drain
A.	In recirc	In recirc
B.	OFF	OFF
C.	Operating	OFF
D.	OFF	Operating

<table>
<tr><td>Answer: B</td></tr>
<tr><td></td></tr>
<tr><td>Explanation:

When Drywell pressure exceeds 1.23 psig a Containment isolation occurs. This action causes Containment and Floor drain system to isolate their discharge

‘B’ is correct</td></tr>
<tr><td></td></tr>
<tr><td>Distracters:

‘A’ is wrong – Plausible, the Drywell Floor and equipment drain system can go into a recirc mode if</td></tr>
</table>

temperature of the water is elevated not the Containment sump systems.

'C' & 'D' are wrong – Both systems will isolate not just one.

K/A Match

This question requires the applicant to have the knowledge of the interrelationship between Containment Floor and Equipment drain system and the Containment isolation system.

Technical References:

04-1-01-P45-1, Rev 20

GLP-OPS-P4500, Rev 20

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-P4500, OBJ 7

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(9)	
Level of Difficulty:	3.0	

PRA Applicability:

None.

<table>
<tr><td>Examination Outline Cross Reference</td><td>Level</td><td>RO</td></tr>
<tr><td rowspan="5">239002 Relief/Safety Valves

K4. Knowledge of RELIEF/SAFETY VALVES design feature(s) and/or interlocks which provide for the following:

K4.02 Minimizes containment fatigue duty cycles resulting from relief valve cycling during decay-heat-dominant period late in an isolation transient (LLS logic): Plant-Specific</td><td>Tier</td><td>2</td></tr>
<tr><td>Group #</td><td>1</td></tr>
<tr><td>K/A</td><td>239002 K4.02</td></tr>
<tr><td>Rating</td><td>3.4</td></tr>
<tr><td>Revision</td><td>1</td></tr>
<tr><td colspan="3">Revision Statement:

Rev 1, NRC Review and comments

Changed “would” to “should” is stem</td></tr>
</table>

Question: 44

The plant is at rated power early in core life, when the MSIVs close causing a reactor scram.

Reactor pressure reaches 1120 psig resulting in multiple SRVs cycling.

Following the initial transient, the SRV(s) should be expected to automatically maintain reactor pressure between:

A. 926 and 1033 psig

B. 936 and 1073 psig

C. 946 and 1113 psig

D. 1013 and 1113 psig

Answer: A
Explanation: Low-Low Set initiates automatically when RPV pressure reaches SRV F051D’s normal lift setpoint of 1103 psig. Once initiated, six SRVs are capable of operating in the Low-Low Set mode with the following adjusted opening and closing setpoints: One SRV, F051D, lifts at 1033 psig and blows down to 926 psig. A second SRV, F051B, lifts at 1073 psig and blows down to 936 psig. Only one SRV is needed to maintain pressure after the initial transient. ‘A’ is correct

Distracters:

‘B’ is wrong – Plausible, the F051B will open and close at these setpoints

‘C’ & ‘D’ are wrong – Both indicate the opening and closing setpoints of other lo-lo set valves and other relief setpoints for SRVs

K/A Match

This question requires the applicant to have the knowledge of the lo-lo set system and the reason for it to provide pressure control to minimize the cycles of SRVs.

Technical References:

GLP-OPS-E2202, Rev 9

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-E2202, OBJ 11.2

Question Source:	**Bank # 1086**	2014 NRC exam
(note changes and attach parent)	**Modified Bank #**	
	New	
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(7)	
Level of Difficulty:	3.0	

PRA Applicability:

None.

Examination Outline Cross Reference	Level	RO
259002 Reactor Water Level Control System A3. Ability to monitor automatic operations of the REACTOR WATER LEVEL CONTROL SYSTEM including: A3.04 Changes in reactor feedwater flow	Tier	2
	Group #	1
	K/A	259002 A3.04
	Rating	3.2
	Revision	1
Revision Statement: **Rev 1, NRC Review and comments** **Changed "will" to "should"**		

Question: 45

The reactor is operating at rated power.

The C34 Feedwater Flow transmitter fails full upscale to 12.75 mlbm/hr.

What effects should directly result from this condition?

A. The feedwater Master level controller will switch to MANUAL.

B. The feedwater system will automatically lower feedwater flow.

C. The feedwater system will automatically transfer to single-element control.

D. An "estimated" feedwater flow signal replaces the failure transmitter with no change in control.

Answer: **C**
Explanation: "Hard-failures" of feedwater flow signals are responded to by immediately de-selecting and disabling three-element control.
Distracters: 'A' is wrong – Plausible because if all four level channels hard fail the level control stations will automatically swap to MANUAL. 'B' is wrong - Plausible if the upscale feed flow signal is sensed and used as a good input to the level control system. High feed flow the system would attempt to reduce feed flow to anticipate a rise in level due to feed flow steam flow mismatch. 'D' is wrong - Plausible because an estimated feedwater flow can replace the normal total feedwater flow

signal and three-element control can be re-selected and returned to service.

K/A Match

This question requires the applicant to have the knowledge of hard failure on the RPV level control system.

Technical References:

GLP-OPS-C3400 Obj 6.3

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-C3400 Obj 6.3

Question Source:	**Bank # 1198**	2015 NRC exam
(note changes and attach parent)	**Modified Bank #**	
	New	
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.41(b)(7)	
Level of Difficulty:	3.0	

PRA Applicability:

None.

Examination Outline Cross Reference	Level	RO
261000 Standby Gas Treatment System K1. Knowledge of the physical connections and/or cause effect relationships between STANDBY GAS TREATMENT SYSTEM and the following: K1.08 Process radiation monitoring system	Tier	2
	Group #	1
	K/A	261000 K1.08
	Rating	2.8
	Revision	1
Revision Statement: **Rev 1, NRC Review and comments (U)** **Restructured answers to A/D, A/C, B/C, and B/D**		

Question: 46

Which of the following will cause a Standby Gas Treatment system 'B' initiation?

A. Fuel Pool Sweep exhaust rad monitor 'A' indicating 31 mrem/hr
Fuel Pool Sweep exhaust rad monitor 'D' INOP trip

B. Fuel Pool Sweep exhaust rad monitor 'A' indicating 31 mrem/hr
Fuel Pool Sweep exhaust rad monitor 'C' indicating 38.1 mrem/hr

C. Fuel Handling Area exhaust rad monitor 'B' indicating 5.3 mrem/hr
Fuel Handling Area exhaust rad monitor 'C' INOP trip

D. Fuel Handling Area exhaust rad monitor 'B' indicating 4.2 mrem/hr
Fuel Handling Area exhaust rad monitor 'D' mode switch in STANDBY

Answer: **C**
Explanation: For SBGT to initiate on process rad monitor system the 'A' systems must see A and D monitors and 'B' must see B and C monitors, for Fuel handling area > 3.6 mrem/hr or INOP OR Fuel pool sweep >30 mrem/hr or INOP.
Distracters: 'A' is wrong, due to A and D monitors will initiate the 'A' system not 'B' 'B' is wrong, due to A and C monitors will not initiate either system. 'D' is wrong, due to B and D monitors will not initiate either system

K/A Match This question requires the applicant to have the knowledge of the interrelationship with process radiation monitoring system and the SBGT system.		
Technical References: GLP-OPS-T4801, Rev. 10		
Handouts to be provided to the Applicants during exam: NONE		
Learning Objective: GLP-OPS-T4801 Obj 8.7		
Question Source:	**Bank # 747**	2014 NRC exam
(note changes and attach parent)	**Modified Bank #**	
	New	
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.41(b)(11)	
Level of Difficulty:	2.0	
PRA Applicability:		
None.		

<table>
<tr><td>Examination Outline Cross Reference</td><td>Level</td><td>RO</td></tr>
<tr><td rowspan="5">262001 A.C. Electrical Distribution

2.4.21 Knowledge of the parameters and logic used to assess the status of safety functions, such as reactivity control, core cooling and heat removal, reactor coolant system integrity, containment conditions, radioactivity release control, etc.</td><td>Tier</td><td>2</td></tr>
<tr><td>Group #</td><td>1</td></tr>
<tr><td>K/A</td><td>262001 2.4.21</td></tr>
<tr><td>Rating</td><td>4.0</td></tr>
<tr><td>Revision</td><td>1</td></tr>
<tr><td colspan="3">Revision Statement:

Rev 1, NRC Review and comments

Changed answers to a 2X2, Added "Without any operator action." to the stem. Due to loss of power ONEP has the bus restored as an immediate action.</td></tr>
</table>

Question: 47

Division 1 Diesel Generator is currently in MAINT MODE.

The voltage to ALL ESF busses drops to 3290 volts.

The voltage transient duration is 10 seconds and then voltage returns to normal.

All other plant parameters are normal.

Without any operator actions:

(1) Which one of the following states the condition of the ESF busses after this voltage transient?

And

(2) What should be the status of the ECCS systems?

A. (1) 16AB and 17AC are energized by their Diesel Generators
(2) Only Division 2 and 3 ECCS systems are available

B. (1) 16AB and 17AC are energized by their Diesel Generators
(2) All ECCS systems are available

C. (1) 16AB is energized by its Diesel Generator only
(2) Only Division 2 and 3 ECCS systems are available

D. (1) 17AC is energized by its Diesel Generator only
(2) All ECCS systems are available

Answer:	**C**	
Explanation: This is a 79% BUV which after 9 seconds will cause a Div 1 & 2 Shed causing those buses to be transferred to the EDG; however, the 88% Div 3 BUV will not shed its bus unless the condition persists for 5 minutes. With Div 1 D/G in MAINT the D/G for 15AA will not start and 15AA will remain deenergized. Therefore Division 2 and 3 ECCS are the only ones available. 'C' is correct		
Distracters: 'A' is wrong, the 17 bus will not energize from its D/G 'B' is wrong, Div. 3 D/G did not get a start signal so 17AC is still energized through normal power feed and Div 1 ECCS is not available. 'D' is wrong, Div. 3 D/G did not get a start signal so 17AC is still energized through normal power feed and 16AB is powered from its DG		
K/A Match This question requires the applicant to have the knowledge to use the given information and determine the status of the ECCS safety systems.		
Technical References: GLP-OPS-R2100 Objective: 12 & 15		
Handouts to be provided to the Applicants during exam: NONE		
Learning Objective: GLP-OPS-R2100 Obj 12 & 15		
Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(7)	
Level of Difficulty:	3.0	

PRA Applicability:
R21 ESF is listed as # 8 on the System Importance to CDF. Div 1& 2 EDGs are listed as #11 on the System Importance to CDF.

Examination Outline Cross Reference	Level	RO
262002 Uninterruptable Power Supply (A.C./D.C.) K6. Knowledge of the effect that a loss or malfunction of the following will have on the UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.) : K6.03 Static inverter	Tier	2
	Group #	1
	K/A	262002 K6.03
	Rating	2.7
	Revision	1
Revision Statement: **Per Ops Rep, Added first part of the question.**		

Question: 48

DC bus 11DD has been lost.

(1) Which of the following describes the actions on the UPS inverter system?

(2) Upon restoration of 11DD to the normal power supply, using 04-1-01-L62-1, Static Inverters SOI, what action should the operator perform to ensure return to normal operation?

A. (1) Inverter 1Y79 swaps to alternate source
(2) Verify 1Y79 automatically swaps to Normal Supply

B. (1) Inverter 1Y79 swaps to alternate source
(2) Depress INVERTER TO LOAD pushbutton on 1Y79 control panel

C. (1) Inverter 1Y88 swaps to alternate source
(2) Verify 1Y88 automatically swaps to Normal Supply

D. (1) Inverter 1Y88 swaps to alternate source
(2) Depress INVERTER TO LOAD pushbutton on 1Y88 control panel

Answer:	**A**
Explanation: The candidate must be able to determine which Inverter has lost power by knowing which is ESF and which is BOP powered. SOI P&L 3.4 The Static (automatic) switch <u>will</u> automatically transfer back to the normal supply (inverter supply load) upon restoration of normal power. The ESF Static Inverters **<u>AND</u>** 1Y99 Inverter do **<u>not</u>** have the Auto Retransfer feature back to DC Power. **<u>If</u>** DC Power is lost, the BATTERY INPUT breaker has an under voltage trip associated with it.	

1Y79 is powered from 11DD and a BOP inverter, 1Y88 is powered from ESF batteries and an ESF inverter.

'A' is correct

Distracters:

'B' is wrong, 1Y79 is a BOP inverter and will auto transfer to normal power

'C' & 'D' are wrong, 1Y88 is an ESF inverter and must be manually transferred back to normal power.

K/A Match

This question requires the applicant to have the knowledge to determine the difference between an ESF and BOP inverter and the effect on loss of an inverter.

Technical References:

04-1-01-L62-1
GLP-OPS-L6200

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-L6200, Obj. 17

Question Source:	**Bank # 1242**	Not used on an NRC exam
(note changes and attach parent)	**Modified Bank #**	
	New	
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(7)	
Level of Difficulty:	3.0	

PRA Applicability:

None.

Examination Outline Cross Reference	Level	RO
263000 D.C. Electrical Distribution K2. Knowledge of electrical power supplies to the following: K2.01 Major D.C. loads	Tier	2
	Group #	1
	K/A	263000 K2.01
	Rating	3.1
	Revision	1
Revision Statement: **Rev 1, NRC Review and comments** **Changed "is" to "should be" in stem**		

Question: 49

The plant is operating at rated conditions.

The following alarms are received:

- P680-9A-D11, GEN SEAL OIL TROUBLE
- P680-10A-C10, GEN H2 SEAL OIL PUMP C FAULT

Which of the following should be the DC Electrical bus that was lost?

A. 11DB

B. 11DD

C. 11DE

D. 11DF

Answer: D
Explanation: 04-1-01-1-L11 Attachment 1F shows the power supply for the DC SEAL OIL PUMP C MOTOR. 11DF is the only 250VDC system 'D' is correct
Distracters: All distractors are other DC buses, but all are 125VDC.

K/A Match This question requires the applicant to have the knowledge of the power supply for major DC loads.		
Technical References: 04-1-01-L11-1, Attachment 1F, 125V DC BUS 11DF LOAD LIST		
Handouts to be provided to the Applicants during exam: NONE		
Learning Objective: GLP-OPS-L1100 Obj 6.2, 8.3		
Question Source:	**Bank # 86**	Not used on an NRC exam
(note changes and attach parent)	**Modified Bank #**	
	New	
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.41(b)(7)	
Level of Difficulty:	2.0	
PRA Applicability:		
125 VDC ESF is listed as #7 on the System Importance to CDF.		

Examination Outline Cross Reference	Level	RO
264000 Emergency Generators (Diesel/Jet) A2. Ability to (a) predict the impacts of the following on the EMERGENCY GENERATORS (DIESEL/JET) ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: A2.01 Parallel operation of emergency generator	Tier	2
	Group #	1
	K/A	264000 A2.01
	Rating	3.5
	Revision	0
Revision Statement:		

Question: 50

Div 1 D/G has auto started due to an inadvertent signal and is running unloaded.

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

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

A. (1) DG GOV MAN CONT
   (2) SLOW

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

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

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

Answer: B
Explanation: Per 04-1-01-P75-1 section 4.2, "**ADJUST** Standby Diesel Generator 11 [12] speed to bring frequency within range of bus frequency by **USING** DG 11[12] GOV MAN CONT so that synchroscope indicator is **ROTATING SLOWLY** in the FAST direction (clockwise). 'B' is correct

Distracters:

‘A’ is wrong, but plausible. The answer states to move in the FAST direction. IF the D/G was carrying the bus alone and an offsite power source was being paralleled to it then the scope would be moving in the FAST direction.

‘C’ is wrong, but plausible. This switch is used but for adjusting voltage and VARS not speed. And The answer states to move in the FAST direction. IF the D/G was carrying the bus alone and an offsite power source was being paralleled to it then the scope would be moving in the FAST direction.

‘D’ is wrong, but plausible. This switch is used but for adjusting voltage and VARS not speed.

K/A Match

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

Technical References:

04-1-01-P75-1, Standby Diesel Generator SOI section 4.2, Rev 112

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-P7500 Obj 17

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(7)	
Level of Difficulty:	3.0	

PRA Applicability:

Div 1 & 2 EDGs are listed as #11 on the System Importance to CDF.

Examination Outline Cross Reference	Level	RO
300000 Instrument Air System (IAS) A4. Ability to manually operate and / or monitor in the control room: A4.01 Pressure gauges	Tier	2
	Group #	1
	K/A	300000 A4.01
	Rating	2.6
	Revision	0
Revision Statement: **NRC Review and comments (U)** **Changed question.**		

Question: 51

If Instrument Air header pressure drops to a__________ psig setpoint on Control Room panel 1H13-P870, the Air Bleed Off Station will vent the remaining air in the Instrument Air header to the Auxiliary Building atmosphere.

A. 110 psig

B. 90 psig

C. 60 psig

D. 30 psig

Answer: D
Explanation: If Instrument Air header pressure drops to ~30 psig, Air Bleed Off Station pilot valve F523 opens to vent the air from the air operator on PV-F531, causing it to fail open This vents the remaining air in the Instrument Air lines to the Auxiliary Building atmosphere. This ensures that any radioactive contamination, from the Containment or Drywell, that may have entered the depressurized and potentially degraded Instrument Air System lines, will be treated by the Standby Gas Treatment System rather than leak to the environment via a leak in the Instrument Air header somewhere outside the Auxiliary Building (Secondary Containment). 'D' is correct
Distracters: 'A' is wrong – Normal Control Room indication, plausible due to the air compressors will auto start on receiver pressure, the normal receiver pressure is 125 to 135 psig 'B' is wrong – This would be approximately the pressure at which the standby air compressor would auto start but the auto start comes from the receivers not the header. Also the low limit on the Control Room

rounds

'C' is wrong – This is the local air pressure at which a type of air operated valves will auto close to preserve closing capability.

K/A Match

This question requires the applicant to have ability to monitor the Instrument Air pressure in the control room by knowing location of indication and normal value and actions of low header pressure.

Technical References:

GLP-OPS-P5100, Plant Air System lesson plan, Rev. 8

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-P5100 Obj 6.6 and 26

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.41(b)(4)	
Level of Difficulty:	2.0	

PRA Applicability:

None.

Examination Outline Cross Reference	Level	RO
400000 Component Cooling Water System (CCWS) A3. Ability to monitor automatic operations of the CCWS including: A3.01 Setpoints on instrument signal levels for normal operations, warnings, and trips that are applicable to the CCWS	Tier	2
	Group #	1
	K/A	400000 A3.01
	Rating	3.0
	Revision	1
Revision Statement: **Rev 1, NRC Review and comments** **Deleted “(1)” in stem, added “should” in question part of stem.**		

Question: 52

The plant is operating at rated conditions.

A transient occurs on the CCW Pumps.

Currently only one CCW pump is running.

CCW header temperature is 96°F.

All subsequent actions from the Loss of CCW ONEP have been completed.

Which of the following, in addition to Reactor Recirculation Pumps, describes the components that should still have CCW flowing through them?

A. Control Rod Drive Pump oil coolers ONLY

B. Fuel Pool Cooling and Cleanup heat Exchangers ONLY

C. RWCU Non Regen Heat Exchanger and Control Rod Drive Pump oil coolers ONLY

D. RWCU Non Regen Heat Exchanger and Fuel Pool Cooling and Cleanup heat Exchangers ONLY

Answer:	**A**
Explanation: Per the ONEP If only one pump is running then isolate the CCW to Fuel Pool HT EX. and isolate CCW to RWCU non-regen Ht EX. This is RO knowledge because it covers the overall mitigating strategy of the ONEP. Also, with only one pump running a low flow condition will occur and automatically cause the	

FPCCU heat exchangers to isolate.

'A' is correct

Distracters:

'B' is wrong – FPCCU would be isolated due to performing the actions of the onep

'C' is wrong – RWCU would be isolated due to performing the actions of the onep

'D' is wrong - RWCU would be isolated due to performing the actions of the onep and FPCCU would already be isolated by performing actions of the onep

K/A Match

This question requires the applicant to have ability to monitor the automatic actions on a loss of a CCW pump.

Technical References:

GLP-OPS-P4200
05-1-02-V-1, Loss of CCW ONEP

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-P4200 Obj 10

Question Source:	**Bank # 1205**	2015 NRC Exam
(note changes and attach parent)	**Modified Bank #**	
	New	
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(10)	
Level of Difficulty:	3.0	

PRA Applicability:

None.

Examination Outline Cross Reference	Level	RO
209002 High Pressure Core Spray System (HPCS) K1. Knowledge of the physical connections and/or cause effect relationships between HIGH PRESSURE CORE SPRAY SYSTEM (HPCS) and the following: K1.02 Suppression Pool: BWR-5,6 .	Tier	2
	Group #	1
	K/A	209002 K1.02
	Rating	3.5
	Revision	1
Revision Statement: **Rev 1, NRC Review and comments** **Changed "wills to should in stem** **Changed (2) wording on which valve must be operated first**		

Question: 53

A valid SUPP POOL LVL HI annunciator is received on the P601 panel at rated power.

Consider the following valves:

- E22-F015, HPCS pump Suppression Pool Suction Valve
- E22-F001, HPCS pump CST Suction Valve

The HPCS system should respond to this event by ________(1)_______ .

After Suppression pool level is returned to normal the operator should realign the HPCS suction by ______(2)_______.

A. (1) opening E22-F015 and then closing E22-F001
   (2) manually closing E22-F015 then open E22-F001

B. (1) opening E22-F015 and then closing E22-F001
   (2) manually opening E22-F001 then close E22-F015

C. (1) closing E22-F001 and then opening E22-F015
   (2) manually closing E22-F015 then open E22-F001

D. (1) closing E22-F001 and then opening E22-F015
   (2) manually opening E22-F001 then close E22-F015

Answer:	**A**
Explanation: Per ARI P601-16A-C5 (SUPP POOL LVL HI), the suction swap occurs by opening E22-F015 and then	

closing E22-F001. Although it is possible to manually realign the suctions at any point, a caution in the ARI reads "Restore normal suppression pool level before realigning HPCS suction to the CST."

Placing the switch momentarily in the OPEN position, will open the valve provided:

- it is in the fully closed position and,
- the HPCS Pump Suction from Suppression Pool Valve is NOT fully open.

F015 must be closed first

‘A’ is correct

Distracters:

‘B’ is wrong – HPCS suctions will not auto realign

‘C’ is wrong – F001 will not close first the system will maintain a suction path, also it receives a close signal from the F015 valve position not Supp Pool Level

‘D’ is wrong - F001 will not close first the system will maintain a suction path, also it receives a close signal from the F015 valve position not Supp Pool Level and HPCS suctions will not auto realign

K/A Match

This question requires the applicant to have the knowledge of the HPCS’s physical connection with the Suppression Pool and the cause and effect of suppression pool level.

Technical References:

GLP-OPS-E2201
ARI P601-16A-C5

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-E2201 Obj 9

Question Source:	**Bank # 563**	Not on previous NRC Exam
(note changes and attach parent)	**Modified Bank #**	
	New	
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(10)	
Level of Difficulty:	3.0	

PRA Applicability:
HPCS is listed as #16 on the System Importance to CDF.

Examination Outline Cross Reference	Level	RO
201001 Control Rod Drive Hydraulic System K4. Knowledge of CONTROL ROD DRIVE HYDRAULIC SYSTEM design feature(s) and/or interlocks which provide for the following: K4.08 Controlling control rod drive header pressure	Tier	2
	Group #	2
	K/A	201001 K4.08
	Rating	3.1
	Revision	1
Revision Statement: **Rev 1, Facility Rep, F002 is mentioned in only one answer, added the mention of F002 in answer 'B'**		

Question: 54

Given the following CRD system components:

C11-F003, CRD DRIVE WTR PRESS CONT VLV
C11-F002, CRD FLOW CONT VLV

Which of the following describes how CRD Drive Water D/P is maintained during reactor pressurization?

A. F003 is adjusted manually ONLY to maintain D/P and flow until rated pressure is achieved in the RPV.

B. F003 automatically adjusts position to maintain D/P while the F002 automatically positions to maintain flow during pressurization.

C. F003 is manually set to provide the required D/P initially then the automatic positioning of the F002 will maintain D/P during pressurization.

D. F003 automatically adjusts position to maintain system flow which in turn will maintain the required D/P until pressurization is complete.

Answer: **C**
Explanation: During a reactor startup and pressurization the drive water D/P is set at 250 psid. As reactor pressure rises CRD flow begins to drop due to higher pressure in the RPV, the CRD FCV F002 will open to maintain set flowrate of 60 gpm. When the F002 opens to provide more flow the D/P will rise back to the set value. Therefore, with the flow control valve in automatic, set at 60 gpm, and the pressure control valve set at 250 psid, as reactor pressure increases, the flow control valve opens to provide more flow and maintain the appropriate drive water Δp. 'C' is correct

Distracters:

'A' is wrong – Only one adjustment is needed to maintain D/P

'B' is wrong – F003 is a manual MOV and does not operate in automatic

'D' is wrong - F003 is a manual MOV and does not operate in automatic

K/A Match

This question requires the applicant to have the knowledge of the design features of the CRD system to control D/P.

Technical References:

GLP-OPS-C111A, CRD Lesson Plan.

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-C111A Obj 9

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(7)	
Level of Difficulty:	3.0	

PRA Applicability:

None.

Examination Outline Cross Reference	Level	RO
202001 Recirculation System K1. Knowledge of the physical connections and/or cause effect relationships between RECIRCULATION SYSTEM and the following: K1.15 Nuclear boiler instrumentation (reactor water level/pressure)	Tier	2
	Group #	2
	K/A	202001 K1.15
	Rating	3.2
	Revision	1
Revision Statement: **Rev 1, NRC Review comments** **Changed "are" to "should be" in stem**		

Question: 55

The plant is operating at rated conditions.

A RPV level control problem causes reactor water level to lower.

A Reactor Scram occurs on low reactor water level.

Reactor water level reached -32 inches and was recovered to normal level.

Which of the following describes which Reactor Recirculation Pump breakers should be CLOSED?

A. CB-3A/B

B. CB-1A/B and 2A/B.

C. CB-1A/B, 2A/B and 5A/B.

D. CB-1A/B, 2A/B, 3A/B and 4A/B.

Answer:	**D**
Explanation: The given conditions indicates level less than +11.4 inches which will cause CB-5A/B to trip. The scram will cause reactor power to lower below the bypass for EOC/RPT, therefore 3 & 4 breakers will not open. Reactor parameters are in transient but did NOT exceed the ATWS/ARI setpoints of -41.6". If this setpoint	

had been reached then no breakers would be closed

'D' is correct

Distracters:

'A' is wrong – This would be correct if a ATWS/ARI system initiation. ATWS/ARI will trip CB-1,2, 4, and 5 leaving 3 closed.

'B' is wrong – CB-3 and CB-4 will still be closed due to +11.4 inches only trips CB-5

'C' is wrong – CB-5 will trip level reaches +11.4 inches

K/A Match

This question requires the applicant to have the knowledge of the Recirc system trip setpoints from the nuclear boiler system (level and pressure).

Technical References:

GLP-OPS-B3300, Reactor Recirc Lesson Plan.

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-B3300 Obj 27.2, 28.2

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(7)	
Level of Difficulty:	3.0	

PRA Applicability:

None.

Examination Outline Cross Reference	Level	RO
202002 Recirculation Flow Control System K6. Knowledge of the effect that a loss or malfunction of the following will have on the RECIRCULATION FLOW CONTROL SYSTEM : K6.05 Reactor water level	Tier	2
	Group #	2
	K/A	202002 K6.05
	Rating	3.1
	Revision	0
Revision Statement:		

Question: 56

Which of the following describes the RPV water level setpoint required to initiate a Reactor Recirc Flow Control Valve Runback signal?

A. +4 inches

B. +11.4 inches

C. +32.7 inches

D. +40.7 inches.

Answer: C
Explanation: A reactor recirc FCV runback will occur if <2 feedpumps are running and Reactor Recirc pumps are in FAST speed and RPV water level reaches 32.7". Which is also the low level (level 4) alarm signal. 'B' is correct
Distracters: 'A' is wrong – Plausible, this is the level signal used after Feedwater level control initiates a Setpoint Setdown. 'B' is wrong – Plausible, this is the level that will cause a Recirc pump downshift to slow speed due to being a cavitation interlock. 'D' is wrong – Plausible, this is the HIGH level alarm setpoint.
K/A Match This question requires the applicant to have the knowledge of a loss of Reactor Water level will have on

the Recirc system FCV runback.		
Technical References: GLP-OPS-B3300, Reactor Recirc Lesson Plan.		
Handouts to be provided to the Applicants during exam: NONE		
Learning Objective: GLP-OPS-B3300 Obj 24.3		
Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.41(b)(7)	
Level of Difficulty:	2.0	
PRA Applicability:		
None.		

Examination Outline Cross Reference	Level	RO
201005 Rod Control and Information System (RCIS) K3. Knowledge of the effect that a loss or malfunction of the ROD CONTROL AND INFORMATION SYSTEM (RCIS) will have on following: K3.01 Control rod drive system: BWR-6	Tier	2
	Group #	2
	K/A	201005 K3.01
	Rating	3.3
	Revision	3
Revision Statement: **Rev 1 – Added in Mode 2 to clarify that Pattern controller is in effect.** **Rev 2 – Added Control rod movement is controlled by the RPC to ensure clarity.** **Rev 3, NRC Review and comments** **Removed period in answer 'D'**		

Question: 57

A Reactor Startup is in progress in Mode 2.

Control Rod movement is being controlled by the RPC.

As a control rod is being moved to position 20, a reed switch fails and a DATA FAULT is received.

Which of the following describes the ability to drive control rods?

A. INSERT block only

B. WITHDRAW block only

C. INSERT AND WITHDRAW block

D. INSERT AND WITHDRAW block for only selected control rod

Answer: C
Explanation: A DATA FAULT will cause an Insert and Withdraw block due to RP&IS doesn't know the position of the control rod and must be corrected prior to continued movement to ensure Rod Pattern Control. 'C' is correct
Distracters:

‘A’ is wrong – Plausible, this will occur but along with a Withdraw block

‘B’ is wrong – Plausible, this will occur but along with a Insert block.

‘D’ is wrong – Plausible, this will occur but for all control rod motion not just one rod.

K/A Match

This question requires the applicant to have the knowledge of a failure of the RCIS system will have on Control Rod Drive system.

Technical References:

GLP-OPS-C1102, RC&IS Lesson Plan.

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-C1102 Obj 13.2, 13.3

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(6)	
Level of Difficulty:	3.0	

PRA Applicability:

None.

<table>
<tr><td>Examination Outline Cross Reference</td><td>Level</td><td>RO</td></tr>
<tr><td rowspan="5">216000 Nuclear Boiler Instrumentation

AI. Ability to predict and/or monitor changes in parameters associated with operating the NUCLEAR BOILER INSTRUMENTATION controls including:

A1.02 Removing or returning a sensor (transmitter) to service</td><td>Tier</td><td>2</td></tr>
<tr><td>Group #</td><td>2</td></tr>
<tr><td>K/A</td><td>216000 A1.02</td></tr>
<tr><td>Rating</td><td>2.9</td></tr>
<tr><td>Revision</td><td>2</td></tr>
<tr><td colspan="3">Revision Statement:

Rev 1 – Validators did not remember the operation of a DP cell, Changed answer ‘C’ for clarification.

Rev 2, NRC Review and comments

Changed “will to “should” in stem,

Added “on the indicator” at the end of the stem

Deleted “of the indicator” in answer ‘D’

Changed answer ‘C’ to “unchanged”</td></tr>
</table>

Question: 58

I&C are working on Div 1 RPV Wide Range level transmitter and inadvertently opens the equalizing valve across the transmitter.

RPV level should indicate __________ on the indicator.

A. full upscale

B. full downscale

C. unchanged

D. mid-range

Answer: A
Explanation: With a level instrument in operation, opening the instrument equalizing valve will cause the instrument to fail full scale as the differential pressure across the transmitter goes to zero.

‘A’ is correct.

Distracters:

‘B’ is wrong – Plausible, this will occur if d/p goes high on the transmitter, such as failure of the variable leg of a transmitter.

‘C’ is wrong – with any change in d/p a change in indication will occur.

‘D’ is wrong – Plausible, if the student thinks that equalizing the pressure on the transmitter will cause an mid-scale reading.

K/A Match

This question requires the applicant to have the knowledge of a failure of a level transmitter when removing from service.

Technical References:

GLP-OPS-B2101, RC&IS Lesson Plan.

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-B2101 Obj 12.2, 12.3

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(7)	
Level of Difficulty:	2.0	

PRA Applicability:

None.

Examination Outline Cross Reference	Level	RO
226001 RHR/LPCI: Containment Spray System Mode K2. Knowledge of electrical power supplies to the following: K2.02 Pumps	Tier	2
	Group #	2
	K/A	226001 K2.02
	Rating	2.9
	Revision	0
Revision Statement:		

Question: 59

A tornado has caused a total loss of offsite power.

Division 1 Diesel Generator has failed to start.

Which of the following indicates the systems available for Containment Spray?

A. RHR ‘A’ only

B. RHR ‘B’ only

C. RHR ‘B’ and ‘C’

D. RHR ‘A’ and ‘B’.

Answer: **B**
Explanation: The only systems that can be used for Containment spray is RHR ‘A’ and ‘B’. With a loss of offsite power and Div 1 Diesel Gen. bus 15AA is not available. Therefore RHR ‘A’ is not available due to 15AA powers RHR ‘A’ and 16AB powers RHR ‘B’. Bus 16AB should be powered by the Div 2 D/G and available for Containment Spray. ‘B’ is correct
Distracters: ‘A’ is wrong –the student must be able to determine with the given losses that RHR ‘A’ has no power. ‘C’ is wrong – RHR ‘C’ does not have a Containment Spray mode.

‘D’ is wrong – RHR ‘A’ has no power		
K/A Match This question requires the applicant to have the knowledge of power supplies to the RHR pumps with a Containment spray mode.		
Technical References: GLP-OPS-E1200, RHR Lesson Plan.		
Handouts to be provided to the Applicants during exam: NONE		
Learning Objective: GLP-OPS-E1200 Obj 7.1		
Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(7)	
Level of Difficulty:	2.0	
PRA Applicability:		
None.		

Examination Outline Cross Reference	Level	RO
245000 Main Turbine Generator and Auxiliary Systems A4. Ability to manually operate and/or monitor in the control room: A4.01 Turbine Lube Oil Pumps	Tier	2
	Group #	2
	K/A	245000 A4.01
	Rating	2.7
	Revision	0
Revision Statement: **NRC Review and comments (Original U)** **NEW QUESTION**		

Question: 60

During a Main Turbine trip and coast down;

Which of the following indicates the lube oil pressure where the Auxiliary Lube Oil Pumps should auto start?

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

B. All pumps start at 95 psig

C. All pumps start at 90 psig

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

Answer: D
Explanation: A and B start at 95 psig. C starts at 90 psig. ‘D’ is correct
Distracters: All distracters are different uses of the auto start number
K/A Match This question requires the applicant to have the ability to monitor the Main Turbine Lube Oil Header and recall the auto start signals for the pumps

Technical References: GLP-OPS-N3402, Lesson Plan.		
Handouts to be provided to the Applicants during exam: NONE		
Learning Objective: GLP-OPS-N3402 Obj 6		
Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.41(b)(5)	
Level of Difficulty:	3.0	
PRA Applicability:		
None.		

Examination Outline Cross Reference	Level	RO
256000 Reactor Condensate System K4. Knowledge of REACTOR CONDENSATE SYSTEM design feature(s) and/or interlocks which provide for the following: K4.08 Dedicated ECCS water supply: Plant-Specific	Tier	2
	Group #	2
	K/A	256000 K4.08
	Rating	3.6
	Revision	1

Revision Statement:

Rev 1, NRC Review and comments

Added Hotwell Makeup / Reject level control system to answers B and D and yes the CST has a level control system using the Hotwell Makeup / Reject Header, corrected question description.

Question: 61

How much water is reserved in the CST for HPCS and RCIC operation?

What design feature ensures this amount is maintained?

A. 50,000 gallons
Hotwell Makeup / Reject level control system

B. 50,000 gallons
Internal CST standpipe

C. 170,000 gallons
Hotwell Makeup / Reject level control system

D. 170,000 gallons
Internal CST standpipe

Answer: D

Explanation:

Of the total CST capacity, approximately 170,000 gallons is reserved for use by HPCS and the Reactor Core Cooling System because of physical piping design and layout. A standpipe is used inside the CST for other systems that use the CST as a source.

'D' is correct

Distracters:

'A' is wrong – Plausible, 50,000 is the amount of water left in the CST when the Low CST level HPCS/RCIC suction swap occurs. The CST level control system is controlled by the Makeup Reject

header controller.

‘B’ is wrong – Plausible, 50,000 is the amount of water left in the CST when the Low CST level HPCS/RCIC suction swap occurs..

‘C’ is wrong – The CST level control system is controlled by the Makeup Reject header controller..

K/A Match

This question requires the applicant to have the knowledge of the dedicated water in the CST for HPCS and RCIC.

Technical References:

GLP-OPS-E2201, HPCS Lesson Plan.
GLP-OPS-P1100, Condensate and Refueling Water Storage and Transfer Lesson Plan

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-E2201 Obj 5

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.41(b)(7)	
Level of Difficulty:	2.0	

PRA Applicability:

HPCS and RCIC are #16 and #17 on the System Importance to CDF.

Examination Outline Cross Reference	Level	RO
259001 Reactor Feedwater System K5. Knowledge of the operational implications of the following concepts as they apply to REACTOR FEEDWATER SYSTEM : K5.03 Turbine operation: TDRFP's-Only	Tier	2
	Group #	2
	K/A	259001 K5.03
	Rating	2.8
	Revision	1
Revision Statement: **Rev 1, NRC Review and comments** **Added less than to answer A**		

Question: 62

The plant is operating at rated thermal power when the Main Turbine trips.

The operator at the controls aligns for startup level control using Reactor Feed Pump 'A'.

10 minutes after the Scram the transient is complete; all plant parameters are stable again.

Which of the following describes the Governor Control Valve position for RFPT 'A'?

A. Less than one-half the original indication due to the Scram.

B. Slightly lower than original indication due to the Scram.

C. Slightly higher than original indication due to more load on the feed pump turbine.

D. More than double the original indication due to a swap to its high pressure steam source.

Answer: **D**
Explanation: Rated position for the governor valve is 20% after a scram. The governor will slowly move to high pressure steam and governor valve will be 60%. 'D' is correct

Distracters:

'A' is wrong – Plausible, just the opposite from the actual .

'B' is wrong – Plausible, Due to the swap to high pressure steam that actually have less energy due to not being superheated the gov will have to be farther open.

'C' is wrong – Due to the swap to high pressure steam that actually have less energy due to not being superheated the gov will have to be farther open by a large amount not slightly.

K/A Match

This question requires the applicant to have the knowledge of the implications of the effects on Reactor feed pump turbines when the steam supply swaps after a plant shutdown.

Technical References:

GLP-OPS-N2100, Feedwater Lesson Plan.
04-1-01-N21-1, Feedwater SOI

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-N2100 Obj 17, 18

Question Source:	**Bank # 761**	2013 NRC Exam Q62
(note changes and attach parent)	**Modified Bank #**	
	New	
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(4)	
Level of Difficulty:	2.0	

PRA Applicability:

None.

Simulator actions.

Examination Outline Cross Reference	Level	RO
272000 Radiation Monitoring System A3. Ability to monitor automatic operations of the RADIATION MONITORING SYSTEM including: A3.06 Ventilation system isolation indications	Tier	2
	Group #	2
	K/A	272000 A3.06
	Rating	3.4
	Revision	0
Revision Statement:		

Question: 63

At time = 0 minutes: All 4 Control Room Vent Radiation Monitors are indicating 6 mR/hr.

At time = 5 minutes: A LOCA occurs.

At time = 10 minutes: Reactor water level reaches a low of -51” wide range before HPCS and RCIC begin to restore it.

Which of the following describes operation of the Fresh Air Inlet valves, Z51-F007 and F016, for the Control Room Standby Fresh Air Ventilation System?

Auto-isolated at time:

A. 10 minutes; can be manually re-opened at time = 20 min.

B. 0 minutes; can be manually re-opened at time = 10 minutes.

C. 0 minutes; are interlocked closed until all signals are manually reset.

D. 10 minutes; are interlocked closed until all signals are manually reset.

Answer: **B**
Explanation: On the receipt of a control room isolation signal, -41.6” Reactor water level or 1.23 psig Drywell pressure or >5mr on Control Room rad monitor, the SBFA unit will auto start in Recirc mode. The fresh air inlet valves F007 and F016 are interlocked closed for 10 min after isolation signal. A subsequent LOCA signal has no effect on the 10 min time delay. ‘B’ is correct
Distracters:

‘A’ is wrong. The isolation occurred at T-0

‘C’ is wrong. The fresh air valves can be opened 10 min after isolation signal.

‘D’ is wrong. The isolation occurred at T-0

K/A Match

This question requires the applicant to have the ability to monitor ventilation isolations from radiation signals.

Technical References:

GLP-OPS-Z5100, Control Room HVAC Lesson Plan.

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-Z5100 Obj 9, 11

Question Source:	**Bank # 251**	Not used on NRC Exam
(note changes and attach parent)	**Modified Bank #**	
	New	
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(7)	
Level of Difficulty:	3.0	

PRA Applicability:

None.

Examination Outline Cross Reference	Level	RO
290002 Reactor Vessel Internals A2. Ability to (a) predict the impacts of the following on the REACTOR VESSEL INTERNALS ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: A2.03 †Control rod drop accident	Tier	2
	Group #	2
	K/A	290002 A2.03
	Rating	3.6
	Revision	0
Revision Statement:		

Question: 64

A Control Rod Drop can occur when a control rod becomes stuck and uncoupled for the CRD mechanism.

(1) When would a Control Rod Drop accident be more of an impact on the Reactor?

(2) What action should be performed by the crew to recouple the Control Rod?

A. (1) 5% power, Startup
(2) Individual control rod scram.

B. (1) 100% Rod Line, end of core life
(2) Normal insertion

C. (1) 100% Rod Line, end of core life
(2) Individual control rod scram

D. (1) 5% power, Startup
(2) Normal insertion

Answer:	D
Explanation: The given indications are a control rod that has become uncoupled from the CRD mech. This could be the setup for a control rod drop accident. Per Tech Specs Bases 3.3.2.1, Compliance with the BPWS, and therefore OPERABILITY of the RPC, is required in MODES 1 and 2 with THERMAL POWER s 10% RTP. When THERMAL POWER is > 18% RTP, there is no possible control rod configuration that results in a control rod worth that could exceed the 280 cal/gm fuel damage limit during a CRDA (Ref. 3). In	

MODES 3 and 4, all control rods are required to be inserted in the core. In MODE 5, since only a single control rod can be withdrawn from a core cell containing fuel assemblies, adequate SDM ensures that the consequences of a CRDA are acceptable, since the reactor will be subcritical. 'D' is correct
Distracters: 'A' is wrong. Per CRD SOI precaution and limitation, 3.13 Scramming control rods <u>Should</u> only be done as part of testing **<u>OR</u>** in emergency situations. 'B' is wrong. When THERMAL POWER is > 18% RTP, there is no possible control rod configuration that results in a control rod worth that could exceed the 280 cal/gm fuel damage limit during a CRD drop. 'C' is wrong. Per CRD SOI precaution and limitation, 3.13 Scramming control rods <u>Should</u> only be done as part of testing **<u>OR</u>** in emergency situations. When THERMAL POWER is > 18% RTP, there is no possible control rod configuration that results in a control rod worth that could exceed the 280 cal/gm fuel damage limit during a CRD drop.
K/A Match This question requires the applicant to have the ability to predict the impact on a Control rod drop accident and have knowledge of the procedure to mitigate.
Technical References: GLP-OPS-C111B, Control Rod Mech Lesson Plan. Tech Specs Bases 3.3.2.1
Handouts to be provided to the Applicants during exam: NONE
Learning Objective: GLP-OPS-C111B Obj 9, 11

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(7)	
Level of Difficulty:	3.0	
PRA Applicability:		

None.

Examination Outline Cross Reference	Level	RO
219000 RHR/LPCI: Torus/Suppression Pool Cooling Mode 2.4.49 Ability to perform without reference to procedures those actions that require immediate operation of system components and controls.	Tier	2
	Group #	2
	K/A	219000 2.4.49
	Rating	4.6
	Revision	0
Revision Statement: **Rev 0, NRC Review and comments (U)** **NEW QUESTION**		

Question: 65

An ATWS has occurred.

Reactor Power is 11%.

All MSIVs have closed

(1) Suppression pool cooling immediate action must be performed within a MAXIMUM of ___________ minutes.

(2) Suppression pool cooling is considered to be MAXIMIZED when RHR pump is running and _______________.

A. (1) 4 minutes
   (2) E12-F024A and B, RHR A/B TEST RTN TO SUPP POOL only is OPEN

B. (1) 4 minutes
   (2) E12-F024A and B, RHR A/B TEST RTN TO SUPP POOL is OPEN and E12-F048A and B, RHR HX A/B BYP VLV is CLOSED

C. (1) 11 minutes
   (2) E12-F024A and B, RHR A/B TEST RTN TO SUPP POOL only is OPEN

D. (1) 11 minutes
   (2) E12-F024A and B, RHR A/B TEST RTN TO SUPP POOL is OPEN and E12-F048A and B, RHR HX A/B BYP VLV is CLOSED

Answer:	**D**
Explanation:	

Per procedure the timed critical action for suppression pool cooling during an ATWS is 11 minutes and for suppression pool cooling to be considered to be in service the pump must be running with the F024A/B must be full OPEN and the F048 full CLOSED.
'D' is correct

Distracters:

'A' is wrong. plausible because 4 minutes is the time delay for the F048 to remain closed when closed time delay for the valve. The F024 being open will provide some cooling but not full cooling until the F048 bypass valve is fully closed.

'B' is wrong. plausible because 4 minutes is the time delay for the F048 to remain closed when closed time delay for the valve

'C' is wrong. The F024 being open will provide some cooling but not full cooling until the F048 bypass valve is fully closed

K/A Match

This question requires the applicant to know immediate operator actions for an ATWS >5% and the timed critical actions .

Technical References:

GLP-OPS-EP2A.
02-S-01-43, Transient Mitigation Strategy, Rev 7
GGNS-NE-16-00004_001

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-EP3 & EP4, OBJ 3

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(7)	
Level of Difficulty:	3.0	

PRA Applicability:

None.

Examination Outline Cross-Reference	Level	RO
2.1.36 – Knowledge of procedures and limitations involved in core alterations.	Tier #	3
	Group #	
	K/A #	**G2.1.36**
	Rating	3.0

Revision Statement:

Rev 0, NRC Review and comments (U)

NEW QUESTION

5-2017 NRC Exam

Question: 66

Which of the following describes when continuous communication must be maintained between the Control Room and Refueling Platform?

A. Moving SRM detector from full out position to full in position.

B. Removing Control Rod 32-33 from an empty cell.

C. Moving a Control Rod to perform One-Rod-Out interlock checks.

D. Removing the CRD Mechanism from Control Rod 48-33.

Answer: C
Explanation: Per Tech Spec Definition, Movement of any fuel, sources, or reactivity control components within the reactor vessel with the head removed. A is wrong. Per Tech Spec Definition movement of source range monitors are not considered to be core alterations. B is wrong. Per Tech Spec Definition, Control Rod movement provided there are no fuel assemblies in the associated core cell. C is correct D is wrong. Per 03-1-01-5, IOI, 2.15, Continuous communication is required with the Control room and Refuel Floor when changing CRD mechanisms.
Technical References:

Tech Specs, Definitions, Core Alteration 03-1-01-5 Rev 137, 2.6		
References to be provided to applicants during exam: NONE		
Learning Objective: Document learning objective if possible. GLP-OPS-IOI05, Objective 2.3		
Question Source:	Bank # 252	X
(note changes; attach parent)	Modified Bank	
	New	
Question History:	Last NRC Exam	5-2017 NRC (Q67)
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
	LOD	2
10CFR Part 55 Content:	55.43(b)(5)	

Examination Outline Cross Reference	Level	RO
2.1.19 Ability to use plant computers to evaluate system or component status.	Tier	3
	Group #	0
	K/A	2.1.19
	Rating	3.9
	Revision	0
Revision Statement:		

Question: 67

Which of the following computer systems can be used to determine current system / component status and trend parameters?

A. PDS

B. EOOS

C. CYCLOPS

D. INFORM

Answer: **A**

Explanation:

PDS, Plant Display System, is used for up to date information and current information on system and components.

'A' is correct

Distracters:

'B' is wrong. EOOS is Equipment Out Of Service, even though it gives information on equipment that is Out of service, the information is manually input and does not supply up to date information.

'C' is wrong. CYCLOPS is used by the STA/RE to determine core information, not system status.

'D' is wrong. INFORM is used to inform the Emergency Response Organization for plant emergency status, no system or component status.

K/A Match

This question requires the applicant to have the knowledge of and use of computer systems that provide Component / System status.

Technical References: 04-1-01-C91-1, Plant Display System (PDS), SOI, Rev. 21
Handouts to be provided to the Applicants during exam: NONE
Learning Objective: GSMS-RO-IN002, Obj. 8

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.41(b)(10)	
Level of Difficulty:	2.0	
PRA Applicability:		
None.		

Examination Outline Cross Reference	Level	RO
2.2.40 Ability to apply Technical Specifications for a system.	Tier	3
	Group #	0
	K/A	2.2.40
	Rating	3.4
	Revision	0
Revision Statement:		

Question: 68

Which of the following requires IMMEDIATE entry into TS 3.4.1 Recirculation Loops Operating?

A. TOT JP FLO 1B33-UR-R613 is 64 MLBM/HR
RECIRC PMP A DRIVING FLOW 1C51-FR-R614 is 24.9 KGPM
RECIRC PMP B DRIVING FLOW 1C51-FR-R614 is 28.9 KGPM

B. TOT JP FLO 1B33-UR-R613 is 74.8 MLBM/HR
RECIRC PMP A DRIVING FLOW 1C51-FR-R614 is 26.4 KGPM
RECIRC PMP B DRIVING FLOW 1C51-FR-R614 is 30.5 KGPM

C. TOT JP FLO 1B33-UR-R613 is 80.5 MLBM/HR
RECIRC PMP A DRIVING FLOW 1C51-FR-R614 is 29.4 KGPM
RECIRC PMP B DRIVING FLOW 1C51-FR-R614 is 32.8 KGPM

D. TOT JP FLO 1B33-UR-R613 is 104 MLBM/HR
RECIRC PMP A DRIVING FLOW 1C51-FR-R614 is 42.5 KGPM
RECIRC PMP B DRIVING FLOW 1C51-FR-R614 is 40.8 KGPM

Answer: C

Explanation:

Tech Specs 3.4.1 states "Verify recirculation loop jet pump flow mismatch with both recirculation pumps in operation is:

< = 10% of rated core flow when operating at <70% core flow
< = 5% of rated core flow when operating at > = 70% core flow

70% core flow = 112.5 * 0.7 = 78.75 mlbm/hr - 5% = 2.2 KGPM, 10% = 4.4 KGPM

'C' is >70% core flow at 80.5 Mlbm/hr and mismatch is 3.4 Kgpm which is >2.2 Kgpm.

'C' is correct

Distracters:

'A' is wrong. The indication is <70% core flow (78.75 Mlbm/hr) so flows must be within 4400 gpm. Actual mismatch is 4000gpm.

'B' is wrong. The indication is <70% core flow (78.75 Mlbm/hr) so flows must be within 4400 gpm. Actual mismatch is 4100gpm..

'D' is wrong. The indication is >70% core flow (78.75 Mlbm/hr) so flows must be within 2200 gpm. Actual mismatch is 1700gpm..

K/A Match

This question requires the applicant to have the knowledge of entry into Tech Specs.

Technical References:

Tech Specs 3.4.1

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-B3300 Objective: 44

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

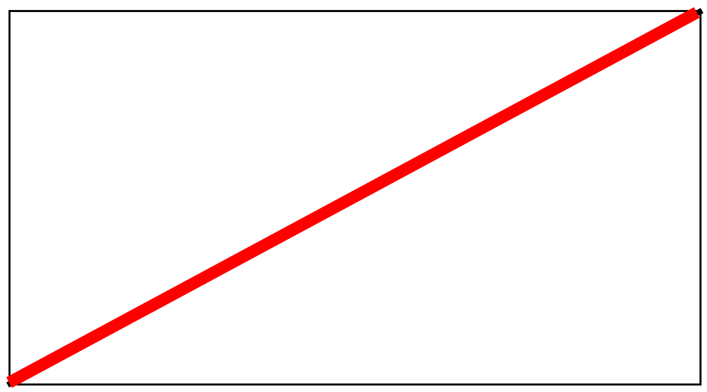
PRA Applicability:

None.

Examination Outline Cross Reference	Level	RO
2.2.43 Knowledge of the process used to track inoperable alarms.	Tier	3
	Group #	0
	K/A	2.2.43
	Rating	3.0
	Revision	0
Revision Statement:		

Question: 69

A Control Room alarm window has a single red tape diagonally across the window.



Which of the following describes the status of the control room annunciator represented in the figure?

A. OPERABLE ANNUNCIATOR – Multiple inputs, one input is bypassed

B. PROBLEM ANNUNCIATOR – Faulty Alarm Circuitry, alarm card removed

C. PROBLEM ANNUNCIATOR – Faulty Alarm Circuitry, alarm card still installed

D. OPERABLE ANNUNCIATOR – Multiple inputs, more than one input is bypassed

Answer: C

Explanation:

With the alarm card still installed, only a single strip of tape (a diagonal) is used (see section 6.2.1.d). (as indicated the above figure). **IDENTIFY** the alarm window that is **NOT** functioning properly by **PLACING** a length of red tape diagonally across the window, except for alarms bypassed per section 6.3, as shown on Attachment IV.

‘C’ is correct

Distracters:

‘B’ is incorrect because Per 02-S-01-25, section 6.2 and Attachment IV. Specifically, section 6.2.1.g…when the Problem Annunciator has its alarm card removed, a second strip of red tape is placed on the window, to form an X

‘A’ & ‘D’ are incorrect because for an OPERABLE ANNUNCIATOR, with any number of multiple inputs bypassed, two vertical strips of tape are used (see section 6.3.5.a and Attachment IV).

K/A Match

This question requires the applicant to have the knowledge of indications of INOP alarms.

Technical References:

02-S-01-25

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-PROC

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.41(b)(10)	
Level of Difficulty:	2.0	

PRA Applicability:

None.

Examination Outline Cross Reference	Level	RO
2.2.44 Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions.	Tier	3
	Group #	0
	K/A	2.2.44
	Rating	4.2
	Revision	0
Revision Statement:		

Question: 70

The following alarm was received.

Per 04-1-01-C82-1, Annunciator System:

(1) Which of the following describes the meaning of the color?

(2) What actions should be taken?

A. (1) Conditions which require Immediate action.
(2) Refer to ARI to determine which input is in alarm due to multiple inputs

B. (1) Conditions which require Immediate action.
(2) Verify automatic actions have occurred

C. (1) Conditions which require corrective action in a timely manner.
(2) Verify automatic actions have occurred

D. (1) Conditions which require corrective action in a timely manner.
(2) Refer to ARI to determine which input is in alarm due to multiple inputs

Answer:	B
Explanation:	

Per 04-1-01-C82-1, Annunciator System , section 4.1

Amber alarms (warning alarms) apply to conditions which require Immediate action

Alarms which have the lower right hand corner blacked out are associated with an automatic isolated condition. **REFER** to the" Automatic Isolated" ONEP 5-1-02-III-5 for initiating conditions and affected components

'B' is correct

Distracters:

'A' is wrong - Per 04-1-01-C82-1, Annunciator System Alarms which have a black square box in the lower right hand corner of the annunciator window have reflash capability and reannunciate upon other conditions which initiate the alarm.

'C' is wrong - Per 04-1-01-C82-1, Annunciator System White alarms (caution alarms) apply to conditions which require corrective action in a timely manner, this is an Amber alarm

'D' is wrong - Per 04-1-01-C82-1, Annunciator System White alarms (caution alarms) apply to conditions which require corrective action in a timely manner, this is an Amber alarm and Annunciator System Alarms which have a black square box in the lower right hand corner of the annunciator window have reflash capability and reannunciate upon other conditions which initiate the alarm.

K/A Match

This question requires the applicant to have the ability to interpret control room alarm colors and designations to determine operator actions and/or procedure entry.

Technical References:

04-1-01-C82-1, Annunciator System SOI, Rev 29

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-PROC

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.41(b)(10)	
Level of Difficulty:	2.0	

PRA Applicability:
None.

Examination Outline Cross Reference	Level	RO
2.3.11 Ability to control radiation releases.	Tier	3
	Group #	0
	K/A	2.3.11
	Rating	3.8
	Revision	1

Revision Statement:

Rev 1, NRC Review and comments

Made stem examples more plant specific.

Moved "during normal plant operations" to the beginning after "processes."

Add "effluents" at the end of 2.

Question: 71

Consider the following processes during normal plant operations:

1. Radwaste Control Room discharging a tank through G17-F355
2. Circulating Water System Blowdown
3. RP Discharge of solid radioactive waste container
4. Fuel Handling Area Exhaust

Which of the above processes are controlled using a Discharge Permit per 01-S-08-11, Radioactive Discharge Controls?

A. 1, only

B. 4, only

C. 1 and 2, only

D. 2 and 3, only

Answer: A

Explanation:

‘A’ is correct

01-S-08-11, Radioactive Discharge Controls, Rev 116

Section 5.4.3 - Batch Liquid Discharge Permit similar to Attachment 1, BATCH LIQUID RADWASTE DISCHARGE PERMIT should be completed for all batch liquid releases.

Distracters:

‘B’ & ‘C’ are wrong because, per Section 5.3, continuous discharges do not require a permit.

‘D’ is wrong per Section 6.2 which states that solid radwaste must not be discharged but rather handled Reference 3.5 (EN-RP-11, Radioactive Material Control).

K/A Match

This question requires the applicant to have the ability to interpret the different types of radioactive releases and ability to control the process.

Technical References:

01-S-08-11, Radioactive Discharge Controls, Rev 116

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-PROC Obj.51

Question Source:	**Bank # 195**	2008 NRC Exam
(note changes and attach parent)	**Modified Bank #**	
	New	
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.41(b)(13)	
Level of Difficulty:	2.0	

PRA Applicability:

None.

Examination Outline Cross Reference	Level	RO
2.3.14 Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities.	**Tier**	3
	Group #	0
	K/A	2.3.14
	Rating	3.4
	Revision	1

Revision Statement:

Rev 1, NRC Review and comments

Retained original question from 10 pre-view with following changes:

Revised stem: An event has occurred that requires entry into E3, "Containment Control," due to rising containment pressure. What is the MINIMUM containment pressure at which venting is procedurally required?

Change "will arise" to "should occur."

Delete "Before reaching" from all four distractors.

Question: 72

An event has occurred that requires entry into EP-3, Containment Control, due to rising containment pressure.

What is the MINIMUM containment pressure at which venting is procedurally required?

What type of radiation/contamination hazard should occur?

A. 15 psig
Filtered release

B. 15 psig
Un-Filtered release

C. 22.6 psig
Filtered release

D. 22.6 psig
Un-Filtered release

Answer: D
Explanation: Per EP-3 If CTMT pressure cannot be maintained below 22.6 psig, THEN Vent the CTMT to control

pressure below 22.6 psig……Att. 13

05-S-01-EP-1, Attachment 13 (page 3 of 16), step 2.9, where direction is given to notify the Emergency Director that an Un-filtered release will occur, not the Radiological Assessment Coordinator and not filtered.

'D' is correct

Distracters:

'A' is wrong - because, 15 psig is the design pressure for the primary containment. Filtered release is a plausible distracter because the smaller 6" containment vent path (Low Volume Purge) for EP Att. 14 vents through the Ctmt exhaust charcoal filter train.

'B' is wrong - 15 psig is the design pressure for the primary containment

'C' is wrong - Filtered release is a plausible distracter because the smaller 6" containment vent path (Low Volume Purge) for EP Att. 14 vents through the Ctmt exhaust charcoal filter train.

K/A Match

This question requires the applicant to have the knowledge of possible radiation hazards during an emergency and results from actions.

Technical References:

05-S-01-EP-1, Att. 1, Rev. 38
02-S-01-40, EP Technical Bases, Rev. 9

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-EPTS26 Obj. 40

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.41(b)(13)	
Level of Difficulty:	3.0	

PRA Applicability:
None.

Examination Outline Cross Reference	Level	RO
2.4.2 Knowledge of system set points, interlocks and automatic actions associated with EOP entry conditions.	Tier	3
	Group #	0
	K/A	2.4.2
	Rating	4.5
	Revision	0
Revision Statement:		

Question: 73

Which of the following coincides with an EOP entry condition?

A. DRWL PRESS HI/LO alarm on P870

B. Any Suppression pool temp high alarms on P870

C. HPCS system automatic suction swap on Suppression Pool Level

D. Initiation of Reactor Recirculation Pump cavitation interlock on Reactor water level.

Answer: **D**

Explanation:

Per EP-2 entry conditions a RPV water level of 11.4"

The Reactor Recirculation Pump Cavitation interlock is 11.4"

'D' is correct

Distracters:

'A' is wrong – P870-9A-D4, DRWL PRESS HI/LO, states the high pressure setpoint 1 psig. Entry condition for EP-2 or EP-3 is 1.23 psig drywell pressure

'B' is wrong – Two high temp alarms for suppression pool temp setpoint is 85.5°F, while the other two is set at 115°F. 85.5 is less than the entry condition for EP-3 of 95°F

'C' is wrong – P601-16A-C5, SUPP POOL LVL HI, corresponds with the HPCS suction swap, states Suppession pool level of 18'9" or 18.75 feet. EP-3 entry condition for High Suppression pool level is 18.81 feet.

K/A Match

This question requires the applicant to have the knowledge of possible radiation hazards during an

emergency and results from actions.

Technical References:

GLP-OPS-B3300, Reactor Recirc System lesson plan, Rev. 30
02-S-01-40, EP Technical Bases, Rev. 9
04-1-02-1H13-P870-9A-D4, Rev 130
04-1-02-1H13-P870-9A-E4, Rev 105

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-EPTS26 Obj. 40

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.41(b)(10)	
Level of Difficulty:	2.0	

PRA Applicability:

None.

Examination Outline Cross Reference	Level	RO
2.4.29 Knowledge of the emergency plan.	**Tier**	3
	Group #	0
	K/A	2.4.29
	Rating	3.1
	Revision	1

Revision Statement:

Rev 1 Per NRC review (10 free view) changed wording in stem of parameters, HPCS system is injecting. Changed the answers to be clear and deleted reference to HPCS.

Per Facility Rep, added "low pressure " to stem before ECCS systems.

NRC Review and comments (U)

Replaced K/A and Question

Question: 74

A Site Area Emergency is declared.

The initial notification is to be completed within _________ minutes of declaration and the follow-up is to be completed within _________ minutes of the initial.

A. (1) 15
   (2) 30

B. (1) 15
   (2) 60

C. (1) 30
   (2) 30

D. (1) 30
   (2) 60

Answer: **B**

Explanation:

Per 10-S-01-6, section 6.1.1.a "Shift Manager / Emergency Director shall ensure that notification to the following agencies is initiated within 15 minutes of an emergency declaration or PAR development.

6.1.7.b For all declarations a follow-up Notification is performed one hour after the initial notification.

'B' is correct

Distracters:

'A' is wrong – Follow up notification is 1 hour not 30 minutes.

'C' is wrong – Initial is 15 minutes from declaration not 30 and follow up is 15 from initial not 30

'D' is wrong – Initial is 15 minutes from declaration not 30.

30 minutes is a plausible distractor for both answers due to declaration is required from 15 minutes of discovery of occurrence and then notification is required with 15 minutes of declaration with equals 30 minutes from occurrence to notification.

K/A Match

This question requires the applicant to have the RO knowledge of the emergency plan

Technical References:

10-S-01-6, Notification of Offsite Agencies and Plant ON-CALL Emergency Personnel

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-EPTS26 Obj. 15

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

PRA Applicability:

None.

Examination Outline Cross Reference	Level	RO
2.4.45 Ability to prioritize and interpret the significance of each annunciator or alarm.	Tier	3
	Group #	0
	K/A	2.4.45
	Rating	4.1
	Revision	0
Revision Statement:		

Question: 75

A Reactor Scram has occurred on MSIV closure.

The CRS has entered EP-2 and implemented transient alarm response.

Which of the following valid alarms should be immediately reported to the CRS?

A. ADS/SRV LEAK, P601-18A-G2

B. DRWL PRESS HI, P601-21A-E7

C. RECIRC PUMP A AUTO TRIP XFER TO LO SP, P680-3A-D4

D. SCRAM PILOT VLV AIR HDR PRESS LO, P680-5A-C4.

Answer: B

Explanation:

Per EN-OP-115-08, Annunciator Response, section 9, The announcement of transient alarms during Abnormal/ONEP and EOP is not required. In such cases, the operators are expected to announce those alarms that are of significance to the implementation of the applicable Abnormal/ONEP and EOP.

The Drywell Pressure Hi alarm should be immediately reported to the CRS. He is required to enter EP-3 and Re-enter EP-2. Also this alarm could be indicative of a LOCA in the drywell.

'B' is correct

Distracters:

'A' is wrong – This alarm will be in due to pressure control on the SRVs from the MSIV closure.

'C' is wrong – This alarms indicates an auto transfer to slow speed and is an action that should have already happened due to turbine trip.

'D' is wrong – This alarm will be in due to a scram signal and will not reset until the scram is reset.

K/A Match

This question requires the applicant to have the ability to recognize, interpret and prioritize control room alarms during a transient.

Technical References:

EN-OP-200, Rev. 5
04-1-02-1H13-P601-18A-G2, Rev. 166
04-1-02-1H13-P601-21A-E7, Rev 110
04-1-02-1H13-P680-5A-C4, Rev. 100
04-1-02-1H13-P680-3A-D4, Rev. 160

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-PROC

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.41(b)(5)	
Level of Difficulty:	3.0	

PRA Applicability:

None.

GGNS ILT 2/2020 **NRC** INITIAL LICENSED OPERATOR WRITTEN EXAMINATION

SRO EXAM

ANSWER KEY

76	A
77	C
78	B
79	B
80	B
81	C
82	C
83	D
84	D
85	C
86	A
87	A
88	D
89	D
90	C
91	B
92	C
93	B
94	D
95	B
96	D
97	B
98	D
99	C
100	A

Examination Outline Cross Reference	Level	SRO
295003 Partial or Complete Loss of A.C. Power AA2. Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER : AA2.02 Reactor power / pressure / and level.	Tier	1
	Group #	1
	K/A	295003 AA2.02
	Rating	4.3
	Revision	3

Revision Statement:

Rev 1 – Validation comment add MSIVs have closed instead of the student determine that they closed

Rev 2 – Validation comment Added No other EP Entry conditions have been exceeded.

Rev 3, NRC Review and comments

Changed stem to state "A grid disturbance causes a loss of offsite power" this will cause all MSIVs to close and a loss of all Condensate / Feedwater.

The addition of No other EP entry conditions is to have them determine that Suppression Pool Temp has not exceeded 110 degrees that would cause a different Level band.

B(2) should now be plausible, the applicant must determine that MSIVs have closed and this is not possible.

Question: 76

The plant is operating at rated conditions.

MSIVs have closed.

Current conditions:

- Rx power is 11%
- Rx pressure is maintaining between 926 and 1073
- Rx water level is +10” and lowering
- No other EP entry conditions have been exceeded

With the given conditions,

(1) What should be the CRS’s direction for RPV level control?

(2) What should be the CRS’s direction for RPV pressure band?

A. (1) Lower level to -70 inches Wide Range

(2) Lower Pressure band 450 to 600 psig

B. (1) Lower level to -70 inches Wide Range
(2) Maintain pressure band 800 to 1060 psig

C. (1) Lower level to -167 inches Fuel Zone Range or <5% power
(2) Lower Pressure band 450 to 600 psig

D. (1) Lower level to -167 inches Fuel Zone Range or <5% power
(2) Maintain pressure band 800 to 1060 psig

Answer: **A**
Explanation: First determine that MSIVs have closed An ATWS occurs due to power at 11% after MSIVs are closed and with water level at 10 inches. SRVs are open due to the pressure being maintained is 926 to 1073 psig which is the opening and closing setpoints for the lo-lo set SRVs. The CRS should enter EP-2 then to EP-2A and lower level to -70” Wide Range. Pressure control should be manually controlled at a band of 450 to 600 psig due to loss of feedwater and lowering band will allow Condensate to feed and control level. Level should be lowered to -70 inches per EP-2A, the lowering to TAF is a Step 8 terminate action and that criteria is not met yet. Step L5 has to have suppression pool temperature >110. ‘A’ is correct.
Distracters: ‘B’ is wrong, the pressure band is incorrect. Pressure must be lowered to ensure the Condensate system is able to feed. ‘C’ is wrong, Level should be lowered to -70 inches per EP-2A, the lowering to TAF is a Step 8 terminate action and that criteria is not met yet. Step L5 has to have suppression pool temperature >110. ‘D’ is wrong, Level should be lowered to -70 inches per EP-2A, the lowering to TAF is a Step 8 terminate action and that criteria is not met yet. Step L5 has to have suppression pool temperature >110. Also, the pressure band is incorrect. Pressure must be lowered to ensure the Condensate system is able to feed.
K/A Match The SRO candidate must be able to interpret the given information with a loss of power and decide on the correct actions to take.
Technical References:

04-1-02-1H13-P601-19A-E4, ARI, Rev. 162
GLP-OPS-B1300, Rev 14, Nuclear Boiler and Vessel Internals Lesson Plan
02-S-01-40, EP Bases, Rev. 9
04-1-01-C71-1, RPS SOI, Rev. 37

Handouts to be provided to the Applicants during exam:

None

Learning Objective:

GLP-OPS-B1300, Rev 14 OBJ. 11.1
GLP-OPS-EP2A

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.43(b)(5)	
Level of Difficulty:	3.0	

PRA Applicability:

R21 ESF is listed as #8 on the System Importance to CDF.
RPS is listed as #5 on the System Importance to CDF.

Examination Outline Cross Reference	Level	SRO
295006 SCRAM AA2. Ability to determine and/or interpret the following as they apply to SCRAM : AA2.03 Reactor water level	Tier	1
	Group #	1
	K/A	295006 AA2.03
	Rating	4.3
	Revision	2
Revision Statement: **Rev 1 - Validation comment, add "RCIC has tripped on overspeed" to ensure SRO knows level is being controlled by a high volume system.** **Rev 2, NRC Review and comments** **Changed answer 'B' to Restarting.**		

Question: 77

A complete loss of Feedwater occurred at rated power.

Both Reactor Feed Pumps are tripped due to a spurious high water level signal that remains sealed in.

RCIC has tripped on overspeed.

The CRS directs level control using HPCS.

Reactor water level is +20 inches Narrow Range.

Which of the following Scram ONEP steps should the CRS direct the RO to **skip**?

A. Inserting IRMs.

B. Restarting RWCU.

C. Resetting RPS Scram logic.

D. Restarting Reactor Recirc Pumps.

Answer:	**C**
Explanation: The level band when HPCS is used for level control is -30" to +50" per 02-S-01-43 and EP-2. Scram	

ONEP 05-1-02-I-1 step 3.7 directs resetting the RPS scram only if level is stable, to ensure subsequent scrams are not received. Since level is expected to go below 11.4” for the current level band, resetting the scram is not appropriate.

‘C’ is correct.

Distracters:

Candidate must assume that RPV level lowered to less than level 2 (-41.6”). At rated conditions and a total loss of Feedwater level will lower to approximately -60” to -80”

All distracters are plausible because they are Scram ONEP subsequent actions, but they are wrong because they are advantageous and are not affected by current conditions.

‘A’ is wrong, Step 3.6.2 of the SCRAM ONEP states to insert IRMs, there is no reason not to insert the IRMs at this time.

‘B’ is wrong, Step 3.10 states in the CAUTION . **IF** no Reactor Recirculation pumps are running, **THEN** section 3.11 **AND** 3.12 **Should** be performed concurrently without delay to prolong the onset of thermal stratification. Step 3.10 states **IF** Reactor Water Cleanup System (RWCU) has tripped, **THEN RESTART** RWCU per Attachment I after Reactor water level **AND** pressure have stabilized.

‘D’ is wrong, See 3.10 CAUTION from above. Step 3.11 states, **IF** no Reactor Recirculation pumps are running, **THEN PERFORM** the following without delay. Step 3.11.4 **START** Both Reactor Recirculation pumps, **IF** available, in slow speed per Attachment II.

K/A Match

The candidate must be able to interpret the given information on reactor water level and conditions and decide on the correct actions to take.

Technical References:

05-1-02-I-1, Reactor Scram ONEP, Rev. 132
02-S-01-43, Transient Mitigation Strategy, Rev. 7

Handouts to be provided to the Applicants during exam:

None

Learning Objective:

GLP-OPS-PROC, OBJ. 11.1
GLP-OPS-ONEP, OBJ. 2, 3

Question Source:	**Bank # 679**	Bank not on any NRC exam
(note changes and attach parent)	**Modified Bank #**	

	New	
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.43(b)(5)	
Level of Difficulty:	3.0	
PRA Applicability:		
None.		

<table>
<tr><td>Examination Outline Cross Reference</td><td>Level</td><td>SRO</td></tr>
<tr><td rowspan="5">295025 High Reactor Pressure

EA2. Ability to determine and/or interpret the following as they apply to HIGH REACTOR PRESSURE:

EA2.02 Reactor power</td><td>Tier</td><td>1</td></tr>
<tr><td>Group #</td><td>1</td></tr>
<tr><td>K/A</td><td>295025 EA2.02</td></tr>
<tr><td>Rating</td><td>4.2</td></tr>
<tr><td>Revision</td><td>2</td></tr>
<tr><td colspan="3">Revision Statement:

Rev. 1, Per Ops Rep, Changed correct answer to match current operations philosophy.

Rev 2, NRC Review and comments

Changed "will to "should" in stem</td></tr>
</table>

May 2017 NRC Exam Q 79

Question: 78

HANDOUT PROVIDED

An ATWS has occurred.

Current conditions are:

- Reactor power is 40%
- Reactor pressure is 1045 psig and steady
- Two SRVs are currently open
- Suppression pool level is at the Tech Spec low level limit
- Suppression pool temperature is 150°F and rising.
- RHR ‘A’ and RHR ‘B’ systems have just been placed in Suppression Pool Cooling

Which of the following describes the next action the CRS should direct?

A. Emergency Depressurize the Reactor when HCTL is exceeded

B. Lower pressure band to prevent entering unsafe zone of HCTL

C. Immediately Emergency Depressurize the Reactor when exceeding HCTL is predicted.

D. Anticipate an Emergency Depressurization and fully open the Main Turbine Bypass valves

Answer: **B**
Explanation: With the plant maintaining current conditions, reactor power exceeds the capacity of the bypass valves therefore SRVs will continue to be opened to maintain pressure. With SRVs open Suppression pool temp will rise. In the pressure branch of EP-2A, it states "If SP temp cannot be maintained in the Safe Zone of the HCTL THEN, maintain RPV pressure in the Safe Zone of the HCTL." SRO must know EP steps without reference. 'B' is correct.
Distracters: **All other distractors are plausible, they all are different paths that could be taken if the student forgets the step in the EOPs and that HCTL can be exceeded if it can be restored to the safe zone.** 'A' is wrong - Emergency Depress is required only if RPV pressure cannot be restored and maintained in the Safe Zone of the HCTL. 'C' is wrong: Emergency Depress is required only if RPV pressure cannot be restored and maintained in the Safe Zone of the HCTL. Since the RHR systems have just been started in Suppression Pool cooling the SRO should wait and determine if the limit can be restored. 'D' is wrong - Reactor power is too high, the Bypass valves are already fully open and procedures do not allow to anticipate an ED while in EP-2A ATWS.
K/A Match The candidate must be able to interpret the effects of the given information with reactor power and reactor pressure to determine the next action.
Technical References: EP-3 Rev date 3/23/2016 02-S-01-40 Rev 8, EP Technical Bases, Attachment V, pages 41, 44, and 45
Handouts to be provided to the Applicants during exam: HCTL Curve

Learning Objective: GLP-OPS-EP01		
Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #1318**	**May 2017 NRC Exam Q79**
	New	
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.43(b)(5)	
Level of Difficulty:	3.0	
PRA Applicability:		
None.		

Examination Outline Cross Reference	Level	SRO
295027 High Containment Temperature (Mark III Containment Only) 2.2.25 Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits.	Tier	1
	Group #	1
	K/A	295027 2.2.25
	Rating	4.2
	Revision	2

Revision Statement:

Rev 1, Facility Rep. grammar on stem question, added “is to” and changed answers to fit the format.

Rev 2, Added clarity to answer ‘B’ by adding “during a DBA.

Question: 79

One bases for the Tech Spec ACTION to immediately suspend RCIC flow testing based on high Suppression Pool temperature is to….

A. Preclude having to enter EP-3, Containment Control.

B. Ensure the Containment temperature limit is not exceeded during a DBA.

C. Prevent RCIC turbine exhaust check valve chatter that can damage RCIC piping.

D. Prevent elevated RCIC turbine lube oil temperature that could damage RCIC turbine.

Answer: **B**

Explanation:

In the stem, RCIC flow testing implies power is above 1%. TS 3.6.2.1 Action C.1 requires immediately suspending testing that adds heat to the Supp Pool when Supp Pool average temperature exceeds 105°F. The implication in the stem is Supp Pool temperature has just exceeded 105°F.

Suppression pool average temperature is allowed to be > 95°F with THERMAL POWER > 1% RTP when testing that adds heat to the suppression pool is being performed. However, if temperature is > 105"F, the testing must be immediately suspended to preserve the pool's heat absorption capability. With the testing suspended, Condition A is entered and the Required Actions and associated Completion Times are applicable.

TS bases 3.6.2.1 states the Supp Pool temperature limit was developed to address technical concerns that include limiting containment average air temperature to < 185°F during the DBA.

‘B’ is correct.

Distracters:

'A' is wrong - is plausible because entry into an Emergency Procedure is undesirable and implies elevated risk and challenge to plant safety, but it is wrong because TS 3.6.2.1 Action C that requires suspending testing that is adding heat to the Supp Pool at 105°F, so the Supp Pool Temperature EP-3 entry condition of 95°F would have already been met, so EP-3 entry cannot be avoided.

'C' is wrong - is plausible because exhaust check valve operation could theoretically be affected mechanistically by Supp Pool parameters, but it is wrong because 04-1-01-E51-1 step 3.2 states this is a concern for operation of RCIC at speeds below 2000 rpm and because RCIC operability is not a concern in the DBA analysis and is not mentioned in the bases for TS 3.6.2.1.

'D' is wrong - is plausible because it describes undesired effects of elevated Supp Pool temperature on RCIC. It is each wrong because 04-1-01-E51-1 step 3.3 states RCIC suction temperature is limited to 140°F for non-emergency operation to prevent bearing/turbine seals overheating, and during emergencies, EP caution 4 states RCIC equipment damage could occur if Supp Pool temperature exceeds 225°F, well above the 105°F condition implied in the stem and because RCIC operability is not a concern in the DBA analysis and is not mentioned in the bases for TS 3.6.2.1.

K/A Match

The candidate must have the knowledge of the bases in Technical Specifications for limiting conditions for operations on Suppression Pool Temp.

Technical References:

TR 3.6.2.1 and bases
EP-3
EP Caution 4

Handouts to be provided to the Applicants during exam:

None

Learning Objective:

GLP-OPS-TS001, OBJ. 39

Question Source:	**Bank # 694**	2013 NRC Exam Q79
(note changes and attach parent)	**Modified Bank #**	
	New	
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.43(b)(2)	
Level of Difficulty:	2.0	

PRA Applicability:
RCIC is listed as #17 on the System Importance to CDF.

Examination Outline Cross Reference	Level	SRO
295031 Reactor Low Water Level EA2.Ability to determine and/or interpret the following as they apply to REACTOR LOW WATER LEVEL : EA2.03 Reactor pressure	Tier	1
	Group #	1
	K/A	295031 EA2.03
	Rating	4.2
	Revision	1
Revision Statement: **Rev 1 Per NRC review (10 free view) changed wording in stem of question, added "procedurally required". Added more explanation about entering alternate level control leg of EOPs.**		

Question: 80

The plant has experienced a LOCA

All High pressure injection systems are not available.

Reactor Water level has reached the lowest indication for Wide Range.

Which of the following describes the procedurally required RPV pressure control being directed by the CRS?

A. LO-LO Set SRVs.

B. Emergency Depressurization.

C. Pressure band 450 – 600 psig using SRVs.

D. Pressure band 800 – 1060 psig using Main Bypass Valves.

Answer: B
Explanation: In the stem, when all high pressure feed is lost the CRS/SRO should enter the alternate control leg of the EOPs. There, other systems are started in an attempt to restore level. If RPV water level continues to lower as Reactor water level reaching the lowest indication is -160" Wide Range. Per EP-2 When level drops to -160 in. Emergency Depressurization is Required. The SRO must recognize that the reactor must be depressurized. 'B' is correct.

Distracters:

'A' is wrong - is plausible because during a LOCA it is permissible to use LO-LO Set system to control pressure within band. However, the given information requires an ED.

'C' is wrong - is plausible because this is a band that is used to feed with Condensate. However, with no high pressure feed bases states not to lower the pressure band with nothing to feed.

'D' is wrong – With level indicated at -160" Wide Range the MSIVs would have closed at -150.3", therefore, Main Bypass Valves are unavailable.

K/A Match

The candidate must have the knowledge of how to control reactor pressure with a low reactor water level.

Technical References:

02-S-01-40, EP Bases, Rev. 9

Handouts to be provided to the Applicants during exam:

None

Learning Objective:

GLP-OPS-EP02

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.43(b)(5)	
Level of Difficulty:	3.0	

PRA Applicability:

None.

Examination Outline Cross Reference	Level	SRO
295038 High Off-Site Release Rate 2.4.4 Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures.	Tier	1
	Group #	1
	K/A	295038 2.4.4
	Rating	4.7
	Revision	0
Revision Statement:		

Question: 81

The Plant was shut down due to indications of fuel failure.

RCIC system experienced a steam leak with a failure to isolate.

All RCIC Max Safe Values have been met.

Which of the following would require the CRS to direct an Emergency Depressurization?

A. FHA or FPS exhaust radiation levels above trip setpoints

B. SGTS FLTR TRN radiation level above the operating limit

C. Dose Calculation calls for a declaration of a General Emergency

D. MSL radiation level is more than 3 times full power background

Answer: C

Explanation:

Per EP-4 step 11 states IF a discharge cannot be isolated Declaration of a General Emergency is expected due to radioactivity release dose calculation, THEN, Emergency Depressurization is required.

'C' is correct.

Distracters:

'A' is wrong - is plausible this is used in a retention step for EP-4 step 1. IF FHA / FPS exhaust radiation level is above 3.6 / 30 mr/hr, THEN Verify SGTS initiation and isolation.

'B' is wrong - is plausible because this would cause a ED however, it requires the Max Safe Value to be

exceeded before an ED is required not just exceeding the operating limit.

'D' is wrong – is plausible this is used in a retention step for EP-4 step 1. MSL radiation level is more that 3 times full power background, THEN, Close the MSIVs and MSL drain valves.

K/A Match

The candidate must have the knowledge to recognize system operating parameters that are entry-level conditions for emergency procedures.

Technical References:

02-S-01-40, EP Bases, Rev. 9

Handouts to be provided to the Applicants during exam:

None

Learning Objective:

GLP-OPS-EP4

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.43(b)(4)	
Level of Difficulty:	3.0	

PRA Applicability:

RCIC is listed as #17 on the System Importance to CDF.

Examination Outline Cross Reference	Level	SRO
600000 Plant Fire On Site 2.4.34 Knowledge of RO tasks performed outside the main control room during an emergency and the resultant operational effects.	Tier	1
	Group #	1
	K/A	600000 2.4.34
	Rating	4.7
	Revision	2

Revision Statement:

Rev 1

Changed stem #1 to ensure clarity of SRO directions

Changed stem #4 to ensure clarity of SRO directions.

Rev 2, NRC Review and comments

Changed #1 to Defeat LSS

Question: 82

Fire in the control room

Which of the following actions should the SRO direct to be performed for this event?

1. Defeat LSS
2. Depressurize the RPV
3. Manually start Div 1 D/G and energize 15AA
4. Perform Upper Containment Pool lineup to RCIC Suction

ONLY:

A. 1, 3 and 4

B. 1, 2 and 4

C. 1, 2 and 3

D. 2, 3 and 4

Answer: C
Explanation:

Per 05-1-02-II-1, Shutdown from the Remote Shutdown panel ONEP, the following actions are time critical action for a fire in the control room.

> **IF** a Main Control Room fire is in progress, the Reactor Operator dispatched to perform Attachment XX Should complete the actions to manually start Div I Diesel Generator (**IF NOT** auto started) within 8 minutes of reactor being scrammed **AND** Should complete actions to Close the Div I Diesel Generator output breaker within 12 minutes of reactor being scrammed.
>
> **IF** a Main Control Room fire is in progress, the CRS **AND** FSS (Remote Shutdown Panel) to perform Attachment XXI will complete the actions to Defeat LSS **AND** open the 15 Bus breakers within 8 minutes of reactor being scrammed.
>
> **IF** a Main Control Room fire is in progress, the CRS **AND** FSS at the Remote Shutdown Panel to perform Attachment XXI will complete the actions to open the SRVs to depressurize the Reactor within 12 minutes.
>
> **IF** a Main Control Room fire is in progress, the CRS **AND** FSS at the Remote Shutdown Panel to perform Attachment XXI will complete the actions to depressurize **AND** inject RHR A to the reactor within a maximum of 14.3 minutes.

Per step 3.1.3 “actions prior to leaving the control room does not include RCIC.

‘C’ is correct.

Distracters:

‘A’ is wrong - Depressurize the RPV is also required.

‘B’ is wrong – Initiating RCIC is not required prior to leaving the Control Room during a Fire but it is required if evacuating for any other reasons.

‘D’ is wrong – This action is from 05-S-01-FSG-002, Alternate RCIC Suction Source, that is to be used during an extended loss of AC power, not from the remote shutdown panel.

K/A Match

The candidate must have the knowledge of RO tasks performed outside the main control room during an emergency and the resultant operational effects.

Technical References:

05-1-02-II-1, Shutdown from the Remote Shutdown Panel ONEP, Rev. 51

Handouts to be provided to the Applicants during exam:

None

Learning Objective:

GLP-OPS-ONEP

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.43(b)(5)	
Level of Difficulty:	3.0	
PRA Applicability:		
None.		

Examination Outline Cross Reference	Level	SRO
295015 Incomplete SCRAM / 1 AA2.01 Ability to determine and/or interpret the following as they apply to INCOMPLETE SCRAM: AA2.01 Reactor power	Tier	1
	Group #	2
	K/A	295015 AA2.01
	Rating	4.3
	Revision	0
Revision Statement: **New Question.**		

Question: 83

An ATWS is in progress with reactor power at 3%.

Which of the following actions must be directed by the CRS?

1. Initiate ARI/RPT system
2. Start Suppression Pool Cooling
3. Initiate Standby Liquid Control system
4. Initiate and override Low Pressure ECCS system

A. 1 and 2 only

B. 1 and 3 only

C. 2 and 4 only

D. 3 and 4 only

Answer: D
Explanation: The SRO candidate must interpret reactor power, because, with an ATWS <5% the RO will perform ‘1’, and ‘2’ per immediate operator actions. All other actions must be directed by the CRS. If reactor power was >5% all actions would be performed by immediate operator actions. ‘D’ is correct.
Distracters:

‘A’ is wrong, these actions are perfomed per immediate operator actions in the SCRAM onep

‘B’ is wrong, #1 is an immediate action and #3 is correct

‘C’ is wrong, #2 is an immediate action and #4 is correct.

K/A Match

The candidate must have the ability to interpret the given information of reactor power then the SRO must decide what actions are required to be directed.

Technical References:

05-S-01-EP-1, Emergency/Severe Accident Procedure Support Documents
05-S-01-EP-2, RPV Control

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-EP2A, OBJ. 1

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.43(b)(5)	
Level of Difficulty:	3.0	

PRA Applicability:

None.

Examination Outline Cross Reference	Level	SRO
295033 High Secondary Containment Area Radiation Levels 2.2.12 Knowledge of surveillance procedures.	Tier	1
	Group #	2
	K/A	295033 2.2.12
	Rating	4.1
	Revision	3

Revision Statement:

Rev 1, Per Ops Rep, Tech Specs now state use the Surveillance Frequency Control Program to determine surveillance frequency times, added times to stem and added 02-S-01-17 as a reference, removed reference material.

Rev 2, Validation, changed stem and gave the surveillance frequency and changed answer A to 36 hours instead of 12.

Rev 3, NRC Review and comments

Changed stem question to "Per Technical Specifications, which of the following is the MAXIMUM amount of time allowed by tech specs to perform the surveillance?"

Changed answers to 4, 8, 12, 24

Question: 84

The plant is operating at rated power.

At 0700 on 2/1/20, while reviewing the logs, the CRS discovers the following:

The last time that a Channel Check was performed on Fuel Handling Area Ventilation radiation monitor D17-K617A was 24 hours ago.

Per the Surveillance Frequency Control Program, the surveillance frequency is 12 hours.

Per Technical Specifications, which of the following is the MAXIMUM amount of time allowed by tech specs to perform the surveillance?

A. 4

B. 8

C. 12

D. 24

Answer: **D**		
Explanation: TS SR 3.0.3 allows up to 24 hours to complete a missed surveillance. Per 02-S-01-17, 6.8.1. **IF** it is discovered that a Surveillance was **NOT** performed within its specified frequency, **THEN** per SR 3.0.3 compliance with the requirement to declare the LCO **NOT** met May be delayed, from the time of discovery, up to 24 hours **OR** up to the limit of the specified Frequency, whichever is greater. This allows time to perform the surveillance. 'D' is correct.		
Distracters: 'A', 'B' and 'C' are incorrect but plausible due to all are completion times per tech specs.		
K/A Match The candidate must have knowledge of Radiation monitor surveillance procedures and Tech Specs.		
Technical References: TS 3.3.6.2 and SR 3.0.3 02-S-01-17, Control of Limiting Conditions for Operation, Rev. 134		
Handouts to be provided to the Applicants during exam: **NONE**		
Learning Objective: GLP-OPS-TS001, OBJ. 40		
Question Source:	**Bank # 698**	2009 NRC Exam Q85
(note changes and attach parent)	**Modified Bank #**	
	New	
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.43(b)(2)	
Level of Difficulty:	3.0	

PRA Applicability:
None.

Examination Outline Cross Reference	Level	SRO
500000 High Containment Hydrogen Concentration 2.1.25 Ability to interpret reference materials, such as graphs, curves, tables, etc.	Tier	1
	Group #	2
	K/A	500000 2.1.25
	Rating	4.2
	Revision	1
Revision Statement: **Rev 1 Per NRC review (10 free view) changed wording in stem of question, all caps on MINIMUM and added the word "procedurally".**		

Question: 85

HANDOUT PROVIDED

The plant has experienced an ATWS with a steam line break in the Containment.

Current Containment pressure is 9.8 psig.

What is the MINIMUM Hydrogen concentration to procedurally require the CRS to direct Hydrogen Recombiners shutdown and Containment Spray initiated?

A. 2.9%

B. 6%

C. 7.8%

D. 9%

Answer:	**C**
Explanation: Per EP-3, at 2.9% CTMT or DW hydrogen, Exit all EPs and Enter SAPs. Once in the SAPS Hydrogen leg, when Hydrogen is in the UNSAFE zone of the HDOL curve then Secure Hydrogen Recombiners and start Containment Spray. 'C' is correct.	
Distracters:	

‘A’ is incorrect This is the concentration to exit the EPs and enter the SAPs. ‘B’ is incorrect This is the concentration for starting the recombiners ‘D’ is incorrect This is the concentration for high limit for igniter operation.
K/A Match The candidate has the knowledge and ability to interpret the given information and using reference materials, such as graphs to determine actions.
Technical References: 05-S-01-SAP-1, Severe Accident Procedure, Rev 10
Handouts to be provided to the Applicants during exam: HDOL Curve
Learning Objective: GLP-OPS-SAPS, OBJ. 40

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.43(b)(5)	
Level of Difficulty:	3.0	
PRA Applicability:		
None.		

<table>
<tr><td>Examination Outline Cross Reference</td><td>Level</td><td>SRO</td></tr>
<tr><td rowspan="5">211000 Standby Liquid Control System

A2. Ability to (a) predict the impacts of the following on the STANDBY LIQUID CONTROL SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:

A2.03 A.C. power failures</td><td>Tier</td><td>2</td></tr>
<tr><td>Group #</td><td>1</td></tr>
<tr><td>K/A</td><td>211000 A2.03</td></tr>
<tr><td>Rating</td><td>3.4</td></tr>
<tr><td>Revision</td><td>1</td></tr>
<tr><td colspan="3">Revision Statement:

Rev 1, NRC Review and comments

Changed (2) to the following: "The CRS must call for _____________?"

Deleted the Nows in the answers.</td></tr>
</table>

Question: 86

Standby Liquid Control (SLC) pump ‘B’ is tagged for Maintenance.

An ATWS has occurred with reactor power at 15%.

The RO performs immediate operator actions.

Electrical bus 15AA has lost power and indicates a fault on the bus.

(1) Which of the following describes the availability of the SLC system?

(2) The CRS must call for _______________?

A. (1) No SLC pumps are available
(2) Attachment 28

B. (1) ‘A’ SLC pump is available
(2) Attachment 28

C. (1) ‘A’ SLC pump is available
(2) Attachments 18, 19, and 20 only

D. (1) No SLC pumps are available
(2) Attachments 18, 19, and 20 only

Answer: A		
Explanation: With a loss of 15AA, power is lost to SLC pump 'A' therefore no SLC pumps are available due to 'B' pump already being OOS. EP-2A calls for Attachment 28 when boron cannot be injected. 'A' is correct.		
Distracters: 'B' is incorrect 'A' pump is not available. 'C' is incorrect 'A' pump is not available, these attachments will be called for but not for the loss of SLC 'D' is incorrect these attachments will be called for but not for the loss of SLC		
K/A Match The candidate must predict the impact on loss of power and determine the correct procedure to use to mitigate.		
Technical References: 04-1-01-1C41-1, SLC SOI, 05-1-02-EP-1, Attachment 28 02-S-01-40, EP Technical Bases		
Handouts to be provided to the Applicants during exam: NONE		
Learning Objective: GLP-OPS-EP2A, OBJ. 40 GLP-OPS-C4100, OBJ.		
Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.43(b)(5)	
Level of Difficulty:	3.0	

PRA Applicability:
None.

Examination Outline Cross Reference	Level	SRO
217000 RCIC A2. 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: A2.01 System Initiation Signal	Tier	2
	Group #	1
	K/A	217000 A2.01
	Rating	3.7
	Revision	1
Revision Statement: **Rev 1, NRC Review and comments,** **Changed "DIRECTS" to "should DIRECT," and "is directing" to "should direct."**		

Question: 87

The plant is operating at rated power.

- RCIC auto initiates on an **invalid** Level 2 signal.
- Initiation logic seals in and will not reset.
- The reactor remains online.

(1) The CRS should DIRECT the RO to ___________.

(2) The CRS should direct actions from ____________.

A. (1) TRIP the RCIC Turbine
   (2) Loss of Feedwater Heating ONEP

B. (1) TRIP the RCIC Turbine
   (2) System Operating Instruction, Shutdown of the RCIC System

C. (1) Secure RCIC using SOI Hardcard
   (2) System Operating Instruction, Shutdown of the RCIC System

D. (1) Secure RCIC using SOI Hardcard
   (2) Loss of Feedwater Heating ONEP

Answer:	A

Explanation:

With RCIC system initiation at power the SRO should enter the Loss of Feedwater Heating ONEP section 'C', C2 states IF RCIC initiation is NOT VALID THEN TRIP RCIC. The ONEP DOES NOT reference shutting the system using the Hardcard.

'A' is correct.

Distracters:

'B' is incorrect, but plausible due to the SOI holds all hardcards, but RCIC initiation at power is an entry into the ONEP

'C' is incorrect but plausible due to the SOI holds all hardcards, but RCIC initiation at power is an entry into the ONEP. If RCIC is not needed during an event the hardcard is used to shutdown but during an inadvertent initiation at power, the cold water injection downstream of the feedwater temp indicators require a more rapid shutdown

'D' is incorrect If RCIC is not needed during an event the hardcard is used to shutdown but during an inadvertent initiation at power, the cold water injection downstream of the feedwater temp indicators require a more rapid shutdown.

K/A Match

The candidate must have the ability to use the given information and determine what actions are to be directed and what procedure is entered.

Technical References:

05-1-02-V-5, Loss of Feedwater Heating ONEP

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-E5100 Obj 8.0 and GLP-OPS-ONEP Obj 35

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.43(b)(2)	
Level of Difficulty:	3.0	

PRA Applicability:
None.

Examination Outline Cross Reference	Level	SRO
239002 SRVs A2 Ability to (a) predict the impacts of the following on the RELIEF/SAFETY VALVES; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: A2.03 Stuck open SRV.	Tier	2
	Group #	1
	K/A	239002 A2.03
	Rating	4.2
	Revision	0
Revision Statement:		

Question: 88

The plant is at 90% power following a sequence exchange.

The following occurs:

- ADS / SRV LEAK. P601-18A-G-2, and SRV/ADS VLV OPEN/DISCH LINE PRESS HI, P601-19A-A5 annunciators are received
- Generator megawatt output has lowered by approximately 75 MWe
- SRV B21-F041D has a red indication on P601 handswitch
- Suppression Pool average water temperature is currently 98°F and rising
- Suppression Pool Water Level is 18.71ft. and rising.
- Handswitches on P601 and P631 for SRV B21-F041D are taken to the CLOSE position
- I&C has been notified to pull associated fuses per ARI.

What is the next required action to correct, control, or mitigate the impacts of this event on the plant?

A. Lower Reactor power

B. Place the Reactor Mode Switch to SHUTDOWN

C. Place one loop of RHR in Suppression Pool Cooling

D. Place both loops of RHR in Suppression Pool Cooling

Answer:	**D**

Explanation:

With the given information the student can determine that a stuck open SRV exsist.

ONEP 05-1-02-V-21, Reactor Pressure Control Malfunctions is entered and section 3.3 refers you to the ARIs to perform actions to get the SRV closed, which are in progress from the given information.
The next action should be enter EP-3 (>95°F Suppression Pool Temp.) and start all available Suppression Pool cooling (not needed for adequate core cooling). The SRO should direct both loop of suppression pool cooling to be started per EP-3. Before SP temp reaches 110°F the SRO will enter EP-2 which requires a Mode Switch to SHUTDOWN. This is based on Tech Spec 3.6.2.1 and bases. If the actions of the ARI do not close the SRV then ONEP directs a plant shutdown.

'D' is correct.

Distracters:

'A' is incorrect – The ONEP does provide actions to lower reactor power but not in this case only if pressure control system is malfunctioning.

'B' is incorrect – IF the actions of the ARI fail and the SRO determines that Supp Pool Temp will reach 110°F then the ONEP directs a mode switch to shutdown.

'C' is incorrect – When the EPs are entered due to supp pool temp the direction is to start all available cooling systems, not just one.

K/A Match

The candidate must have the ability to predict the impact of a stuck open SRV and determine the correct procedure and action to mitigate the event.

Technical References:

05-1-02-V-21, Reactor Pressure Control Malfunctions
02-S-01-40, EP Technical Bases

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-EP3, OBJ.8
GLP-OPS-ONEP, OBJ. 7

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

Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.43(b)(5)	
Level of Difficulty:	3.0	
PRA Applicability:		

Examination Outline Cross Reference	Level	SRO
259002 Reactor Water Level Control System 2.4.50 Ability to verify system alarm setpoints and operate controls identified in the alarm response manual.	Tier	2
	Group #	1
	K/A	259002 2.4.50
	Rating	4.0
	Revision	0
Revision Statement:		

Question: 89

The plant is at rated thermal power when the following alarms are received:

- RFPT A TRIP
- RX LVL 40"/32" HI/LO

The plant has the following conditions:

- Reactor power – 82%
- Core Flow - 85 mlbm/hr
- RFPT B speed – 5850 rpm
- Reactor water level is slowly lowering

(1) Given the alarms and conditions, what event has occurred?

(2) The CRS should transition to _________________ procedure that contains the actions to mitigate these conditions.

A. (1) Complete Reactor Recirc Flow Control Valve Runback
(2) ONEP 05-1-02-V-7, Feedwater System Malfunctions.

B. (1) Complete Reactor Recirc Flow Control Valve Runback
(2) ARI 04-1-02-1H13-P680-2A-A2, RFPT A TRIP.

C. (1) Partial Reactor Recirc Flow Control Valve Runback
(2) ARI 04-1-02-1H13-P680-2A-A2, RFPT A TRIP.

D. (1) Partial Reactor Recirc Flow Control Valve Runback
(2) ONEP 05-1-02-V-7, Feedwater System Malfunctions

Answer:	D

Explanation: With the indications given the Recirc FCVs did not completely runback due to high core flow and power, Reactor power is too high for one Feedpump to handle. ONEP 05-1-02-V-7, Feedwater System Malfunctions contains the action to **ENSURE** Reactor Recirculation System Flow Control Valve runback occurs (62 Mlbm/hr). The term Ensure means if it did not occur then perform actions to make it happen. 'D' is correct.
Distracters: 'A' is incorrect – If a complete runback would have occurred then core flow would be approximately 62 Mlbm/hr. 'B' is incorrect - If a complete runback would have occurred then core flow would be approximately 62 Mlbm/hr. ARI directs to refer to the ONEP 'C' is incorrect - ARI directs to refer to the ONEP
K/A Match The candidate uses the given alarms and parameters to verify system actions are in concurrence with the alarm response manual and transition to another procedure to mitigate the event.
Technical References: 05-1-02-V-7, Feedwater System Malfunctions ONEP, Rev. 31
Handouts to be provided to the Applicants during exam: NONE
Learning Objective: GLP-OPS-ONEP, OBJ.

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.43(b)(5)	
Level of Difficulty:	3.0	
PRA Applicability:		
None.		

Examination Outline Cross Reference	Level	SRO
262001 A.C. Electrical Distribution 2.4.41 Knowledge of the emergency action level thresholds and classifications.	Tier	2
	Group #	1
	K/A	262001 2.4.41
	Rating	4.6
	Revision	0
Revision Statement:		

Question: 90

HANDOUT PROVIDED

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 +50"
- 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: 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.

'D' is correct.

Distracters:

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 'D' plausible, but wrong.

The plausibility of distracters '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.

K/A Match

The candidate must have the knowledge of the emergency action level thresholds and classifications.

Technical References:

10-S-01-1, Section EPP 01-02

Handouts to be provided to the Applicants during exam:

EAL Emergency Classification Flowcharts

Learning Objective:

GLP-OPS-EPTS6, OBJ.

Question Source:	**Bank # 380**	2011 NRC Exam Q77
(note changes and attach parent)	**Modified Bank #**	
	New	
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.43(b)(5)	
Level of Difficulty:	3.0	

PRA Applicability:

ESF R20 is listed as #4 on the System Importance to CDF.
EDGs are listed as #11 on the System importance to CDF.

Examination Outline Cross Reference	Level	SRO
201001 Control Rod Drive Hydraulic System A2. Ability to (a) predict the impacts of the following on the CONTROL ROD DRIVE HYDRAULIC SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: A2.08 Inadequate system flow	**Tier**	2
	Group #	2
	K/A	201001 A2.08
	Rating	2.8
	Revision	2
Revision Statement: **Rev 1, Ops review – added units in stem, changed stem to correct indication of full upscale on flow meter.** **Rev 2, NRC Review and comments** **Changed "CRS" in 4th line to "CRD"** **Added "required action" to question part of the stem**		

Question: 91

The Plant has experienced an ATWS.

The CRS calls for the Nominal Band for Reactor Water level and pressure.

Current RPV water level is -50 inches and slowly lowering.

Attachments are installed for control rod insertion and CRD has been maximized for pressure.

The RO inserting control rods reports control rod movement has stopped and CRD system flow rate is **pegged full upscale** with C11-F002A, CRD FLO CONT VLV indicating green only indication.

Which of the following describes the required action by the CRS?

A. Maximize CRD for Flow

B. Restore the Aux Building

C. Call for Attachment 24, Vent CRD Overpiston Volumes, to be installed

D. Call for Attachment 22, Place Individual Scram Test Switches in TEST, to be installed

Answer: B

Explanation:

With the given indications an Aux building/Containment isolation occurred at -41.6 inches and therefore causing instrument air to isolate to the containment. The failure of the Flow control valve is loss of air, the FCV will fail as is but drift close due to instrument air leaks. As the FCV closes control rod motion will stop. To mitigate this event the CRS must call for "Restore the Aux Building" per EP-2A and restore Air to the Aux building and Containment.

'B' is correct.

Distracters:

'A' is wrong – This action is performed during an ATWS prior to installation of Attachments 18, 19 and 20. This should cause control rods to drift in due to high cooling water pressure. This action will have no effect on the FCV due to the loss of Air.

'C' is wrong – Attachment 24 should be used when control rods can't be moved due to hydraulic block, but not in this situation, this also requires an entry into the Containment, not advised during an ATWS.

'D' is wrong - Attachment 24 should be used when control rods can't be moved due to failure to de-energize scram solenoids, but not in this situation, this also requires an entry into the Containment, not advised during an ATWS

K/A Match

The candidate must be able to predict the impacts low system flow and using procedures mitigate the event.

Technical References:

02-S-01-40, EP Technical Bases, Rev. 9
04-1-01-C11-1, CRD SOI, Rev. 155

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-EP2A, OBJ.

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

	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.43(b)(6)	
Level of Difficulty:	3.0	
PRA Applicability:		
None.		

Examination Outline Cross Reference	Level	SRO
223001 Primary Containment System and Auxiliaries 2.4.35 Knowledge of local auxiliary operator tasks during an emergency and the resultant operational effects.	Tier	2
	Group #	2
	K/A	223001 2.4.35
	Rating	4.0
	Revision	1

Revision Statement:

Rev 1, NRC Review and comments

Changed “will” to “should” in stem

Question: 92

The plant is operating at rated conditions.

An inadvertent initiation of Division 1 ECCS occurs.

The initiating signal will not clear.

Actual Drywell Pressure is rising.

(1) Which of the following should the CRS direct to secure the ‘A’ Drywell Purge Compressors?

(2) What is the availability of the ‘A’ Drywell Purge Compressor?

A. (1) EOP Attachment 15, Defeating Drywell Purge Compressors Start Signals
(2) Purge compressor will start using control room handswitch

B. (1) EOP Attachment 15, Defeating Drywell Purge Compressors Start Signals
(2) Purge compressor will not start from any remote location

C. (1) Combustible Gas Control SOI, Recovery from Manual or Automatic Initiation
(2) Purge compressor will not start from any remote location

D. (1) Combustible Gas Control SOI, Recovery from Manual or Automatic Initiation
(2) Purge compressor will start using control room handswitch

Answer:	C

Explanation:

An initiation signal that will not reset has a hard start signal on the Drywell Purge Compressor. The only way to stop this compressor to **locally** trip the breaker and prevent it from reclosing. This action is performed in the Combustible Gas Control System SOI, Recovery from Manual or Automatic Initiation.

‘C’ is correct.

Distracters:

‘A’ is wrong – This action is performing an EOP Attachment which can only be done when the EOP have been entered, current conditions have no EOP entry conditions. Purge compressor will not start except for **local** breaker manipulation.

‘B’ is wrong – This action is performing an EOP Attachment which can only be done when the EOP have been entered, current conditions have no EOP entry conditions. .

‘D’ is wrong - . Purge compressor will not start except for **local** breaker manipulation.

K/A Match

The candidate must have the knowledge of local auxiliary operator tasks during an emergency and the resultant operational effects.

Technical References:

05-S-01-EP-1, Emergency/Severe Accident Procedure Support Documents, Rev. 38
04-1-01-E61-1, CGCS SOI, Rev. 41

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-EP2, OBJ.

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.43(b)(5)	
Level of Difficulty:	3.0	

PRA Applicability:
None.

Examination Outline Cross Reference	Level	SRO
259001 Reactor Feedwater System A2. Ability to (a) predict the impacts of the following on the REACTOR FEEDWATER SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: A2.03 Loss of condensate pump(s)	Tier	2
	Group #	2
	K/A	259001 A2.03
	Rating	3.6
	Revision	3
Revision Statement: **Rev 1 Per NRC review (10 free view) changed wording of entire question.** **Rev 2, changed wording of question #2 to clarify.** **Rev 3, NRC Review and comments** **Changed stem to clarify alarm by adding "due to 2 of the 3 Hotwell level switches failing"**		

Question: 93

The plant is operating at 65% power for a sequence exchange.

RCIC system is INOP

The following alarm is received due to 2 of the 3 Hotwell level switches failing:

- CNDSR HTWL LVL LO, P680-2A-E9

ATC operator reports normal level in the Hotwell.

The CRS directs the crew to manually scram the reactor, enters EP-2 and Reactor Scram ONEP.

(1) What level band should the CRS direct to the crew?

(2) What procedure should be transitioned to by the CRS to restore Condensate/Feedwater for normal level control?

A. (1) +11.4 inches to +53.5 inches
   (2) CNDSR HTWL LVL LO, P680-2A-E9 Alarm Response Instruction

B. (1. -30 inches to +50 inches
   (2) CNDSR HTWL LVL LO, P680-2A-E9 Alarm Response Instruction

C. (1) +11.4 inches to +53.5 inches
   (2) 05-1-02-V-7, Feedwater System Malfunctions ONEP

D. (1) -30 inches to +50 inches
   (2) 05-1-02-V-7, Feedwater System Malfunctions ONEP

Answer:	**B**

Explanation:

The alarm given will cause all Condensate pumps to trip at the same time.

RPV water level will lower to the scram setpoint and both Feedpumps will trip.

Per 02-S-01-43, Transient Mitigation Strategy, step 6.6.6.d Level Control table, the Widened (Expanded) ECCS OR SRVs (non-ATWS) -30 inches to +50 inches should be used due to RCIC is INOP and ECCS (HPCS) will be used to control level.

To restore normal level control the ARI should be used First, with normal level indicated, the CRS can have the relay for low condenser hotwell level be removed and the Condensate System be restored.

'B' is correct.

Distracters:

'A' is wrong – the level band given is only used during normal situations, in this case ECCS system is being used for level control, therefore the expanded band should be used.

'C' is wrong – the level band given is only used during normal situations, in this case ECCS system is being used for level control, therefore the expanded band should be used and the actions to remove the relay is in the ARI not the ONEP.

'D' is wrong - the actions to remove the relay is in the ARI not the ONEP.

K/A Match

The candidate must have the knowledge to predict the impacts of a loss of condensate pump on the Feedwater system and determine the correct procedure to mitigate the event.

Technical References:

04-1-02-1H13-P680-2A-E9, CONDSR HTWL LVL LO, ARI Rev. 218
05-1-02-V-7, Feedwater System Malfunctions ONEP
02-S-01-43, Transient Mitigation Strategy

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective: GLP-OPS-ONEP, OBJ.		
Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.43(b)(5)	
Level of Difficulty:	3.0	
PRA Applicability: Condensate is listed as #10 on the System Importance to CDF.		

Examination Outline Cross Reference	Level	SRO
2.1.2 Knowledge of operator responsibilities during all modes of plant operation.	Tier	3
	Group #	
	K/A	2.1.2
	Rating	4.4
	Revision	4

Revision Statement:

Rev 1 Per NRC review (10 free view) changed stem to state “IAW with procedure” and changed the answers to ensure all numbers are used the same amount.

Rev 2 Per Ops Rep, changed wording of #4, changed stem to state station procedures instead of a specific one.

Rev 3 Per Ops review, added “initial” to #2 due to the acting ERO Emergency Director is not active licensed.

Rev 4, NRC Review and comments,

Changed #3 in stem, to "required to be in the main control room at all times when fuel is in the reactor"

Added more in the explanation for #3

Question: 94

IAW with station procedures, which of the following can ONLY be a responsibility of an active licensed SRO?

1. Ensures shift manning is met
2. Assume the initial duties of the Emergency Director
3. Required to be present in the Main Control Room at all time when fuel is in the reactor.
4. Concurs with the release and closure of outage and system work windows

A. 1, 2 and 3 only

B. 1, 3 and 4 only

C. 2, 3 and 4 only

D. 1, 2 and 4 only

Answer: D
Explanation:

#1 Per the procedure EN-OP-115-03, step 5.2.1 "In the case of illness or unexpected absence, SM **ENSURE** a shift member is held over and replacement personnel arranged to restore the shift complement within two hours."

#2 EN-OP-115 states the Shift Manager, "Assumes the role of Emergency Director until relieved by the Emergency Response Organization IAW Station Emergency Plan" and the CRS Assume the responsibilities and authority of the SM if the SM becomes incapacitated. Immediately notify Operations Management.

#4 EN-OP-115 states the Shift Manger Concurs with the release and closure of outage and system work windows that have an impact on the shutdown safety functions (SOER 09-01)."

'D' is correct.

Distracters:

#3 – A Senior Reactor Operator shall be present at GGNS at all times when fuel is in the reactor. A Senior Reactor Operator, normally the SM, will be present in the Control Room when the reactor is in Plant Mode 1, 2, or 3, or during emergencies. During emergencies, the SRO directing activities in the Control Room will be the Control Room Supervisor (WMC SRO prohibited). The procedure states "shall be present at GGNS" NOT in the CONTROL ROOM, this is only in modes 1, 2, and 3. The SRO is NOT required to be in the CONTROL ROOM in modes 4 or 5.

'A' is wrong – #4 is also a SRO responsibility per EN-OP-115.

'B' is wrong – #3 is not the SROs responsibility and #1 and #2 are.

'C' is wrong - #3 is not the SROs responsibility and #1 is.

K/A Match

The candidate must have the knowledge of the responsibilities of the SRO.

Technical References:

EN-OP-115, Conduct of Operations, Rev. 26
EN-OP-115-02, Control Room Conduct and Access Controls, Rev. 6
EN-OP-115-03, Shift Turnover and Relief, Rev 004

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-PROC, OBJ.

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

	New	X
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.43(b)(5)	
Level of Difficulty:	2.0	
PRA Applicability:		
None.		

Examination Outline Cross Reference	Level	SRO
2.1.37 Knowledge of procedures, guidelines, or limitations associated with reactivity management.	Tier	3
	Group #	
	K/A	2.1.37
	Rating	4.6
	Revision	0
Revision Statement: **NEW QUESTION.**		

Question: 95

Per EN-OP-115-14, Reactivity Management, which of the following would require dedicated SRO oversight other than the CRS with no concurrent duties?

A. Reducing power using Recirc flow to 85% for surveillance test.

B. Performing a sequence exchange after lowering power to 65%.

C. Performing 06-OP-C11-M-0001, Control Rod Operability Surveillance.

D. Raising power using Recirc flow from 80% to rated conditions at a ramp rate given by the RE.

Answer: B
Explanation: Using Attachement 1 of EN-OP-115-14, Reactivity Management, the SRO must determine the risk level of the manipulation. Sequence exchange or power maneuver where power is lowered to <70% is considered a High Risk Reactivity Manipulation. The Required Minimum Controls for Reactivity Risk Level for a High Risk Reactivity call for Reactivity SRO Oversight (Dedicated SRO, other than the CRS, with no concurrent duties) 'B' is correct.
Distracters: 'A' is wrong – Procedure states This would be a Low risk manipulation. Sequences exchanges or power maneuvers where power is not lowered to less than 70% 'C' is wrong – Procedure states This would be a Normal Risk Reactivity Manipulation. Control rod exercise testing 'D' is wrong - Procedure states This would be a Normal Risk Reactivity Manipulation. Power ascension

that is limited by the ramp rate

K/A Match

The candidate must have knowledge of procedures, guidelines, or limitations associated with reactivity management.

Technical References:

EN-OP-115-14, Reactivity Management, Rev 1

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-PROC, OBJ. 4.10

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.43(b)(5)	
Level of Difficulty:	2.0	

PRA Applicability:

None.

Examination Outline Cross Reference	Level	SRO
2.2.7 Knowledge of the process for conducting special or infrequent tests.	**Tier**	3
	Group #	
	K/A	2.2.7
	Rating	3.6
	Revision	1

Revision Statement:

Rev 1 Per NRC review (10 free view) changed the answers to ensure all numbers are used the same amount.

Question: 96

Consider the following:

1. Reactor startup
2. Containment entry at power
3. CORE ALTERATIONS due to Fuel movement.
4. Intentionally draining the reactor cavity level down to the RPV upper flange in MODE 5

Per EN-OP-116, Infrequently Performed Tests or Evolutions (IPTEs), which of the above is/are **required** to be controlled as IPTEs?

A. 1, 2 and 3 only.

B. 2, 3 and 4 only.

C. 1, 2 and 4 only.

D. 1, 3 and 4 only

Answer: **D**

Explanation:

See EN-OP-116, Attachment 9.1 (Pre-identified IPTEs). Items 1 and 3 are contained under the section labeled All Units. Item 4 is contained under the section labeled BWR Units.

'D' is correct.

Distracters:

‘A’ is wrong - 4 is also an IPTE per the procedure.

‘B’ is wrong – 3 is not a IPTE but a normal action for GGNS and 1 & 2 are IPTEs per the procedure

‘C’ is wrong – 3 is not a IPTE but a normal action for GGNS and 4 is listed per the procedure as an IPTE.

K/A Match

The candidate must have the knowledge of what is listed as IPTEs in the procedure.

Technical References:

EN-OP-116, IPTEs

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-PROC, OBJ. 44.5

Question Source:	**Bank # 179**	Not used on a NRC Exam
(note changes and attach parent)	**Modified Bank #**	
	New	
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.43(b)(7)	
Level of Difficulty:	3.0	

PRA Applicability:

None.

Examination Outline Cross Reference	Level	SRO
2.2.21 Knowledge of pre- and post-maintenance operability requirements.	**Tier**	3
	Group #	
	K/A	2.2.21
	Rating	4.1
	Revision	1

Revision Statement:

Rev 1, NRC Review and comments

Remove reference

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Question: 97

A small packing leak was discovered on P11-F130, REFUEL WTR XFER PMP SUCT FM SUPP POOL.

Maintenance has tightened the packing on the valve per a Work Order.

A maintenance leak check was performed and the leak was stopped.

What post-maintenance testing is required to be performed by Operations before the Work Order may be closed?

A. Functional stroke of valve IAW P11 SOI

B. Timed stroke of valve IAW P11 valve operability surveillance test

C. Functional stroke of valve with local visual observation IAW P11 SOI

D. Local leak rate test (LLRT) to verify Suppression Pool leakage within allowable limits IAW Engineering LLRT procedure

Answer: B

Explanation:

The candidate is expected to recognize that this is a safety–related primary containment isolation valve in the BOP condensate and refueling water transfer system.

Maintenance activities to tighten the packing of this valve represent a potential to affect the valve stroke time. A timed valve stroke per the appropriate surveillance procedure is required to demonstrate the valve meets the requirements of Tech Specs and the IST program.

‘B’ is correct.

Distracters:

‘A’ is wrong, but plausible. Although this will demonstrate functionality of the valve post–maintenance, Tech Specs and the IST program impose valve stroke time requirements that must be determined by a timed valve stroke performed IAW the applicable surveillance procedure.

‘C’ is wrong, but plausible. Although this will demonstrate functionality of the valve post–maintenance, Tech Specs and the IST program impose valve stroke time requirements that must be determined by a timed valve stroke performed IAW the applicable surveillance procedure and local observation is not required.

‘D’ is wrong, but plausible. Tightening of the valve packing does not present the potential to affect how leak–tight the valve is. While the original packing leak did present a concern for challenging limits on allowable Suppression Pool leakage into the Secondary Containment, a Maintenance leak check PMT is adequate to show that the leakage is stopped. A full LLRT of the penetration is not required.

K/A Match

The applicant must recognize the correct post maintenance test requirements

Technical References:

TS 3.6.1.3, Primary Containment Isolation Valves (PCIVs), Amendment 120
01–S–07–2, Test Control, Rev. 109, steps 2.6.1b, 5.12, 6.2.4b
EN-WM-107, Post Maintenance Testing, Attachment 3 Pg 2

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-OPS-PROC, OBJ. 27.2

Question Source:	**Bank # 556**	**Dec 2017 NRC Exam Q97**
(note changes and attach parent)	**Modified Bank #**	
	New	
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.43(b)(5)	
Level of Difficulty:	3.0	

PRA Applicability:
None.

SRO Only Justification:

This question requires the applicant to first determine this valve is a PCIV within a BOP system. Then determine retest requirements.

Examination Outline Cross Reference	Level	SRO
2.3.4 Knowledge of radiation exposure limits under normal or emergency conditions.	**Tier**	3
	Group #	
	K/A	2.3.4
	Rating	3.7
	Revision	0
Revision Statement:		

Question: 98

A Site Area Emergency is in progress.

A control room RO needs to be sent into the plant to perform a task.

Per 10-S-01-17, against what dose limits does the CRS (without additional concurrence) compare the current dose to determine if there is sufficient exposure margin?

A. 10CFR100 Reactor Site Criteria

B. GGNS Administrative Dose Limits

C. Authorized Emergency Exposure Dose Limits

D. 10CFR20 Standards for Protection Against Radiation

Answer: D
Explanation: See 10-S-01-17, sections 6.1.2 and 6.1.3. By definition, the administrative dose limits are automatically suspended and replaced by the 10CFR20 Federal dose limits at the declaration of an Alert emergency or higher. See 10-S-01-17, sections 6.2.1 and 6.7.1.b. The latter section clearly states that if the limit is likely to be exceeded (i.e., after determining if sufficient exposure margin exists), then obtain an Exposure Extension Authorization per section 6.1. Clearly then, the SRO compares the RO's current dose against the 10CFR20 Federal limits. 'D' is correct.
Distracters: 'A' is wrong – CFR 100 deals with Reactor Site Criteria.

‘B’ is wrong – the administrative dose limits are automatically suspended and replaced by the 10CFR20 Federal dose limits at the declaration of an Alert emergency or higher. ‘C’ is wrong - Section 6.1 deals with the limits suggested by this answer, these limits are for protecting equipment and life.
K/A Match
Technical References: 10-S-01-17, Emergency Personnel Exposure Control
Handouts to be provided to the Applicants during exam: NONE
Learning Objective: GLP-OPS-EPTS6, OBJ. 17

Question Source:	**Bank #**	2010 NRC Exam
(note changes and attach parent)	**Modified Bank #**	
	New	
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.43(b)(4)	
Level of Difficulty:	2.0	
PRA Applicability:		
None.		

Examination Outline Cross Reference	Level	SRO
2.4.29 Knowledge of the emergency plan.	Tier	3
	Group #	
	K/A	2.4.29
	Rating	4.4
	Revision	1

Revision Statement:

Rev 1, Per Ops Rep, added alarm FP LEVEL TROUBLE, this alarm is the basis of the EP-4 entry condition.

Question: 99

HANDOUT PROVIDED

The plant is in Day 5 of a Refueling Outage.

The Reactor cavity pool is drained and preps are being made to remove the Drywell Head.

New fuel is being transferred to the Containment Pool.

The Reactor Cavity gate seal fails.

P680-4A2-A6, FP LEVEL TROUBLE alarm is received

Fuel Pool Cooling and Cleanup pumps have tripped on low Drain Tank level.

RO reports ARM D21-K622, Aux Bldg Fuel Hdlg Area, indication is rising.

(1) What is the current emergency classification?

(2) Which procedure(s) should be entered to mitigate this event?

A. (1) Unusual Event.
(2) High Radiation During Fuel Handling ONEP only

B. (1) Alert.
(2) EP-4 and High Radiation During Fuel Handling ONEP

C. (1) Unusual Event.
(2) EP-4 and High Radiation During Fuel Handling ONEP

D. (1) Alert.
(2) High Radiation During Fuel Handling ONEP only

Answer: **C**

Explanation:

Per EAL flow charts Modes 4 through De-Fuel

UNUSUAL EVENT – AU2

UNPLANNED water level drop in reactor refueling pathways as indicated by water level drop in Upper Ctmt Pools, Aux Bldg Fuel Pools or the Fuel Transfer Canal, personnel observation or indication on area camera.

AND

VALID Area Radiation Monitor reading rises on any of the following:
Ctmt 209 Airlock --------------------------------- **(1D21K630)**
Ctmt Fuel Hdlg Area ---------------------------- **(1D21K626)**
Aux Bldg Fuel Hdlg Area ---------------------- **(1D21K622)**

The given information is lowering pool levels and a rising radiation on D21K622, therefore an Unusual Event is required.

When the Fuel Pool Cooling and Cleanup pump tripped on low drain tank level meant that the "scuppers" are uncovered due to the lowering level the scuppers are located below level of 207.49 ft. which is an entry condition for EP-4.

The CRS should enter EP-4 and also enter High Rad During Fuel Handling ONEP.

'C' is correct

Distracters:

'A' is wrong – EP-4 has met an entry condition and is required to be entered. Plausible due to the candidate must have the knowledge of relative levels of the Spent Fuel Pools vs. EP-4 entry condition.

'B' is wrong – Only an UE is declared, the upgrade to an ALERT requires level to reach the fuel which with the given information will not happen due to the fuel is recessed within the pool and level will not reach them and a high rad alarm which has not come in yet.

'D' is wrong – EP-4 has met an entry condition and is required to be entered. Plausible due to the candidate must have the knowledge of relative levels of the Spent Fuel Pools vs. EP-4 entry condition. Only an UE is declared, the upgrade to an ALERT requires level to reach the fuel which with the given information will not happen due to the fuel is recessed within the pool and level will not reach them and a high rad alarm which has not come in yet.

K/A Match
Candidate must have knowledge of the Emergency Play to correctly classify this event.

Technical References: 10-S-01-1, Activation of Emergency Plan EP-4 05-1-02-II-8, High Radiation During Fuel Handling
Handouts to be provided to the Applicants during exam: EAL Flow Charts
Learning Objective: GLP-OPS-EPTS6, OBJ. 1 GLP-OPS-EP-4

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.43(b)(7)	
Level of Difficulty:	3.0	
PRA Applicability:		
None.		

Examination Outline Cross Reference	Level	SRO
2.4.42 Knowledge of emergency response facilities.	**Tier**	3
	Group #	
	K/A	2.4.42
	Rating	3.8
	Revision	1

Revision Statement:

Rev 1, NRC Review and comments

Changed "will" to "should" in stem

Question: 100

After the Shift Manager is relieved, the Emergency Director reports to the ___________ facility.

What facility, after declared operational, should provide offsite notifications during an ERO emergency?

A. EOF
EOF

B. TSC
TSC

C. EOF
TSC

D. TSC
EOF

Answer: **A**

Explanation:

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

2.1.2 The Shift Manager assumes the role of Emergency Director upon initial classification of an emergency, and resumes normal Control Room duties when relieved by the EOF Emergency Director.

2.4.2 Assuming the duties of Emergency Director after the EOF is declared operational.

Per EN-EP-609, Emergency Operations Facility (EOF) Operations.

Offsite Communicator arrives at EOF and signs in to staffing board.

Assumes the position then obtain copies of all Emergency Notification forms that have been transmitted prior to EOF activation from the Control Room.

'A' is correct.

Distracters:

'B' is wrong – The Emergency Director responded to the TSC at one time but now, he responds to the EOF. The Emergency Plant Manager responds to the TSC

The TSC does have a Communicator but the positon only communicates with the other facilities not the state and local agencies.

'C' is wrong – The TSC does have a Communicator but the positon only communicates with the other facilities not the state and local agencies.

'D' is wrong - The Emergency Director responded to the TSC at one time but now, he responds to the EOF. The Emergency Plant Manager responds to the TSC

K/A Match

To understand the roles and responsibilities of ERO personnel the SRO must have the knowledge of all facilities within the ERO.

Technical References:

10-S-01-1, Activation of the Emergency Plan
EN-EP-609, Emergency Operations Facility (EOF) Operations
EN-EP-610, Technical Support Center (TSC) Operations

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

GLP-EP-EPTS26, OBJ.

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

10CFR Part 55 Content:	55.43(b)(5)	
Level of Difficulty:	3.0	
PRA Applicability:		
None.		

NRC INITIAL LICENSED OPERATOR WRITTEN EXAMINATION
OPEN-REFERENCES TABLE OF CONTENTS
SRO EXAM

TAB	PROVIDED HANDOUTS
1	05-S-01-EP-1, Emergency/Severe Accident Procedure Support Documents (Rev 039), HCTL Figure (Figure 1)
2	05-S-01-EP-1, Emergency/Severe Accident Procedure Support Documents (Rev 039), HDOL Figure (Figure 5)

Additional References Provided Outside of This Binder:

- Electrical Drawing E-1173-19
- Emergency Classification Flowcharts: (10-S-01-1, EPP 01-02), dated 11-21-2013
    - Modes 1 through 3 (page 1 of 2)
    - Modes 4 through De-Fuel (page 2 of 2)
- Steam Tables

TAB 1

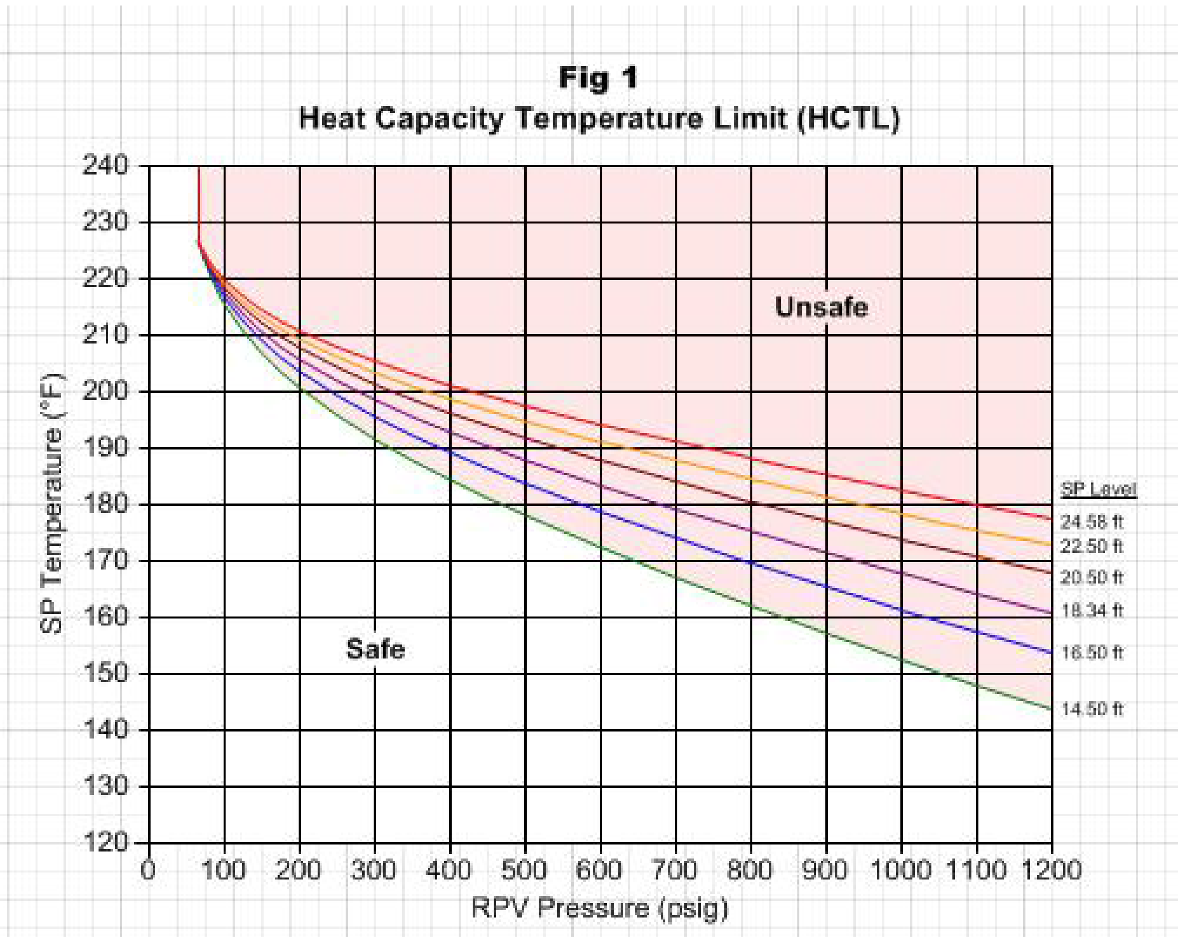
Fig 1
Heat Capacity Temperature Limit (HCTL)
Unsafe
Safe
SP Temperature (°F)
RPV Pressure (psig)
240
230
220
210
200
190
180
170
160
150
140
130
120
0
100
200
300
400
500
600
700
800
900
1000
1100
1200
SP Level
24.58 ft
22.50 ft
20.50 ft
18.34 ft
16.50 ft
14.50 ft

TAB 2

Figure 5
Hydrogen Deflagration Overpressure Limit (HDOL)