Examination Outline Cross-Reference	Level	RO
209001 Low Pressure Core Spray System **Knowledge of electrical power supplies to the following: K2.02 Valve power**	**Tier #**	2
	Group #	1
	K/A #	209001 K2.02
	Rating	2.5

Question 1

The breaker which connects electrical power to the motor (for motive force) on LPCS Injection Valve E21-F005 is located at…

A. 15AA

B. 15BA1

C. 15B11

D. 15P11

Answer: C

Explanation:

See the LPCI SOI, Attachment III, page 1 of 2 (Electrical Lineup). 15AA supplies 4160 VAC to the pump motor. 15BA1 is the 480 VAC LCC (not shown explicitly on Attachment III) that feeds the 480 VAC MCC 15B11. It is 15B11 that directly powers the motor operator for E21-F005. 15P11 supplies 120 VAC to the motor heater.

For these reasons, only ‘C’ is correct.

Technical References:

04-1-01-E21-1 LPCS SOI

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-E1200, OBJ. 7.1

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

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

Examination Outline Cross-Reference	Level	RO
300000 Instrument Air System (IAS) **Knowledge of the connections and / or cause effect relationships between INSTRUMENT AIR SYSTEM and the following: K1.05 Main Steam Isolation Valve air**	**Tier #**	2
	Group #	1
	K/A #	300000 K1.05
	Rating	3.1

Question 2

Within the drywell, which of the following describes how air gets from Instrument Air System piping to the control unit for a given MSIV?

From the piping…

A. directly to the control unit

B. through an air dryer and then to the control unit

C. to an air-driven compressor which discharges to the control unit

D. to an accumulator common to all 4 MSIVs and then to the control unit

Answer: A

Explanation:

See GLP-OPS-B1300, pages 31-32 and GFIG-OPS-B1300, Figure 29. References clearly show that arriving Instrument Air connects <u>directly</u> to the control unit. The control unit is the combination of solenoid-operated valves that are used to "control" the routing of the air to the valve's operating cylinder. As the arriving Instrument Air connects to the control unit, it <u>also</u> fills a dedicated accumulator (one per MSIV, not shared); that accumulator is there as a backup supply to provide one closing stroke of the MSIV in the event of an Instrument Air loss. Thus, 'A' is correct and 'D' is wrong.

'B' is wrong; there is no air dryer. It's especially plausible to the Candidate who is unsure of this arrangement, yet considers the importance of supplying dry air to a safety-related stroking component like an MSIV.

'C' is wrong; it's especially plausible to the Candidate who confuses the MSIV / Instrument Air interface with the ADS-SRVs/Instrument Air interface (i.e., where there are air-driven ADS Booster Compressors).

Technical References:

GLP-OPS-B1300; GFIG-OPS-B1300 FIGURE 29

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-N1136, OBJ. 9.5

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

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

Question 3

Due to heavy smoke, operators are forced to evacuate the control room before the reactor can be manually scrammed.

What method are the operators directed to use, initially, to shut down the reactor?

A. Start SLC locally.

B. Open the RPS bus breakers CB-2's and CB-8's.

C. Manually initiate ARI at local panel 1H22-P076.

D. Open the ESF inverter breakers that power the RPS Trip Channels.

Answer: B

Explanation:

Per the Remote Shutdown ONEP, section 3.3.1, only choice 'B' is correct.

'A' is wrong. Although SLC can be injected locally, this is not the initial method, prescribed by procedures, for shutting down the reactor.

'C' is wrong. ARI can only be tested at local panel P076, not initiated.

'D' is wrong. See the explanation for correct choice 'B'.

Technical References:

05-1-02-II-1, Remote S/D ONEP

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-ONEP, OBJ. 2.0

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

Examination Outline Cross-Reference	Level	RO
2.2.7 Knowledge of the process for conducting special or infrequent tests.	**Tier #**	3
	Group #	
	K/A #	2.2.7
	Rating	2.9

Question 4

Per EN-OP-116, Infrequently Performed Tests or Evolutions (IPTEs), every IPTE must have a “Controlling Document” with which to actually perform the test or evolution.

Controlling Document types include Work Order Instructions, existing Plant Procedures, and…

A. Engineering Change Instructions.

B. Performance Engineering Instructions.

C. Special Test Instructions.

D. Special Process Instructions.

Answer: C

Explanation:

Per EN-OP-116, section 3.0[7], only ‘C’ is correct.

‘A’ is wrong; has strong face plausibility.

‘B’ is wrong; this is a partial label for the series “17” procedures at GGNS.

‘D’ is wrong; these are a subset of the series “07” procedures at GGNS.

Technical References:

EN-OP-116, IPTEs

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-PROC, OBJ. 80.0

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

Examination Outline Cross-Reference	Level	RO
295038 High Off-Site Release Rate **Knowledge of the reasons for the following responses as they apply to HIGH OFF-SITE RELEASE RATE: EK3.02 System isolations**	**Tier #**	1
	Group #	1
	K/A #	295038 EK3.02
	Rating	3.9

Question 5

Which of the following is a reason for the automatic isolation feature of normal Secondary Containment Ventilation?

A. Prevent the release of airborne radioactivity beyond ODCM limits from within the Turbine Building.

B. Prevent untreated airborne radioactivity from being released to the outside.

C. Ensure Auxiliary Building d/p is maintained negative.

D. Ensure Auxiliary Building d/p is maintained positive.

Answer: B

Explanation:

“Normal” Secondary Containment ventilation is actually the combined Aux Bldg Ventilation (T41) and Fuel Handling Area Ventilation (T42) systems. These auto-isolate (and SGTS auto-initiates) on the associated vent exhaust high-high radiation conditions. In doing so, this isolation feature “prevents untreated airborne radioactivity from being released to the outside environment.” This implies that although T41/T42 releases are untreated, SGTS is treated before its release. Therefore, ‘B’ is correct.

‘C’ and ‘D’ are wrong mainly because they do not speak directly to the fundamental “reason for the automatic isolation” described in the stem. They provide sufficient plausibility based on the fact that both the “normal” ventilation systems and SGTS also serve to maintain a negative building d/p.

‘A’ is wrong because the Turbine Building is not part of the Secondary CTMT boundary.

Technical References:

GLP-OPS-T4100; GLP-OPS-T4200; GLP-OPS-T4800; EP-2; EP-4

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-T4200, OBJ. 4A

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

Examination Outline Cross-Reference	Level	RO
295001 Partial or Complete Loss of Forced Core Flow Circulation **Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION: AA1.07 Nuclear boiler instrumentation system**	**Tier #**	1
	Group #	1
	K/A #	295001 AA1.07
	Rating	3.1

Question 6

The plant is operating at rated power when Recirc Pump ‘A’ trips.

The plant is now stable again, including Recirc Loop flows and Total Core Flow.

To determine actual Total Core Flow, the operator…

A. uses the Total Core Flow indicated on the P680 recorder, as is.

B. sums the Loop ‘A’ and Loop ‘B’ jet pump loop flows.

C. subtracts Loop ‘A’ jet pump flow from Loop ‘B’ jet pump flow.

D. subtracts Loop ‘A’ jet pump flow from the Total Core Flow indicated on the P680 recorder.

Answer: A

Explanation:

Per 04-1-01-B33-1, section 3.18, only when Total Core Flow (on the P680 recorder) is indicating less than 38 Mlbm/hr (~34% of rated) is the recorder indication inaccurate due to jet pump reverse flow. In that case, Total Core Flow would be determined by manually summing the individual loop jet pump flows. This is what answer choice ‘B’ represents. However, this question’s stem has the plant operating at rated power before the trip of Recirc Pump ‘A’. The Candidate is therefore expected to recognize that a single Recirc Pump trip from these rated conditions would leave Total Core Flow still well above 34% of rated (well above 38 Mlbm/hr). As such, the operator would determine Total Core Flow simply by reading the P680 recorder, as is.

For the reasons above, ‘A’ is correct and ‘B’ is wrong.

‘C’ is wrong. There is no Recirc loop configuration (either running-running or running-idle) where core flow is determined by finding the <u>difference</u> between the two jet pump flows in each loop.

‘D’ is wrong. See Lesson Plan pages 12-13. Flow summer B33-FY-K613 has already subtracted out total jet pump flow signal coming from loop ‘A’ (the idle loop). Only loop ‘B’ jet pump flow is being shown on the Total Core Flow recorder. To suggest that the operator substract loop ‘A’ jet pump flow from the recorder flow indication would be to suggest that the loop ‘A’ flow be subtracted twice.

Technical References:

04-1-01-B33-1, Recirculation System SOI
GLP-OPS-B2104, RPV Flow Instrumentation

References to be provided to applicants during exam: None

Learning Objective:

GLP-OPS-B3300, Obj. 8, 38.1
GLP-OPS-B2104, Obj. 5.0

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

Examination Outline Cross-Reference	Level	RO
295027 High Containment Temperature **2.4.18 Knowledge of the specific bases for EOPs**	**Tier #**	1
	Group #	1
	K/A #	295027 2.4.18
	Rating	3.3

Question 7

Per 02-S-01-40, EP Technical Bases, which of the following is a basis for Emergency Depressurizing the RPV before CTMT temperature reaches 185°F?

A. Ensure the RPV is depressurized while the SRVs are still functional.

B. Ensure the RPV is depressurized before CTMT temperature gets high enough to damage the CTMT.

C. Limit the release of energy into CTMT in order to keep from exceeding the CTMT temperature LCO limit.

D. Limit the release of energy into CTMT in order to preserve the pressure suppression capacity of the suppression pool.

Answer: B

Explanation:

See EP Tech Bases, Attachment VI, page 17 of 34, bottom-most paragraph. From this reference, clearly 'B' is correct.

'A' is wrong. It suggests the basis for ED before exceeding a DW temperature of 330°F (see Attachment VI, page 12 of 34).

'C' is wrong. The Tech Spec LCO limit for CTMT temperature is 95°F (see Attachment VI, page 13 of 34).

'D' is wrong, but has sufficient face plausibility for the uncertain Candidate.
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Technical References:

02-S-01-40, EP Technical Bases

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-EP3, OBJ. 7

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

Examination Outline Cross-Reference	Level	RO
215003 Intermediate Range Monitor (IRM) System **Knowledge of the effect that a loss or malfunction of the following will have on the INTERMEDIATE RANGE MONITOR (IRM) SYSTEM: K6.04 Detectors**	**Tier #**	2
	Group #	1
	K/A #	215003 K6.04
	Rating	3.0

Question 8

A reactor startup is in progress with all IRMs on Range 5.

A dry tube leak results in a water-filled (shorted) detector for IRM F.

What is the plant/system response?

A. RPS ‘A’ half-scram and rod block

B. Rod block, only

C. RPS ‘B’ half-scram and rod block

D. Alarms, only

Answer: C

Explanation:

A water-filled detector is essentially a “saturated” detector to the IRM F instrument; i.e., sends a MAX (upscale) signal to the drawer. This results in both a rod block (108/125 scale) and an RPS ‘B’ (in the case of IRM F) half-scram (120/125 scale). Therefore, only choice ‘C’ is correct.

Technical References:

GLP-OPS-C5102, IRM System

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-C5102, OBJ. 21

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

Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	 X
10CFR Part 55 Content:	55.41(b)(7) 55.43.	

Examination Outline Cross-Reference	Level	RO
271000 Offgas System **Knowledge of the effect that a loss or malfunction of the OFFGAS SYSTEM will have on following:** **K3.01 Condenser vacuum**	**Tier #**	2
	Group #	2
	K/A #	271000 K3.01
	Rating	3.5

Question 9

The plant is operating at rated power with Offgas Train ‘A’ in service.

A major tube rupture occurs inside Offgas Condenser ‘A’.

Which of the following parameters **LOWERS** in response to this event?

A. Offgas Preheater ‘A’ Inlet Pressure

B. Offgas Condenser ‘A’ Level

C. Offgas Condenser ‘A’ Outlet Temperature

D. Main condenser vacuum

Answer: D

Explanation:

See the following: P&ID M-1092A, and ARIs P845-1A-A6, B1, and C6.

TBCW flows through the Offgas Condenser tube-side. Thus, a tube rupture results in a flood-up (high level) on the condenser shell-side (at much lower Offgas system pressure). For this reason, ‘B’ is wrong.

‘A’ is wrong for the same reason. See ARI P845-1A-B1, which shows that Preheater Inlet Pressure goes high (not low) as a result of a flooded Offgas condenser.

‘C’ is wrong. This is not quite so obvious as the first two. See ARI P845-1A-C6. When the condenser floods, TBCW recirculation (i.e., heat transfer of BTUs away from the condenser) essentially stops; the “pool” of water in the shell-side becomes ineffective, causing a rise in the Offgas outlet temperature (i.e., less cooling of the Offgas stream through the condenser shell).

‘D’ is correct. The reason is simply due to the “back-pressure” placed on the Offgas stream flow felt all the way back through the Offgas system to the Condenser Air Removal System (SJAEs) as a result of the flooded condenser. This is the mechanism for a main condenser degrading (lowering) vacuum.

Technical References:

M-1092A; ARIs P845-1A-A6, B1, C6

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-N6465, OBJ. 25

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

Examination Outline Cross-Reference	Level	RO
209002 High Pressure Core Spray System (HPCS) **Ability to monitor automatic operations of the HIGH PRESSURE CORE SPRAY SYSTEM (HPCS) including: A3.04 System flow**	**Tier #**	2
	Group #	1
	K/A #	209002 A3.04
	Rating	3.7

Question 10

From its normal standby alignment HPCS has automatically initiated on a valid initiation signal.

Which of the following identifies the control room panel P601 indications that could be used to confirm HPCS injection into the reactor?

A. Only injection valve position can be used.

B. Only testable check valve position can be used.

C. Only injection valve position or pump discharge flow can be used.

D. Only testable check valve position or pump discharge flow can be used.

Answer: D

Explanation:

See M-1086.

E22-F004 Injection Valve position says nothing about status of injection flow; the pump could be tripped or running only on min flow while F004 is still indicating open as a result of an initiation signal. For this reason, choices ‘A’ and ‘C’ are wrong.

The only way to physically open (i.e., move the disc) the testable check valve, E22-F005, is by way of passing flow through it...on the way directly into the RPV. Additionally, the Pump Discharge Flow element (feeding FI-R603 on P601) is located on the HPCS discharge piping where it sees both injection flow to the RPV as well as test return line flow to the supp pool. There is no test return line flow (i.e., E22–F010/F011 and E22-F023 valves are closed in standby lineup and received an auto-close signal on the initiation); therefore, all indicated flow on R603 is evidence of RPV injection flow. For this reason, choice ‘D’ is correct and ‘B’ is wrong.

Technical References:

M-1086, HPCS P&ID

References to be provided to applicants during exam: None

Learning Objective: GLP–OPS–E2201, Obj. 4, 5.6, 8.3, 12, 20, 23

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

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

Question 11

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

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

B. RCIC steam supply line cannot be isolated.

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

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

Answer: D

Explanation:

‘A’ is wrong because the SLC squibs are 120 VAC powered via the 480 VAC MCC breaker powering the associated SLC pump; they have no reliance on DC power, either directly or indirectly.

‘B’ is wrong. RCIC steam line has two supply isolations, E51-F064 (Div 1) which uses 11DA for Div 1 isolation logic, and E51-F063 (Div 2) which uses 11DB for Div 2 isolation logic. So long as 11DB is available, the F063 valve will close to effect a successful RCIC steam line isolation.

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

‘D’ is correct. 11DA (Div 1) powers the normally de-energized solenoid for one of the 3 Backup Scram Valves (the other two are Div 2 powered from 11DB). Without 11DA, the Div 1 valve cannot reposition. <u>All 3 Backup Scram Valves must reposition</u> in order to depressurize the scram air header via the Backup Scram Valves.

Technical References:

E-1023
E-1185
E-1257

E-1169

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-L1100, OBJ. 17

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

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

Question 12

Which of the following is the electrical power supply to the LPCI initiation logic for RHR ‘C’?

A. 1Y88

B. 1Y95

C. 11DB

D. 11DC

Answer: C

Explanation:

RHR ‘C’ is a Division 2 ECCS system.

LPCI initiation logic is powered from ESF 125V DC power.

Answers ‘A’ and ‘B’ identify the ESF Division 2 120V AC inverters for the ‘B’ and ‘D’ logic channels of NSSSS and other systems and are therefore wrong.

‘C’ is the ESF Division 2 125V DC power bus and is therefore the correct answer.

‘D’ is the ESF Division 3 125V DC power bus, which is wrong.

Technical References:

E-1181, Sheets 66, 68
04-1-01-E12-1, Att. IIIB

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-E1200, Obj. 9.2

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

Examination Outline Cross-Reference	Level	RO
295005 Main Turbine Generator Trip **Ability to operate and/or monitor the following as they apply to MAIN TURBINE GENERATOR TRIP: AA1.02 RPS**	**Tier #**	1
	Group #	1
	K/A #	295005 AA1.02
	Rating	3.6

Question 13

Immediately after a controlled reactor power change, the following P680 annunciator has cleared:

- TURB CV/SV CLOSE TRIP BYP

Which of the following are included among the Immediate Operator Actions if an automatic turbine trip were to occur?

A. Place the mode switch in SHUTDOWN.
Run the MHC Start Device to zero to close the Turbine Stop and Control Valves.

B. Ensure both generator output breakers are open.
Run the MHC Start Device to zero to close the Turbine Stop and Control Valves.

C. Place the mode switch in SHUTDOWN.
Ensure Turbine Stop and Control Valves are closed.

D. Confirm reactor remains operating with reactor power stabilized.
Ensure Turbine Stop and Control Valves are closed.

Answer: C

Explanation:

'A' and 'B' are wrong because of the second part, which is a Subsequent Action of the Turbine Trip ONEP required only in the event that the turbine has not tripped, a condition which is contrary to the stem condition for this question ("automatic turbine trip were to occur").

'C' is correct. This annunciator cleared after a power rise above the Turbine Trip Scram enabled setpoint of 27%. Thus an automatic turbine trip results in a reactor scram, requiring entry into both ONEPs…Scram and Turbine Trip. The first part of the answer choice comes from the IOAs for the Scram ONEP (and satisfies this KA…to operate RPS; i.e., the mode switch placed in SHUTDOWN). The second part of the answer comes from the IOAs of the Turbine Trip ONEP…the turbine valves should auto-close.

‘D’ is wrong because of the first part which suggests the reactor should not have scrammed. This would be the case if the annunciator had alarmed and sealed-in, rather than having cleared.

Technical References:

Scram ONEP; Turbine Trip ONEP

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-ONEP, OBJ. 1.0

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

Examination Outline Cross-Reference	Level	RO
295026 Suppression Pool High Water Temperature **Ability to determine and/or interpret the following as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE: EA2.02 Suppression pool level**	**Tier #**	1
	Group #	1
	K/A #	295026 EA2.02
	Rating	3.8

Question 14

Operators are in EP-3, Containment Control, with a rising suppression pool temperature.

SPDS and PDS are not working.

You have been directed to peer check a point on HCTL Figure 1.

Indicated suppression pool level is 18.0 feet, which is between the 16.5 feet and 18.34 feet curves provided on Figure 1.

Per 02-S-01-27, Operations Philosophy, which suppression level is to be used to determine the point on HCTL Figure 1?

A. 18.34 feet

B. 18.0 feet

C. 16.5 feet

D. 14.5 feet

Answer: C

Explanation:

NOTE – Regarding KA Match – We've chosen to focus this question on the KA in the following way: Demonstrate the **ability to determine** where operators should **conservatively** recognize **suppression pool level** to be for a given **suppression pool temperature, and use the same for plotting the margin to HCTL**, when the indicated suppression pool level falls in between two curves drawn on HCTL Figure 1. We consider this not only a best-fit KA match, but one that is extremely important with respect to operational validity.

See Ops Philosophy, section 6.1.7. When actual level falls in between two provided curves, default to using the lower curve to plot the HCTL point. In this case, the lower curve is 16.5 feet…only answer 'C' is correct. For information: 14.5 feet is the lowest level curve provided on Figure 1; thus, provides sufficient plausibility as a 4th answer choice.

NOTE – Classifying this as a Higher Cognitive item because it is presented as an “application” of one procedure’s criteria (Ops Philosophy directive to use the “lowest…curve”) to a specific situation (that given in the stem), requiring the Candidate to deduce that the 16.5 feet curve applies in this particular case.

Technical References:

HCTL Figure 1; 02-S-01-27, Operations Philosophy

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-PROC, OBJ. 57.8

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

Examination Outline Cross-Reference	**Level**	RO
295007 High Reactor Pressure	**Tier #**	1
	Group #	2
2.2.42 Ability to recognize system parameters that are entry-level conditions for Technical Specifications.	**K/A #**	295007 2.2.42
	Rating	3.9

Question 15

Which of the following requires a Tech Spec LCO entry (i.e., <u>actual</u> implementation of a Required Action)?

With the plant operating at 90% power…

A. recirc loop flows are found to be mismatched by 3% of rated core flow.

B. reactor pressure rises and stabilizes at 1048 psig due to a pressure regulator problem.

C. the Baxter-Wilson transmission line connection to GGNS is taken out of service.

D. one SRV (not an ADS valve) is declared inoperable for its Relief function.

Answer: B

Explanation:

'A' is wrong. Per SR 3.4.1.1, the mismatch limit is 5% when operating >70% of rated core flow.

'B' is correct per LCO 3.4.12. The limit for LCO entry is 1045 psig in Modes 1 and 2.

'C' is wrong per LCO 3.8.1.a. Only two of the three (Baxter-Wilson, Franklin, Port Gibson) qualified AC offsite circuits are required to be OPERABLE.

'D' is wrong per LCO 3.4.4. This LCO is very generous (for non-ADS function) before any LCO entry is required. Of the 20 total SRVs, the Safety function of only 7 is required to be OPERABLE, and the Relief function of only an additional 6 is required. Therefore, the choice 'D' suggestion for a single non-ADS SRV is far removed from the limit for an LCO 3.4.4 entry.

Technical References:

SR 3.4.1.1; LCO 3.4.12; LCO 3.8.1; LCO 3.4.4

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-B2102, OBJ. 11

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

Examination Outline Cross-Reference	Level	RO
233000 Fuel Pool Cooling and Clean-up **Ability to (a) predict the impacts of the following on the FUEL POOL COOLING AND CLEAN-UP ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:** **A2.03 Low surge tank level/high level**	**Tier #**	2
	Group #	2
	K/A #	233000 A2.03
	Rating	2.8

Question 16

A refueling outage is in progress.

Fuel offload is about to commence when the in-service shutdown cooling system trips.

Operators are maximizing FPCC flow per the Inadequate Decay Heat Removal ONEP.

The second FPCC Pump is available in standby.

Which of the following describes the action required before starting the second FPCC Pump?

A. Isolate flow to the Upper Containment Pool.

B. Ensure the heat exchanger bypass valve is fully open.

C. Ensure FPCC drain tank level is <33%.

D. Ensure FPCC drain tank level is >90%.

Answer: D

Explanation:

G41 SOI requirements regarding start of a standby FPCC pump are required to be followed, as the Inadequate Decay Heat Removal ONEP directs “maximizing” FPCC flow.

Prerequisites for start of a FPCC pump is that fuel pool drain tank level is greater than 90% (Note to Step 5.15.2.a(5)). The drain tank high-level requirement is to provide adequate margin to the pump trip on low drain tank level of 17%. The candidate must be able to predict the effects of a pump start, that inventory is promptly removed from the drain tank on the pump start while return flow is significantly delayed, resulting in a lowering drain tank level trend. A pump trip on low tank level is expected if drain tank level is not high enough before pump start.

‘D’ is the correct answer as it meets the procedural requirements and requires accurate prediction of the change in drain tank level on start of a pump.

‘C’ is wrong; drain tank level will be inadequate to prevent pump trip at this low level.

‘A’ is wrong, but represents a misunderstanding of SOI P&L 3.28, which requires only 1 pump in service to valve in/out the Upper Containment Pools. It does not require a 2nd pump to be started while aligned to the SFP only.

‘B’ is wrong; bypassing a heat exchanger is not required, although bypassing any in–service filter/demins would be desired.

Technical References:

04-1-01-G41-1
05-1-02-III-1
04-1-01-E12-2

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-G4100 Obj. 7.2, 25
GLP-OPS-ONEP Obj. 20

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

Examination Outline Cross-Reference	Level	RO
295034 Secondary Containment Ventilation High Radiation **2.4.4 Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures.**	**Tier #**	1
	Group #	2
	K/A #	295034 2.4.4
	Rating	4.5

Question 17

Which of the following control room P601 annunciators (if valid) is, by itself, an entry condition for EP-4, Auxiliary Building Control?

A. FP EXH RAD HI

B. FH AREA EXH RAD HI

C. FH AREA VENT RAD HI-HI / INOP

D. FP EXH DIV 1, 4 RAD HI-HI / INOP

Answer: D

Explanation:

See EP-4, Table 10 and the entry conditions listed at the top of the flowchart.

Of the answer choices, only 'D' is a Table 10 EP-4 entry condition Secondary CTMT Ventilation High Rad entry condition-related alarm (i.e., annunciator P601-19A-B10), with an alarm setpoint of 30 mr/hr.

Although choices 'A' and 'B' are P601 annunciators related to the same parameters, they do not correspond to EP-4 entries; therefore, they are wrong.

Answer choice 'C' is wrong. The FH AREA VENT RAD HI–HI / INOP alarm is referenced by EAL AA2 and is indicative of an offsite rad release, which could potentially satisfy the EP–4 entry condition for an Alert due to offsite rad release. However, the Candidate cannot assume this Alert declaration given no information for such in the question stem. Declaring an Alert from AA2 for abnormal rad levels in the Sec CTMT in order to satisfy the EP–4 entry condition would require a concurrent fuel handling accident or fuel pool inventory loss. Similarly, assumption of EAL AA1 is inappropriate; to meet AA1 the rad levels must be significantly greater than the alarm setpoint (200 times) and for an extended duration (15 minutes)…again no such stem information.

NOTE - While these Secondary CTMT Vent Exhaust rad monitor readings (in mr/hr) are readily available on the PDS display at P680 (as well as P844), we've chosen, instead, to capture this KA by way of "recognizing the P601 annunciator…as an EP-4 entry required situation".   This more accurately depicts how we train both our ILT Candidates and our incumbent control room operators at GGNS in this important task. What's more, presenting the question in the form of annunciator recognition provides better discrimination validity (given the set of answer choices) than would a set of 4 choices that target more-easily distinguished parameter values.

Technical References:

EP-4
ARIs P601-19A-B10, C11, D9 and D10
EAL flowcharts

References to be provided to applicants during exam: None

Learning Objective: GLP–OPS–EP4, Obj. 5

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

Examination Outline Cross-Reference	Level	RO
204000 Reactor Water Cleanup System **Knowledge of REACTOR WATER CLEANUP SYSTEM design feature(s) and/or interlocks which provide for the following: K4.03 Over temperature protection for system components**	**Tier #**	2
	Group #	2
	K/A #	204000 K4.03
	Rating	2.9

Question 18

Which of the following describes the design feature that protects the resin in the RWCU Filter-Demins from overheating?

When Non-Regenerative Heat Exchanger outlet temperature reaches 140°F…

A. RWCU Pump Suction Valve G33-F004 automatically closes.

B. RWCU F/D Outlet Valves G36-F004A/B automatically close.

C. RWCU Filter-Demin Bypass Valve G33-F044 automatically opens.

D. RWCU Blowdown Flow Control Valve G33-F033 automatically opens.

Answer: A

Explanation:

See ARI P680-11A-C6 and lesson plan GLP-OPS-G3336, page 26 of 41. Only ‘A’ is correct.

‘B’ is wrong. These F/D valves are controlled only by the MODICON programmable logic controller that controls the F/D backwash/precoat sequencing.

‘C’ is wrong. The F044 valve is an MOV that has no automatic features; it is only manually controlled at P680.

‘D’ is wrong. Although the F033 (once manually opened at P680) does have an auto-close feature, it has no auto-open feature and is in no way associated with NRHX outlet temperature.

Technical References:

GLP-OPS-G3336; ARI P680-11A-C6

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-G3336, OBJ. **8.6**

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

Examination Outline Cross-Reference	Level	RO
2.3.13 Knowledge of radiological safety procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.	**Tier #**	3
	Group #	
	K/A #	2.3.13
	Rating	3.4

Question 19

A refueling outage is in progress.

Fuel is being offloaded from the reactor vessel to the Upper Containment and Spent Fuel pools.

CONTAINMENT FUEL HDLG AREA RADIATION HIGH annunciator comes into alarm at P844.

What is the expected control room operator response to this alarm?

A. Contact the Refuel Floor SRO to determine if an evacuation is required.

B. Ensure CTMT ventilation isolates.

C. Isolate the Upper Containment Pools from the FPCC system.

D. Initiate CRFA system and isolate Control Room ventilation.

Answer: A

Explanation:

Receipt of this alarm during fuel handling activities in the Containment is indicative of a potentially significant radiological safety concern in the Containment due to a fuel handling accident or fuel pool inventory loss. Prompt action by the operator is required to ensure the safety of personnel in the Containment. The ACRO maintains constant contact with the Refueling SRO during core alterations and can rapidly determine the severity of the event. 'A' is the correct answer.

B is wrong. This is a D21 area radiation monitor not associated with the primary containment ventilation isolation logic, which is driven by D17 process radiation monitors required by TS 3.3.6.1. Additionally, primary containment operability is not required in Mode 5, so containment ventilation isolation would be ineffective in limiting the potential spread of radioactivity out of CTMT.

C is wrong. This answer choice suggests Subsequent Action 3.5.2 of the High Radiation

During Fuel Handling ONEP. Even if the operators were to enter that ONEP solely based on the single annunciator (in the stem) being a sufficient "symptom" to do so (see ONEP Symptom 4.1 on page 2 of the ONEP), the 3.5.2 action is not required because the question's stem conditions do not give any indication of there being a release of fission products into any of the pools (see section 3.5).

D is wrong. There is no immediate operator action to initiate a train of CRFA based on this alarm. Concern over Control Room dose is appropriate due to the potential for a fuel handling accident, but determining the severity of the event and taking action to protect personnel in the Containment takes precedence.

Technical References:

04-1-02-1H13-P844 1A-A3
17-S-06-5
05-1-02-II-8

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-ONEP Obj. 1
GLP-OPS-D1721 Obj. 1, 15
GLP-OPS-EP04 Obj. 5

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b)(11,12)	
	55.43	

Examination Outline Cross-Reference	Level	RO
295037 SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown **Ability to determine and/or interpret the following as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN: EA2.03 SBLC tank level**	**Tier #**	1
	Group #	1
	K/A #	295037 EA2.03
	Rating	4.3

Question 20

An ATWS is in progress with reactor pressure at 900 psig.

Only SLC Pump ‘B’ has been started; SLC Pump ‘A’ is unavailable.

Starting SLC Tank level is 4,800 gallons.

How long should it take to inject enough boron to maintain the reactor shut down under all conditions?

(Assume reactor pressure remains at 900 psig; assume the SLC Pump flow rate is the Tech Spec minimum.)

Approximately…

A. 15 minutes

B. 29 minutes

C. 44 minutes

D. 88 minutes

Answer: D

Explanation:

NOTE – Regarding KA Match – We’ve chosen to focus this question on the KA in the following way: First, recognize the SLC Tank level at which the reactor is expected “remain shutdown under all conditions” (i.e., less than 1200 gallons remaining in the tank – Cold S/D Boron Weight); secondly, given that only one SLC Pump is available, **determine** the total time it should take to achieve that target **level,** which is **interpreted** as having achieved the desired state**: remaining shutdown under all conditions.** We feel this approach to the question is not only a great fit for the KA but makes this type of KA very “genuine” for the RO Candidate.

Per EP-2A, step P-5 and Table 7, when Tank level reaches 1200 gallons, the RO/SRO

Candidate should interpret this level as having injected the “Cold Shutdown Boron Weight” implicit (but not explicit) in the question itself. This is the “interpret” portion of the KA match. The running SLC Pump has to empty 3600 gallons from the tank (4800 – 1200 = 3600) to get to the Cold Shutdown Boron Weight condition. At a flow rate of 41 gpm (the approximate Tech Spec SR 3.1.7.7 minimum (41.2 gpm)), it should take about 88 minutes…making answer choice ‘D’ correct.

‘A’, ‘B’ and ‘C’ are wrong; they suggest variants based on either both SLC Pumps running and/or recalling a Tech Spec minimum pump flow rate that is too high (i.e., 82 gpm rather than 41 gpm).

NOTE – Although the SRO Candidates do have access to the EP-2A flowchart (for their SRO Exam requirements), they must still make the cognitive connection between EP-2A step P-5 (with Table 7) and the question itself (“maintain the reactor shutdown under all conditions”). Additionally, like the RO Candidates, the SROs must also recall the TS minimum flow rate spec for the SLC Pump.

Technical References:

EP-2A
TS SR 3.1.7.7

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-EP02A, OBJ. 3

Question Source:	Bank # GGNS-OPS-08038a	X
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b)(10)	
	55.43	

Examination Outline Cross-Reference	Level	RO
295023 Refueling Accidents **Knowledge of the interrelations between REFUELING ACCIDENTS and the following: AK2.04 RMCS/Rod control and information system**	**Tier #**	1
	Group #	1
	K/A #	295023 AK2.04
	Rating	3.2

Question 21

How does RC&IS prevent an inadvertent criticality during refueling operations in Mode 5 with the Mode Switch in REFUEL?

A. By having the Rod Pattern Controller enforce control rod movements in accordance with a prescribed sequence.

B. By inserting a rod withdrawal block if the Refueling Platform is moved over the core to remove a fuel assembly.

C. By inserting a rod withdrawal block if a rod is selected and fuel is loaded in that control cell.

D. By producing a rod selection block if an operator attempts to select a rod when a different rod is already withdrawn.

Answer: D

Explanation:

RC&IS is the system for normal positioning of the control rods. During refueling operations requirements for RC&IS operability are specified by TS 3.9.1 and TS 3.9.2.

'D' is correct. This is the One-Rod-Out interlock required to be operable by TS 3.9.2 to prevent withdrawal of more than one control rod during refueling.

'A' is wrong. The referenced approved spiral sequence refers to the fuel loading sequence required when complying with TS 3.10.6, for reloading the core after a full core offload when all control rods in de-fueled cells may not be inserted. Additionally, the spiral reload sequence is enforced administratively, no automatic actions or interlocks enforce the loading sequence.

'B' is wrong. A control rod withdrawal block is not enforced by RC&IS until the Refueling Platform is positioned over the core with a loaded hoist. The purpose of this interlock is to prevent loading fuel into a control cell where the control rod may be withdrawn.

'C' is wrong. RC&IS does not receive input for the status of fuel loading in a control cell.

Technical References:

TS 3.9.2 Bases
04-1-01-F11-1, Att. V
04-1-02-1H13-P680-4A2-C5

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-C1102 Obj. 13, 14.7, 15

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

Examination Outline Cross-Reference	Level	RO
Knowledge of facility requirements for controlling vital/controlled access.	**Tier #**	3
	Group #	
	K/A #	2.1.13
	Rating	2.5

Question 22

Refer to the below figure to answer this question.

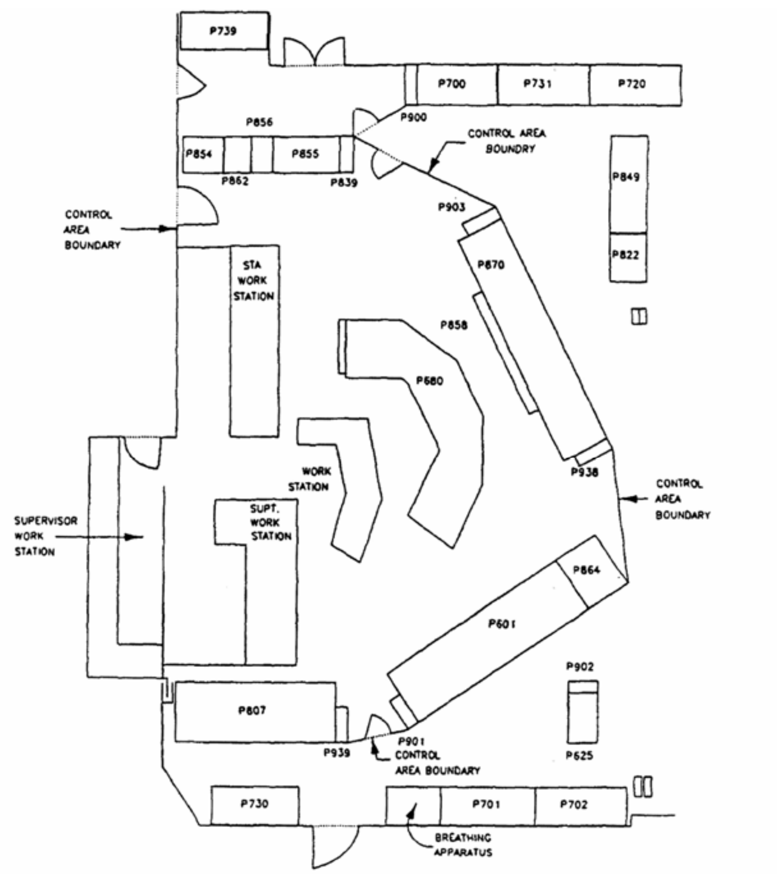
Per 01-S-06-4, Access and Conduct in the Control Room, who may authorize non-shift personnel entry into the area defined by the “Control Area Boundary”?

A. ACRO only

B. ACRO or CRS only

C. CRS or Shift Manager only

D. Any on-shift control room watch-stander



Answer: D

Explanation:

See 01-S-06-4 Step 6.3.2. Control Area is defined in Step 5.1 and Attachment I.

Only choice 'D' is correct.

Technical References:

01-S-06-4

References to be provided to applicants during exam: None

Learning Objective: GLP–OPS–PROC, Obj. 11.2

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

Examination Outline Cross-Reference	**Level**	RO
	Tier #	3
	Group #	
	K/A #	2.1.18
	Rating	3.6

Question 23

EN-OP-115, Conduct of Operations, groups control room annunciators into four categories: Expected; Unexpected; Repeat; Nuisance.

Per that procedure, how should the RO respond to a “Repeat” annunciator that is not due to testing or a planned evolution, when it alarms?

A. Announce it every time it alarms; reference the alarm response instruction every time it alarms.

B. Announce it every time it alarms; reference to the alarm response instruction is required only for the first time it alarms on each shift.

C. Announce it only the first time it alarms on each shift; reference the alarm response instruction every time it alarms.

D. Announce it only the first time it alarms on each shift; reference to the alarm response instruction is required only for the first time it alarms on each shift.

Answer: B

Explanation:

See EN-OP-1115, section 5.10[4](c). Per that section, only choice ‘B’ is correct.

Technical References:

EN-OP-115, Conduct of Operations

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-PROC, Obj. 8.8

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

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

Examination Outline Cross-Reference	Level	RO
295024 High Drywell Pressure **Ability to determine and/or interpret the following as they apply to HIGH DRYWELL PRESSURE:** **EA2.03 Suppression pool level**	**Tier #**	1
	Group #	1
	K/A #	295024 EA2.03
	Rating	3.8

Question 24

Refer to the below figure to answer this question.

Which of the following describes the relationship between suppression pool level and evidence of steam in the CTMT airspace during a LOCA?

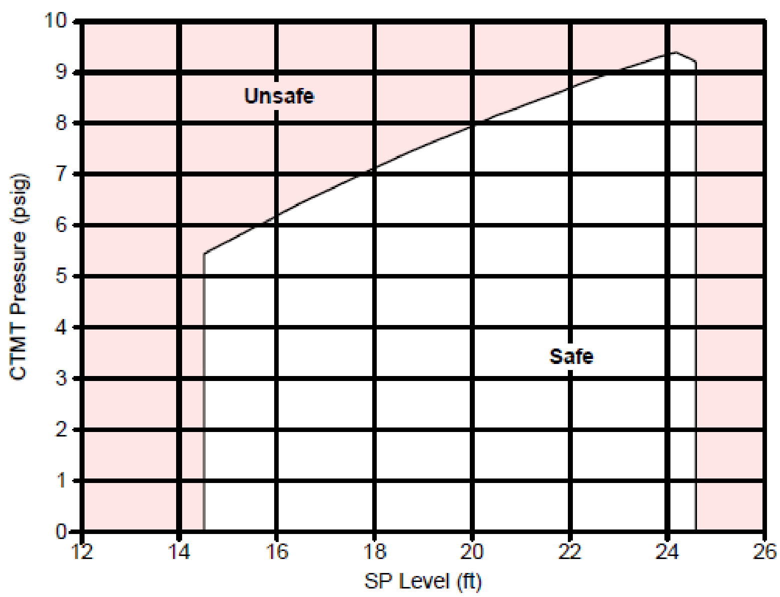
A. With CTMT pressure at 7 psig there is evidence at a pool level of 16 feet but no evidence at a pool level of 20 feet.

B. With any CTMT pressure there is no evidence at a pool level of 14 feet.

C. Opening 8 ADS SRVs at a pool level of 22 feet is more likely to produce such evidence than would opening the SRVs at a pool level of 15 feet.

D. With CTMT pressure at 7 psig there is evidence at pool levels above 24.3 feet.

Fig 4
Pressure Suppression Pressure (PSP)



Answer: A

Explanation:

NOTE – For GGNS (BWR-6) we're substituting CTMT Pressure for the Drywell Pressure specified in this KA. The CTMT Pressure substitution allows us to apply the Pressure Suppression Pressure (PSP) concept to this KA, making for an operationally valid RO-level exam question.

See EP Technical Bases (02-S-01-40), Attachment X, pages 34-36 of 45, for explanation of PSP Figure 4. For the sake of the question, the bottom-line is: PSP is a function of supp pool level and is used to ensure sufficient pressure suppression capacity for an ED while the RPV is at pressure. UNSAFE areas generally represent relationships between CTMT pressure and supp pool level where pressure suppression is in sufficient; therefore, steam in the CTMT airspace is more likely.

'A' is correct. Plotting these points on Figure 4 reveals we're already UNSAFE (i.e., evidence of steam in CTMT airspace) at 16 feet but still SAFE at 20 feet (i.e., no evidence).

'B' is wrong. In fact, just the opposite is true. At all CTMT pressures, we're UNSAFE at this low pool level of 14 feet. Levels to the left-side of the 14.5 vertical line represent (by analysis) pool levels where the RPV may not be kept in a pressurized state since steam discharged through the vents may not be condensed. Therefore, uncondensed (unsuppressed) steam may enter the CTMT airspace.

'C' is wrong. In fact, the opposite is true. As compared to 19 feet, a 22 feet pool level has that much more water volume with which to absorb the heat energy from the ED. Thus, it offers a greater pressure suppression capacity, not a lesser one…i.e., is less likely to produce an outcome of steam in the CTMT airspace post-ED than would an ED at a much lower pool level.

'D' is wrong. Although everything to the right-side of this 24.3 feet vertical line is UNSAFE it is not because of steam in the CTMT airspace concerns. This part of PSP Figure 4 focuses solely on the Maximum Pressure Suppression Primary CTMT Water Level. Above this 24.3 feet level, the pressure suppression function of CTMT itself cannot be assured.

Technical References:

EP-3, PC Control
EP Technical Bases (02-S-01-40)

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-EP3, OBJ. 7

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

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

Examination Outline Cross-Reference	Level	RO
295018 Partial or Complete Loss of Component Cooling Water	**Tier #**	1
Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER: AK1.01 Effects on component/system operations	**Group #**	1
	K/A #	295018 AK1.01
	Rating	3.5

Question 25

Which of the following would be an appropriate operator action to mitigate a reduction in CCW system flowrate while keeping the plant operating at power?

A. Isolate Drywell equipment drain sumps.

B. Isolate CCW flow to the Containment and Drywell.

C. Transfer both Reactor Recirc Pumps to SLOW speed.

D. Maintain RWCU cooled by CCW but with its Filter-Demins isolated.

Answer: C

Explanation:

'C' is the correct answer. Transferring the recirc pumps to slow speed will reduce load on the CCW system and has procedural guidance per the Loss of CCW ONEP, section 3.1.4

'A' is incorrect. There is no procedural guidance on isolation of DW equipment drains sumps. Isolating the sumps without affecting recirc pump operation is not possible without entering the drywell.

B is incorrect. This would be an appropriate action only on evidence of a break in the Containment or Drywell. Additionally, there is no way for the plant to remain operating at power with CCW to the CTMT/DW having been isolated. CCW ONEP section 3.3 directs a manual scram prior to the manual isolation.

D is incorrect. Isolating the RWCU filter-demins without also separating the RWCU system from CCW does nothing to reduce the heat load on CCW.

Technical References:

05-1-02-V-1

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-ONEP Obj. 1, GLP-OPS-P4200 Obj. 6.7, 10, 15

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

Examination Outline Cross-Reference	Level	RO
600000 Plant Fire On Site **2.1.30 Ability to locate and operate components, including local controls.**	**Tier #**	1
	Group #	1
	K/A #	600000 2.1.30
	Rating	4.4

Question 26

A fire has started in the 'B' RPS MG Set Room.

Automatic CO_2 has initiated but no CO_2 is going into the room.

The Fire Brigade Leader orders you to manually initiate the CO_2 system to flood the room with CO_2.

Which of the following describes one of the ways you can perform this task?

A. Near the 10-ton CO2 Storage Unit, remove the pin and place the Master Valve Manual Release Lever in the OPEN position. Outside the 'B' RPS MG Set Room, break the glass on the Electro-Manual Station and rotate the arm to the OPEN position for a specified time, then back to CLOSE.

B. Near the 10-ton CO_2 Storage Unit, locate the remote pushbutton station for the 'B' RPS MG Set Room; pull the handle on the station to break the glass and momentarily depress the pushbutton to initiate a timed release of CO_2.

C. Outside the 'B' RPS MG Set Room, break the glass on the Electro-Manual Station and momentarily rotate the arm to the OPEN position to initiate a timed release of CO_2.

D. Outside the 'B' RPS MG Set Room, pull the handle on pushbutton station to break the glass, and depress and hold the pushbutton for a specified time, then release.

Answer: A

Explanation:

See Cardox SOI, Section 5.2.

'A' is correct; it describes the Electro-Manual Initiation method of section 5.2.2.b.

'B' is wrong; there is no remote pushbutton station for room near the 10-ton storage unit; the station is local, outside the room (see section 5.2.2.a).

‘C’ is wrong; the Electro-Manual Station arm must be held in the OPEN position; there is no timed-release feature with this station (see section 5.2.2.b.2).

‘D’ is wrong; the local pushbutton station is a momentary pushbutton with a timed-release feature (see section 5.2.2.a.1).

Technical References:

04-1-01-P64-3, CARDOX SOI

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-P6400, OBJ. 6.6

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

Examination Outline Cross-Reference	Level	RO
295036 Secondary Containment High Sump/Area Water Level **Ability to operate and/or monitor the following as they apply to SECONDARY CONTAINMENT HIGH SUMP/AREAWATER LEVEL: EA1.01 Secondary containment equipment and floor drain systems**	**Tier #**	1
	Group #	2
	K/A #	295036 EA1.01
	Rating	3.2

Question 27

The plant scrammed due to a Feedwater leak in the Auxiliary Building Steam Tunnel.

Reactor water level was restored automatically by HPCS and RCIC.

An operator is controlling reactor water level in the appropriate band using RCIC and CRD.

The scram has been reset.

EP-4 has been entered on multiple entry conditions.

Which of the following describes the capability of restoring ECCS room water levels to within the EP-4 Operating Limits?

A. The ECCS room sump pumps will operate automatically to remove water from the ECCS rooms.

B. Operators must align the ECCS room sump pumps to the suppression pool by opening P45-F273/274 AUX BLDG EQ/FL DR PMPBK TO SUPP POOL.

C. Operators must reset NSSSS and re-open isolation valves, then the ECCS room sump pumps will operate automatically to remove water from the ECCS rooms.

D. ECCS room sump pumps cannot be made available under these conditions.

Answer: A

Explanation:

RPV level dropped to at least -41.6 inches due to the auto start of HPCS and RCIC. While this resulted in an automatic isolation of the Auxiliary Building P45 valves, this does not affect operation of the ECCS room sump pumps. The ECCS room sump pumps are not interlocked with the P45 Aux Bldg isolation valves or with high levels in the Aux Bldg floor/equip drain transfer tanks. Therefore the ECCS room sump pumps will continue to operate. The Aux Bldg transfer tanks will overflow to the Aux Bldg floor drain sumps and flood the lower elevation of the Aux Building. The ECCS room watertight doors will maintain ECCS room integrity and satsify the EP–4 requirement to preserve the functionality of the equipment located in the room. Answer ‘A’ is correct.

‘B’ is wrong because no procedure exists for this action. Additionally, the sump pumpback valves do not directly affect ECCS room sump pump operation as noted above. The pumpback valves re–align the discharge of the Aux Bldg floor/equip drain transfer tank pumps from Radwaste to the Suppression Pool.

‘C’ is wrong because no operator action is required to restore ECCS room water levels within limits, the ECCS room sump pumps will operate automatically.

‘D’ is wrong; see explanation for ‘A’.

Technical References:

04-1-01-M71-1
05-S-01-EP-4 step 4
04-1-02-1H13-P870 3A-B3; 2A-A1; 9A-B3
M-1098A

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-M7101, objective 7.8
GLP-OPS-P4500, objective 7

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b)(1) & (9)	
	55.43	

<table>
<tr><td>Examination Outline Cross-Reference</td><td>Level</td><td>RO</td></tr>
<tr><td rowspan="5">215005 Average Power Range Monitor/Local Power Range Monitor System
A1. Ability to predict and/or monitor changes in parameters associated with operating the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM controls including:
A1.07 APRM (gain adjustment factor)</td><td>Tier #</td><td>2</td></tr>
<tr><td>Group #</td><td>1</td></tr>
<tr><td>K/A #</td><td>215005 A1.07</td></tr>
<tr><td>Rating</td><td>3.0</td></tr>
<tr><td></td><td></td></tr>
</table>

Question 28

The plant is operating at power with AGAFs indicating 1.0 on CYCLOPS.

The ACRO notices that the calculated Core Thermal Power displayed on PDS is lowering.

The STA reports that the RWCU system flow computer point input to the heat balance has failed downscale.

Actual RWCU system flow is unchanged.

Which of the following describes how the AGAFs and the APRMs are affected?

A. AGAFs will indicate < 1.0.
APRM scrams on reactor power will occur at lower reactor power.

B. AGAFs will indicate < 1.0.
APRM scram functions are unaffected.

C. AGAFs will indicate > 1.0.
APRM scrams on reactor power will occur at higher reactor power.

D. AGAFs will indicate > 1.0
APRM scram functions are unaffected.

Answer: B

Explanation:

APRM AGAF is the ratio of calculated reactor power (i.e., power calculated by the heat balance) to indicated reactor power (i.e., APRM power). The downscale failure of the RWCU flow input to the heat balance results in a decrease in calculated core thermal power. This is because the heat balance “calorimetric” equation has lost one of its positive (+) mathematical terms…i.e., the “heat loss out of the reactor through RWCU.” By definition, all “heat loss” terms are mathematically (+) terms in the calorimetric equation. Because the calculated power has lowered (i.e., the numerator of the AGAF ratio) and the indicated APRM power has not changed (i.e., the denominator of the

AGAF ratio), the resulting AGAF value will be smaller than it was before the RWCU flow input failure. (i.e, AGAF < 1.0).

Actual reactor power is unchanged by the event, as the malfunction is explicitly a computer issue affecting the heat balance calculation only. In other words, although the computer will "indicate that AGAFs are < 1.0", the AGAFs have in fact not physically been adjusted (changed). Therefore, the APRMs are unaffected by this RWCU computer point failure; thus, the scram functions are unaffected.

For the reasons above, only choice 'B' is correct.

Technical References:

GLP–OPS–C5104 pg. 11–12
GLP–OPS–HFT07

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-C5104, Obj. 4
GLP–OPS–HFT07, Obj. 14

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

Examination Outline Cross-Reference	Level	RO
217000 Reactor Core Isolation Cooling System (RCIC) **Knowledge of the effect that a loss or malfunction of the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) will have on following: K3.04 Adequate core cooling**	**Tier #**	2
	Group #	1
	K/A #	217000 K3.04
	Rating	3.6

Question 29

For which of the following situations would the loss of RCIC, by itself, have the **MOST ADVERSE IMPACT** on the ability to adequately cool the reactor core?

(For each situation, assume sufficient steam pressure to otherwise operate RCIC.)

A. Station Blackout

B. Complete loss of feedwater coincident with MSIV closure when operating at rated power

C. Complete loss of feedwater coincident with MSIV closure when operating at low power

D. Steam line break in drywell

Answer: A

Explanation:

In a Station Blackout (assuming sufficient steam pressure still available) only RCIC would be available for RPV makeup and adequate core cooling (i.e., independent of AC power needs). This situation is suggested by choice ‘A’.

Whether from operating at rated power, or low power, RCIC’s design purpose is to provide RPV inventory makeup and adequate core cooling in the case of a core isolation coincident with loss of feedwater…as suggested by choices ‘B’ and ‘C’. However, a loss of RCIC, by itself, in either of these situations is likely to be moot considering the availability of HPCS.

A steam line break in the drywell, as suggested by choice ‘D’, is just a variety of an RCS leak. By itself, this situation suggests nothing in particular about the loss of RCIC with respect to adequate core cooling, especially with respect to the several other ECCS systems, as well as feedwater, that might well be available as backup.

When comparing all 4 situations among the answer choices, ‘A’ clearly represents the situation where the “loss of RCIC, by itself, would have the **MOST** adverse impact on the ability to adequately cool the reactor core." Thus, 'A' is correct.

Technical References:

Tech Spec LCO 3.5.3 (RCIC) Bases

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-SECUR, OBJ. 8

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

Examination Outline Cross-Reference	Level	RO
295031 Reactor Low Water Level **Ability to operate and/or monitor the following as they apply to REACTOR LOW WATER LEVEL: EA1.04 High pressure core spray**	**Tier #**	1
	Group #	1
	K/A #	295031 EA1.04
	Rating	4.3

Question 30

Following an auto-initiation of HPCS on low reactor water level, the HPCS injection valve E22-F004 automatically closes on high reactor water level.

Reactor water level subsequently lowers to -10" and the CRS directs the RO to re-inject with HPCS.

No operator actions have yet been taken with respect to HPCS at control room panel P601.

Which of the following operator actions, by itself, will cause HPCS to re-inject to the RPV?

A. Depress the HPCS INIT RESET pushbutton.

B. Depress the HPCS HI LVL RESET pushbutton.

C. Momentarily place the control switch for E22-F004 in the OPEN position, then release.

D. While holding the control switch for E22-F004 in the CLOSE position, arm and depress the HPCS MAN INIT pushbutton, then release the control switch.

Answer: B

Explanation:

Per the stem conditions, E22-F004 auto-closed on a Level 8 signal. Thereafter, the only way to energize the 42F contactor used to open the valve is by breaking the seal-in circuit of the Level 8 auto-close signal. Only two things can do this: 1) water level again lowers to Level 2 (-41.6"), OR 2) an operator manually depresses the HPCS HI LVL RESET pushbutton at P601. Since the stem conditions indicate that HPCS originally auto-initiated on a Level 2 signal, and that no other operator action has been taken (e.g., the HPCS INIT RESET pb has not been depressed), then the Level 2 signal is still present in the HPCS initiation logic. Thus, the single action suggested by choice 'B' will, by itself, result in E22-F004 automatically re-opening to recommence injection to the RPV. Thus, 'B' is correct.

‘A’ is wrong. Even though this action will in fact reset the initiating condition (being that the stem condition indicates current water level is -10”…above the Level 2 initiation setpoint), this action will not, by itself, cause F004 to auto-open. For more, see explanation for correct answer ‘B’.

‘C’ is wrong. So long as the Level 8 auto-closure signal is still sealed-in (i.e., has not been manually reset using the pushbutton), there is no way to manually stroke open the valve using its handswitch at P601. For more, see explanation for correct answer ‘B’.

‘D’ is wrong for all of the reasons already discussed. This distracter is especially plausible to the uncertain Candidate because it represents the action used to “initiate and override (closed)” F004…such as in an ATWS event per EP-2A.

Technical References:

E-1183-003 and 023

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-E2201, OBJ. 9.4

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

Examination Outline Cross-Reference	Level	RO
295014 Inadvertent Reactivity Addition **Ability to determine and/or interpret the following as they apply to INADVERTENT REACTIVITY ADDITION: AA2.02 Reactor period**	**Tier #**	1
	Group #	2
	K/A #	295014 AA2.02
	Rating	3.9

Question 31

A reactor startup is in progress, with the ACRO as the person withdrawing control rods.

Inadvertently, the ACRO continuously withdraws the next control rod 1 notch past its target position, where the rod then settles.

A stable reactor period of 55 seconds results from this error.

Which of the following describes the action to be taken by the ACRO?

A. Immediately insert the control rod 1 notch to its target position.

B. Immediately insert the control rod as necessary to slow the reactor period.

C. After being given permission by the Reactivity SRO, insert the control rod 1 notch to its target position.

D. After being given permission by the Reactivity SRO, insert the control rod as necessary to slow the reactor period.

Answer: C

Explanation:

The only true result of the ACRO error is the rod mis-position event itself. The resulting "55-second period" is already slower than the "maintain > 50 seconds" allowed by IOI-1, section 5.36.

For the above reason, choice 'B' is wrong. It is plausible to the Candidate who mistakenly interprets the 55-second period as being too short a period. If it were, 'B' would be the required ACRO action…directed by IOI-1, step 2.1.4 (implemented as an Immediate Operator Action <u>without</u> prior SRO direction).

Similarly, choice 'A' is wrong. This distracter's plausibility is rooted in a similar Immediate Operator Action found in the CRD Malfunctions ONEP, section 2.5. That action, however, is for moving a rod <u>more than 1 notch</u> past its intended position. The stem conditions ensure that choice 'A' is wrong with regard to the ONEP required action.

Because there is no need to “slow the reactor period” (as already discussed), choice ‘D’ is also wrong.

‘C’ is correct. In fact, after informing the CRS and consulting with the Reactivity SRO, the ACRO will insert the rod to its target position only after being directed to do so by the SRO.

Technical References:

03-1-01-1 (IOI-1); ARI P680-7A-C10; 05-1-02-IV-1 (CRD ONEP)

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-PROC, OBJ. 8.11, 73.2

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

Examination Outline Cross-Reference	Level	RO
295029 High Suppression Pool Water Level **2.4.45 Ability to prioritize and interpret the significance of each annunciator or alarm.**	**Tier #**	1
	Group #	2
	K/A #	295029 2.4.45
	Rating	4.1

Question 32

A small–break LOCA occurred 30 minutes ago.

HPCS automatically initiated but has now been returned to its standby lineup.

Operators are maintaining level in the nominal level band with Feedwater.

SUPP POOL LVL HI/LO is in alarm on P870. Suppression Pool level is observed to be 18.9'

Subsequently, the following alarms are received on P870:

- SPMU DIV I VLV OPEN
- SPMU DIV 2 VLV OPEN

Which of the following is the proper interpretation of these alarms and the appropriate operator response?

A. SPMU has initiated on low suppression pool level; verify SPMU initiation goes to completion.

B. SPMU has initiated spuriously; override SPMU initiation by placing the MODE SEL switch to OFF.

C. SPMU has initiated spuriously; suppression pool water should be transferred to Radwaste using RHR.

D. SPMU has initiated on high drywell pressure; level control should be transferred over to HPCS.

Answer: D

Explanation:

A small–break LOCA will result in high drywell pressure actuation of ECCS systems and Primary CTMT and Auxiliary Building isolations. With the availability of the Feedwater system and a minor break severity, the HPCS logic can be reset with DW pressure still high and the system returned to a standby lineup. A Suppression Pool Makeup initiation occurs on the Aux Bldg isolation 30-minute TD signal. The SPMU initiation is the source of the SPMU DIV 1 and 2 VLV OPEN alarms.

With an SPMU initiation, Supp Pool level will be high.

'A' is incorrect; low Suppression Pool level is just one of 2 initiation signals. No indication of a SP leak or reduction in SP inventory is suggested in the stem.

'B' is incorrect; the initiation is not spurious. Placing the SPMU MODE SEL switch to OFF will stop any further valve travel, but the valves will not automatically re-close. The valves cannot be manually re-closed with an initiation signal present, in this case the Aux Bldg isolation 30-min TD. The Upper Containment Pool ultimately will be transferred to the Suppression Pool.

'C' is incorrect; the initiation is not spurious. EP–3 identifies a number of options for reducing SP level, but none are available with a high DW pressure signal present. RHR cannot be aligned due to the Aux Bldg isolation closure signal to the E12–F203 transfer to Radwaste valve.

EP–3 requires termination of systems with injection from outside the Containment if not needed for adequate core cooling. With the availability of the HPCS system to maintain reactor level, answer 'D' represents the appropriate interpretation of the alarms and operator response.

Technical References:

05-1-02-III–5, Automatic Isolations ONEP
04–1–01–E22–1 step 6.3.2.a
05-S-01-EP–3
E–1220, sht 1, 2

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-EP03 Obj. 7, 21
GLP-OPS-E3000 Obj. 7.1, 8

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

Examination Outline Cross-Reference	Level	RO
239001 Main and Reheat Steam System **2.4.4 Ability to recognize abnormal indications for system operating |parameters that are entry-level conditions for emergency and abnormal operating procedures.**	**Tier #**	2
	Group #	2
	K/A #	239001 2.4.4
	Rating	4.5

Question 33

What is the value specified in EP-4, Auxiliary Building Control, as the entry condition for Main Steam Line high radiation?

A. 1.5 x Normal Full-Power Background

B. 2.0 x Normal Full-Power Background

C. 2.5 x Normal Full-Power Background

D. 3.0 x Normal Full-Power Background

Answer: A

Explanation:

Per EP-4, entry conditions and Table 10, choice 'A' is correct, and the 3 distracters are wrong.

Technical References:

EP-4, Aux Bldg Control

References to be provided to applicants during exam: None

IMPORTANT NOTE – Because SRO Candidates will have the EP-4 flowchart, both the Entry Condition statements and the Table 10 line item for MSL radiation will be redacted.

Learning Objective: GLP-OPS-EP4, OBJ. 5

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

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

Examination Outline Cross-Reference	Level	RO
245000 Main Turbine Generator and Auxiliary Systems **Ability to predict and/or monitor changes in parameters associated with operating the MAIN TURBINE GENERATOR AND AUXILIARY SYSTEMS controls including: A1.04 Steam flow**	**Tier #**	2
	Group #	2
	K/A #	245000 A1.04
	Rating	2.7

Question 34

The plant is operating at 70% power.

The ACRO momentarily depresses (for 1 second) the LOWER pushbutton for PRESSURE REFERENCE.

Which of the following P680 indications will result from this action?

Initially…

A. BYPASS STM FLO DEMAND will rise.

B. LOAD – ACTUAL will lower.

C. TURB STM FLO will rise.

D. PRESS CONT DEV will lower.

Answer: C

Explanation:

‘A’ is wrong. This is “%” indicator for the monitoring EHC demand for steam flow through the Bypass Valves. The Bypass Valves are currently closed (0% steam flow demand) and will remain closed because, at 70% power, there is plenty of room for the turbine Control Valves to stroke open more in an effort to lower reactor pressure.

‘B’ is wrong. As the turbine Control Valves open more, more steam is admitted into the turbine, allowing the generator to pickup more real load (MWe). Thus, LOAD –ACTUAL rises, not lowers.

‘C’ is correct. Depressing the PRESS REF Lower pb for 1 second lowers the “demanded” pressure signal ~15 psig, placing the EHC demand for reactor pressure this much below the current actual reactor pressure. In other words, EHC now sees actual reactor pressure as “too high.” The system responds by throttling open the Control Valves to lower steam pressure and, with it, reactor pressure. As the CVs open, the steam flow into the turbine rises; thus, the indicated TURB STM FLO rises.

‘D’ is wrong. This indicator shows the real-time deviation between the Pressure Demand output signal and the Actual Pressure signal. The “initial” response of this indicator (when the PRESS REF Lower pb is depressed for 1 second) is to show an increasing deviation (i.e., “will rise”) between the two signals. The deviation will become smaller (decrease; “will lower”) as reactor pressure lowers in response to the lowered output signal from the pressure controller. This fact is the reason it’s necessary to preface the answer choices with the word…”Initially”…, so as to ensure that this distracter is wrong.

Technical References:

GLP-OPS-N3202

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-N3202, OBJ. 4.0

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

Examination Outline Cross-Reference	Level	RO
201003 Control Rod and Drive Mechanism **Ability to manually operate and/or monitor in the control room: A4.01 CRD mechanism temperature**	**Tier #**	2
	Group #	2
	K/A #	201003 A4.01
	Rating	2.6

Question 35

The plant is operating at power when the following alarm is received at P680:

- CRD HYD TEMP HI sealed-in

How do the control room operators determine which CRD Mechanism has the high temperature?

A. By scanning the System C11 alarm page on either of the PDS displays at P680

B. By looking for an illuminated LED on either of the RACS panels, P651 or P652

C. By dispatching an operator to check each HCU for presence of an alarm light

D. By dispatching an operator to check for an alarm on a local panel recorder

Answer: D

Explanation:

See ARI P680-4A2-A4. Only 'D' is correct. Individual CRDM temperatures are available only at this local panel recorder. 'A'.'B', and 'C' are each plausible distracters to the Candidate who is unsure of this fact.

Technical References:

ARI P680-4A2-A4

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-C111B, OBJ. 9

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

Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b)(6) 55.43	

Examination Outline Cross-Reference	Level	RO
226001 RHR/LPCI: Containment Spray System Mode **Knowledge of the physical connections and/or cause effect relationships between RHR/LPCI: CONTAINMENT SPRAY SYSTEM MODE and the following: K1.13 Containment instrumentation**	**Tier #**	2
	Group #	2
	K/A #	226001 K1.13
	Rating	3.1

Question 36

A valid Div 1 LOCA signal is generated.

8 minutes later, CTMT pressure is 5.0 psig and reactor water level is -155”.

What would happen if the CRO were to arm and depress the CTMT SPRAY A MAN INIT pushbutton right now, and why?

A. CTMT Spray ‘A’ would <u>not</u> initiate because CTMT pressure is still too low.

B. CTMT Spray ‘A’ would initiate because Drywell pressure is high enough.

C. CTMT Spray ‘A’ would <u>not</u> initiate because the LOCA signal has not been present long enough.

D. CTMT Spray ‘A’ would initiate because reactor water level is low enough.

Answer: B

Explanation:

See GLP-OPS-E1200, pages 41-42. Only a high drywell pressure signal (1.39 psig) is required in order to manually initiate CTMT Spray. With CTMT pressure at 5.0 psig, drywell pressure has to be well above 1.39 psig. Thus, ‘B’ is correct.

‘A’ is wrong. It suggests the 7.84 psig CTMT pressure signal requirement which applies only to auto-initiation of CTMT Spray.

‘C’ is wrong. It suggests the 10.85 minute time delay requirement (for presence of the LOCA signal) which applies only to auto-initiation of CTMT Spray.

‘D’ is wrong. It suggests that CTMT Spray <u>manual</u> initiation logic has any reliance upon a reactor water Level 1 (-150.3”) signal condition; it does not.

Technical References:

GLP-OPS-E1200, RHR System

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-E1200, OBJ. 10

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

Examination Outline Cross-Reference	Level	RO
290002 Reactor Vessel Internals **Knowledge of REACTOR VESSEL INTERNALS design feature(s) and/or interlocks which provide for the following:K4.02 Separation of fluid flow paths within the vessel**	**Tier #**	2
	Group #	2
	K/A #	290002 K4.02
	Rating	3.1

Question 37

Which of the following RPV internal components has the purpose of separating the water that is already flowing upward through the core from the water that is flowing downward through the downcomer annulus?

A. Core Shroud

B. Separator

C. Lower Plenum

D. Baffle Plate

Answer: A

Explanation:

See GLP-OPS-B1300, page 13. Only the Core Shroud fits this description purpose. Thus, 'A' is correct.

'B' is wrong. This is the steam separator at the upper plenum area.

'C' is wrong. This area redirects the downcomer water to flow upward through the core plate and into the core, but does not provide separation between the two flows.

'D' is wrong. This component separates the downcomer annulus from the below core plate area, but does not provide separation between the two flows.

Technical References:

GLP-OPS-B1300, RPV Internals

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-B1300, OBJ. 4.1

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

	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b)(3) 55.43	

Examination Outline Cross-Reference	Level	RO
295003 Partial or Complete Loss of A.C. Power **Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER: AK3.05 Reactor SCRAM**	**Tier #**	1
	Group #	1
	K/A #	295003 K3.05
	Rating	3.7

Question 38

The plant is operating at rated 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 that will occur as the MSIVs close due to the loss of an Instrument Air supply to them.

B. Is a conservative action based on the unavailability of Plant Air Compressor ‘A’ as being unacceptable to sustained plant operation.

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

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

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 are loads on that header. They will begin to drift closed as air pressure lowers in their control units. An auto-scram will result from the MSIV closure. Because this question is asking for just “a reason”, this single failure mechanism is enough to justify the correct answer ‘A’. Pre-empting the auto-scram by inserting a manual scram conforms to the “Conservative Decision-Making” requirements of EN-OP-115, Conduct of Operations.

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

‘C’ 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.

‘D’ is wrong. Strongly plausible to the weaker Candidate, but not true.

NOTE – Classifying this question as Higher Cognitive based on the fact that no such “reason” or “bases” is found anywhere in GGNS controlled documents or its lesson plans/training materials (therefore, it is not a “from memory” lower cognitive question). The Candidate will have to use higher cognitive processes to derive the reason as claimed in the correct answer.

Technical References:

EN-OP-115; 05-1-02-I-4 (Loss of AC ONEP); TS LCOs 3.5.1 and 3.8.1

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-ONEP, OBJ. 2.0

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

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: AK2.06 Offgas system**	**Tier #**	1
	Group #	1
	K/A #	295019 AK2.06
	Rating	2.8

Question 39

Which of the following describes a consequence of losing the Instrument Air supply to the Offgas system?

A. Increased radioactivity release due to bypassing of the charcoal adsorber beds

B. Increased radioactivity release due to process flow leakage from the system

C. Increased concentration of combustible gases due to decrease in recombiner inlet temperature

D. Decreased condenser vacuum due to interruption of process flow

Answer: B

Explanation:

Per the Loss of Instrument Air ONEP, Offgas AOVs typically fail as–is. Important process valves either fail open, such as the adsorber inlets and system outlet, fail closed, such as the adsorber bed bypass valves, or fail as–is, to preserve system functionality.

Answer B is correct. Instrument Air is supplied to many valves in the process flowpath as stem sealing air to prevent process flow leakage into the Offgas Building. Loss of Instrument Air allows process flow to escape the system, resulting in increased airborne radiation levels in the Offgas Building. This radioactivity is then released into the atmosphere via the Offgas/Radwaste Building ventilation system.

Answer A is wrong. The charcoal adsorber bypass valves (F045, F340) fail closed on a loss of Instrument Air. The charcoal adsorber beds remain in service. The candidate may recall that the F051C(D) valves do fail open,. but these only bypass the first of four in–series charcoal adsorbers.

Answer C is wrong. The offgas preheater steam supply control valve fails open on a loss of instrument air to maintain process flow preheating for recombiner operation.

Answer D is wrong. None of the system process flow valves fail closed on a loss of instrument air to interrupt process flow.

Technical References:

05–1–02–V–9, Loss of Instrument Air ONEP
M–1092

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-N6465, Obj. 14.2

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

Examination Outline Cross-Reference	Level	RO
295021 Loss of Shutdown Cooling **Knowledge of the operational implications of the following concepts as they apply to LOSS OF SHUTDOWN COOLING: AK1.03 Adequate core cooling**	**Tier #**	1
	Group #	1
	K/A #	295021 K1.03
	Rating	3.9

Question 40

The plant is in MODE 4 cooling down for a refueling outage.

Currently:

- RHR ‘A’ is operating in Shutdown Cooling
- Both Recirc Pumps are running in SLOW speed
- Reactor coolant temperature is 199°F
- Reactor water level is steady at +50”

RHR Pump ‘A’ trips on motor overload.

RHR Pump ‘B will not start.

Which of the following methods is available for operators to re-establish decay heat removal?

A. Place ADHR in service.

B. Raise reactor water level to +82”.

C. Align RWCU for Alternate Shutdown Cooling.

D. Establish a flowpath to/from the suppression pool with SRVs and LPCS.

Answer: D

Explanation:

See ONEP Section 3.3 (RPV Head Installed) and the top-most CAUTION of Section 3.0.

‘A’ is wrong. Even if ADHR were already unisolated at this high a temperature (doubtful), it must be isolated again prior to RCS temperature reaching 200°F (MODE 3).

‘B’ is wrong. So long as forced coolant flow exists (Recirc pumps are still running), there is no reason to raise level to +82”. This +82” mark is only for establishing natural circulation in the event of there being no Recirc pumps running, and has no function with respect to restoring adequate decay heat removal capability.

‘C’ is wrong. Per the stem conditions, we’re still way too early heading into the outage (i.e., far too much decay heat being produced) for an RWCU Alternate Shutdown Cooling lineup to be available for DHR. This method is **absolutely not** available.

‘D’ is correct per section 3.3.3.f. A feed/bleed flowpath is to be established to/from the suppression pool via SRVs and LPCS (since there is no working RHR pump).

Technical References:

05-1-02-III-1, Inadequate DHR ONEP

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-ONEP, OBJ. 20.0

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

Examination Outline Cross-Reference	Level	RO
2.3.7 Ability to comply with radiation work permit requirements during normal or abnormal conditions.	**Tier #**	3
	Group #	
	K/A #	2.3.7
	Rating	3.5

Question 41

Per EN-RP-105, Radiological Work Permits (RWP), a radiation worker must participate in an RWP Pre-Job Brief prior to entering the…

A. Aux Building Steam Tunnel.

B. RHR ‘C’ Pump Room.

C. RCIC Pump Room.

D. El. 119’ Piping Penetration Room.

Answer: A

Explanation:

See EN-RP-105, section 5.3[8], 4th bullet…VHRA or LHRA entry requires the RWP Pre-Job Brief.

Of the 4 areas among the answer choices, only the Aux Bldg Steam Tunnel is a Locked HRA. The other 3 areas are only Radiation Areas, for which the RWPs require no pre-job briefs.

Technical References:

EN-RP-105, RWPs

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-PROC, OBJ. 32.2

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

	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b)(12) 55.43	

Examination Outline Cross-Reference	Level	RO
295025 High Reactor Pressure **Knowledge of the reasons for the following responses as they apply to HIGH REACTOR PRESSURE: EK3.09 Low-low set initiation**	**Tier #**	1
	Group #	1
	K/A #	295025 EK3.09
	Rating	3.7

Question 42

Which of the following is an advantage of allowing SRV Lo-Lo Set to control reactor pressure following a condition requiring a scram?

A. Avoids pressure-induced power oscillations during an ATWS.

B. Minimizes SRV cycling without the need for an operator.

C. More evenly distributes the heat input to the suppression pool.

D. Provides the best way to avoid reactor pressure transients.

Answer: B

Explanation:

‘A’ is wrong. 02-S-01-27, section 6.2.4 does not allow using Lo-Lo Set during an ATWS.

‘B’ is correct. Lo-Lo Set uses only 6 SRVs of 20 and opens them in a sustained pressure-reduction way so as to minimize SRV cycling…automatically, without the need for manual SRV operation by an operator.

‘C’ is wrong. A disadvantage is that the 6 SRVs are used again and again, dumping heat into the same few areas of the suppression pool.

‘D’ is wrong. Manual SRV control is always the most stable form of pressure control.

Technical References:

Operations Philosophy (02-S-01-27), section 6.2.4.
EP Technical Bases, Attachment IV, page 31 of 51 discussion for “SRVs”.

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-EP02, OBJ. 7

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b)(3) & (5)	
	55.43	

Examination Outline Cross-Reference	Level	RO
295030 Low Suppression Pool Water Level **Knowledge of the reasons for the following responses as they apply to LOW SUPPRESSION POOL WATER LEVEL: EK3.03 RCIC operation**	**Tier #**	1
	Group #	1
	K/A #	295030 EK3.03
	Rating	3.6

Question 43

RCIC has been placed in operation as a source of makeup to the suppression pool using the E51 SOI.

Why do we ensure the RCIC Inboard and Outboard Test Return Valves to the CST (E51-F022 and F059) are closed for this lineup?

A. Protect the RCIC Pump from operating in a runout condition.

B. Protect the RCIC Pump from overheating due to operating at a shutoff head.

C. Avoid pumping down the suppression pool to an even lower water level.

D. Ensure that water gets from the CST into the suppression pool.

Answer: D

Explanation:

See E51 SOI, section 6.4. This lineup simply places RCIC is operation with a suction from the CST, but with its CST Return line gagged off (i.e. F022/F059 closed)...forcing the pumped water through the min flow valve (F019) and into the supp pool…thereby raising pool level. If the CST Return line were open, RCIC would pump CST-to-CST, bypassing the supp pool (i.e., not an effective way to raise pool level with RCIC). Thus, ‘D’ is correct.

‘A’ is wrong. F022/F059 would be open for a normal full-flow pump run, presenting no concern for operating at “runout”. What’s more, this is a turbine-driven pump, not a motor-driven one. Operating at so-called “runout” (with its concern for excessive motor current) is not a concern.

‘B’ is wrong. It simply provides face plausibility for the weaker Candidate.

‘C’ is wrong. This choice is meant to suggest that the pump might be taking a suction from the supp pool, rather than from the CST, for this particular lineup. This, of course, is not the case and would make no sense in terms of trying to raise supp pool level.

Technical References:

04-1-01-E51-1 RCIC SOI

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-EP01, OBJ. 5

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

Examination Outline Cross-Reference	Level	RO
700000 Generator Voltage and Electric Grid Disturbances **Ability to operate and/or monitor the following as they apply to GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES: AA1.01 Grid frequency and voltage**	**Tier #**	1
	Group #	1
	K/A #	700000 AA1.01
	Rating	3.6

Question 44

Per the Loss of AC Power ONEP, which of the following is a possible symptom of Grid Instability?

A. 500 KV VOLTAGE recorder is reading > 525 KV at P807.

B. 500 KV FREQ recorder is reading < 58.3 Hz at P807.

C. GEN MVARs recorder is reading more negative than -170 MVARs at P680.

D. Jackson Dispatcher reports that grid voltage on the 500 KV system is > 525 KV.

Answer: B

Explanation:

See ONEP section 3.4 NOTE.

'A' is wrong because only a low or oscillating 500 KV system voltage is a possible symptom, not a high voltage.

'B' is correct. Taken directly from the ONEP, section 3.4 NOTE.

'C' is wrong because excessive MVARs ("IN" in this case) can be a concern for generator stator bar overheating, but is not in itself a symptom of Grid Instability. See the section 3.4 NOTE (MVARs is not mentioned there); also, see the section 3.4.3 CAUTION.

'D' is wrong. Although it is true that the Jackson Dispatcher report is the only acceptable record of 500 KV system voltage (hence, the plausibility of this answer choice), the voltage being reported is, again, a high voltage (rather than a low one). The Jackson Dispatcher report being the only acceptable grid voltage indication is found in the Plant AC/DC Surveillance, 06-OP-1R20-W-0001.

Technical References:

05-1-02-I-4, Loss of AC Power ONEP
06-OP-1R20-W-0001, Plant AC/DC Electrical Power Distribution Weekly

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-ONEP, OBJ. 56.0

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

Examination Outline Cross-Reference	Level	RO
205000 Shutdown Cooling System (RHR Shutdown Cooling Mode) **Ability to manually operate and/or monitor in the control room: A4.03 SDC/RHR discharge valves**	**Tier #**	2
	Group #	1
	K/A #	205000 A4.03
	Rating	3.6

Question 45

RHR 'A' Shutdown Cooling loop has been flushed and warmed using the E12-2 SOI.

A common suction flowpath has been established.

The loop is ready for operation.

Which of the following describes how the RO places the loop in operation with return flow via the LPCI injection line?

A. Stroke LPCI Injection Valve E12-F042A to OPEN and release; immediately start RHR Pump 'A'.

B. Start RHR Pump 'A'; count 8 seconds; open LPCI Injection Valve E12-F042A.

C. Start RHR Pump 'A'; immediately open LPCI Injection Valve E12-F042A.

D. Fully open LPCI Injection Valve E12-F042A; count 8 seconds; start RHR Pump 'A'.

Answer: C

Explanation:

See E12-2 SOI, step 4.1.2c(17) and its CAUTION.

Only 'C' is correct per this step. The CAUTION explains the importance of immediately opening F042A.

'A', 'B', and 'D' are all plausible variations to the Candidate who has not grasped the concept the 'globally' critical concept of avoiding an RPV inventory loss while operating controls associated with a Shutdown Cooling lineup.

NOTE – Classifying this question a Higher Cognitive for the following reasons: 1) GGNS does not expect its Candidates to memorize and recall SOI "normal operations" section steps (which is from where this question is taken); 2) nonetheless, we expect our Candidates to be able to derive the correct answer by "reason" (cognitive analysis)…they must recognize that, when compared to the distracters, choice 'C' appears to be the only method that both prevents loss of reactor water inventory to the suppression pool via the RHR Pump min flow valve and does not first depressurize the discharge piping (by opening F042A) before starting the pump.

Technical References:

04-1-01-E12-2, Shutdown Cooling SOI

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-E1200, OBJ. 14

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

Examination Outline Cross-Reference	Level	RO
211000 Standby Liquid Control System **Ability to monitor automatic operations of the STANDBY LIQUID CONTROL SYSTEM including: A3.01 Pump discharge pressure**	**Tier #**	2
	Group #	1
	K/A #	211000 A3.01
	Rating	3.5

Question 46

An ATWS is in progress.

Both SLC Pumps are injecting boron to the RPV.

The reactor is at rated pressure when 8 ADS Valves are opened for Emergency Depressurization.

As the reactor depressurizes, what is the expected response of the SLC PUMP A/B DISCH PRESS indication on P601?

A. Remains at approximately 1500 psig.

B. Remains at approximately 1200 psig.

C. Trends downward remaining approximately 500 psig above reactor pressure.

D. Trends downward remaining approximately 200 psig above reactor pressure.

Answer: D

Explanation:

SLC Pump discharge pressure is always ~200 psig greater than reactor pressure.

'A' is wrong because this 1500 psig value exceeds the pump discharge pressure response even at rated reactor pressure.

'B' is wrong. It would be the expected reading at rated reactor pressure.

'D' is correct (and therefore 'C' is wrong) because as reactor pressure lowers during the ED, these PD pumps will simply continue to overcome the existing reactor pressure by ~200 psig.

Technical References:

C41 SOI; 06-OP-1C41-Q-0001, Attachment 1 Data Sheet for ‘A’ SLC Pump Quarterly

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-C4100, OBJ. 12

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

Examination Outline Cross-Reference	Level	RO
212000 Reactor Protection System **Ability to (a) predict the impacts of the following on the REACTOR PROTECTION SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: A2.02 RPS bus power supply failure**	**Tier #**	2
	Group #	1
	K/A #	212000 A2.02
	Rating	3.7

Question 47

Refer to the figure below for this question. The lights have been numbered 1 through 8 (left-to-right) for the purposes of this question, only.

The plant is operating at rated power with the following:

- RPS Bus ‘A’ is on its NORMAL power supply
- RPS Bus ‘B’ is on its ALTERNATE power supply

The MCC feeder breaker to RPS MG Set ‘A’ trips (breaker internal fault) and cannot be re-closed.

Which scram solenoid power (white) indicating lights will extinguish on P680 <u>and</u> what is the appropriate action before the RPS ‘A’ half-scram can be reset?

A. All odd-numbered lights will extinguish.
Place the MG SET A TRANSFER switch in the ALT A position.

B. All odd-numbered lights will extinguish.
Make RPS MG Set ‘A’ power available again to RPS Bus ‘A’.

C. Lights 1 through 4 will extinguish.
Place the MG SET A TRANSFER switch in the ALT A position.

D. Lights 1 through 4 will extinguish.
Make RPS MG Set ‘A’ power available again to RPS Bus ‘A’.

P680
RPS scram solenoid power indicating lights

Answer: B

Explanation:

The ONEP for Loss of One or Both RPS Buses directs that only one of the two buses may be on its Alternate power source when in MODEs 1 or 2 (question stem condition indicates MODE 1). Because RPS Bus ‘B’ is already on its Alternate source, we cannot also place RPS Bus ‘A’ on its Alternate source, per the ONEP. Therefore, the only way to reset the ‘A’ half-scram is by making RPS MG Set ‘A’ power available again to RPS Bus ‘A’. Thus, answer choices ‘A’ and ‘C’ are wrong.

The 8 white lights are associated as follows: the odd-numbered lights are in the circuits that bring RPS ‘A’ bus power to all of the ‘A’ scram solenoids; the even-numbered lights are in the circuits that bring RPS ‘B’ bus power to all of the ‘B’ scram solenoids. When RPS MG Set ‘A’ tripped (in the stem conditions), de-energizing RPS Bus ‘A’, all of the ‘A’ scram solenoid circuits (including the associated power monitoring lights) lost power. Therefore. For this reason, along with the reason described above, the correct answer is ‘B’.

‘D’ is wrong for the reason ‘B’ is correct. This is a strong distracter to the uncertain Candidate who is still confused by the P680 layout of the scram power solenoid indicating lights/labeling (see figure below). RPS Trip System ‘A’ is in fact comprised of the “Div 1” and “Div 3” Trip Channels, which are located on the left side of the P680 apron (as shown and as labeled). A standard RPS ‘A’ channel half-scram (not an MG Set power loss) would in fact be generated through a trip of Div 1 and Div 3…nonetheless, all of the odd-numbered lights (all the way across P680) would extinguish.

Technical References:

05-1-02-III-2, RPS Bus Loss ONEP

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-C7100, OBJ. 12

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

Examination Outline Cross-Reference	Level	RO
215004 Source Range Monitor (SRM) System **Ability to (a) predict the impacts of the following on the SOURCE RANGE MONITOR (SRM) SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: A2.02 SRM inop condition**	**Tier #**	2
	Group #	1
	K/A #	215004
	Rating	3.4

Question 48

A reactor startup is in progress.

All IRMs are on Range 4.

All SRMs are still being withdrawn as necessary to maintain them between 10^2 and 10^5 CPS.

A SRM high voltage power supply (in control room panel P672) fails to zero volts output.

This power supply is the one powered from 1Y95.

Which of the following describes the rod block(s) that should result, <u>and</u> describes the **MINIMUM** action necessary to continue the reactor startup?

A. SRM INOP rod block, only
Bypass SRM ‘D’ using the joystick at P680.

B. SRM INOP rod block <u>and</u> SRM DOWNSCALE rod block
Bypass SRM ‘D’ using the joystick at P680.

C. SRM INOP rod block, only
Repair/replace the failed power supply.

D. SRM INOP rod block <u>and</u> SRM DOWNSCALE rod block
Repair/replace the failed power supply.

Answer: A

Explanation:

The SRM Downscale rod block is automatically bypassed when associated IRMs are on Range 3 or higher. Stem condition states IRMs are all on Range 4; therefore, <u>no</u> SRM Downscale rod block will occur. This makes ‘B’ and ‘D’ wrong.

The power supply in P672 powers only a single SRM…SRM ‘D’. Therefore, all we have to do is bypass SRM ‘D’ with the associated joystick at P680, in order to clear the rod block and continue rod withdrawals

For the reasons already discussed, ‘A’ is correct.

‘C’ is wrong. The 2nd part (necessary action) is meant to suggest that the P672 power supply one of the two power supplies that affect dual-SRM channels (such as SRM B & F which are both supplied from the power supply in P670. If this were the case, the joystick would not be a solution…SRM B & F (along with SRM D) share a common joystick. The joystick can only be placed in one SRM position at a time.

Technical References:

GLP-OPS-C5101, pages 19, 20, 23, and 30.
04-1-01-C51-1, NMS SOI, Section 5.1

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-C5101, OBJ. 20

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

Examination Outline Cross-Reference	Level	RO
217000 Reactor Core Isolation Cooling System (RCIC) **Knowledge of the operational implications of the following concepts as they apply to REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) : Flow indication**	**Tier #**	2
	Group #	1
	K/A #	217000 K5.02
	Rating	3.1

Question 49

The plant has scrammed due to a loss of feedwater.

HPCS is not available.

The RCIC Flow Controller at P601 will not respond.

The Rover is directed to establish reactor water level control with RCIC at Remote Shutdown Panel P150.

Which of the following describes the Rover's operation of RCIC at P150?

A. The only way to prevent over-fill as RCIC makes up to the RPV is by the Rover closing E51-F045 RCIC Turbine Steam Supply at the panel.

B. The only way to determine RCIC flow at the panel is by the Rover nulling out the flow controller and reading the flow off the controller vertical tape.

C. The Rover will have to manually swap the RCIC suction over to the suppression pool if it becomes necessary.

D. The control room will have to keep the Rover aware of RCIC turbine speed; if a trip is necessary because of inadequate speed, closing the Trip/Throttle Valve is the only method available.

Answer: B

Explanation:

'A' and 'C' are wrong. These two choices suggest facts that would be true only if operators had evacuated the control room due to a control room fire or security threat and Attachment III of the Remote Shutdown ONEP had already been performed (see Remote Shutdown ONEP, section 3.5.1 and Attachment V, page 1 of 2 (CAUTION). The CAUTION shows that when Att. III is performed RCIC automatic functions are disabled, including the Level 8 turbine shutdown (i.e., auto-closure of F045 and auto-swap of pump suction to supp pool).

'D' is wrong. Although it is true that the only way to trip RCIC at P150 is by closing the

T/T Valve (i.e., the panel has no Trip pushbutton), there is in fact Turbine Speed indication at the panel. There is no need for the control room to keep the Rover informed of speed.

‘B’ is correct. P150 has no dedicated RCIC flow indicator (such as exists in the control room at P601). Only by nulling out the P150 RCIC flow controller can the Rover then read flow off the controller vertical tape.

NOTE – Classifying this question as Higher Cognitive for the following reasons: 1) GGNS history shows that LOT Candidates often don’t recall the detail about having to null out the RCIC flow controller at P150 to read flow there; 2) as such, they would be inclined to <u>strongly</u> consider answer choices ‘A’ and ‘C’; however, the Candidate has to consider the stem condition that got the Rover to P150…we did <u>not</u> evacuate the control room; therefore we are not in the Remote Shutdown ONEP…i.e., we have <u>not</u> performed ONEP Attachment III. This is where the higher cognitive thinking process is employed for the question.
.

Technical References:

05-1-02-II-1, Remote Shutdown Panel ONEP

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-C6100, OBJ. 6

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

Examination Outline Cross-Reference	Level	RO
295028 High Drywell Temperature **Ability to determine and/or interpret the following as they apply to HIGH DRYWELL TEMPERATURE: EA2.04 Drywell pressure**	**Tier #**	1
	Group #	1
	K/A #	295028 EA2.04
	Rating	4.1

Question 50

A severe accident is in progress.

It is suspected that a core melt has caused a breach of the RPV.

Drywell temperature is rising.

Which of the following should also be expected if an RPV breach has in fact occurred?

A. An increasing differential pressure between the RPV and the drywell

B. An increasing differential pressure between the RPV and CTMT

C. A rising drywell pressure coincident with a falling RPV pressure

D. A rising drywell pressure coincident with a rising RPV pressure

Answer: C

Explanation:

The elements of the core breach signature are specified by the Technical Support Guidelines of 05–S–01–SAP–1. The signature requires coincident observation of all of the following: DW temperature rise, DW pressure rise, and DW–to–RPV differential pressure fall.

‘A’ is wrong because it suggests a greater (not a lesser) DW-to-RPV d/p.

‘B’ is wrong because it suggests a greater (not a lesser) RPV-to-CTMT d/p; also, not a part of the Core Breach signature, but face plausible.

‘C’ is correct per the Core Breach signature specified above.

‘D’ is wrong for the reason ‘C’ is correct.

Technical References:

05-S-01-SAP–1

References to be provided to applicants during exam: None

Learning Objective: GLP-EP–EPT19, Obj. 6.1

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

Examination Outline Cross-Reference	Level	RO
218000 Automatic Depressurization System **Knowledge of the effect that a loss or malfunction of the following will have on the AUTOMATIC DEPRESSURIZATION SYSTEM: K6.03 Nuclear boiler instrument system (level indication)**	**Tier #**	2
	Group #	1
	K/A #	218000 K6.03
	Rating	3.8

Question 51

A valid LOCA signal is present.

Both Divisions of ADS automatically initiated and all ADS SRVs have automatically opened.

ADS SRV control switches are still in their AUTO position at P601.

One of the Narrow Range Level transmitters that input to the ADS initiation logics (transmitter B21-N695A) fails upscale.

What impact does this level transmitter failure have on ADS SRV operation <u>and</u> why?

A. No impact, because Div 1 ADS logic has already tripped.

B. No impact, because Div 2 ADS logic is unaffected by this transmitter; if it were, all ADS SRVs would automatically close.

C. Only 4 ADS SRVs automatically close, because only one division of ADS logic is affected.

D. All ADS SRVs automatically close, because affecting even one division of ADS logic impacts all of the ADS SRVs.

Answer: A

Explanation:

After at least one ADS channel has tripped (i.e., as evident by the ADS SRVs having already automatically opened), the +11.4” Narrow Range water level signal input to the ADS initiation logic is no longer relevant (i.e., the level signal does not seal-in). Therefore, an upscale transmitter failure has no impact and the ADS SRVs remain open. For this reason, ‘A’ is correct.

‘B’ is wrong for the reason ‘A’ is correct. The Candidate is expected to deduce that the N695A transmitter label is a Div 1 ADS component; therefore, <u>only</u> that would be the reason why the ADS SRVs are unaffected. This of course is irrelevant for the reason already described. Even if it were the N695B transmitter that failed upscale, the SRVs would still remain open, as already discussed.

‘C’ is wrong for the reason ‘A’ is correct. It also suggests a relationship between ½ of the ADS SRVs and a given ADS Division; there is no such relationship.

‘D’ is wrong for the reason ‘A’ is correct.

Technical References:

GLP-OPS-E2202, page 28 of 47

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-E2202, OBJ. 28

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

Examination Outline Cross-Reference	Level	RO
223002 Primary Containment Isolation System / Nuclear Steam Supply Shut-Off **Knowledge of the physical connections and/or cause- effect relationships between PRIMARY CONTAINMENT ISOLATION SYSTEM / NUCLEAR STEAM SUPPLY SHUT-OFF and the following: K1.02 Reactor water cleanup**	**Tier #**	2
	Group #	1
	K/A #	223002 K1.02
	Rating	3.3

Question 52

SLC ‘A’ has been initiated from the control room.

Which RWCU isolation valve receives an automatic isolation signal?

A. G33-F001, RWCU PUMP SUCTION DRYWELL INBOARD

B. G33-F004, RWCU PUMP SUCTION CTMT OUTBOARD

C. G33-F251, RWCU SUPPLY TO RWCU HXs

D. G33-F252, RWCU RETURN TO RWCU PUMPS

Answer: B

Explanation:

See Automatic Isolations ONEP, pages 17-18, for Group 8 (RWCU) isolation signals.

SLC ‘A’ initiation closes G33-F004, only. Therefore, ‘B’ is correct.

SLC ‘B’ initiation closes G33-F001 and F251. Therefore, ‘A’ and ‘C’ are wrong.

G33-F252 is not affected by SLC initiation. Therefore, ‘D’ is wrong.
.

Technical References:

Automatic Isolations ONEP

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-G3336, OBJ. 9.1

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

Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b)(7) 55.43	

Examination Outline Cross-Reference	Level	RO
239002 Relief/Safety Valves **Knowledge of RELIEF/SAFETY VALVES design feature(s) and/or interlocks which provide for the following: K4.09 Manual opening of the SRV**	**Tier #**	2
	Group #	1
	K/A #	239002 K4.09
	Rating	3.7

Question 53

An operator can manually open a Lo-Lo Set SRV at…

A. Remote Shutdown Panel P150, without any prior switch manipulation.

B. Remote Shutdown Panel P150, but only after placing the Lockout Transfer Relay switch to ON at Alternate Shutdown Panel P152.

C. Alternate Shutdown Panel P152, without any prior switch manipulation.

D. Alternate Shutdown Panel P152, but only after placing the Lockout Transfer Relay switch to ON at Remote Shutdown Panel P150.

Answer: A

Explanation:

All 6 Lo-Lo Set SRVs can manually opened at Remote Shutdown Panel P150, without the need for any prior switch manipulation.

'A' is correct and 'B' is wrong for the reason stated above. The P152 Lockout Transfer Switch in ON would disable the control room controls for the Lo-Lo Set SRVs, but is not necessary in order to simply open them from the Remote Shutdown Panel P150.

'C' and 'D' are wrong because none of the SRVs can be operated from Alternate Shutdown Panel P152.

Technical References:

GLP-OPS-E2202, page 35

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-C6100, OBJ. 12.2

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

	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b)(7) 55.43	

Examination Outline Cross-Reference	Level	RO
259002 Reactor Water Level Control System **Ability to manually operate and/or monitor in the control room: A4.11 High level lockout reset controls**	**Tier #**	2
	Group #	1
	K/A #	259002 A4.11
	Rating	3.5

Question 54

Post-scram level control problems have resulted in a Level 9 trip of both RFPTs.

Currently at P680:

- RFPT A TRIP annunciator is sealed-in
- RX LVL 40"/32" HI/LO annunciator is sealed-in
- RFPT/MN TURB LVL 9 TRIP annunciator is sealed-in

To reset the trip on RFPT 'A' the operator must wait for the…

A. RX LVL 40"/32" HI/LO annunciator to clear, then depress the RFPT A TRIP RESET pushbutton.

B. RFPT/MN TURB LVL 9 TRIP annunciator to clear, then depress the RFPT A TRIP RESET pushbutton.

C. RX LVL 40"/32" HI/LO annunciator to clear, then depress the LEVEL A, B, C SELECTED/DISABLED pushbuttons.

D. RFPT/MN TURB LVL 9 TRIP annunciator to clear, then depress the LEVEL A, B, C SELECTED/DISABLED pushbuttons.

Answer: B

Explanation:

The RFPT/MN TURB LVL 9 TRIP annunciator clearing is indicative of the Level 9 trip signal clearing, at which point the RFPT trip can be reset as soon as the operator depresses the RFPT A TRIP RESET pushbutton. Therefore, 'B' is correct.

'A' is wrong because it suggests that the operator must "Wait" until water level lowers to the point of clearing the RX LVL 40"/32" HI/LO annunciator before being able to reset the RFPT trip by depressing the RFPT A TRIP RESET pushbutton. This is not true.

'C' and 'D' are wrong because the Level A, B, C, Selected/Disabled pushbuttons are used to manually select the level instruments used by the Digital Feed Control System.

Technical References:

ARIs P680-2A-A2, 3A-A3, and 4A2-D1
GLP-OPS-N2100, page 21

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-N2100, OBJ. 13.0

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

Examination Outline Cross-Reference	Level	RO
261000 Standby Gas Treatment System **Ability to monitor automatic operations of the STANDBY GAS TREATMENT SYSTEM including: A3.01 System flow**	**Tier #**	2
	Group #	1
	K/A #	261000 A3.01
	Rating	3.2

Question 55

Both SGTS trains have automatically started on a valid initiation signal.

2 minutes have passed and flows in both SGTS trains are stable.

Which of the following describes how the operator can verify proper SGTS operation on P870?

A. Observe all SGTS FLOW CONTROL DAMPERS indicate a full open position.

B. Observe approximately 7800 scfm on a recorder for SGTS 'A' and approximately 7800 scfm on a recorder for SGTS 'B'.

C. Observe approximately 3900 scfm on a recorder for SGTS 'A' and approximately 3900 scfm on a recorder for SGTS 'B'.

D. Observe all SGTS FLOW CONTROL DAMPERS indicate an intermediate position.

Answer: C

Explanation:

Each train has its own flow recorder on P870. Each train's Exhaust Filter Train Control Vanes regulate flow at 3875 scfm ("approximately 3900 scfm").

For the above reason, 'C' is correct and 'B' is wrong.

'A' and 'D' are wrong. See SGTS SOI, section 5.2.1 NOTE on page 5. Bottom-line, after 90-seconds past the initiation, these flow control dampers are neither <u>all</u> open, nor <u>al</u>l intermediate...rather, they are a combination of the two positions.

Technical References:

SGTS SOI (04-1-01-T48-2), pages 5-6

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-T4801, OBJ. 8.7

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

Examination Outline Cross-Reference	Level	RO
262001 A.C. Electrical Distribution **Knowledge of A.C. ELECTRICAL DISTRIBUTION design feature(s) and/or interlocks which provide for the following: K4.06 Redundant power sources to vital buses**	**Tier #**	2
	Group #	1
	K/A #	262001 K4.06
	Rating	3.6

Question 56

The plant is operating with the following:

- Div 3 DG tagged out for maintenance
- 15AA, 16AB, and 17AC energized

The voltage on all ESF 4160V AC buses drops to 3800 volts for 10 seconds then returns to normal.

Which of the following describes an ESF bus status one minute later?

A. 17AC is de-energized.

B. 16AB is powered from the Div 2 DG.

C. 15AA and 16AB are the only buses powered from offsite sources.

D. 15AA, 16AB, and 17AC are powered from offsite sources.

Answer: D

Explanation:

Initially, all 3 buses are being powered from offsite via the ESF Transformers. The voltage setpoint for the Div 1 and Div 2 LSS 90% BUV trip of offsite bus feeder breakers is 3744 volts for 9 seconds. However, the stem indicates that volts did not drop that low (3800 volts). Therefore, no Div 1 and 2 BUV shed occurs, leaving 15AA and 16AB powered from offsite just as they were before the voltage drop. Div 3 has no LSS; instead, it uses under-voltage relaying to produce a Div 3 BUV trip of the offsite feeders to 17AC and an auto-start of the Div 3 DG (if it were available). This setpoint is 3661 volts for 4 seconds (88% BUV), or 3045 volts with no time delay. Again, with voltage dropping no lower than 3800 volts, the Div 3 BUV does not occur, leaving 17AC powered from offsite just as before the voltage drop.

For the reasons described above, ‘D’ is correct and ‘A’, ‘B’, ‘C’ are wrong.

Technical References:

GLP-OPS-R2100, pages 19, 28

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-R2100, OBJ. 12, 13

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

Examination Outline Cross-Reference	Level	RO
262002 Uninterruptable Power Supply (A.C./D.C.) **Knowledge of UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.) design feature(s) and/or interlocks which provide for the following: K4.01 Transfer from preferred power to alternate power supplies**	**Tier #**	2
	Group #	1
	K/A #	262002 K4.01
	Rating	3.1

Question 57

A UPS static switch will automatically transfer load from the inverter to the alternate source in response to exceeding the …

A. over-frequency setpoint.

B. over-voltage setpoint

C. over-current setpoint.

D. over-temperature setpoint.

Answer: C

Explanation:

Of the choices, the inverter has only an over-current setting for auto-transfer of its load to the alternate power source. Therefore, choice ‘C’ is correct.

Note – We’ve applied the “setpoint” phrase to each answer choice to allow the use of over-temperature as the ‘D’ distracter. Otherwise, one could argue that an unchecked over-temperature condition could ultimately lead to inverter circuit failure, which in turn could lead to short-circuited over-current conditions…thus making ‘D’ a 2nd correct answer. However, as presented here, ‘D’ suggests there is an “over-temperature setpoint” (i.e., some given inverter internal (cabinet/panel) temperature at which the UPS will auto-transfer); no such auto-transfer function exists.

Technical References:

GLP-OPS-L6200, page 11

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-L6200, OBJ. 4.1

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

Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b)(4) 55.43	

Examination Outline Cross-Reference	Level	RO
400000 Component Cooling Water System (CCWS) **Ability to manually operate and / or monitor in the control room: A4.01 CCW indications and control**	**Tier #**	2
	Group #	1
	K/A #	400000 A4.01
	Rating	3.1

Question 58

Operators are starting up the CCW system.

The Rover has started two of the CCW Pumps and intends to place the third CCW Pump in Standby.

In error, the Rover depresses the STANDBY pushbutton for one of the two running pumps.

What additional operator action is required to recover from this operator error (i.e., have two pumps running and only one pump in Standby)?

A. Depress the STANDBY pushbutton for the idle pump, only.

B. Depress the STANDBY pushbutton for the running pump a second time then depress the STANDBY pushbutton for the idle pump.

C. Start the third pump, then stop the running pump that was mistakenly placed in Standby; depress the STANDBY pushbutton for the pump that was just stopped.

D. Depress the STANDBY pushbutton for the idle pump then depress the STANDBY pushbutton for the running pump a second time.

Answer: C

Explanation:

A running CCW Pump <u>will</u> go into Standby when its STANDBY pb is depressed. Only two conditions will take a CCW Pump out of the Standby mode: 1) an auto-trip signal to that pump; 2) stopping the pump with its control switch. The STANDBY pb is also used to place an idle pump in Standby mode.

'A' is wrong; it suggests that the running pump never actually went into Standby…it did. This action would result in two pumps being left in Standby.

'B' is wrong; depressing the STANDBY pb a second time does nothing to take a pump back out of Standby. This action would result in two pumps being left in Standby.

‘C’ is correct for the reasons described above.

‘D’ is wrong; again, depressing the STANDBY pb on the running pump a second time will not remove it from a Standby mode. The action suggested by this choice would result in two pumps being left in Standby.

Technical References:

GLP-OPS-P4200, pages 12-13
P42 SOI (04-1-01-P42-1), page 4
E-1226, sheets 1, 2, 3

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-P4200, OBJ. 7.1

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

Examination Outline Cross-Reference	Level	RO
290003 Control Room HVAC **Ability to monitor automatic operations of the CONTROL ROOM HVAC including: A3.01 Initiation/reconfiguration**	**Tier #**	2
	Group #	2
	K/A #	290003 A3.01
	Rating	3.3

Question 59

The plant is operating at rated power when a valid Control Room HVAC isolation occurs.

No operator action has yet been taken with respect to Control Room HVAC (Z51).

Which of the following are <u>among</u> the resulting Z51 damper/valve indications the operator should observe at control room panel P855?

A. F010 CONT RM NORM O/A INBD INL VLV – **CLOSED**
F007 CONT RM FRESH AIR UNIT A INL/RECIRC VLV – **OPEN**
F005 CONT RM FRESH AIR UNIT A OUTL DMPR – **OPEN**

B. F011 CONT RM NORM O/A OTBD INL VLV – **OPEN**
F014 CONT RM FRESH AIR UNIT B RECIRC VLV – **OPEN**
F013 CONT RM FRESH AIR UNIT B OUTL DMPR – **OPEN**

C. F003 CONT RM UTILITY EXH FAN INBD INL VLV – **CLOSED**
F016 CONT RM FRESH AIR UNIT B INL/RECIRC VLV – **CLOSED**
F013 CONT RM FRESH AIR UNIT B OUTL DMPR – **OPEN**

D. F002 CONT RM PURGE INBD EXH VLV – **CLOSED**
F008 CONT RM FRESH AIR UNIT A RECIRC VLV – **CLOSED**
F005 CONT RM FRESH AIR UNIT A OUTL DMPR – **CLOSED**

Answer: C

Explanation:

See 04-S-01-Z51-1, section 5.4.2.b & c.

‘A’ is wrong because the Fresh Air Inlet (F007) auto-closes on the isolation signal.

‘B’ is wrong because the Control Room Normal Outside Air Outboard (F011) auto-closes on the isolation signal.

‘C’ is correct per sections 5.4.2.b & c.

‘D’ is wrong because Fresh Air Unit ‘A’ Outlet Damper (F005) auto-opens on the isolation signal.

Technical References:

04-S-01-Z51-1, sections 5.4.2.b & c, pages 11-12

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-Z5100, OBJ. 8

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

Examination Outline Cross-Reference	Level	RO
2.1.32 Ability to explain and apply system limits and precautions.	**Tier #**	3
	Group #	
	K/A #	2.1.32
	Rating	3.8

Question 60

A plant startup is in progress with all IRMs on range 6.

A plant problem has caused a 20-minute interruption in control rod withdrawals.

The 20-minute interruption has allowed the reactor to go subcritical again.

The next step in the control rod sequence is designated as continuous gang mode withdrawal.

Per IOI-1, Cold Shutdown To Generator Carrying Minimum Load, how should control rod withdrawal proceed, and why?

A. Individually notch control rods at least to the point of re-achieving criticality, because there is no void feedback available to turn power.

B. Individually notch control rods at least to the point of adding heat, because there is no temperature feedback available to turn power.

C. Continuously withdraw the next control rod gang, because the control rod sequence already accounts for the possibility of having to re-achieve criticality.

D. Continuously withdraw the next control rod gang, because the interruption time for rod withdrawals has been less than 30 minutes.

Answer: A

Explanation:

See IOI-1, P/Ls 2.1.6, 2.1.16, 2.1.17. 2.1.6 tells us not to take the reactor critical in Continuous Withdrawal mode. 2.1.17 speaks to short-term (<30 minute) delays in low rod worth rod withdrawals and the effect of the negative temperature coefficient…subcriticality. It allows for recommencing rod withdrawals to achieve re-criticality only by using notch withdrawals until criticality is achieved. 2.1.16 is the P/L that speaks to the "Why" for all of this…no available VOID (coefficient of reactivity) feedback available to turn power when we're operating close to the point of criticality.

For the above reasons, ‘A’ is correct and ‘B’, ‘C’, ‘D’ are wrong.

Technical References:

IOI-1, Cold Shutdown To Generator Carrying Minimum Load

References to be provided to applicants during exam: None

Learning Objective:

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

Examination Outline Cross-Reference	Level	RO
2.2.6 Knowledge of the process for making changes to procedures.	**Tier #**	3
	Group #	
	K/A #	2.2.6
	Rating	3.0

Question 61

Which of the following describes an EN-AD-102, Procedure Adherence and Level of Use, requirement concerning an editorial error identified in a surveillance procedure?

A. Can make a pen & ink correction regardless of when such an error is identified.

B. Can make a pen & ink correction only if the error is identified while the procedure is in use.

C. Can make a pen & ink correction only to correct a typo; editorial changes beyond that require a procedure change request.

D. Cannot correct such an error in a surveillance procedure using pen & ink; must submit a procedure change request.

Answer: B

Explanation:

See EN-AD-102, section 5.2.4. This section describes the process for making pen & ink corrections to editorial errors in technical procedures identified while the procedure is in use. A surveillance procedure is such a technical procedure. This section's exclusivity make it understood that using such correction method is acceptable only while the procedure is in use and not before-hand.

For the above reasons, 'B' is correct and 'A' is wrong.

'C' is wrong: EN-AD-102 does not differentiate between typos and other types of editorial changes.

'D' is wrong; per section 5.2.4[1(c), the change request is to be submitted as a follow-up to the pen & ink correction.

Technical References:

EN-AD-102, Procedure Adherence and Level of Use

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-PROC, OBJ. 81.6

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

Examination Outline Cross-Reference	Level	RO
400000 Component Cooling Water System (CCWS) **Ability to predict and / or monitor changes in parameters associated with operating the CCWS controls including: A1.02 CCW temperature**	**Tier #**	2
	Group #	1
	K/A #	400000 A1.02
	Rating	2.8

Question 62

A rising CCW system temperature has resulted from the PSW to CCW Temperature Control Valve failing closed.

Per the Subsequent Actions of the Loss of CCW ONEP, CCW flow to the FPCC Heat Exchangers has been isolated.

After 10 minutes, CCW system temperature has risen to Spent Fuel Pool temperature.

Per the ONEP, operators are re-establishing CCW flow to the FPCC Heat Exchangers.

The Rover re-opens P42-F105 CCW SUPPLY TO FPHX A/B and P42-F205 CCW RETURN FROM FPHX A/B using their handswitches at control room panel P870.

After stable CCW flow is established through the FPCC Heat Exchangers, what should be the response of CCW system temperature?

A. Temperature rapidly turns and begins to lower.

B. Rate of temperature rise decreases but temperature continues to rise.

C. Temperature rapidly levels off and stabilizes.

D. Rate of temperature rise increases while temperature continues to rise.

Answer: B

Explanation:

The ONEP revision that added step 3.4 and its NOTE resulted from a GGNS Engineering Evaluation that shows how the Spent Fuel Pool water volume actually becomes a heat-sink (rather than a heat load) for the CCW system after CCW system temperature reaches pool temperature. At that time, the pool water volume dramatically reduces the rate of CCW system rise, although CCW system temperature does in fact continue to rise. For this reason, only 'B' is correct.

Technical References:

05-1-02-V-1, Loss of CCW ONEP
"Engineering Evaluation of the Loss of CCW Transient Simulation" (and its "Chart 1")

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-ONEP, OBJ. 2.0

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

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

Question 63

A Site Area Emergency is declared at 1411.

Which of the following complies with the time limit requirements for the initial and first follow-up notifications to State/Local Agencies?

A. Initial is completed at 1426; first follow-up is completed at 1531.

B. Initial is completed at 1430; first follow-up is completed at 1525.

C. Initial is completed at 1425; first follow-up is completed at 1527.

D. Initial is completed at 1424; first-follow-up is completed at 1524.

Answer: D

Explanation:

Per 10-S-01-6, sections 6.1.1.c and f, the initial notification is to be completed within 15 minutes of declaration (within 15 minutes of 1411) and the follow-up is to be completed within one hour of the initial.

Of the answer choices, only 'D' satisfies the requirements; thus, 'D' is correct.

'A' is wrong; both the initial and follow-up are late.

'B' is wrong; the initial is late.

'C' is wrong; the follow-up is late.

NOTE – At GGNS, the RO is a Control Room ERO Communicator, responsible for making the Initial and Follow-up Notifications tested in this question. As such he/she would also be responsible for ensuring such notifications are made within the required time limits. Therefore, this question is in fact a legitimate RO-level question for GGNS, and not an SRO-only one.

NOTE – Classifying this question as Higher Cognitive on the basis of "analysis". The

Candidate must use the same cognitive thinking process to determine that choice ‘D’ is acceptable to E-Plan requirements for initial/follow-up notification as would, say, the NRC were they to observe an actual pair of such notifications during a real plant emergency. That observer would have to consider the 15-minute and 1-hour “fundamental” procedural requirements and analyze the times the actual notifications are completed to determine if the requirements were met. Admittedly not an extremely difficult “thinking” task, but a higher cognitive one nonetheless.

Technical References:

10-S-01-6, Notification of Offsite Agencies

References to be provided to applicants during exam: None

Learning Objective: GLP-EP-EPTS6, OBJ. 3

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

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

Question 64

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

Per the CRD Malfunctions ONEP, which of the following describes an Immediate Operator Action?

A. If a scram accumulator alarm is received on one withdrawn control rod, place the Mode Switch in SHUTDOWN.

B. Place the CRD Flow Controller in Manual, reduce its output to zero, start the other CRD pump, slowly raise CRD flow to about 60 gpm in Manual, return the Controller to Automatic.

C. If a scram accumulator alarm is received on more than one withdrawn control rod, place the Mode Switch in SHUTDOWN.

D. With the CRD Flow Controller in Automatic reduce its output to zero, start the other CRD pump, slowly raise CRD flow to about 60 gpm with the Controller in Automatic.

Answer: B

Explanation:

See ONEP section 2.1 (CRD Pump Trip).

‘A’ and ‘C’ are wrong; an immediate scram is a consideration only when reactor pressure is less than 600 psig (we’re at rated power for this question).

‘B’ is correct (and thus, ‘D’ is wrong); per step 2.1.2, the controller is in Manual for the start of the standby pump, not in automatic.

Technical References:

05-1-02-IV-1, CRD Malfunctions

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-ONEP, OBJ. 1.0

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

Examination Outline Cross-Reference	Level	RO
203000 RHR/LPCI: Injection Mode (Plant Specific) **Ability to predict and/or monitor changes in parameters associated with operating the RHR/LPCI: INJECTION MODE (PLANT SPECIFIC) controls including: A1.08 †Emergency generator loading**	**Tier #**	2
	Group #	1
	K/A #	203000 A1.08
	Rating	3.7

Question 65

A LOCA is in progress.

LPCI ‘B’ and ‘C’ are injecting to the RPV when a complete loss of offsite power occurs.

Which of the following describes the loading of the Div 2 DG as RHR Pumps ‘B’ and ‘C’ are restarted?

Div 2 DG first picks up about…

A. 800 KW when RHR Pump ‘B’ restarts then another 800 KW when RHR Pump ‘C’ restarts.

B. 800 KW when RHR Pump ‘C’ restarts then another 800 KW when RHR Pump ‘B’ restarts.

C. 200 KW when RHR Pump ‘B’ restarts then another 200 KW when RHR Pump ‘C’ restarts.

D. 200 KW when RHR Pump ‘C’ restarts then another 200 KW when RHR Pump ‘B’ restarts.

Answer: B

Explanation:

Each RHR Pump in LPCI full-flow draws about 800 KW (see Loss of AC Power ONEP, Section 3.2, Table 1). Thus, ‘C’ and ‘D’ are wrong.

See the LSS SOI, Table 1 for the Div 2 Load Shed & Sequencing Timing of the RHR Pumps B and C. With a LOCA signal present (stem conditions), the ‘C’ RHR Pump auto-restarts 2.5 seconds after the DG re-powers the bus, and the ‘B’ RHR Pump auto-restarts 7.5 seconds after the DG re-powers the bus. Thus, ‘B’ is correct and ‘A’ is wrong.

Technical References:

04-1-01-R21-1, LSS SOI
LOSS OF AC POWER ONEP, 05-1-02-I-4

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-R2100, OBJ. 15, 16

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

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

Question 66

The plant is operating at rated power when the Div 2 battery charger (1B4) DC output breaker trips open (breaker fault).

Which of the following identifies the initial response of 11DB bus voltage as indicated on control room panel P864?

A. Remains constant.

B. Lowers by 5 to 10 volts.

C. Lowers by 60 to 65 volts.

D. Goes to zero volts.

Answer: A

Explanation:

Div 2 125 VDC Bus 11DB has two load-sharing chargers (1B4 and 1B5) that are always connected to the bus and handling the bus load at approximately 130 VDC. If one charger is lost (as in the case of this question's stem conditions), the other connected charger simply continues to handle the load by itself without any impact on indicated bus voltage. Thus, answer choice 'A' is correct and 'B', 'C', 'D' are wrong.

'B' is plausible in that is represents the voltage response if the Div 3 bus charger (a single charger only) were to fail…in this case the bus battery would take over, where battery terminal volts are about 5-10 volts lower than charger output voltage.

'C' and 'D' provide sufficient face plausibility to the weaker Candidate who has never fully grasped basic GFE electrical concepts and/or GGNS DC electrical distribution fully.

Technical References:

04-1-01-L11-1, PLANT DC SOI

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-L1100, OBJ. 19

Question Source:	Bank # GGNS-OPS-07461a	X
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b)(4)	
	55.43	

Examination Outline Cross-Reference	Level	RO
201005 Rod Control and Information System (RCIS) **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 #**	2
	Group #	2
	K/A #	201005 2.2.44
	Rating	4.2

Question 67

A power ascension is in progress.

Reactor power is stable as indicated by:

- APRMs reading 40%
- Core Thermal Power reading 1560 MWt

The current RPC MODE display (P680) is shown below.

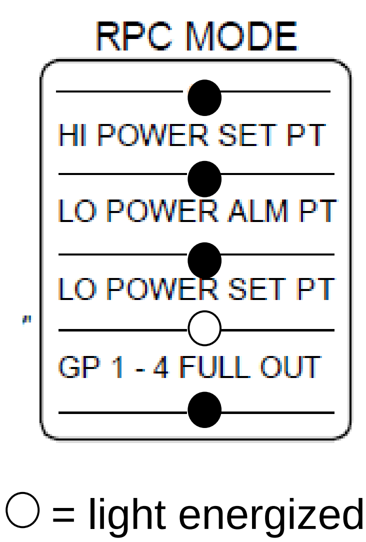
Considering the RPC MODE display,…

A. it appears that RC&IS is operating per design.

B. it appears that the APRM Gain Adjustment Factors (AGAFs) are out-of-spec.

C. the ACRO should check the position of the Bypass Valves.

D. the ACRO should recommend that the Reactor Engineer verify the correct sequence is selected for RC&IS.



Answer: C

Explanation:

Any amount that Bypass Valves are open robs steam away from the HP Turbine. If actual reactor power (as indicated by APRMs and calculated CTP) is 40%, the HP Turbine 1st Stage pressure will, consequently, be less than that which corresponds to 40% because of the steam going through the Bypass Valves. RC&IS "sees" power lower than it actually is (recall that RC&IS only knows reactor power in terms of HP 1st Stage pressure). Because it does, the lamp directly above the LO POWER SET PT (setpoint =25%) label on the RPC MODE display is still de-energized. In other words, RC&IS doesn't yet "believe" that power has risen above the LPSP.

For these reasons, RC&IS is certainly not operating per design…making choice 'A' wrong.

The ACRO should check to see if the Bypass Valves are in fact open some amount (the likely cause of this symptom)…making choice 'C' correct.

'B' is wrong; it is plausible to the Candidate who forgets that RC&IS does not depend on ARPMs for its power input. APRMs are clearly not the problem in this given situation because their 40% reading is exactly consistent with the 40% calculated CTP (i.e., 1560 MWt = 40% of 3898 MWt).

'D' is simply wrong; it has face plausibility to the Candidate who forgets that the RC&IS sequence is irrelevant at 40% power.

Technical References:

T.S. LCO Bases 3.3.2.1
03-1-01-2, Power Ascension IOI, Att. I, Section 5.3 (provides the directive to ensure Bypass Valves are closed before raising power above the LPSP)

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-C1102, Objective 14.3

Question Source:	Bank # GGNS-OPS-08688	X
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b)(7)	
	55.43	

Examination Outline Cross-Reference	**Level**	RO
234000 Fuel Handling Equipment	**Tier #**	2
Knowledge of the physical connections and/or cause effect relationships between FUEL HANDLING EQUIPMENT and the following: K1.01 Fuel	**Group #**	2
	K/A #	234000 K1.01
	Rating	3.2

Question 68

Core Alterations are in progress.

All control rods are full-in.

The refueling platform is currently moving toward the core; its main grapple is loaded with a fuel bundle.

Which of the following will cause Refueling Platform motion to automatically stop when it gets close enough to the core?

A. RC&IS sensing one control rod not full-in

B. Placing the Mode Switch in REFUEL

C. Selecting a control rod on P680

D. Selecting Individual Drive mode for RC&IS

Answer: A

Explanation:

Question tests the Bridge Reverse Stop #1: With the M/S in REFUEL **and** the main grapple loaded, the platform is allowed to move over the core **unless** one control rod is not full-in. For this reason, only choice 'A' is correct.

'B' is wrong. The M/S must be in REFUEL; otherwise, the platform will stop.

'C' is wrong. Per the Bridge Reverse Stop #2 interlock, only when there is **no** control rod that is selected would there be an automatic platform stop.

'D' is wrong. Although the GANG Drive mode of RC&IS does factor into Stop #2, there is no such relationship between the Individual Drive mode of RC&IS and either one of the platform motion interlocks.

Technical References:

GLP-RF-F1101, pages 40-41

References to be provided to applicants during exam: None

Learning Objective: GLP-RF-RF1101, OBJ. 11.0

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

Examination Outline Cross-Reference	Level	RO
264000 Emergency Generators (Diesel/Jet) **Ability to predict and/or monitor changes in parameters associated with operating the EMERGENCY GENERATORS (DIESEL/JET) controls including: A1.09 Maintaining minimum load on emergency generator (to prevent reverse power)**	**Tier #**	2
	Group #	1
	K/A #	264000 A1.09
	Rating	3.0

Question 69

DG12 is about to be paralleled to the 16AB bus for its monthly surveillance run.

The Rover synchronizes the two power sources and successfully closes the DG output breaker.

As a <u>result</u> of the Rover performing **NEXT** required step in the procedure, what P864 indication should he **FIRST** observe for DG12?

A. VARs reach +0.25 MVARs and stabilize there.

B. VARs reach -0.25 MVARs and stabilize there.

C. WATTS reach 350 KW and stabilize there.

D. WATTS reach 1000 KW and stabilize there.

Answer: C

Explanation:

See the monthly surveillance, 06-OP-1P75-M-0002, pages 28-29.

‘A’ and ‘B’ are wrong. Whether we choose to pick up leading or lagging VARs on the machine, in either case, this step <u>follows</u> the step requiring us to **first** establish a minimum load of 350 KW (in order to prevent a reverse power trip of the DG).

‘C’ is correct and ‘D’ is wrong. Step 5.2.21.h requires 350 KW. The 1000 KW load is the first normal increment toward establishing the full load for the monthly run (see step 5.2.21.k).

NOTE – Classifying this question as Higher Cognitive based on the following: Candidates are not expected to recall (from memory) such detail from procedures such as surveillance procedures. Therefore, the Candidate will not necessarily recognize the 350 KW value as opposed to the 1000 KW value. The MVAR distracters to consider make this recognition even more so difficult a task. The collection of answer choices forces a thinking process that is certainly a higher cognitive one.

Technical References:

06-OP-1P75-M-0002, SDG 12 Functional Test

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-P7500, OBJ. 19

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

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

Question 70

RHR ‘A’ is operating in suppression pool cooling per the E12-1 RHR SOI.

Contrary to the SOI requirements, the Rover has both the RHR A HX OUTLET (E12-F003A) <u>and</u> RHR A HX BYPASS (E12-F048A) throttled to control suppression pool temperature.

What is the operational concern for using the heat exchanger valves in this manner to cool the suppression pool?

A. May result in lowering the pool temperature to where the ‘B’ feedwater flued head (RCIC line) low temperature limit is approached.

B. May produce excessive head loss that results in pump cavitation.

C. May exceed the maximum allowable heatup rate for the heat exchanger tube sheet.

D. May produce cavitation in the valves that could lead to piping failure from excessive vibration.

Answer: D

Explanation:

‘A’ is wrong. This choice is taken from the E12-1 SOI P/L 3.7. There is a FSAR low temperature supp pool limit of 65°F for the ‘B’ feedwater line flued head (a brittle fracture concern). RCIC (which takes supp pool water suction) injects via the ‘B’ feedwater line near that component. However, the simultaneous throttling of these two valves has no bearing on the ability to control the pool cooldown.

‘B’ is wrong; the two valves partially throttled produce no more head loss than does a single one of them throttled that much more. Besides, they would have to produce a lot of head loss in order to cavitate an RHR pump that is taking a suction on a pool of water some 18-19’ feet deep, especially when the pump’s NPSH requirement is only 14.5’.

‘C’ is wrong; again, there is no evidence that this throttling combination would do anything to produce an excessive heatup rate of the HX tube sheet…assuming there even is a specified “max allowable heatup rate” for it.

‘D’ is correct. This is based on actual GGNS O.E. (see References below). This operating experience occurred in 2002. A follow-up Engineering Analysis in 2002 confirmed the concern as being just as described in answer choice ‘D’.

Technical References:

04-1-01-E12-1, P/L 3.6.1, CAUTION on page 23
ER-GG-2002-0400-000-0, Engineering Analysis for Throttling Restrictions on E12-F003 and F048

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-E1200, OBJ. 14

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

Examination Outline Cross-Reference	Level	RO
2.2.1 Ability to perform pre-startup procedures for the facility, including operating those controls associated with plant equipment that could affect reactivity.	**Tier #**	3
	Group #	
	K/A #	2.2.1
	Rating	4.5

Question 71

A reactor startup is in progress.

Control rod withdrawals have begun.

Per IOI-1, Cold Shutdown to Generator Carrying Minimum Load, which of the following identifies 1) an acceptable Verifier for the rod movements, and 2) the preferred method for placekeeping each step on the pull sheet?

A. 1) Reactor Engineer
   2) initials

B. 1) Licensed Operator
   2) circle-slash

C. 1) Licensed Operator Candidate
   2) circle-slash

D. 1) AOM (does not require license)
   2) initials

Answer: A

Explanation:

See IOI-1, P/L 2.1.13. The Verifier must one of: RE, STA, or Licensed Operator. Instead of the usual circle-slash method (for almost everything else at GGNS), initialing has been substituting as the placekeeping preferred method for the pull sheet.

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

Note – We've chosen not to use the 'STA' in our distracters so as not to confuse our Candidates. GGNS does not have non-licensed (stand-alone) STAs. We're concerned that an answer choice that suggests such might be 'more distracting' than what is required by discrimination validity itself.

Technical References:

IOI-1

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-IOI01, Obj. 3

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

Examination Outline Cross-Reference	Level	RO
2.4.18 Knowledge of the specific bases for EOPs.	**Tier #**	3
	Group #	
	K/A #	2.4.18
	Rating	3.3

Question 72

Table 5
Alternate ATWS Systems

- HPCS
  - ☛ Use CST suction if available.
  - ☛ OK to defeat interlocks.................... **Att 4**
- LPCS
- LPCI
  - ☛ Inject through the HX as soon as possible.
- RHR Service Water Crosstie
- Condensate Transfer..........................**Att 25**
- Fire System.......................................**Att 26**
- ECCS jockey pumps
- SLC test tank...................................**Att 27**
- SLC boron tank

The EP-2A Level leg does not allow us to consider using Table 5 Alternate ATWS Systems to restore and maintain level above -191” until <u>after</u> an Emergency Depressurization is performed.

Which of the following is the basis for this strategy?

A. To reduce reactor power as much as possible by voiding before injecting with these systems.

B. To establish a low-enough reactor pressure so that all available systems are able to inject.

C. To avoid having to inject with low-quality water systems until absolutely necessary.

D. To ensure sufficient time has been allotted for the Power leg Attachments before diverting resources to Table 5 Attachments.

Answer: A

Explanation:

See EP Technical Bases (02-S-01-40), Attachment V, page 31 of 62, bottom-most 3 bullets.

‘A’ is correct per the reference cited above. The ED rapidly depressurized the core, significantly reducing power in advance of injecting with Table 5 systems…depending upon which of these systems is used to inject, large power excursions may results…the lower the power beforehand, the less the impact of the power excursion. This is the fundamental reason for the pre-emptive ED in step L-9 pf EP-2A.

‘B’ is wrong. This is a very strong distracter that may be argued as a second correct answer…it is NOT! While it is true that if level cannot be restored using Table 4 systems and all we have available beyond those is LPCI (Table 5), low-pressure ECCS, then we will have to ED in order for LPCI to be able to inject at all, this is not the “basis” for this step of EP-2A. This is evident when we consider all of the Table 5 systems, which include high-pressure injection systems as well: HPCS, SLC.

‘C’ is wrong. While it is true that some of the Table 5 systems are of low-quality water (e.g., Fire Water), we would never delay the preservation of adequate core cooling simply for the sake of a concern for the quality of the water with which we cool the core.

‘D’ is wrong; it has only face plausibility considering Table 5 and what Candidates (both RO and SRO) know about the Power leg Attachments (especially Attachments 18, 19, 20) of EP-2A during an ATWS.

Technical References:

EP Technical Bases, Attachment X, Section 11.17

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-EP02A, OBJ. 7

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

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

Question 73

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

Assuming post-scram reactor water level never rises above the scram setpoint, what will be the setpoint on the FW Master Level Controller 7 seconds after the scram?

A. 54.0”

B. 36.0”

C. 18.0”

D. 12.4”

Answer: A

Explanation:

The scram setpoint on low level is 11.4”. DFCS (Water Level Control) also initiates “Setpoint Setdown” at this same water level (11.4”). Once initiated, the FW Master Level Controller setpoint changes to 54.0” and remains there for either 10 seconds, or until 12.4” is reached, at which time the setpoint is automatically changed to 18.0”. Since the stem conditions “assume” that water level never rises above the scram setpoint of 11.4”, and asks what the setpoint of the Controller is only 7 seconds after Setpoint Setdown was first initiated, this means that the 54.0” setpoint is still in effect 7 seconds after the scram.

For these reasons, ‘A’ is correct and ‘B’, ‘C’, ‘D’ are wrong.

Technical References:

GLP-OPS-C3401 lesson plan

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-C3401, Objective 12

Question Source:	Bank # GGNS-LORQT-06088	X
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b)(10)	
	55.43	

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

Question 74

The plant is operating at rated power when a ground develops on the 11DB bus (on the bus itself).

The ground current is steady at 0.550 ma.

What alarm/indication, if any, should be present in the control room related to this ground?

A. No alarm/indication

B. Alarm sealed-in at P864

C. Slight drop in 11DB bus volts at P864

D. Slightly fluctuating 11DB bus volts at P864

Answer: B

Explanation:

‘B’ is correct; per ARI P864-2A-H4, this annunciator alarms at a bus ground current of 0.412 ma (+ or -). As such, ‘A’ is also wrong.

‘C’ and ‘D’ are wrong. At this level of ground current, we are still a long way away from any adverse impact on bus voltage; there is no reason to expect any change in indicated bus volts. The P864-2A-H4 essentially validates this claim by the fact that it doesn’t even call for “action” until a ground current of 0.9 ma is reached.

Technical References:

ARI P864-2A-H4

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-P7500, OBJ. 12.0

Question Source: Bank #

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

Examination Outline Cross-Reference	Level	RO
264000 Emergency Generators (Diesel/Jet) **Knowledge of the purpose and function of major system components and controls.**	**Tier #**	2
	Group #	1
	K/A #	264000 2.1.28
	Rating	4.1

Question 75

Per the SOI, operators have placed DG 11 in MAINTENANCE mode using the local pushbutton and the DG 11 MAINT PERM pushbutton at control room panel P864.

Which of the following describes the operation of DG 11 in this mode?

A. Cannot be manually started; will automatically start on BUV or LOCA.

B. Cannot be manually started; will automatically start on LOCA, only.

C. Can only be manually started.

D. Will not start, manually or automatically.

Answer: D

Explanation:

In Maintenance, the engine is totally locked out and incapable of starting, manually or automatically. Therefore, 'D' is correct and 'A', 'B', 'C' are wrong.

Technical References:

E-1110, sheet 12

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-P7500, tbd

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

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

Examination Outline Cross-Reference	Level	SRO
295024 High Drywell Pressure	**Tier #**	1
	Group #	1
2.2.25 Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits.	**K/A #**	295024 2.2.25
	Rating	4.2

Question 76

Per the Technical Specification Bases, which of the following is a basis for the High Drywell Pressure scram function of RPS?

A. Ensures the reactor vessel water level Safety Limit is not exceeded.

B. Reduces the amount of energy added to the reactor coolant.

C. Initiates low-pressure ECCS systems in response to a LOCA.

D. Ensures suppression pool water is not forced into the drywell during a LOCA.

Answer: B

Explanation:

See Tech Spec Bases discussion B 3.3.1.1, Function 7 (page B 3.3-13 &13.a). Of the answer choices, only 'B' is one of the Bases for this particular scram function.

Technical References:

Bases for Tech Spec LCO 3.3.1.1, Function 7, and LCO 3.6.5.4

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-TS001, OBJ. 26

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

Examination Outline Cross-Reference	Level	SRO
600000 Plant Fire On Site **AA2.14 Equipment that will be affected by fire suppression activities in each zone**	**Tier #**	1
	Group #	1
	K/A #	600000 AA2.14
	Rating	3.6

Question 77

Use your provided references to answer this question.

A fire has started in fire zone 1A208, ESF Switchgear Room.

The fire brigade has requested all electrical switchgear located in the room be de–energized to aid in suppressing the fire.

For which of the following systems will this action result in the loss of equipment required to achieve and maintain cold shutdown?

A. RHR ‘A’

B. RCIC

C. LPCS

D. RWCU

Answer: A

Explanation:

10–S–03–2 identifies a number of components affected by fires in the area, including valves associated with RCIC and RWCU. However, neither of those systems are credited in the UFSAR safe shutdown analysis for achieving and maintaining cold shutdown conditions in the event of a fire, making answers ‘B’ and ‘D’ wrong.

Answer ‘C’ is wrong; LPCS is a Division I ECCS system but is not credited in the safe shutdown analysis and will not be directly affected.

Answer ‘A’ is correct; RHR ‘A’ is a safe shutdown system and will be unable to be aligned in any mode to assist in establishing and maintaining cold shutdown on de–energization of the affected switchgear.

Technical References:

10-S–03–2
UFSAR Appendix 9C, Safe Shutdown Analysis

References to be provided to applicants during exam:

10-S–03–2, Response to Plant Fires: cover page; page 9; Att. II; Att. III; Att. IV pages 47-50

Learning Objective: GLP-OPS-P6400 Obj. 2
GLP–OPS–PROC Obj. 65.1

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

Examination Outline Cross-Reference	Level	SRO
295032 High Secondary Containment Area Temperature **Knowledge of surveillance procedures.**	**Tier #**	1
	Group #	2
	K/A #	295032 2.2.12
	Rating	4.1

Question 78

The plant is operating at rated power.

Daily Operations Log Surveillance Data Sheet II (Day Shift) is completed at 0800 for the Secondary Containment Area Temperatures Channel Check required by Tech Spec SR 3.3.6.1.1.

This is a 12-hour surveillance requirement.

At 2000 that same day, the CRS is reviewing the Daily Operations Log Surveillance Data Sheet III (Night Shift).

The CRS notes that the surveillance has not been performed since 0800 that day.

What is the **LATEST** time by which this surveillance can be completed and comply with the NRC Late Date time?

A. 2000 hours that day

B. 2115 hours that day

C. 2300 hours that day

D. 0800 hours the next day

Answer: C

Explanation:

The maximum allowed surveillance frequency is 1.25 times the completion time. For a 12 hour completion time the maximum allowed completion time is 15 hours per GGNS Tech Specs SR3.0.2 and Administrative Procedure 01-S-06-12 section 6.4.4. 15 hours added to 0800 would be 2300. Therefore, the correct answer is ‘C’.

‘A’ is wrong. It suggests there is no “25% grace period” allowed; there is.

‘B’ is wrong. This choice is for the SRO Candidate who cannot accurately recall the 1.25 x frequency…instead, he/she adds 1.25 hours to the surveillance interval (12 + 1.25 hours =

13.25…0800 + 13.25 = 2115).

‘D’ is wrong. It suggests a grace period of twice the interval, which is wrong.

Technical References:

06-OP-1000-D-0001
Tech Spec 3.3.6.1
Tech Spec SR3.0.2
Administrative Procedure 01-S-06-12 section 6.4.4a.

References to be provided to applicants during exam: None

Learning Objective:

GLP-OPS-PROC Objective 15.5
GLP-OPS-TS001 Objectives 8 & 9

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

Examination Outline Cross-Reference	Level	SRO
2.1.34 Knowledge of primary and secondary plant chemistry limits.	**Tier #**	3
	Group #	
	K/A #	2.1.34
	Rating	3.5

Question 79

TRM 6.4.1, Reactor Coolant System Chemistry, specifies the limit(s) for…

A. Specific Activity for Dose Equivalent Iodine.

B. Total Activity for dissolved noble gases.

C. Conductivity, Silica, and pH.

D. Chlorides, Conductivity, and pH.

Answer: D

Explanation:

Per TRM 6.4.1, only 'D' is correct.
'C' is wrong because of the Silica claim.
'A' is wrong; it represents the RCS Specific Activity limits of Tech Spec LCO 3.4.8.
'B' is wrong; it's taken from ODCM 6.11.1.

Technical References:

TRM 6.4.1; TS 3.4.8; ODCM 6.11.1

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-B3300, OBJ. 44

Question Source:	Bank # LOT 308 AUDIT	X
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41	
	55.43(b)(2)	

Examination Outline Cross-Reference	Level	SRO
295009 Low Reactor Water Level **Ability to determine and/or interpret the following as they apply to LOW REACTOR WATER LEVEL: AA2.01 Reactor water level**	**Tier #**	1
	Group #	2
	K/A #	295009 AA2.01
	Rating	4.2

Question 80

Use your provided references to answer this question.

An ATWS is in progress with the following:

- Reactor power is < 4%
- SLC is injecting
- Reactor pressure is 800 psig
- Suppression Pool level is 16.5 feet
- Suppression Pool temperature is 152 °F
- Reactor water level is -210” and steady
- All available Table 4 systems are injecting

What is the **NEXT** required action?

A. Inject to the RPV via the RHR SDC return line.

B. Enter the Emergency Depressurization leg.

C. Lower the reactor pressure band.

D. Exit the EPs and enter the SAPs.

Answer: B

Explanation:

See EP-2A and EP-3.

‘A’ is wrong. Although injection via the SDC return line is a Table 4 system, stem conditions indicate that reactor pressure is still too high (800 psig) to go in through SDC (low-pressure system).

‘C’ is wrong. This choice considers HCTL Figure 1, where with a 16.5’ supp pool level and reactor pressure at 800 psig, HCTL is still SAFE at 152°F supp pool temperature. However, the choice suggests a lowering of the reactor pressure band in order to avoid having to Emergency Depressurize should supp pool temp reach ~155°F.

‘D’ is wrong. Given these stem conditions, while in EP-2A we can’t get to the SAPs other than by way of step L-6…i.e., after we’ve Emergency Depressurized.

‘B’ is correct. With level “steady” at -210”, we’re clearly unable to restore it to >-191” using the available Table 4 systems that are currently injecting. Therefore, per step L-6 we must perform an ED.

Technical References:

EP-2A, HCTL Figure 1

References to be provided to applicants during exam:

EP flowcharts

Learning Objective: GLP-OPS-EP02A, OBJ. 3

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

Examination Outline Cross-Reference	Level	SRO
295037 SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown 2.2.36 Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operations.	Tier #	1
	Group #	1
	K/A #	295037 2.2.36
	Rating	4.2

Question 81

Use your provided references to answer the following question.

A plant power ascension is in progress.

Reactor power is 40%.

Engineering has determined the control power fuses installed in the LFMG drive motor breakers (CB-1A and CB-1B) for both Recirc Pumps are defective and must be replaced.

What is the Completion Time for the applicable Tech Spec Required Action, if any?

A. 14 days

B. 72 hours

C. 1 hour

D. No applicable action

Answer: A

Explanation:

The Bases (see page B 3.3-79) describes that a given Channel's OPERABILITY relies on its associated Recirc Pump drive motor breakers, as well. The easiest reference to seeing the association for the CB-1's is found in the lesson plan (GLP-OPS-B3300, page 54 of 99)…where we find that ATWS Channel 2 includes CB-1 and CB-4 for each pump.

As such, given that this question is proposing that we declare the tripping capability for both CB-1A and CB-1B inoperable (starting from the time that Engineering discovered the fuses to be defective), we are in fact **declaring ONLY one ATWS-RPT Channel (Channel 2) inoperable for each of the two Recirc Pumps.** With Channel 1 still OPERABLE, the ATWS-RPT Function (i.e., to trip the pumps on Level 2 and/or high reactor pressure) remains in place.

Therefore, Action 3.3.4.2.A.1 applies and the Completion Time is 14 days.

For the above reasons, choice ‘A’ is correct.

‘B’ and ‘C’ are wrong. These choices suggest the Completion Times associated with 3.3.4.2.B and C, respectively, which have only to do with inoperable ATWS Functions (not Channels), as defined by the LCO Statement itself…Reactor High Pressure and Reactor Lo-Lo Water Level.

‘D’ is wrong for the reason ‘A’ is correct. It is very plausible to the SRO Candidate who mistakenly concludes the following: Since the reactor is at 40% power, Recirc pumps are in Fast speed and so the CB-1’s are already open anyway. Therefore, the ATWS-RPT trip action has already occurred and LCO need not be entered.

Technical References:

TS LCO 3.3.4.2 and its Bases
GLP-OPS-B3300

References to be provided to applicants during exam:

TS LCO 3.3.4.2, entire ***(WITH Bases)***

Learning Objective: GLP-OPS-B33 Obj. 14, 28

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

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

Question 82

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 has been performed per the Work Order and has verified that the leak is stopped.

What post-maintenance testing, if any, is required to be performed by Operations before the Work Order may be closed?

A. No post–maintenance testing is required to be performed by Operations.

B. Operations must perform a local leak rate test (LLRT) to verify Suppression Pool leakage is within allowable limits.

C. Operations must perform a functional stroke of the valve using the P11 SOI.

D. Operations must perform a timed stroke of the valve using the P11 valve operability surveillance test

Answer: D

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. Therefore, ‘D’ is correct.

‘A’ is incorrect. Typically, Operations does not perform post–maintenance testing (PMT) on BOP systems. However, this is a safety–related component required to be OPERABLE by Tech Specs. Maintenance tests are therefore inadequate to demonstrate operability.

‘B’ is incorrect. 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.

‘C’ is incorrect; 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.

Technical References:

TS 3.6.1.3
01–S–07–2 steps 2.6.1b, 5.12, 6.2.4b

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-PROC, OBJ. 27.2

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

Examination Outline Cross-Reference	Level	SRO
295016 Control Room Abandonment **2.4.41 Knowledge of the emergency action level thresholds and classifications.**	**Tier #**	1
	Group #	1
	K/A #	295016 2.4.41
	Rating	4.6

Question 83

Use your provided references to answer this question.

The plant is operating at rated power when a fire starts in the control room photo-copier.

The fire spreads to consumables throughout that same area causing operators to evacuate the control room due to smoke.

13 minutes later, operators establish control of the plant from the Remote Shutdown Panels.

20 minutes after it started, the fire is extinguished by the Fire Brigade.

Fire Brigade Leader reports there is no visible damage to any permanent structures or plant equipment.

Based on this sequence of events, the SRO should make an emergency declaration based on the following EAL…

A. HU4

B. HA3

C. HA4

D. HS3

Answer: B

Explanation:

See EAL flowcharts.

‘A’ is wrong. With the fire taking >15 minutes to extinguish, HU4 would apply were it not for the higher level EAL (HA3) taking priority.

‘B’ is correct. Simply evacuating the control room and establishing Remote Shutdown Panel control per the ONEP is enough to require this EAL.

‘C’ is wrong. This EAL does not apply because there is no visible damage to permanent plant structures or plant equipment.

‘D’ is wrong. This EAL does not apply because Remote Shutdown Panel control of the plant was established within 15 minutes of control room evacuation.

Technical References:

EAL flowcharts

References to be provided to applicants during exam:

EAL flowcharts

Learning Objective: GLP-EP-EPTS6, OBJ. 1

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

Examination Outline Cross-Reference	Level	SRO
295038 High Off-Site Release Rate **2.2.40 Ability to apply Technical Specifications for a system.**	**Tier #**	1
	Group #	1
	K/A #	295038 2.2.40
	Rating	4.7

Question 84

Use your provided references to answer this question.

Power is lost to Containment Ventilation Radiation Monitor Noble Gas Pallet D17-P019.

No other subsystems are affected.

What is the Tech Spec / TRM / ODCM Required Action?

A. Restore the AXM to OPERABLE within 7 days.

B. Take grab samples once per 8 hours and analyze them within 24 hours.

C. Estimate effluent flow rate once per 8 hours.

D. Immediately establish alternate sampling and estimate alternate sample flow once per 8 hours.

Answer: A

Explanation:

The Candidate first reviews Tech Spec Loop Logics (17-S-06-5) and locates subject pallet D17-P019 on Attachment I, page 15 of 22. That line item refers the Candidate to both Attachment II, page 73 and to TRM Table 6.3.11-1, Function (1).

The greater help is the TRM reference itself. TRM Table 6.3.11-1, Function (1) references ACTION B for this channel. Action B directs us to restore the channel to OPERABLE within 7 days. Thus 'A' is correct. The use of "AXM" in the answer choice is taken from the Loop Logic Attachment II, page 73 label for the D17-P019 being one of 6 "AXM Noble Gas Pallets". This label is familiar language to GGNS operators and students/Candidates.

'B', 'C', and 'D' are wrong. These are all taken from the Required Actions of ODCM LCO 6.3.10 and its Table 6.3.10-1, Instrument (2). This Table does not include the AXM pallet addressed in the stem of this question. Rather, it addresses the Iodine/Particulate Sampler (D17-P002), the Effluent System Flow Rate channel (D17-P011A), and the Alarm channel (D17-RITS-K603). It is from the Actions related to these that distracters

‘B’, ‘C’ and ‘D’ are suggested. These are very plausible to the Candidate who fails to first review and make good use of the Loop Logic reference before proceeding to the TRM/ODCM.

Technical References:

TRM 6.3.11; ODCM LCO 6.3.10; 17-S-06-5 (TS Loop Logics)

References to be provided to applicants during exam:

TRM 6.3.11, all
ODCM LCO 6.3.10, all
17-S-06-5 (TS Loop Logics): cover page; Att.I, pages 15-16; Att.III, pages 72-74

Learning Objective: GLP-OPS-D1721, OBJ. 27

Question Source:	Bank # LORQT-06337	X
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41	
	55.43(b)(2)	

Examination Outline Cross-Reference	Level	SRO
2.2.40 Ability to apply Technical Specifications for a system.	**Tier #**	3
	Group #	
	K/A #	2.2.40
	Rating	4.7

Question 85

The plant is operating at rated power.

HPCS Jockey Pump has been tagged out for 2 days for pump and motor replacement.

Alarms and reports are received indicating the LPCS Jockey Pump has tripped and has extensive motor damage.

What is the Tech Spec Required Action?

A. Restore LPCS to an OPERABLE status within 7 days.

B. Restore LPCS or HPCS to an OPERABLE status within 72 hours.

C. Verify RCIC is OPERABLE within one hour and restore HPCS to OPERABLE status within 14 days.

D. Within 1 hour initiate action to place the unit in Mode 2 within 7 hours, Mode 3 within 13 hours and Mode 4 within 37 hours.

Answer: D

Explanation:

The loss of both ECCS Spray subsystems is a special case in Tech Specs versus the loss of a low pressure injection subsystem (LPCI A, B, C) and HPCS.

Per Tech Spec 3.5.1, HPCS and LPCS systems Inoperable (Jockey pump loss makes the system INOP) 3.5.1 Condition H requires entry into LCO 3.0.3 immediately. LCO 3.0.3 requires within 1 hour to initiate action to place the unit in mode 2 within 7 hours, mode 3 within 13 hours and mode 4 within 37 hours. Therefore Answer D is correct.

Answer A is wrong; it’s only one Low Pressure Injection subsystem or Low Pressure Core Spray System INOP, Condition A.

Answer C is wrong; it’s HPCS only INOP, Condition B.

Answer B is wrong; it’s two ECCS injection subsystems INOP, or 1 injection and one spray system INOP, Condition C.

Technical References:

TS LCOs 3.5.1, 3.0.3

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-TS001 Objectives 8; 16

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

Examination Outline Cross-Reference	Level	SRO
2.4.30 Knowledge of events related to system operation/status that must be reported to internal organizations or external agencies, such as the State, the NRC, or the transmission system operator.	**Tier #**	3
	Group #	
	K/A #	2.4.30
	Rating	4.1

Question 86

Per 01-S-06-5, Reportable Events or Conditions, which of the following is an NRC 4-Hour Notification Requirement?

A. Tech Spec deviation authorized per 10CFR50.54(x)

B. Entry of an unauthorized person into the Protected Area

C. Loss of Special Nuclear Material (SNM)

D. Actuation of RPS (full-scram signal) while critical

Answer: D

Explanation:

‘A’, ‘B’ and ‘C’ are wrong. See 01-S-06-5, Attachment III, page 1; these are all 1-Hour Reports.

‘D’ is correct. Per Attachment III, page 2, Item II.3, this is a 4-Hour Report.

Technical References:

01-S-06-5, Reportable Events or Conditions

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-PROC, Objectives 11.8, 11.9

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

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

Examination Outline Cross-Reference	Level	SRO
295004 Partial or Complete Loss of D.C. Power **Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER: AA2.02 Extent of partial or complete loss of D.C. power**	**Tier #**	1
	Group #	1
	K/A #	295004 AA2.02
	Rating	3.9

Question 87

Use your provided references to answer this question.

The plant is in MODE 5 with spent fuel currently being off-loaded from the core.

RHR ‘A’ is OPERABLE for Shutdown Cooling and is operating in that mode.

Div 1 and Div 2 DGs are OPERABLE.

The Bus 11DA feeder breaker to Distribution Panel 1DA2 trips (internal breaker fault) and cannot be re-closed.

Which of the following identifies the two Tech Spec LCOs that are the **MINIMUM** LCOs requiring entry for actual implementation of a Required Action?

A. 3.8.2 and 3.8.5

B. 3.8.2 and 3.8.8

C. 3.8.5 and 3.9.8

D. 3.8.8 and 3.9.8

Answer: D

Explanation:

‘A’ is wrong. LCO 3.8.5 (DC Sources – Shutdown) is not the applicable LCO for a DC Distribution Panel loss. The scope of 3.8.5 is limited to the battery and the battery bus (see the Bases discussion for LCO 3.8.5. Rather, LCO 3.8.8 applies for this Distribution Panel loss; see the Bases discussion for LCO 3.8.8 (and 3.8.7) on pages B 3.8-81 and 3.8-79).

See the LCO 3.8.8 Actions… If the SRO chooses to skip Action 3.8.8.A.1 in favor of 3.8.8.A.2, **there is no reason to enter 3.8.2 for declaring the Div 1 DG inoperable**. **Thus, 3.8.2 is not absolutely essential for entry.** See the Bases discussion that justifies this approach on page B 3.8-82.

‘B’ is wrong. 3.8.5 doesn’t apply for the reason already discussed. LCO 3.8.8 does apply for the reason already discussed.

'C' is wrong. LCO 3.8.5 doesn't apply for the reason discussed in choice 'A'.

'D' is correct. 3.8.8 applies as discussed for choice 'A'. LCO 3.9.8 applies as follows…the LCO Action from 3.8.8.A.2.5 has us declare RHR 'A' SDC inoperable immediately, which means we then enter its applicable LCO, which is LCO 3.9.8, RHR – High Water Level, because we're flooded up off-loading fuel from the core.

Technical References:

LCOs 3.8.2; 3.8.5; 3.8.8; 3.9.8; L11 SOI

References to be provided to applicants during exam:

LCO 3.8.2, all (no Bases)
LCO 3.8.5, all (no Bases)
LCO 3.8.8, all (no Bases)
LCO 3.9.8, all (no Bases)
L11 SOI: cover page; Att.IA, all

Learning Objective: GLP-OPS-L1100, OBJ. 18

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

Examination Outline Cross-Reference	Level	SRO
262001 A.C. Electrical Distribution **Knowledge of the emergency action level thresholds and classifications.**	**Tier #**	2
	Group #	1
	K/A #	262001 2.4.41
	Rating	4.6

Question 88

Use your provided references to answer this question.

The plant is in MODE 3 with the Div 1 and Div 2 DGs inoperable and unavailable.

Div 3 DG is OPERABLE.

At 0130, a severe storm results in the loss of both the Baxter-Wilson line and the Port Gibson line.

At 0200, capability to get power from the Port Gibson line to the Div 1 and Div 2 ESF buses is restored.

What is the required emergency classification?

A. Unusual Event

B. Alert

C. Site Area Emergency

D. General Emergency

Answer: B

Explanation:

See EAL flowcharts, Loss of AC Power section.

When the severe storm takes 2 of the 3 offsite power sources away (at time 0130), it leaves the 500 KV Franklin as the only remaining power source to the Div 1, 2 ESF buses…currently, it's powering bus 15AA via ST-11. Although the Div 3 DG is OPERABLE, it's irrelevant to the Station Blackout (SBO) definition. As such, at time 0130, the 15-minute clock of EAL "SA1" started. Reason?...an additional single failure (in this case, loss of the Franklin line) would result in an SBO. Although we recovered a second power source (Port Gibson line) at 0200, we still went beyond the 15-minute "threshold" clock of EAL SA1 and are therefore required to make that declaration.

For the reasons above, only choice ‘B’ is correct.

Technical References:

EAL flowcharts.

References to be provided to applicants during exam:

EAL flowcharts

Learning Objective: GLP-EP-EPTS6, OBJ. 1

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

Examination Outline Cross-Reference	Level	SRO
300000 Instrument Air System **2.4.6 Knowledge of EOP mitigation strategies.**	**Tier #**	2
	Group #	1
	K/A #	300000 2.4.6
	Rating	4.7

Question 89

The Instrument Air header in the Water Treatment Building ruptures.

A manual scram results in an ATWS.

Reactor power is 20%.

SLC fails to initiate.

Condensate pumps trip due to the minimum flow valve failing closed.

Which of the following describes a strategy that will be required by EP–2A under these conditions?

A. Lower reactor pressure to 450–600 psig with the main turbine bypass valves manual jack.

B. Lower reactor pressure to 450–600 psig with MSIV upstream drains B21–F016 and B21–F019.

C. Perform an Emergency Depressurization using ADS SRVs.

D. Perform an Emergency Depressurization using RWCU in recirc mode.

Answer: C

Explanation:

An ATWS is in progress with a total loss of instrument air. This will result in eventual closure of the MSIVs and loss of feedwater. The condensate system has been lost. With a 20% power ATWS, available injection systems (RCIC, CRD) will not be able to maintain reactor level and an Emergency Depressurization will be required by EP–2A. ‘C’ is therefore correct.

‘A’ is wrong; the loss of air has closed the MSIVs and MSL drain Aux Bldg isolation valves. The main turbine bypass valves will be unavailable for reactor pressure control. Additionally, the reduced pressure band is only appropriate in an ATWS to make available low–pressure injection systems. In this case condensate has been lost, so a pressure reduction is inappropriate.

‘B’ is wrong; although the MSIV upstream drains are MOVs, the MSL drain Aux Bldg isolation valves will fail closed on loss of air. The pressure reduction is also inappropriate for the same reason as in choice ‘A’.

‘D’ is wrong; no cooling is available to the RWCU non–regenerative heat exchangers due to the loss of air (PSW isolated, P42–F103 CCW supply to RWCU NRHX fails closed). Otherwise, the plausibility of this distracter is based on the Candidate incorrectly predicting the inability to open ADS SRVs due to the complete loss of air loss. In that case, performing an ED with RWCU in recirc mode is one of the Table 3 Alternate Depressurization Systems allowed by EP-2A for the ATWS.

NOTE – Although the SRO Candidates have the EP-2A flowchart for their exam, we consider this question to be a closed-reference one. Reason…this question is more of a deductive reasoning type than it is a “drive through the EP flowcharts” type. The body of knowledge used to derive the correct answer first depends heavily on a sound understanding of systems/integrated plant before it then proceeds to the SRO-only domain (at GGNS) of how to apply that understanding to the EP-2A strategy.

Technical References:

05–S–01–EP–2 (EP2A)
M1063B
M1077A

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-EP2A Obj. 7, 21

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

Examination Outline Cross-Reference	Level	SRO
205000 Shutdown Cooling System (RHR Shutdown Cooling Mode) **Knowledge of limiting conditions for operations and safety limits.**	**Tier #**	2
	Group #	1
	K/A #	205000 2.2.22
	Rating	4.7

Question 90

The plant is in MODE 5 with the reactor cavity drained down.

What decay heat removal systems/subsystems must be OPERABLE?

A. Any single decay heat removal system/subsystem.

B. Both RHR shutdown cooling loops; no substitutions are permitted.

C. Both RHR shutdown cooling loops; ADHR may be substituted for one of them.

D. Any two decay heat removal systems/subsystems; RWCU can be one of them.

Answer: C

Explanation:

See Tech Spec LCO 3.9.9 and its Bases (page B 3.9-30, LCO Discussion). Even though the LCO Statement suggests answer choice 'D' might be correct, the SRO Candidate's knowledge of the Bases for this LCO will prove otherwise…where we find that in fact both RHR SDC loops are required to OPERABLE, yet ADHR may be substituted for one of them. For this reason, answer choice 'C' is correct and 'D' is wrong.

'A' is wrong. There is no LCO where "**any single decay heat removal system/subsystem" (which would imply ADHR as well)** is acceptable.

'B' is wrong. This would be correct for the Modes 3 and 4 LCOs 3.4.9 and 3.4.10.

Technical References:

Tech Spec LCOs 3.4.9, 3.4.10, 3.9.9 and its Bases

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-E1200, OBJ. 17

Question Source: Bank # GGNS-LORQT-06516a X

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

Examination Outline Cross-Reference	Level	SRO
Knowledge of the fuel-handling responsibilities of SROs.	**Tier #**	3
	Group #	
	K/A #	2.1.35
	Rating	3.9

Question 91

Per 01-S-06-50, Control of Fuel Services Operations, the Refuel SRO is a required Fuel Handling Team position…

A. only when irradiated fuel is being moved, either on the Refueling Floor or in the Fuel Handling Area.

B. when any fuel, new or irradiated, is being moved, either on the Refueling Floor or in the Fuel Handling Area.

C. only when Core Alterations are being performed directly on the Refueling Floor.

D. when any Core Alterations are being performed, either directly on the Refueling Floor or from the control room panels.

Answer: D

Explanation:

See 01-S-06-50, sections 2.8 and 6.6.1. Although not written explicitly, the Refuel SRO is in fact only a requirement for CORE ALTERATIONS, which by definition eliminates activities in the Fuel Handling Area…thus making answer choices 'A' and 'B' wrong.

Section 6.6.1 describes why 'D' is correct and 'C' is wrong.

Technical References:

01-S-06-50, Control of Fuel Services Operations

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-PROC, OBJ. 9.0

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

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

Examination Outline Cross-Reference	Level	SRO
Knowledge of radiation exposure limits under normal or emergency conditions.	**Tier #**	3
	Group #	
	K/A #	2.3.4
	Rating	3.7

Question 92

A Site Area Emergency is in progress.

A control room RO needs to be sent into the plant to perform a task.

RP has estimated the total whole-body dose the RO will receive.

Per 10-S-01-17, Emergency Personnel Exposure Control, a control room SRO is responsible to ensure the RO has sufficient exposure margin to complete the task.

The SRO reviews the Daily Dose Margin Report and determines the RO's current dose.

Per 10-S-01-17, against what dose limits does the SRO compare the current dose to determine if there is sufficient exposure margin?

A. GGNS Administrative Dose Limits

B. 10CFR20 Federal Dose Limits

C. 10CFR100 Federal Dose Limits

D. Authorized Emergency Exposure Dose Limits

Answer: B

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. Therefore, choice 'A' is wrong.

'C' is wrong because 10CFR100 specifies dose limits for offsite releases.

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. Section 6.1 deals with the limits suggested by answer choice 'D'. Clearly then, the SRO compares the RO's current dose against the 10CFR20 Federal limits. Thus, answer choice 'B' is correct.

NOTE – This is an SRO-only question for the following reasons: 1) clearly, it relates an SRO-only responsibility in the said procedure (i.e., to determine sufficient exposure margin); 2) although the actual question being asked is a fairly elementary one (i.e., "What dose limit is to be used?"), this item of knowledge is not at all common knowledge among the non-Radiation Protection Personnel population, most especially among the RO population. Nonetheless, because this "elementary" information is associated with an SRO's responsibility during a plant emergency situation, it is reasonable to expect an SRO Candidate to simply know that we suspend the usual GGNS Admin Dose Limits during a Site Area Emergency. Answering this question correctly is not about recalling a fact; rather, it is a test of how well our SRO Candidates can use deductive reasoning…doesn't it make sense that we would automatically suspend the overly-conservative dose limits that are in place during normal plant operations to give us the Federal exposure limit freedom we need, without the need for authorizing admin extensions?

Technical References:

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

References to be provided to applicants during exam: None

Learning Objective: GLP-EP-EPTS6, OBJ. 17

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b)(12)	
	55.43(b)(4)	

Examination Outline Cross-Reference	Level	SRO
Knowledge of emergency plan protective action recommendations.	**Tier #**	3
	Group #	
	K/A #	2.4.44
	Rating	4.4

Question 93

Severe storms have degraded plant conditions to where a General Emergency has been declared.

The ERO is preparing the first Protective Action Recommendation (PAR) to be issued; for the current dose projections, it will be a Standard PAR.

Offsite survey teams report that all roads that provide evacuation passage out of one of the downwind sectors are flooded out and impassable by motor vehicle.

Per 10-S-01-12, Radiological Assessment and PARs, how should the Emergency Director respond to this problem, with respect to issuing the PAR?

A. Default to preparing and issuing an Ad-hoc PAR instead of the Standard PAR.

B. Notify state/local agencies of the problem and inform them that a PAR will not be issued.

C. Issue the Standard PAR, with the usual evacuation of all downwind sectors.

D. Issue the Standard PAR with one exception: recommend that the flooded out sector be sheltered.

Answer: D

Explanation:

See 10-S-01-12, section 6.1.7.e(2)(a)(2). This procedure section directs the ED to consider recommending that the affected sector be sheltered rather than evacuated. Other than this one exception, the Standard PAR can be issued as planned. For this reason, 'D' is correct and 'C' is wrong.

'A' is wrong. See section 6.2.2. Ad-hoc PARs apply outside the 10-mile EPZ.

'B' is wrong. This sector evacuation problem is not a warrant to avoid issuance of the PAR.

Technical References:

10-S-01-12, Radiological Assessment and PARs

References to be provided to applicants during exam: None

Learning Objective: GLP-EP-EPTS6, OBJ. 2

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

Examination Outline Cross-Reference	Level	SRO
295002 Loss of Main Condenser Vacuum **2.4.6 Knowledge of EOP mitigation strategies.**	**Tier #**	1
	Group #	2
	K/A #	295002 2.4.6
	Rating	4.7

Question 94

Use your provided references to answer this question.

From rated power operators have inserted a manual scram in response to a degrading main condenser vacuum.

Operators enter EP-2A for an ATWS while condenser vacuum continues to degrade.

As EP-2A is entered:

- 4 control rods are stuck at position 12
- RC&IS has failed (not functioning)
- Reactor power is < 4%
- Reactor water level is +5” and slowly lowering
- RCIC is not available
- Condenser vacuum is 21” Hg Vac and lowering
- MSIVs are open

To what band should the CRS direct the restoration of reactor water level <u>and</u> using what method(s)?

A. 11.4” to 53.5”
Use Feedwater; when RFPT trips on low vacuum, direct RO to reset RFPT Vacuum Trip. May also maximize CRD for flow.

B. 11.4 to 53.5”
<u>Only</u> use CRD maximized for flow.

C. -30” to +30”
Use Feedwater; when RFPT trips on low vacuum, direct RO to reset RFPT Vacuum Trip. May also maximize CRD for flow.

D. -30” to +30”
<u>Only</u> use CRD maximized for flow.

Answer: A

Explanation:

See EP-2A, ATWS RPV Control. With reactor power < 4% upon EP-2A entry, the CRS proceeds to step L-6 and restores level to the “nominal” level band of +11.4” to 53.5” per the guidance of Operations Philosophy, section 6.6.8.d (ATWS <4%). Step L-6 allows use of available Table 4 injection systems for level restoration; in this case, Feedwater and CRD (maximized for flow). See ARI P680-2A-A2 (RFPT A TRIP), page 2 of 2. Step 4.3 allows us to bypass a low vacuum trip of the RFPT so long as we’re above the 9” Hg Vac MSIV isolation setpoint. Implementation of the ARI Subsequent Action requires CRS (SRO) direction.

For the above reasons, choice ‘A’ is correct and ‘B’, ‘C’, ‘D’ are wrong.

Technical References:

EP-2A, ATWS RPV Control
02-S-01-27, Operations Philosophy
ARI P680-2A-A2

References to be provided to applicants during exam:

EP flowcharts

Learning Objective: GLP-OPS-EP02A, OBJ. 3

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

Examination Outline Cross-Reference	Level	SRO
223001 Primary Containment System and Auxiliaries **2.2.40 Ability to apply Technical Specifications for a system.**	**Tier #**	2
	Group #	2
	K/A #	2.2.40
	Rating	4.7

Question 95

Use the figure below (<u>not</u> a Main Steam Line) to answer this question.

Consider the Required Actions of Tech Spec LCO 3.6.1.3, Primary Containment Isolation Valves (PCIVs).

The LCO specifies Required Actions for declaring one PCIV in the penetration inoperable and Required Actions for declaring both PCIVs in the penetration flowpath inoperable.

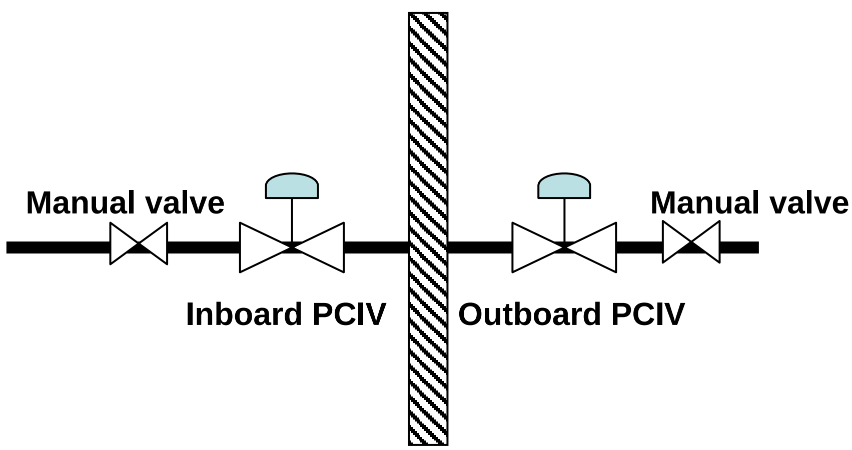
Most significantly, how do these Required Actions differ?

A. In the Completion Times for closing a valve to isolate the penetration flowpath.

B. In the number of closed valves necessary to isolate the penetration flowpath.

C. In whether or not a PCIV must be de-activated in order to isolate the penetration flowpath.

D. In whether or not immediate entry into LCO 3.0.3 is required.



Answer: A

Explanation:

See LCO 3.6.1.3, Actions A.1 and B.1. A.1 addresses a single PCIV inoperable and allows 4 hours to isolate the penetration flowpath with at least one closed and de-activated automatic valve, closed manual valve, blind flange, or check valve. B.1 addresses two PCIVs inoperable but allows only 1 hour to do the same (except by using a check valve). For this reason, 'A' is correct and 'B' is wrong.

'C' is wrong. For either of the two Required Actions, the requirement to both close <u>and</u> de-activate the automatic valve (if one is to be used to satisfy the isolation criteria) is the same.

'D' is wrong. Strong plausibility to the weaker SRO Candidate but no such requirement given this question.

Technical References:

TS LCO 3.6.1.3, PCIVs

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-M4101, OBJ. 11

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

Examination Outline Cross-Reference	**Level**	SRO
241000 Reactor/Turbine Pressure Regulating System	**Tier #**	2
	Group #	2
Ability to apply Technical Specifications for a system.	**K/A #**	241000 2.2.40
	Rating	4.7

Question 96

Use your provided references to answer this question.

The plant is operating at 80% power.

The latest Turbine Stop and Control Valve Operability Test (06-OP-1N32-V-0001) has just failed.

Consequently, the following valves must be declared inoperable:

- HP Stop Valve N11-F026A
- HP Control Valve N11-F026C

Which of the following is a required action?

A. Immediately enter LCO 3.0.3.

B. Isolate the main turbine from its steam supply within 6 hours.

C. Develop and implement compensatory actions for the inoperable valves.

D. Verify that a safety function is not compromised by the inoperable valves; if a compromise is discovered then enter LCO 3.0.3.

Answer: C

Explanation:

O6-OP-1N31-V-0001 implements SR 6.3.8.1 of TRM LCO 6.3.8.1, “Turbine Overspeed Protection System.”

At GGNS, each of 4 Main Steam Lines (MSLs) feeds the HP turbine steam chest via a combined HP Control/HP Stop Valve assembly (otherwise called a “Main Stop Control Valve” (MSCV)). The surveillance cycles (closed/open) each of these assemblies using the Automatic Turbine Tester (ATT).

Stem conditions indicate that one Stop Valve in the ‘A’ MSL has been declared inoperable and one Control Valve in the ‘C’ MSL. As such, LCO 6.3.8.1, Condition A applies…the

Required Action is to “Immediately enter LCO 6.0.1.” LCO 6.0.1 directs us to develop and implement compensatory actions (for the inoperabilities). For this reason, choice ‘C’ is correct.

‘A’ is wrong. Per the Applicability statement for TRM LCO 6.3.8.1, LCO 3.0.3 does not apply.

‘B’ is wrong. This choice suggests TRM LCO 6.3.8.1 Condition B, Action B.1. Since we have only “one stop or one control valve per steam line inoperable”, Condition B doesn’t apply.

‘D’ is wrong. It’s taken from Item #2 of LCO 6.0.1. Although 6.0.1 does direct us to verify that a safety function is not compromised, there is no link from 6.0.1 to LCO 3.0.3.

Technical References:

TRM LCO 6.0.1
TRM LCO 6.3.8.1
Turbine Stop and Control Valve Operability Test, 06-OP-1N32-V-0001

References to be provided to applicants during exam:

TRM LCO 6.0.1
TRM LCO 6.3.8.1

Learning Objective: GLP-OPS-TS001 Objectives 8; 16

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

Examination Outline Cross-Reference	Level	SRO
209002 High Pressure Core Spray System **A2. Ability to (a) predict the impacts of the following on the HIGH PRESSURE CORE SPRAY SYSTEM (HPCS) ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: A2.08 Inadequate system flow**	**Tier #**	2
	Group #	1
	K/A #	209002 A2.08
	Rating	3.2

Question 97

Use your provided references to answer the following question.

A design–basis LOCA has occurred coincident with a loss of offsite power.

An Emergency Depressurization was performed.

All low–pressure ECCS systems failed.

Only HPCS is injecting, at a flow rate of 3500 gpm.

To mitigate this event, operators must…

A. remain in EP-2 Alternate Level Control until level can be recovered.

B. enter EP-2 Steam Cooling until injection can be established, then return to Alternate Level Control.

C. enter SAP-1 and execute the reactor vessel injection strategy for maintaining RPV integrity.

D. enter SAP-1 and execute the containment injection strategy for an expected RPV breach.

Answer: C

Explanation:

In a DBA LOP–LOCA, at least 3 ECCS systems in 2 divisions are required to successfully mitigate the event and satisfy the ECCS acceptance criteria. In this question, only HPCS is available, and it is operating with an inadequate flow rate (<7000 gpm). Adequate core cooling via core submergence will be lost and reactor level will fall below –191”. HPCS at the reduced flow rate cannot provide adequate core cooling via spray cooling due to inadequacy of the spray pattern over the core. With all adequate core cooling lost, the SAPs must be entered and EP-2 exited. This makes answers ‘A’ and ‘B’ wrong.

‘C’ is correct. HPCS at the reduced flow rate (3500 gpm) is capable of satisfying the

Minimum Debris Retention Injection Rate (MDRIR) shown on SAP-1. Therefore the appropriate strategy in SAP–1 is the RPV flooding strategy of Column 4.

‘D’ is wrong. No RPV breach is expected because the 3500 gpm HPCS flow rate satisfies MDRIR.

NOTE - This question does not directly evaluate the candidate’s ability to predict the effects of low system flow on the HPCS system (the (a) portion of the K/A). Incorporation of a means to evaluate the candidate’s predictive ability was not possible without an inordinate amount of effort. The question does satisfactorily evaluate the (b) part of the K/A at a higher cognitive level, which is allowed by NUREG–1021, ES–401, Section D.2.a.

Technical References:

05-S-01-EP-2
02-S-01-40

References to be provided to applicants during exam:

EP flowcharts
SAP flowcharts

Learning Objective: GLP-OPS-E2201 Obj. 3
GLP–OPS–EP02 Obj. 7
GLP–EP–EPT19 Obj. 2

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

Examination Outline Cross-Reference	Level	SRO
288000 Plant Ventilation Systems **Ability to (a) predict the impacts of the following on the PLANT VENTILATION SYSTEMS ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: A2.02 Low reactor water level**	**Tier #**	2
	Group #	2
	K/A #	288000 A2.02
	Rating	3.6

Question 98

Use your provided references to answer this question.

The plant is operating at 70% power when an inadvertent reactivity addition occurs.

Offgas Pre-Treatment Radiation Monitor and Main Steam Line (MSL) Radiation Monitor readings begin to rise.

Operators conservatively insert a manual scram.

A short-duration reactor pressure spike momentarily lifts an SRV on the scram.

Suppression pool temperature peaks at 90°F.

Reactor water level reaches -45” before recovering to the normal band.

The CRS enters EP-4 on MSL high radiation; he enters all other applicable EPs as well.

The CRS should direct operators to…

A. leave one SGTS train running and place the other in STANDBY with its Mode Select switch using the SOI.

B. shutdown both SGTS trains and place normal Secondary Containment Ventilation systems back in service.

C. place both loops of RHR in suppression pool cooling.

D. close the MSIVs and MSL drains.

Answer: B

Explanation:

‘C’ is wrong. This choice presumes an entry condition for EP-3 (Primary Containment Control). The suppression pool temperature entry is at 95°F; therefore, the CRS did not enter EP-3 and so he has no reason to implement the action of this answer choices (taken from the SP Temperature control leg of EP-3).

‘D’ is wrong. This distracter suggests the action of the bottom-most part of the top-most override step of EP-4. However, this action only applies when the MSL rad monitors reach 3 X NFPB. There is nothing in the stem to suggest that the MSL radiation monitors have risen to these levels.

‘A’ is wrong. If in fact SGTS were required to remain in service (i.e., it had auto-initiated on a high radiation condition in Secondary Containment), this answer choice would be correct. This choice is taken from two sources: 1) EP-4, step 1 (the top-most part of the override step has us “verify SGTS initiation”); and 2) the T48 (SGTS) SOI, section 5.2 (Operation after Automatic or Manual Initiation), where we intentionally leave one train in-service and place the other in a STANDBY status (capable of auto-starting only in the event of poor operating performance from the running train).

However, given the stem conditions for this question, there is no reason to suspect a Secondary Containment high radiation condition. In fact, the SRO Candidate, being told that the CRS has entered EP-4, should conclude that it could only have been on an entry for MSL high radiation. As such, the SRO should implement the second part of the EP-4, step 1 override…direct the operator to secure all of SGTS and restore normal ventilation to service. The Candidate is expected to recognize that the post-scram level drop to below -41.6” is the only reason that SGTS auto-initiated. For this reason, answer choice ‘B’ is correct.

Technical References:

EP-4, Auxiliary Building Control
04-1-01-T48-1, SGTS SOI

References to be provided to applicants during exam:

EP flowcharts

Learning Objective: GLP-OPS-EP4, OBJ. 3

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

Examination Outline Cross-Reference	**Level**	SRO
295031 Reactor Low Water Level **Knowledge of EOP implementation hierarchy and coordination with other support procedures or guidelines such as, operating procedures, abnormal operating procedures, and severe accident management guidelines.**	**Tier #**	1
	Group #	1
	K/A #	295031 2.4.16
	Rating	4.4

Question 99

A station blackout is in progress.

Reactor water level is -10” and slowly lowering.

The reactor is at rated pressure and stable.

Div 3 Diesel Generator failed to start.

The plant DC buses have experienced faults and excessive use such that DC voltages are less than 80 volts DC on all buses.

The **NEXT** place the SRO should direct his/her attention to is…

A. exiting the EPs and entering the SAPs.

B. performing Steam Cooling using EP-2.

C. conducting a black start of RCIC using the E51 SOI and restoring level to its normal band.

D. aligning Fire Water to inject through the ECCS pathways using the EP Attachment and restoring level to its normal band.

Answer: C

Explanation:

‘A’ is wrong; stem conditions do not indicate an EP exit/SAP entry point.

‘B’ is wrong; Steam Cooling is driven from the Alternate Level Control leg of EP-2 and would result only from a LOCA strategy. There is no indication of a LOCA in the stem conditions.

‘D’ is wrong; with the reactor still at rated pressure it’s impossible to inject with Fire Water, besides the fact that the EP Attachment 26 used is an Alternate Injection (Table 2) System of EP-2 that would be used only in the event that level could not otherwise be restored.

‘C’ is correct. Oddly enough, although adequate core cooling is far from being threatened at this point (given the stem conditions), the absence of AC and DC (excessively low DC bus volts) power places us in a position outside of design bases. As such, we’ll implement the hardcard for a RCIC black start found in the E51 (RCIC) SOI to start RCIC without the need for DC power. Once started, we can simply restore RPV level to its normal band.

Technical References:

EP-2, RPV Control
EP-1, Attachment 26
E51 SOI, RCIC

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-EP02, OBJ. 3

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

Examination Outline Cross-Reference	Level	SRO
262002 Uninterruptable Power Supply (A.C./D.C.) **Knowledge of maintenance work order requirements.**	**Tier #**	2
	Group #	1
	K/A #	262002 2.2.19
	Rating	3.4

Question 100

The plant is operating at rated power.

The inverter section of 1Y87 failed and the static switch remained selected to the inverter output (i.e., failed to auto-transfer).

As a result, panel 1Y89 is de-energized and a half-scram is present.

Per EN-WM-100, Work Request Generation, Screening and Classification, what is the maintenance classification for restoring inverter 1Y87 to its normal configuration?

A. Corrective Maintenance

B. Emergency Maintenance

C. Minor Maintenance

D. Tool Pouch Maintenance

Answer: A

Explanation:

See EN-WM-100, section 3.0[2] and [5] definitions of maintenance classifications. <u>Emergency Maintenance</u> is immediate and direct threat to the health and safety of the public, requires immediate attention to prevent deterioration of plant conditions to a possible unsafe or unstable level, and poses a significant industrial hazard that must be corrected immediately to prevent or mitigate actual serious injury or death. <u>Minor Maintenance</u> is work within the Skill of the Craft and can be done without detailed written instructions and meets a list of criteria in the procedure. <u>Tool Pouch Maintenance</u> is work that requires no documentation in plant records and considered to be within the skill of craft. **<u>Corrective Maintenance</u>** is work on Systems, Structures or Components (SSC) that has failed or significantly degraded such that it can no longer perform its design function.

For the above reasons, ‘A’ is correct because the static inverter is no longer performing its function resulting in a loss of power to ESF equipment. The loss of power does not pose a threat to safe shutdown capability or a threat to the health and safety of the plant or public. Repair of the static inverter and associated Static Switch is beyond skill of craft without documentation and does require documentation since 1Y87 is an ESF Inverter.

The Shift Manager determines whether maintenance is Emergency Maintenance.

Therefore answers B, C, and D are wrong.

Technical References:

EN-WM-100, Work Request Generation, Screening and Classification
E-1026
EN-MA-125, Trouble Shooting Control of Maintenance Activities

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-PROC, Objective 24.1

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

Technical References:

EN-WM-100, Work Request Generation, Screening and Classification
E-1026
EN-MA-125, Trouble Shooting Control of Maintenance Activities

References to be provided to applicants during exam: None

Learning Objective: GLP-OPS-PROC, Objective 24.1

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

GRAND GULF NUCLEAR STATION
LOT-610 NRC WRITTEN EXAM - RO

EXAM KEY

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

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EXAM KEY

Question	Answer
76	B
77	A
78	C
79	D
80	B
81	A
82	D
83	B
84	A
85	D
86	D
87	D
88	B
89	C
90	C
91	D
92	B
93	D
94	A
95	A
96	C
97	C
98	B
99	C
100	A

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OPEN REFERENCES
TABLE OF CONTENTS

References provided outside of this book:

- EP flowcharts
- SAP flowcharts
- EAL flowcharts