Examination Outline Cross-Reference	Level	RO
Ability to operate and/or monitor the Neutron monitoring system as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION (CFR: 41.7 / 45.6).	Tier #	1
	Group #	1
	K/A #	295001 AA1.06
	Rating	3.3

Question 1

The plant conditions are as follows:

- Reactor power is 70%
- The Flow Control Valves for Recirculation Loops A and B are open to their 70% position
- The Flow Control Valve for Recirculation Loop B fails closed

Verification of the current thermal scram set points can be obtained at:

A. The Jet Pump P619 panel

B. The APRM recorders on the P680 panel

C. The FCTR flowcards on the P669-P672 panels

D. The PBDS displays on the P669 and P670 panels

Answer: B

Explanation:

A. This is incorrect because the jet pump differential pressure transmitters do not serve the function of input to the flow-biased scram setpoints.

B. This is correct because the APRM recorders provide display output of reactor power and thermal scram set points.

C. This is incorrect the FCTR flow cards provide input of the status of the cards, and whether a thermal scram set point has been reached, but there is no visible indication of what the scram set points are at the time.

D. This is incorrect because PBDS measures LPRM input only, and does not interface with recirculation flow input/output at all.

Technical References:

GLP-OPS-B3300, GLP-OPS-C5104, GLP-OPS-C5106, GFIG-OPS-B3300

References to be provided to applicants during exam:

None

Learning Objective: To be determined

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:	55.41.7, 55.45.6	

Examination Outline Cross-Reference	Level	RO
Knowledge of the operational implications of Load shedding as it applies to PARTIAL OR COMPLETE LOSS OF A.C. POWER (CFR: 41.8 to 41.10).	Tier #	1
	Group #	1
	K/A #	295003 AK1.02
	Rating	3.1

Question 2

Following a Loss of Offsite Power, the Load Shedding and Sequencing System will:

A. Isolate PSW to Control Room air conditioning and ESF switchgear room coolers.

B. Isolate SSW to Control Room air conditioning and ESF switchgear room coolers.

C. Isolate PCW to the RWCU NRHX.

D. Align PSW to P52/P53.

Answer: A

Explanation:

A. This is correct because loss of offsite power does cause this load shedding action per GFIG-OPS-R2100.

B. This is incorrect because SSW is actually aligned to provide service to these coolers in the event of this situation.

C. This is incorrect because CCW actually provides the cooling to the RWCU NRHX.

D. This is incorrect because SSW is the cooling supply to these panels that is manually aligned in this situation.

Technical References:

GLP-OPS-R2100, GFIG-OPS-R2100, FSAR Table 8.3-9

References to be provided to applicants during exam:

None

Learning Objective: To be determined

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:	55.41.8 to 55.41.10	

Examination Outline Cross-Reference	Level	RO
Ability to determine and/or interpret the extent of partial or complete loss of D.C. power as it applies to PARTIAL OR COMPLETE LOSS OF D.C. POWER (CFR: 41.10 / 43.5 / 45.13).	Tier #	1
	Group #	1
	K/A #	295004 AA2.02
	Rating	3.5

Question 3

The plant is in MODE 1 in a normal electrical configuration with the exception of Division II EDG being in a lockout condition.

Following a loss of ESF Transformer 21, which of the following battery chargers would lose power?

A. 11DL

B. 11DC

C. 11DD

D. 11DH

Answer: A

Explanation:

A. This is correct because the bus will be deenergized as part of LSSS for Division II (BUV), and power will not be restored with the diesel generator.

B. This is incorrect because the Division III diesel generator starts and energizes the chargers on this bus within 2-3 seconds (73% BUV).

C. This is incorrect because this bus is deenergized by LSSS for Division I ESF loads, but not Division II.

D. This is incorrect because the 24 V DC bus is supplied by ESF Division I.

Technical References:

GLP-OPS-R2100, GLP-OPS-L1100, GFIG-OPS-R2100, GFIG-OPS-L1100

References to be provided to applicants during exam:

None

Learning Objective: To be determined

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:	55.41.10, 55.43.5, 55.45.13	

Examination Outline Cross-Reference	Level	RO
Knowledge of the operational implications of Pressure effects on reactor level as it applies to MAIN TURBINE GENERATOR TRIP (CFR: 41.8 to 41.10).	Tier #	1
	Group #	1
	K/A #	295005 AK1.03
	Rating	3.5

Question 4

The reactor is at rated thermal power at the beginning of the fuel cycle (BOC).

Following a main turbine trip, reactor water level would:

A. shrink until it reaches Level 2 and HPCS/RCIC initiation restores reactor level.

B. oscillate due to counter effects from pressure increase and recirculation pump speed decrease but stabilize close to the original level.

C. shrink and swell due to SRV cycling, until feedwater control stabilizes level below the original level.

D. oscillate downward at a rate that requires low pressure means (LPCI/LPCS) to restore level.

Answer: B

Explanation:

A. This is incorrect because a main turbine trip would not result in a level decrease to the extent of initiation of HPCS or RCIC systems. This is characteristic for Loss of Vacuum or MSIV Closure events.

B. This is correct because the level decrease due to the pressure increase (shrink) is countered by the reduction core flow/increase in downcomer level caused by the recirculation pump transfer to slow speed.

C. This is incorrect because this is indicative of IPC failure with a turbine control valve closure. In this scenario, the bypass control valves do not open to mitigate pressure increase, and recirculation pumps are tripped versus being placed in slow speed due to reaching the ATWS-ARI/RPT high reactor pressure setpoint.

D. This is incorrect because even in the worst case pressure increase transient event (MSIV closure), HPCS/RCIC are shown to be able to provide water to restore reactor vessel level.

Technical References:

GLP-OPS-MCD12, FSAR Section 15.2.3

References to be provided to applicants during exam:

None

Learning Objective: To be determined

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:	55.41.8 to 55.41.10	

Examination Outline Cross-Reference	Level	RO
SCRAM - Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation (CFR: 41.5 / 43.5 / 45.12 / 45.13).	Tier #	1
	Group #	1
	K/A #	295006 Generic - 2.1.7
	Rating	4.4

Question 5

Following a reactor scram, the operating crew observes:

- Reactor power is 0% and decreasing
- All control rod bottom lights are lit
- Reactor steam dome temperature is 450^0F
- RWCU bottom head suction line temperature is 370^0F
- Recirculation Loop A discharge temperature is 405^0F
- Recirculation Loop B discharge temperature is 395^0F

According to procedure 05-1-02-i-1, Reactor Scram, a reactor cooldown:

A. Should be initiated because the difference between the reactor steam dome and the reactor bottom head drain line temperatures exceeds 50^0F

B. Is not needed because the difference between the reactor bottom head drain line and Recirculation Loops A or B temperatures do not exceed 50^0F

C. Is not needed because the difference between the reactor steam dome and the reactor bottom head drain line temperatures do not exceed 100^0F

D. Should be initiated because the difference between the reactor steam dome and Recirculation Loop B temperatures exceeds 50^0F

Answer: D

Explanation:

A. This is incorrect because the temperature differential between the reactor steam dome and the reactor bottom head drain line needs to exceed 100 F to require a reactor cooldown following the scram.

B. This is incorrect because there is no specific temperature differential requirement between the reactor bottom head drain line and the recirculation loops associated with reactor cooldown initiation following a scram.

C. This is incorrect because although the temperature differential between the reactor steam dome and the reactor bottom head drain line does not exceed 100^0F (a reactor cooldown requirement following a scram), the situation in Answer D requires a reactor cooldown.

D. This is correct because the temperature differential exceeds one of the criteria for initiating a reactor cooldown following a scram per Procedure 05-1-02-I-1, Section 3.9.3.

Technical References:

Procedure 05-1-02-I-1, “Reactor Scram”

References to be provided to applicants during exam:

None

Learning Objective: To be determined

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:	55.41.5, 55.43.5, 55.45.12, 55.45.13	

Examination Outline Cross-Reference	Level	RO
Control Room Abandonment - Knowledge of limiting conditions for operations and safety limits (CFR: 41.5 / 43.2 / 45.2).	Tier #	1
	Group #	1
	K/A #	295016 General-2.2.22
	Rating	4.0

Question 6

The Remote Shutdown System is required to be operable in MODE(S):

A. 1, 2, 3, and 4

B. 1, 2, and 3 ONLY

C. 1 ONLY

D. 1 and 2 ONLY

Answer: D

Explanation:

A. This is incorrect because it does not match the Applicability statement in Technical Specification 3.3.3.2.

B. See Explanation for Answer A.

C. See Explanation for Answer A.

D. This is correct based on the Applicability statement in Technical Specification 3.3.3.2.

Technical References:

Technical Specifications Section 3.3.3.2, Technical Specifications Bases Section 3.3.3.2, GLP-OPS-C6100, ONEP 10-S-01-11

References to be provided to applicants during exam:

None

Learning Objective: To be determined

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:	55.41.5, 55.43.2, 55.45.2	

Examination Outline Cross-Reference	Level	RO
Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER and system loads (CFR: 41.7 / 45.8).	Tier #	1
	Group #	1
	K/A #	295018 AK2.01
	Rating	3.3

Question 7

A loss of Component Cooling Water (CCW) system would result in the loss of cooling to the :

A. CRD pump seals.

B. Offgas vault refrigeration units.

C. Heater drain pump coolers.

D. RWCU non-regenerative heat exchanger.

Answer: D

Explanation:

A. This is incorrect because the CCW system cools the CRD pump oil coolers, but not the pump seals.

B. This is incorrect because this is a cooling load for the TBCW system.

C. This is incorrect because this is a cooling load for the TBCW system.

D. This is correct because it is one of the cooling loads for the CCW system listed in both GLP-OPS-P4200 and ONEP 05-1-02-V-1.

Technical References:

GLP-OPS-P4200, GLP-OPS-P4300, ONEP 05-1-02-V-1

References to be provided to applicants during exam:

None

Learning Objective: To be determined

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:	55.41.7, 55.45.8	

Examination Outline Cross-Reference	Level	RO
PARTIAL OR TOTAL LOSS OF INST. AIR - Ability to manage the control room crew during plant transients (CFR: 41.10 / 43.5 / 45.12 / 45.13).	Tier #	1
	Group #	1
	K/A #	295019 Generic - 2.1.6
	Rating	3.8

Question 8

The reactor is at 90% rated thermal power and the operating crews observes the following plant indications:

- Reactor power is 95%
- Drywell pressure is 1.15 psig and slowly increasing
- Drywell temperature is 85^0F
- INST AIR RCVR PRESS LO alarm on Panel P870 has annunciated
- Control Rod 03-21 has drifted from position 42 to 40 over the last three minutes
- All other control rods are in their expected positions

According to procedure 05-1-02-iv-1, Control Rod/Drive Malfunctions, the operating crew should:

A. Scram the reactor

B. Fully insert control rod 03-21

C. Verify the scram air header pressure regulating valve is operating correctly

D. Restore rod 03-21 to position 42

Answer: B

Explanation:

A. This is incorrect because a scram is not required unless there is more than one rod drifting.

B. This is correct because per Procedure 05-1-02-V-9, after the immediate action to find and attempt to restore air pressure, it directs the user to Procedure 05-1-02-IV-1 if there are any control rod drifts. In this procedure, if there is one rod drifting, the direction is to manually insert the one rod (Section 2.2 of procedure).

C. This is incorrect because although the rod is drifting as the result of low scram air header pressure, the problem is upstream of the pressure regulator as indicated by the low air pressure alarm. The input for this alarm is sensed upstream of the scram air header pressure regulating valve.

D. This is incorrect because the action is to drive the rod so it is fully inserted. accumulators. The present indications are not indicative of this situation.

Technical References:

Procedure 05-1-02-V-9, “Loss of Instrument Air;” Procedure 05-1-02-IV-1, “Control Rod/Drive Malfunctions;” GLP-OPS-P5300

References to be provided to applicants during exam:

None

Learning Objective: To be determined

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:	55.41.10	

Examination Outline Cross-Reference	Level	RO
Ability to operate and/or monitor Alternate heat removal methods as they apply to LOSS OF SHUTDOWN COOLING (CFR: 41.7 / 45.6).	Tier #	1
	Group #	1
	K/A #	295021 AA1.04
	Rating	3.7

Question 9

The plant indications are as follows:

- The reactor is in Mode 4
- It has been 20 days since the reactor was shut down, during which a shuffle of some of the fuel has taken place
- Reactor coolant temperature is 120^0F
- RHR train "A" is in service

The RHR "A" pump fails. Per procedure 05-1-02-III-1, the maximum amount of time available to establish an alternative means of decay heat removal is:

A. 5 hours

B. 7 hours

C. 8 hours

D. 11 hours

Answer: D

Explanation:

A. This is incorrect based on the Explanation to Answer D.

B. This is incorrect based on the Explanation to Answer D.

C. This is incorrect based on the Explanation to Answer D.

D. This is correct because Attachment I, Figure 2, the graph shows that with 20 days shut down, post-shuffle, that it takes 11 hours to reach 200^0F based on an initial temperature of 120^0F. This is related to considerations taken in Section 3.3 of the procedure.

Technical References:

Procedure 05-1-02-III-1, “Inadequate Decay Heat Removal”

References to be provided to applicants during exam:

Procedure 05-1-02-III-1

Learning Objective: To be determined

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:	55.41.7, 55.45.6	

Examination Outline Cross-Reference	Level	RO
Knowledge of the reasons for responses of interlocks associated with fuel handling equipment as they apply to REFUELING ACCIDENTS (CFR: 41.5 / 45.6).	Tier #	1
	Group #	1
	K/A #	295023 EK3.02
	Rating	3.4

Question 10

For fuel handling operations, interlocks that prevent adding excessive reactivity to the core are the:

A. Fuel handling bridge and trolley interlocks

B. Refueling platform interlocks

C. Fuel Handling Bridge Main hoist interlocks

D. Refueling Platform Frame/monorail mounted auxiliary hoist interlocks

Answer: B

Explanation:

A. This is incorrect based on the Explanation for Answer B. Reactivity control is not one of the reasons stated for this interlock set.

B. This is correct because this is stated as one of the purposes for this set of interlocks in GLP-RF-F1101.

C. This is incorrect based on the Explanation for Answer B. Reactivity control is not one of the reasons stated for this interlock set.

D. This is incorrect based on the Explanation for Answer B. Reactivity control is not one of the reasons stated for this interlock set.

Technical References:

GLP-RF-F1101

References to be provided to applicants during exam:

None

Learning Objective: To be determined

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:	55.41.5, 55.45.6	

Examination Outline Cross-Reference	Level	RO
Ability to operate and/or monitor the emergency generators as they apply to HIGH DRYWELL PRESSURE (CFR: 41.7 / 45.6).	Tier #	1
	Group #	1
	K/A #	295024 EA1.06
	Rating	3.7

Question 11

Following a transient, plant conditions are as follows:

- Reactor power is 0%
- Drywell pressure is 1.40 psig
- Reactor vessel level is 42"

Which of the following would result in a trip of a running EDG?

A. Generator current of 1580 Amps

B. Generator speed of 520 rpm

C. Lube oil temperature of 220^0F

D. Generator frequency of 59 Hz

Answer: B

Explanation:

A. This is incorrect because it is not a diesel generator parameter that would result in a trip in the given situation (normally would at 1575 A). See Explanation for Answer B.

B. This is correct because when a LOCA signal is present (Drywell pressure greater than 1.39 psig), only two trips are active for the diesel generators (speed ›517.5 rpm and generator differential overcurrent ›75 A). Other trip parameters are not in effect in this situation. A diesel generator speed of 520 rpm would result in a trip.

C. This is incorrect because it is not a diesel generator parameter that would result in a trip in the given situation (normally would at 200^0F). See Explanation for Answer B.

D. This is incorrect because it is not a diesel generator parameter that would result in a trip in the given situation (normally would at 59.5 Hz). See Explanation for Answer B.

Technical References:

GLP-OPS-P7500, GLP-OPS-MCD16, GLP-OPS-C7100, FSAR Figure 6.3-14

References to be provided to applicants during exam:

None

Learning Objective: To be determined

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:	55.41.7, 55.45.6	

Examination Outline Cross-Reference	Level	RO
Knowledge of the interrelations between HIGH REACTOR PRESSURE and the reactor/turbine pressure regulating system (Plant Specific) (CFR: 41.7 / 45.8) (EK2.08 - RO)	Tier #	1
	Group #	1
	K/A #	295025 EK2.08
	Rating	3.7

Question 12

The plant is operating at rated power when the A Recirc Loop Flow controller malfunction causes reactor power to begin to slowly rise.

Before operators take any action, what should be the initial automatic response of the Main Turbine HP Control Valves and Bypass Control Valves?

	HP Control Valves	Bypass Control Valves
A.	Open farther	Begin to open
B.	Open farther	Remain as is
C.	Remain as is	Begin to open
D.	Remain as is	Remain as is

Answer: B

Explanation:

As reactor power rises so too does reactor pressure, causing the HP Control valves to open farther (from approximately 50-60% open at rated pressure) in order to control the rising turbine inlet steam pressure. Initially, the Bypass Control Valves have no reason to move (from fully closed), while the HP Control Valves still have opening room to control the rising pressure. Therefore, only choice 'B' is correct.

Technical References:

GLP-OPS-N3202, EHC Control Oil lesson plan

References to be provided to applicants during exam:

None

Learning Objective: GLP-OPS-N3202, objectives 3.1 and 4.2

Question Source:	Bank #	X
(note changes; attach parent)	Modified Bank #	
	New	
Question History:	Last NRC Exam	2008
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.5, 55.45.6	

Examination Outline Cross-Reference	Level	RO
Knowledge of conservative decision making practices as it applies to Suppression Pool High Water Temp. (CFR: 41.10 / 43.5 / 45.12) (2.1.39 – RO)	Tier #	1
	Group #	1
	K/A #	295026 G2.1.39
	Rating	3.7

Question 13

While performing an extended post-maintenance test on RCIC, the crew notes the suppression pool average temperature is approaching the Technical Specification (TS) limit.

In this situation, the conservative action for the operating crew would be to:

A. Immediately enter TS 3.6.2.1, Suppression Pool Average Temperature.

B. Begin reducing reactor power.

C. Suspend the RCIC test.

D. Notify the NRC Senior Resident Inspector within 1 hour.

Answer: C

Explanation:

Based on a rising suppression pool temperature that is approaching the TS limit, the conservative action would be to suspend the RCIC test. None of the other actions listed will stop adding heat to the suppression pool.

Technical References:

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	
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.10	

Examination Outline Cross-Reference	Level	RO
Knowledge of the interrelations between HIGH CONTAINMENT TEMPERATURE (MARK III CONTAINMENT ONLY) and the containment spray (plant-specific): (CFR: 41.7 / 45.8).	Tier #	1
	Group #	1
	K/A #	295027 EK2.01
	Rating	3.2

Question 14

A LOCA has occurred and the reactor operator notes the containment is in the “Unsafe to Initiate” part of the Containment Spray Initiation Pressure Limit (CSPIL) figure.

Containment sprays should not be initiated in this situation because there is the potential for:

A. the thermal shock causing a failure of the containment spray header.

B. a rapid increase in containment pressure forcing water from the suppression pool into the drywell and a subsequent loss of ECCS injection.

C. losing all ECCS injection flow due to the loss of suppression pool inventory.

D. drawing a vacuum in containment and a loss of containment integrity.

Answer: D

Explanation:

A. This is incorrect because the water traveling the CS system from the suppression pool would be at an elevated temperature and there would be minimal thermal shock potential.

B. This is incorrect because while the containment is at a high temperature and low pressure, the sprays would not evaporate and increase pressure but would actually condense steam and reduce pressure.

C. This is incorrect because following a LOCA SP inventory would increase and any water used for CS would return to the suppression pool.

D. This is correct because per the EOP bases for Figure 3 of 05-S-01-EP-1, Containment Spray Initiation Pressure Limit (CSIPL), is the evaporative cooling would result in a rapid drop in pressure and could go sub-atmospheric and cause a loss of containment integrity.

Technical References:

Procedure 05-S-01-EP-1 and the EP Technical Bases Document

References to be provided to applicants during exam:
None

Learning Objective: To be determined

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:	55.41.7, 55.45.8	

Examination Outline Cross-Reference	Level	RO
Ability to determine and/or interpret the Drywell pressure as it applies to HIGH DRYWELL TEMPERATURE (CFR: 41.10 / 43.5 / 45.13).	Tier #	1
	Group #	1
	K/A #	295028 EA2.04
	Rating	4.1

Question 15

The reactor is at 450 psia and a steam line rupture in the drywell has driven drywell temperature to 330 F. Even under these worst case conditions, the operator would expect the design maximum drywell pressure of _____ psig not to be exceeded.

A. 15 psig

B. 21 psig

C. 30 psig

D. 40 psig

Answer: C

Explanation:

A. This is incorrect because i) of the discussion cited in the Explanation for Answer C, and ii) this is the maximum Containment pressure.

B. This is incorrect because i) of the discussion cited in the Explanation for Answer C, and ii) this is the maximum Drywell design differential pressure.

C. This is correct because DBA analysis of a steam line rupture in the Drywell shows that with worst case conditions (maximum steam enthalpy at RPV pressure of 450 psia), the peak temperature stated would result in a maximum design pressure of 30 psig.

D. This is incorrect because of the discussion cited in the Explanation for Answer C.

Technical References:

GLP-OPS-M4101, GLP-OPS-MCD07

References to be provided to applicants during exam:

None

Learning Objective: To be determined

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:	55.41.10, 55.43.5, 55.45.13	

Examination Outline Cross-Reference	Level	RO
Ability to determine and/or interpret the Suppression pool temperature as it applies to LOW SUPPRESSION POOL WATER LEVEL (CFR: 41.10 / 43.5 / 45.13).	Tier #	1
	Group #	1
	K/A #	295030 EA2.02
	Rating	3.9

Question 16

A medium break LOCA has occurred following an earthquake. Currently reactor pressure is steady at 750 psig, RPV level is slowly rising, and suppression pool level is lowering. If suppression pool water level continues to lower, then an emergency depressurization would be required:

A. at a lower suppression pool temperature.

B. before suppression pool level reaches 15.5 feet.

C. at a lower containment temperature.

D. at a higher containment pressure.

Answer: A

Explanation:

A. This is correct based on the HCTL limit. With RPV pressure constant, as SP level decreases, the ED on suppression pool temperature decreases.

B. This is incorrect because the minimum suppression level allowed is 14.5 (not 15.5) feet and this does not change.

C. This is incorrect because the ED for containment temperature is a function of containment pressure. Suppression pool level does not influence the CSPIL.

D. This is incorrect because the PSP curve shows as the SP level lowers, the limit on containment pressure lowers.

Technical References:

Procedures 05-S-01-EP-1, 05-S-01-EP-3

References to be provided to applicants during exam:

None

Learning Objective: To be determined

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:	55.41.10	

Examination Outline Cross-Reference	Level	RO
Ability to operate and/or monitor low pressure core spray as it applies to REACTOR LOW WATER LEVEL (CFR: 41.7 / 45.6).	Tier #	1
	Group #	1
	K/A #	295031 EA1.03
	Rating	4.4

Question 17

A small break LOCA has occurred in the drywell and currently:

- Drywell pressure is 1.50 psig and rising
- RPV level is -140 inches and lowering
- RPV pressure is 400 psig
- RHR train "A" is in service

Based on these conditions, the flow indicated on FI-R600, LPCS Pump Discharge Flow, would be expected to be approximately:

A. 0 gpm

B. 1250 gpm

C. 1380 gpm

D. 7200 gpm

Answer: A REVISE Distractors

Explanation:

A. This is incorrect because although reactor vessel level has not reached the LPCS initiation point (-150.3"), LPCS has initiated due to Drywell pressure (1.39 psig).

B. This is correct because i) LPCS has initiated due to Drywell pressure, ii) LPCS Injection Shutoff Valve F-005 opened at 476 psig, but iii) the discharge pressure of the LPCS pump is 330 psig. Flow will be controlled by the Minimum Flow Control Valve, F-011, which is set to recirculate flow at about 1250 gpm.

C. This is incorrect because the reactor pressure is not low enough for LPCS flow to begin entering the reactor vessel. This is the threshold where the Minimum Flow Control Valve fully closes to ensure all LPCS flow goes to the reactor vessel.

D. This is incorrect because of the Explanation for Answer B. This is the full LPCS flow capacity to the reactor vessel.

Technical References:

GLP-OPS-E2100

References to be provided to applicants during exam:

None

Learning Objective: To be determined

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:	55.41.7, 55.45.6	

Examination Outline Cross-Reference	Level	RO
Knowledge of the operational implications of the reactor pressure effects on reactor power as it applies to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN: (CFR: 41.8 to 41.10) (EK1.01)	Tier #	1
	Group #	1
	K/A #	295037 EK1.01
	Rating	4.1

Question 18

A transient resulted in a reactor scram and currently;

- Reactor power is 8 percent and stable
- Reactor pressure is being maintained with the bypass valves at 900 psig
- Suppression pool temperature is 87 degrees and stable
- Reactor water level is -70 inches and slowly lowering

In this situation, RPV pressure reductions should be made:

A. Rapidly to minimize the loss of RCS water inventory.

B. Slowly to minimize the heat up of the suppression pool.

C. Only after the hot shutdown boron weight has been injected into the reactor to ensure the reactor remains shutdown.

D. In a manner that does not result in exceeding the Technical Specification cooldown limit to avoid a rapid positive reactivity addition.

Answer: D

Explanation:

A. This is incorrect because a rapid depressurization will reduce the amount of inventory loss, it will also result in an increase in reactor power. With reactor power stable and no threat to the fuel or containment building, an emergency depressurization is not desirable.
B. This is incorrect because with the bypass valves in operation there is no heat being rejected to the suppression pool.
C. This is incorrect because pressure reductions should be minimized until the cold shutdown boron weight has been injected. Otherwise, there is the potential for an inadvertent criticality event with the reduction in RCS temperature.
D. This is correct because by limiting the rate of pressure reduction, the rate of RCS temperature reduction is controlled so that there is not a rapid positive reactivity addition.

Technical References: EOP Bases Document

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	
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.10	

Examination Outline Cross-Reference	Level	RO
Knowledge of the reasons for emergency depressurization as it applies to HIGH OFF-SITE RELEASE RATE (CFR: 41.5 / 45.6).	Tier #	1
	Group #	1
	K/A #	295038 EK3.04
	Rating	3.6

Question 19

Procedure 05-S-01-EP-4, Auxiliary Building Control, requires an emergency depressurization if a system is discharging outside of primary/secondary containment that cannot be isolated and a "General Emergency" declaration is expected due to a radioactive release.

The reason for emergency depressurizing is to:

A. Reduce the leak rate that is being released outside containment

B. Maintain adequate core cooling

C. Minimize the loss of RCS inventory

D. Ensure RPV level control is maintained using the preferred injection systems

Answer: A

Explanation:

A. This is correct because of the bases for entering Emergency Depressurization with a high radioactive release rate. The reason for taking this course of action is to place the primary system in the lowest possible energy state. This reduces the release rate potential to the outside.

B. This is incorrect because of the reason for entering the emergency depressurization stated for Answer A. That is the primary reason for entrance based on this situation.

C. This is incorrect because of the reason for entering the emergency depressurization stated for Answer A. That is the primary reason for entrance based on this situation.

D. This is incorrect because of the reason for entering the emergency depressurization stated for Answer A. That is the primary reason for entrance based on this situation.

Technical References:

Procedure 05-S-1-EP-4, [PNPP PEI Bases Document, Section on PEI-D17, “Radioactivity Release Control”]

NOTE: Need to see if this is the same basis stated in some sort of EP basis document provided by Grand Gulf. None provided as of yet.

References to be provided to applicants during exam:

EP Documents

Learning Objective: To be determined

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:	55.41.5, 55.45.6	

Examination Outline Cross-Reference	Level	RO
Knowledge of the reasons for actions contained in the abnormal procedure for plant fire on site as it applies to PLANT FIRE ON SITE.	Tier #	1
	Group #	1
	K/A #	600000 – AK3.04
	Rating	2.8

Question 20

A fire starts in Room OC407, one of the RPS MG Set rooms. As a fire response action, the control building fan coil unit (Z17B002) is secured. This action is taken in order to:

A. Aide in Fire Brigade accessibility to the RPS MG Set Room

B. Prevent smoke from migrating to non-affected rooms

C. Deenergize all electrical equipment that could be damaged by the fire

D. Ensure the discharged CO_2 is not evacuated from the room

Answer: B

Explanation:

A. This is incorrect based on the Explanation for Answer B).

B. This is correct because per Procedure 10-S-03-2, Section 6.2.2, Sub-part f), fire in a list of rooms require securing of this fan coil unit for this reason.

C. This is incorrect based on the Explanation for Answer B).

D. This is incorrect based on the Explanation for Answer B).

Technical References:

Procedure 10-S-03-2, “Response to Fires”

References to be provided to applicants during exam:

None

Learning Objective: To be determined

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:	None	

Examination Outline Cross-Reference	Level	RO
Ability to identify and interpret diverse indications to validate the response of another indication (Loss of Main Condenser Vacuum) (CFR: 41.7 / 43.5 / 45.4).	Tier #	1
	Group #	2
	K/A #	295002 2.1.45
	Rating	4.3

Question 21

The reactor is at 80 percent power when the reactor operator observes main condenser vacuum indicates 26” Hg and is lowing (trending toward no vacuum). Lowering vacuum would be expected if:

A. Extraction steam to feedwater heater 5A has isolated.

B. Seal Steam Generator Level indicates 9 inches above centerline on P680.

C. OG POST-TREAT RAD MON DNSC alarm is actuated

D. SJAE A(B) STM SPLY FLO LO alarm is actuated

Answer: D

Explanation:

A. This is incorrect because although feedwater temperature would go down the temperature and pressure of the steam supply from the reactor would not change. Therefore this would not affect condenser vacuum (see GLP-OPS-N2335).

B. This is incorrect because although increasing seal steam generator level is a concern, isolation of the seal steam system does not occur until the High-High-High alarm at 12” above centerline (on Panel P680, see GLP-OPS-N3300). At 9 inches, seal steam would not be lost and therefore this would not result in a change that would affect condenser vacuum.

C. This is incorrect because the having the alarm on Panel P601 confirms one of the two system trip channels (A or B) caused an alarm to actuate. Both channels need to have trip signals to indicate that the system is isolated. Confirmation of this would be to verify that the position of Offgas Isolation Valve N64-F060 is closed (see GLP-OPS-D1721).

D. This is correct because the reception of this alarm on Panel P680 will result in the automatic closing of isolation valves between the LP condenser and the Steam Jet Air Ejectors, causing a loss of vacuum (GLP-OPS-N6200).

Technical References:

GLP-OPS-MCD12, GLP-OPS-N2335, GLP-OPS-N6200, GLP-OPS-N3300, GLP-OPS-D1721, GLP-OPS-N6465

References to be provided to applicants during exam:

None

Learning Objective: To be determined

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:	55.41.7, 55.43.5, 55.45.4	

Examination Outline Cross-Reference	Level	RO
Ability to determine and/or interpret Reactor water level as it applies to LOW REACTOR WATER LEVEL (CFR: 41.10 / 43.5 / 45.13).	Tier #	1
	Group #	2
	K/A #	295009 AA2.01
	Rating	4.2

Question 22

The current plant conditions are as follows:

- Reactor vessel pressure 1025 psig
- Recirculation Loops A and B flow is 44500 gpm each
- Reactor Pressure Vessel level at +32" on Narrow Range level instrumentation

Under these conditions, the low reactor water level condition would result in a level reading on Wide Range level instrumentation of approximately:

A. 17"

B. 31"

C. 41"

D. 49"

Answer: A

Explanation:

A. This is correct because based on the calibration conditions of the Wide Range level instrumentation (no recirculation flow), the level indication reads low at high recirculation flows. Per GLP-OPS-B2101, Page 19, at 100% recirculation flow, level reads about 15" lower than actual level. Since the indicated recirculation flow is around 100%, seeing the level indication around 15" lower than indicated is expected.

B. This is incorrect due to the effects described in the Explanation to Answer A.

C. This is incorrect due to the effects described in the Explanation to Answer A.

D. This is incorrect due to the effects described in the Explanation to Answer A.

Technical References:

GLP-OPS-B2101, GLP-OPS-B3300

References to be provided to applicants during exam:

None

Learning Objective: To be determined

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:	55.41.10, 55.43.5, 55.45.13	

Examination Outline Cross-Reference	Level	RO
Knowledge of the operational implications of temperature increases as it applies to HIGH DRYWELL PRESSURE (CFR: 41.8 to 41.10).	Tier #	1
	Group #	2
	K/A #	295010 AK1.03
	Rating	3.2

Question 23

The plant is at rated thermal power when Plant Service Water is lost to the drywell. In this situation, drywell pressure would be controlled by:

A. Placing the Containment Cooling System in the drywell purge mode of operation.

B. Manually initiating the Standby Gas Treatment System.

C. Manually initiating the Drywell Purge Subsystem of the Combustible Gas Control System.

D. Aligning SSW to the drywell chillers.

Answer: D

Explanation:

A. This is incorrect it is an acceptable means of manipulating Drywell pressure only in plant Modes 4 and 5.

B. This is incorrect because SGTS is not used for a backup function such as this. SGTS initiation isolations Containment Ventilation from interfacing with it, so its operation could not have any effect on Drywell conditions.

C. This is incorrect because this subsystem is used as a means to increase pressure in the Drywell and reduce Drywell hydrogen concentration during a LOCA. It does not have a backup function to help maintain normal Drywell pressure.

D. This is correct because the primary means of controlling drywell pressure is by controlling drywell temperature. This is done by using the Drywell Cooling and Drywell Chilled Water Systems. PSW is the primary cooling supply to the Drywell Chillers, with SSW as a backup.

Technical References:

GLP-OPS-M5100, GLP-OPS-M4100, GLP-OPS-T4801, GLP-OPS-E6100

References to be provided to applicants during exam:

None

Learning Objective: To be determined

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:	55.41.8 to 55.41.10	

Examination Outline Cross-Reference	Level	RO
Knowledge of the reasons for the increased containment cooling (Mark-III) response as it applies to HIGH CONTAINMENT TEMPERATURE (MARK III CONTAINMENT ONLY) (CFR: 41.5 / 45.6).	Tier #	1
	Group #	2
	K/A #	295011 AK3.01
	Rating	3.6

Question 24

As containment temperature continues to increase above 95 degrees, EP-3, Containment Control, drives the crew to use more severe methods to cool the containment. An emergency depressurization is required before reaching 185 degrees in the containment. The reason for this is to:

A. keep the containment temperature below the design limit.

B. ensure the reactor pressure and water level instrumentation remain valid.

C. to ensure the RPV is depressurized while the suppression pool is still capable of absorbing the energy of a blowdown.

D. ensure the containment sprays can be used to cool the containment without violating the containment spray temperature limit.

Answer: A

Explanation:

A. This is correct because the primary containment design temperature limit of 185 degrees.

B. This is incorrect because although the instrumentation may be affected it will still be valid well above 185 degrees.

C. This is incorrect because the containment temperature is not related to blowdown capability. This is a function of suppression pool temperature.

D. This is incorrect because ability to use containment sprays is a function of both temperature and pressure. A lower temperature may still violate the CSIPL.

Technical References:

GLP-OPS-M4100, GLP-OPS-M4101 (Statement of Technical Specification 3.6.1.5),

GFIG-OPS-M4100, GLP-OPS-G3336

References to be provided to applicants during exam:

None

Learning Objective: To be determined

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:	55.41.5, 55.45.6	

Examination Outline Cross-Reference	Level	RO
Knowledge of the interrelations between HIGH OFF-SITE RELEASE RATE and the standby gas treatment/FRVS (CFR: 41.7 / 45.8).	**Tier #**	1
	Group #	1
	K/A #	295017 AK2.12
	Rating	3.4

Question 25

The Standby Gas Treatment System is set to automatically initiate if there is:

A. 20 mR/hr sensed at the Fuel Pool Sweep Exhaust

B. 3.5 mR/hr sensed at the Containment Vent Exhaust

C. 3.6 mR/hr sensed at the Fuel Handling Area Exhaust

D. 3 times the normal radiation level on any of the Main Steam Lines

Answer: C

Explanation:

A. This is incorrect because the threshold for SGTS trains starting based on FPS Exhaust radiation readings is 30 mR/hr.

B. This is incorrect because the radiation alarm set point for Containment Vent Exhaust does not provide an input to STGS.

C. This is correct this is one of the initiators for SGTS Trains “A” and “B” per GLP-OPS-T4801 and GLP-OPS-D1721.

D. This is incorrect because the radiation alarm set point for Main Steam Lines does not provide an input to STGS.

Technical References:

GLP-OPS-T4801, GLP-OPS-D1721, GFIG-OPS-D1721

References to be provided to applicants during exam:

None

Learning Objective: To be determined

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:	55.41.7, 55.45.8	

Examination Outline Cross-Reference	Level	RO
Knowledge of the operational implications of the Reactivity control concepts as they apply to LOSS OF CRD PUMPS: (CFR: 41.8 to 41.10).	Tier #	1
	Group #	2
	K/A #	295022 AK1.02
	Rating	3.6

Question 26

Following the loss of the running Control Rod Drive Hydraulic pump, ONEP 05-1-02-IV-1, "Control Rod/Drive Malfunctions," directs the operator to start the standby pump. Prior to starting the standby pump, the procedure directs the operator to close CRD Flow Control Valves.

Comment [s1]:

The reason for closing the flow control valve prior to starting the standby pump is to:

A. reduce the possibility of rod drifts.

B. minimize the thermal shock on the control rod drive piston seals.

C. reduce the starting current on the CRD pump motor.

D. prevent water hammer damage to the CRD system.

Answer: A

Explanation:

A. This is correct because the CRD Flow Control Station is the means of controlling the transient effects on the system, and the downstream pressure sensed at the Pressure Control Station can cause rod drifts when large pressure differentials between the system and reactor pressure. Per GLP-OPS-C111A, cooling water pressures greater than 20 PSID can cause rod drifts. This step in the procedure allows for controlled return to service of both flow and pressure control functions in a manner that reactivity is not inadvertently manipulated.

B. This is incorrect because although this action will reduce thermal stresses on the piston seals, it is not the basis for the action in the off-normal event procedure.

C. This is incorrect because although this action will reduce CRD pump starting current, it is not the basis for the action in the off-normal event procedure.

D. This is incorrect because although this action will reduce the chances of adverse pressure stresses on the system, it is not the basis for the action in the off-normal event procedure.

Technical References:

GLP-OPS-C111A, GFIG-OPS-C111A, ONEP 05-1-02-IV-1, TBD

References to be provided to applicants during exam:

None

Learning Objective: To be determined

Question Source: (note changes; attach parent)	Bank #	
	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.8, 55.41.9, 55.41.10	

Examination Outline Cross-Reference	Level	RO
Ability to determine and / or interpret the Hydrogen monitoring system availability as it applies to HIGH PRIMARY CONTAINMENT HYDROGEN CONCENTRATIONS (CFR: 41.10 / 43.5 / 45.13).	Tier #	1
	Group #	2
	K/A #	500000 EA2.01
	Rating	3.1

Question 27

Plant conditions are as follows:

- Drywell pressure is 4.5 psig
- Hydrogen analyzer "A" and "B" control switches are in AUTO
- CNTMNT H2 HI alarm is actuated
- Hydrogen recorder blue pen shows steadily increasing hydrogen concentration which stabilized at 10%
- Hydrogen recorder red pen shows stable concentration at 2%

Based on these conditions and 05-S-01-EP-3, "Containment Control," what is the correct action(s) for Hydrogen Control?

A. Verify all hydrogen analyzers operating and energize the Division I and II Hydrogen Igniters

B. Sample Containment for hydrogen levels

C. Enter SAP-2 for Hydrogen control

D. Sample Containment for hydrogen levels and enter SAP-2 for Hydrogen control

Answer: D

Explanation:

A. This is incorrect because the Containment hydrogen analyzers are "unavailable," and the Containment hydrogen concentration has indicated in the past greater than 2.9%. This will cause an exit of EP-3 before entering these actions.

B. This is incorrect because the entry into SAP-2 actions based on Containment hydrogen concentrations greater than 2.9% needs to be addressed as well.

C. This is incorrect because the sampling of Containment hydrogen levels needs to be addressed as well.

D. This is correct because although there indication of Containment hydrogen concentration above 2.9% (entry to SAP-2), there is no valid indication of Containment's actual hydrogen concentrations. The top end of the analyzer's indicating scale is 10%. This means the Containment hydrogen analyzers cannot perform the function specified in the EP/SAP, so they are "unavailable" (see text on "system availability" in Procedure 05-S-01-SAP-1, Section 1.4.1). Therefore, both actions need to be addressed concurrently.

Technical References:

GLP-OPS-E6100, Procedure 05-S-01-SAP-1, Procedure 05-S-01-PSTG, Procedure 05-S-01-EP-3

References to be provided to applicants during exam:

None

Learning Objective: To be determined

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:	55.41.10, 55.43.5, 55.45.13	

Examination Outline Cross-Reference	Level	RO
Knowledge of how to conduct system lineups, such as valves, breakers, switches, etc. (CFR: 41.10 / 45.1 / 45.12) (2.1.29)	Tier #	2
	Group #	1
	K/A #	Generic – 2.1.29
	Rating	4.1

Question 28

Comment [s2]: The LO/TO procedure, EN-OP-102, does not address the standard practices for verifying the position of a valve. It says that it must be verified in the denoted position numerous times, but does not state specific practices. It lists a number of qualification programs for people working in the LO/TO program, so I assume that these practices are covered in their specific training.

Therefore, a question has been drafted based on their mechanical tagout practices per EN-OP-102.

A valve associated with a tagout is located in a High Radiation Area. What is an acceptable approach to hanging a tag on and verifying the position of the valve?

A. Entry into the area of the valve must be completed under an approved RWP to hang the tag and conduct the valve position verification.

B. A Danger tag stating that the valve is part of a Tagout boundary may be installed at the entrance to the area with the valve, with access controls instituted by the Shift Manager.

C. A tag hanging and valve position verification are not needed if the valve will not have work performed on it and it will remain inaccessible for the duration of the Tagout.

D. Hanging a tag on the valve can be forgone if it can be verified in the required position by alternate methods, the valve will not have work performed on it, and it will remain inaccessible during the Tagout implementation.

Answer: D

Explanation:

A. This is incorrect because it is not mandatory to enter area of the valve to apply the tag. There are exception criteria stated in Section 2.5 of Attachment 9.2, EN-OP-102.

B. This is incorrect because the method stated involving applying a tag to the entrance to the area containing the valve involves a Caution tag vice a Danger tag.

C. This is incorrect because it only meets a portion of one of the methods of addressing the tagout in EN-OP-102, Attachment 9.2. Verification of valve position must be provided for if this approach is used, and it is not stated as part of the answer.

D. This is correct because it meets acceptance criteria for an exception to hanging a tag/conducting valve lineup verification per Section 2.5 of Attachment 9.2, EN-OP-102. Having the valve in a High Radiation Area qualifies as an inaccessible area.

Technical References:

EN-OP-102, Attachment 9.2

References to be provided to applicants during exam:

None

Learning Objective: To be determined

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:	55.41.10	

Examination Outline Cross-Reference	Level	RO
Knowledge of the effect that a loss or malfunction of the SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE) will have on reactor water level (plant-specific): (CFR: 41.7 / 45.4) (K3.02 – RO)	Tier #	2
	Group #	1
	K/A #	205000 K.2
	Rating	3.2

Question 29

The reactor has been shutdown for 3 weeks and is in Mode 4. The current plant conditions are as follows:

- Shutdown cooling is being provided by RHR A through the feedwater header return line
- Reactor Recirc Pump A is in operation
- Feed and Condensate are not in service
- Both RWCU pump motors are off-site being refurbished
- Reactor water level is stable

A scaffold pole falls on the MOV for E12-F053A, RHR A Shutdown Cooling Return to Feedwater, and E12-F053A closes. With no further action;

A. reactor water level will decrease.

B. RHR Pump A will automatically trip.

C. CST level will increase.

D. reactor water level will increase.

Answer: A

Explanation:

A. When the shutdown cooling return valve closes the min flow valve (E12-F064A) will open due to the low flow condition. This will result in water being pumped from the RPV to the suppression pool. Consequently, RPV level will decrease and suppression pool level will increase.

B. This is incorrect, the RHR pump will automatically trip on a loss of flowpath but this is sensed on the suction flow-path and not the discharge flow[path. The RHR pump will continue running.

C. is incorrect however the candidate may choose this answer if the student believes the min-flow valve discharges to the CST.

D. is incorrect because water inventory is being lost through the min-flow valve. Some candidates may choose this answer if they don't consider the opening of the min flow valve or the new discharge flow-path. If these are not considered, indicated level would slowly go up due to decay heat induced swell.

Technical References:

Lesson Plan GLP-OPS-E1200

References to be provided to applicants during exam:

None

Learning Objective: GLP-OPS-E1200, objectives 3.4, 8.1, and 8.2

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:	55.41.7	

Examination Outline Cross-Reference	Level	RO
Knowledge of the process for conducting special or infrequent tests as it relates to LPCS. (CFR: 41.10 / 43.3 / 45.13) (2.2.7)	Tier #	2
	Group #	1
	K/A #	209001 G2.2.7
	Rating	2.9

Question 30

A modification to the Low Pressure Core Spray motor windings requires a special test be performed by electrical maintenance before it is declared operable.

Prior to performing the special test, Procedure 01-S-07-2, Test Control, requires a Pre-Test Briefing be conducted by the:

A. Individual supervising the test.

B. Control Room Supervisor.

C. Shift Manager.

D. Superintendent of Electrical Maintenance.

Answer: A – Are these the correct titles?

Explanation:

E. This is correct because the referenced procedure requires the Pre-Test Briefing by held by the person supervising the test.
F. This is incorrect because the referenced procedure requires the Pre-Test Briefing by held by the person supervising the test.
G. This is incorrect because the referenced procedure requires the Pre-Test Briefing by held by the person supervising the test.
H. This is incorrect because the referenced procedure requires the Pre-Test Briefing by held by the person supervising the test.

Technical References: Procedure 01-S-07-2, Test Control

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	
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.10	

Examination Outline Cross-Reference	Level	RO
Knowledge of the effect that a loss or malfunction of the Suppression pool suction strainer (BWR-5, 6) will have on the HIGH PRESSURE CORE SPRAY SYSTEM (HPCS) (CFR: 41.7 / 45.7).	Tier #	2
	Group #	1
	K/A #	209002 K6.04
	Rating	2.5

Question 31

High Pressure Core Spray (HPCS) is in service taking a suction from the suppression pool and discharging back to the suppression pool for a post maintenance test. During the test, a sheet of plastic is sucked into the suppression pool suction strainer blocking it.

One of the effects of this blockage would be HPCS:

A. pump suction pressure will increase.

B. motor current will decrease.

C. pump discharge pressure would increase.

D. motor current will increase.

Answer: B

Explanation:

A. This is incorrect because HPCS pump suction pressure will decrease.

B. This is correct because the amount of work being performed by the pump decreases as flow goes down. The reduction in pump work will cause a decrease in motor current.

C. This is incorrect because as pump flow and suction pressure go down, pump discharge pressure will also go down.

D. See explanation for answer B.

Technical References:

GLP-OPS-E2201

References to be provided to applicants during exam:

None

Learning Objective: To be determined

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:	55.41.7, 55.45.7	

Examination Outline Cross-Reference	Level	RO
Knowledge of the effect that a loss or malfunction of the STANDBY LIQUID CONTROL SYSTEM will have on the core spray line break detection system (plant-specific) (CFR: 41.7 / 45.4).	Tier #	2
	Group #	1
	K/A #	211000 K3.02
	Rating	3.0

Question 32

The plant is at rated thermal power when the Drywell Equipment Floor Sump high leakage alarm annunciates. The alarm followup investigation determines there is a significant amount of valve body leakage coming from C41-F008, SLC Manual Isolation Valve..

This condition could also result in a:

A. RCIC leakage alarm.

B. HPCS Out of Service alarm.

C. RPV head leakage alarm.

D. RWCU line break alarm.

Answer: B

Explanation:

A. This is incorrect because a) leak detection in the RCIC system is provided by monitoring equipment area temperatures, equipment area ventilation differential temperatures; steam flow/pressure to the turbine, and high pump room sump levels; and b) the SLC system does not directly interface with the RCIC system.

B. This is correct because the SLC discharge line is connected to the HPCS discharge line downstream of the HPCS testable check valve. Reducing the pressure sensed in this line compared to the pressure sensor located above the Core Plate will result in an increased differential pressure. A differential pressure of +/- 1.2 PSID between these points will result in both a LINE BREAK status light on the HPCS status panel, and a HPCS SYS OOSVC alarm will annunciate.

C. This is incorrect because a) leak detection for the RPV head is provided by comparing the pressure between the double O-ring seals, and b) the SLC system does not directly interface with the RPV head.

D. This is incorrect because a) leak detection in the RWCU system is provided by comparing the system inlet and outlet flows, and b) the SLC system does not directly interface with the RWCU system.

Technical References:

GLP-OPS-E3100, GFIG-OPS-E3100, GFIG-OPS-E2201, GLP-OPS-C4100

References to be provided to applicants during exam:

None

Learning Objective: To be determined

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:	55.41.7, 55.45.4	

Examination Outline Cross-Reference	Level	RO
Knowledge of electrical power supplies to the RPS motor-generator sets (CFR: 41.7).	Tier #	2
	Group #	1
	K/A #	212000 K2.01
	Rating	3.2

Question 33

The electrical power supply to RPS Motor Generator Set “B” is:

A. 13B22

B. 16B42

C. 15B42

D. 14B22

Answer: D

Explanation:

A. This is incorrect because it is not the correct power supply.

B. See Explanation for Answer A.

C. See Explanation for Answer A.

D. This is correct because it is the power supply to the RPS “B” motor generator set.

Technical References:

GLP-OPS-C7100

References to be provided to applicants during exam:

None

Learning Objective: To be determined

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:	55.41.7	

Examination Outline Cross-Reference	Level	RO
Knowledge of INTERMEDIATE RANGE MONITOR (IRM) SYSTEM design feature(s) and/or interlocks which provide for the rod withdrawal blocks (CFR: 41.7).	Tier #	2
	Group #	1
	K/A #	215003 K4.01
	Rating	3.7

Question 34

The reactor is in Mode 2 with a startup in progress. All the Intermediate Range Monitors (IRMs) are indicating mid-scale on Range 5 when IRM "A" fails downscale.

With no other action;

A. a rod withdraw block ONLY is in effect.

B. a scram will occur if the IRM B switch is taken out of "OPERATE"

C. a rod withdraw AND insert block are in effect

D. control rods can still be inserted or withdrawn

Answer: A

Explanation:

A. This is correct because with any one IRM downscale on range 5 a rod withdraw block is in effect.

B. This is incorrect because the IRM downscale does not create a scram signal. Therefore taking IRM B out of operate will result in a half scram.

C. This is incorrect because the IRM rod blocks do not affect the ability to insert rods.

D. This is incorrect because a single downscale signal causes a withdraw rod block.

Technical References:

GLP-OPS-C5102, GFIG-OPS-C5102

References to be provided to applicants during exam:

None

Learning Objective: To be determined

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:	55.41.7	

Examination Outline Cross-Reference	Level	RO
Knowledge of the effect that a loss or malfunction of the RPS will have on the SOURCE RANGE MONITOR (SRM) SYSTEM (CFR: 41.7 / 45.7).	Tier #	2
	Group #	1
	K/A #	215004 K6.01
	Rating	3.2

Question 35

With the plant in Mode 5 and the RPS shorting links removed, the Electrical Protection Assembly associated with RPS Bus A motor-generator trips on undervoltage.

In this situation, SRM A will:

A. deenergize and result in a full scram.

B. deenergize and result in a half-scram.

C. remain energized and result in a half-scram.

D. remain energized with no scram signal provided.

Comment [s3]: We can't use the term "present" here because there is a half-scram signal in place due to the loss of bus power, not associated with the SRM.

Answer: D

Explanation:

A. This is incorrect because i) power is not lost to the SRMs based on the situation, and ii) the loss of power to the SRMs would not meet either of the two input conditions needed for a scram trip signal (SRM Mode Switch not in OPERATE, or reactor power ≥ 2×10^5 cps).

B. See Explanation for Answer A.

C. This is incorrect because it is always the case that any scram trip signal from any neutron monitoring device (SRM, IRM, or APRM) will cause a full scram with the shorting links removed. No coincidence logic is needed for SRM scrams, and in this case, for any neutron monitoring scrams. Therefore, the association of the SRMs with a given RPS trip system is not relevant to their function.

D. This is correct because i) the power supply to the SRMs is different than the RPS buses, so power is maintained, and ii) it is always the case that any scram trip signal from any neutron monitoring device (SRM, IRM, or APRM) will cause a full scram with the shorting links removed. No coincidence logic is needed for SRM scrams, and in this case, for any neutron monitoring scrams.

Technical References:

GLP-OPS-C7100, GLP-OPS-C5101

References to be provided to applicants during exam:

None

Learning Objective: To be determined

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:	55.41.7, 55.45.7	

Examination Outline Cross-Reference	Level	RO
Knowledge of the operational implications of assignment of LPRM's to specific APRM channels as they apply to AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM (CFR: 41.5 / 45.3).	Tier #	2
	Group #	1
	K/A #	215005 K5.06
	Rating	2.5

Question 36

LPRM detectors are assigned to a specific APRM channel in order to:

A. maintain axial and rotational power symmetry.

B. maintain rotational and mirror-image power symmetry

C. minimize wiring function problems associated with the detectors observed based on industry experience

D. ensure that each APRM channel receives an adequate representation of the local flux throughout the core, both radially and axially.

Answer: D

Explanation:

A. This is incorrect because it ties into the basis for LPRM placement in the core (symmetrical monitoring method), but it is not the basis for the assignment of the LPRM detectors to the APRMs. The axial element addresses part of the LPRM/APRM association issue, but it is out of context with the rest of the answer.

B. This is incorrect it ties into the basis for LPRM placement in the core (symmetrical monitoring method), but it is not the basis for the assignment of the LPRM detectors to the APRMs.

C. This is incorrect because there is no industry feedback that would serve as the major reason for not changing LPRM detector assignments.

D. This is correct because it is the basis stated in GLP-OPS-C5103, Page 15, and GLP-OPS-C5104, Page 17.

Technical References:

GLP-OPS-C5104, GLP-OPS-C5103

References to be provided to applicants during exam:

None

Learning Objective: To be determined

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:	55.41.5, 55.45.3	

Examination Outline Cross-Reference	Level	RO
Knowledge of electrical power supplies to the gland seal compressor (vacuum pump): (CFR: 41.7)	Tier #	2
	Group #	1
	K/A #	217000 K2.04
	Rating	2.6

Question 37

The power supply to the RCIC gland seal compressor is:

A. 1DA2

B. 11DA

C. 1DA1

D. 1DC1

Answer: A

Explanation:

A. This is correct because it is the power supply to the RCIC gland seal compressor per GLP-OPS-E5100.

B. This is incorrect because it is not the correct power supply.

C. See Explanation for Answer B.

D. See Explanation for Answer B.

Technical References:

GLP-OPS-E5100, GLP-OPS-E2201

References to be provided to applicants during exam:

None

Learning Objective: To be determined

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:	55.41.7	

Examination Outline Cross-Reference	Level	RO
Knowledge of the effect that a loss or malfunction of the condensate storage and transfer system will have on the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) (CFR: 41.7 / 45.7).	Tier #	2
	Group #	1
	K/A #	217000 K6.04
	Rating	3.5

Question 38

An identified leak in the Condensate Storage Tank requires that it be drained for repairs. This will affect the function of the RCIC system by eliminating the:

A. ability to flush the system.

B. back-up supply of water for reactor vessel injection.

C. ability to perform flow tests on the system.

D. flow path for the system minimum flow line.

Answer: C

Explanation:

A. This is incorrect because the CST does not provide a flushing water volume for the RCIC system. The CST provides flushing capability for other systems (see GFIG-OPS-P1100, Figure 1), but not for the RCIC system.

B. This is incorrect because the CST is the primary supply of water to the RCIC system, not the back-up supply.

C. This is correct because the RCIC flow testing discharge path only flows to the CST, so this would be affected by this maintenance. This is one of the three inter-system functions stated in GLP-OPS-E5100.

D. This is incorrect because the flow path for the RCIC system minimum flow line discharges to the Suppression Pool, not the CST.

Technical References:

GLP-OPS-E5100, GFIG-OPS-E5100, GLP-OPS-P1100, GFIG-OPS-P1100

References to be provided to applicants during exam:

None (To be determined – check system books)

Learning Objective: To be determined

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:	55.41.7, 55.45.7	

Examination Outline Cross-Reference	Level	RO
Knowledge of the physical connections and/or cause-effect relationships between AUTOMATIC DEPRESSURIZATION SYSTEM and the nuclear boiler instrument system (CFR: 41.2 to 41.9 / 45.7 to 45.8).	Tier #	2
	Group #	1
	K/A #	218000 K1.03
	Rating	3.7

Question 39

The following are plant conditions:

- Reactor Power is 0%
- Reactor vessel level is -140" and decreasing slowly
- Drywell pressure is 1.3 psig
- RPV pressure is 650 psig
- RCIC is running
- LPCS and all RHR pumps are in standby
- A LOCA has been diagnosed for approximately 10 minutes

One of the operators just manually started the LPCS pump. Based on these conditions, the ADS:

A. Timer has not yet initiated.

B. Timer has started the 105 second time delay following the LPCS pump start.

C. Valves opened immediately following the LPCS pump start.

D. Will immediately open all ADS valves when RPV level reaches Level 1.

Answer: A

Explanation:

A. This is correct because the Level 1 initiation signal has not been satisfied.

B. This is incorrect the 105 second time delay does not start until three indications are provided (reactor vessel ≤ +11.4", and reactor vessel ≤ -150.3", and Drywell pressure ≥ 1.39 psig).

C. This is incorrect because although one low pressure ECCS pump must be running, ADS will not initiate until RPV level reaches Level 1.

D. This is incorrect because when level 1 is reached, the 105 second time delay will initiate.

Technical References:

GLP-OPS-E2202

References to be provided to applicants during exam:

None

Learning Objective: To be determined

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:	55.41.2 to 55.41.9; 55.45.7 to 55.45.8	

Examination Outline Cross-Reference	Level	RO
Ability to (a) predict the impacts of the Loss of A.C. or D.C. power to ADS valves on the AUTOMATIC DEPRESSURIZATION SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5 / 45.6).	Tier #	2
	Group #	1
	K/A #	218000 A2.05
	Rating	3.4

Question 40

The plant is at 75% power when a loss of LCC 15BA6 and 15BA3 occur. This would result in a loss of :

A. A loss of Division 1 and 2 ADS logic functions in 2 hours and require entering LCO 3.5.1, ECCS – Operating

B. A loss of Division 2 ADS logic and “B” valve functions in 2 hours and require entering LCO 3.8.4, DC Sources – Operating

C. A loss of RCIC and require entering LCO 3.5.3, RCIC System

D. A loss of Division 1 ADS logic and “A” valve functions in 4 hours and require entering LCO 3.8.4, DC Sources – Operating

Answer: D

Explanation:

A. This is incorrect because the affected power supplies have an impact on Division 1/ “A” ADS valves only. See Explanation for Answer D.

B. This is incorrect because the power supplies to Division 2/ “B” ADS valves are LCC 16BB6 and 16BB3.

C. This is incorrect because the affected power supplies would not have an impact on RCIC.

D. This is correct based on the capacity of the ESF battery banks stated in GLP-OPS-L1100. Without AC power, ESF battery banks for Division I and II ESF can supply 125 VDC loads for up to 4 hours. The two LCCs indicated are the AC supplies for DC Bus 11DA. This supplies power to ADS Division 1 logic and “A” ADS/SRV pilot solenoid valves per GLP-OPS-E2202. This would also require entering LCO 3.8.4, DC Sources – Operating

Technical References:

GLP-OPS-E2202, GLP-OPS-L1100, FSAR Figures 8.3-1 and 8.3-10

References to be provided to applicants during exam:

None

Learning Objective: To be determined

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:	55.41.5, 55.45.6	

Examination Outline Cross-Reference	Level	RO
Ability to predict and/or monitor changes in parameters associated with operating the PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF controls including valve closures: (CFR: 41.5 / 45.5).	Tier #	2
	Group #	1
	K/A #	223002 A1.02
	Rating	3.7

Question 41

The plant has experienced a loss of all high pressure injection and plant parameters are normal except RPV water level is +20 inches and slowly lowering.

If RPV level continues to slowly lower, the first automatic isolation would be the:

A. Main steam isolation valves (MSIVs)

B. Reactor Water Cleanup isolation valves

C. RHR to Radwaste isolation valves

D. Service air isolation valves

Answer: C

Explanation:

A. This is incorrect because the NSSSS isolation would occur at level 1 (- 150 incles).

B. This is incorrect because the NSSSS isolation would occur at level 2 (- 41.6 inches).

C. This is correct because the isolation would occur at + 11.4 inches. Of the 4 choices, this would be the first setpoint reached as level decreased.

D. This is incorrect because this isolation would occur at level 2 (- 41.6 inches).

Technical References:

GLP-OPS-M7101

References to be provided to applicants during exam:

None

Learning Objective: To be determined

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:	55.41.5, 55.45.5	

Examination Outline Cross-Reference	Level	RO
Ability to (a) predict the impacts of the following on the RELIEF/SAFETY VALVES; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: ADS Actuation (CFR: 41.5 / 45.6) (A2.04)	Tier #	1
	Group #	1
	K/A #	239002 A2.04
	Rating	4.1

Question 42

The reactor is operating at 100 percent power with no work activities in progress when 6 ADS valves unexpectedly open and will not reclose. Current plant conditions are:

- HI-HI Levels in Feedwater Heaters 6A and 6B.
- Feedwater injection temperature is decreasing.
- There is a Level 8 scram signal on RPS Division 2 due to the initial RPV water level swell
- Core flow has increased approximately 4 percent
- APRM Upscale alarms have annunciated and reactor power is now stable
- Suppression Pool temperature is 106 degrees and rising rapidly

Based on these conditions the control room crew should now:

A. Take manual control of feedwater and return RPV water level to the normal band.

B. Reduce Reactor Recirc flow to reduce reactor power to less than 100 percent.

C. Place the Mode Switch in SHUTDOWN to preclude exceeding the Technical Specification limit of 110 degrees.

D. Restore feedwater heaters 6A and 6B to service to keep feedwater injection temperature in Region I of the Feedwater Temperature to Power Map.

Answer: D

Explanation:

A. This is incorrect because there is no indication of a problem with the feedwater level control system.
B. This is incorrect because with suppression pool temperature rapidly rising with no success path to maintain temperature below the TS limit, reducing reactor power would serve no purpose.
C. This is correct because with suppression pool temperature at 106 degrees, rapidly rising, and no means to close the ADS valves, the TS limit of 110 degrees will be exceeded. The TS required action is to place the Mode Switch in shutdown.

D. This is incorrect because the suppression pool temperature will be exceeded before feedwater temperature can be restored and this requires placing the Mode Switch in Shutdown.

Technical References: OE 772 – Manual Scram Due to Multiple Lifts of SRV Valves at Grand Gulf

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	
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.10	

Examination Outline Cross-Reference	Level	RO
Ability to monitor automatic operations of the REACTOR WATER LEVEL CONTROL SYSTEM including runout flow control (plant-specific): (CFR: 41.7 / 45.7) (A3.01)	Tier #	1
	Group #	1
	K/A #	259002 A3.01
	Rating	3.0

Question 43

The reactor is in MODE 1 and:

- Reactor Feed Pump A flow is 5 mlbm/hr.
- Reactor Feed Pump B is being placed in service and flow is 3 mlbm/hr.
- The Reactor Water Level Control System is in 3 element control.
- Reactor water level is +40"
- Both Reactor Recirc Pumps are in fast speed.

Following a trip of Reactor Feed Pump A, the FeedWater Level Control System (FWLCS) Net Positive Suction Head (NPSH) interlock will cause the:

A. Reactor recirc pumps to downshift to slow speed.

B. Reactor recirc flow control valves to ramp to the minimum flow position.

C. Feedwater level control system to shift to single element control.

D. FWLCS flow signal to transfer from normal to estimated.

Answer: D

Explanation:

A. This is correct because if total feedwater flow falls below 3.84 mlbm/hr, the recirc pumps will downshift to slow speed.
B. This is incorrect because the flow control valve runback is bypassed if RPV level is greater than 32.7".
C. This is incorrect because this will occur only on a failure of the total steam flow or feedflow inputs to the FWLCS.
D. This is incorrect because this transfer only occurs if there is a failure of one of the flow signal inputs to the FWLCS.

Technical References: Lesson Plans C34 and B33

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	
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.5	

Examination Outline Cross-Reference	Level	RO
Ability to predict and/or monitor changes in parameters associated with operating the STANDBY GAS TREATMENT SYSTEM controls including system flow (CFR: 41.5 / 45.5).	Tier #	2
	Group #	1
	K/A #	261000 A1.01
	Rating	2.9

Question 44

The plant indications are as follows:

- Reactor power is 0%
- Reactor vessel level is -44”
- Drywell pressure is 1.20 psig
- Annulus pressure is -.10” wc
- Standby Gas Treatment System Trains “A” and “B” are in AUTO

Placing the hand switch for Train “B” exhaust fan in the STOP position will result in a steady-state SGTS system flow rate of approximately:

A. 0 scfm

B. 1850 scfm

C. 3850 scfm

D. 7700 scfm

Answer: D

Explanation:

A. This is incorrect because the Train “B” exhaust fan would not secure from operation. See Explanation for Answer B.

B. This is incorrect based on the Explanation for Answer D.

C. This is incorrect and is based on only one fan running. See the Explanation for Answer D.

D. This is correct because the initiation logic for both SGTS trains has been met with reactor vessel level less than -41.6.” With this initiation signal present, placing the exhaust fan control hand switch for either train in the STOP position will not secure then fan. Therefore, both fans will continue to operate and maintain the discharge flow that it is set to provide in normal operation. The approximate system flow-rate is 3875 scfm per fan. (GLP-OPS-T4801, Page 20.)

Technical References:

GLP-OPS-T4801

References to be provided to applicants during exam:

None

Learning Objective: To be determined

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:	55.41.5, 55.45.5	

Examination Outline Cross-Reference	Level	RO
Knowledge of the operational implications of the following concepts as they apply to A.C. ELECTRICAL DISTRIBUTION: principle involved with paralleling two A.C. sources (CFR: 41.5 / 45.3).	Tier #	2
	Group #	1
	K/A #	262001 K5.01
	Rating	3.1

Question 45

The plant is operating with bus 16AB being powered from ESF Transformer 21. A failure of ESF Transformer 21 results in a loss of power to the bus. Assuming no other malfunctions of related equipment, it is desired to power the bus with ESF Transformer 12. If the operator moves to close breaker 152-1611 with no further action, what is the expected response?

A. Bus 16AB is deenergized because closing this breaker supplies power to bus 15AA

B. Bus 16AB is energized by DG 12

C. Bus 16AB is deenergized because breaker 152-1614 must be manually opened first before breaker 152-1611 will close

D. Bus 16AB is energized by ESF Transformer 21

Answer: B

Explanation:

A. This is incorrect because breaker 152-1611 is an appropriate breaker to supply power to bus 16AB.

B. This is correct because breaker 152-1611 would not close without the Synch Switch being turned on. Therefore, DG 12 continues to supply power to bus 16AB.

C. This is incorrect because DG 12 continues to supply power to bus 16AB. Also, “make-before-break” transfer feature in the incoming offsite feeder breakers will ensure that breaker 152-1614 opens automatically before 152-1611 closes.

D. This is incorrect because the operator failed to recognize that DG 12 started and automatically supplied power to bus 16AB. Due to this, the operator should turn on the Synch Switch for breaker 152-1611, synchronize between the two supplies on either side of the breaker, and then close the breaker. Failing to do so results in no operation of breaker 152-1611.

Technical References:

GLP-OPS-R2100, GFIG-OPS-R2100, GLP-OPS-P7500

References to be provided to applicants during exam:

None
NOTE: Need to check the validity of the assumptions made here.

Learning Objective: To be determined

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:	55.41.5, 55.45.3	

Examination Outline Cross-Reference	Level	RO
Ability to manually operate and/or monitor in the control room: voltage, current, power, and frequency on A.C. buses (CFR: 41.7 / 45.5 to 45.8)	Tier #	2
	Group #	1
	K/A #	262001 A4.05
	Rating	3.3

Question 46

The Division 3 Diesel Generator (DG) is carrying the Division 3 electrical distribution system following a loss of off-site power.

Following placing the DG speed control switch in the RAISE position, the operator would expect:

A. Generator MW to increase.

B. Generator MVARS to increase.

C. Generator voltage to increase.

D. Generator frequency to increase.

Answer: A

Explanation:

A. This answer is incorrect because with the diesel operating alone the MW loading is strictly a function of the loads operating on the bus and raising speed will have no affect on the operating loads.

B. This answer is incorrect because with the diesel operating alone the MVAR loading is strictly a function of the types of loads operating on the bus and raising speed will have no affect on the operating loads.

C. This answer is incorrect because operating the engine speed control will have no affect on generator voltage.

D. This answer is correct because with the diesel operating alone the engine speed control will change the speed of the engine and thereby change the operating frequency of the generator.

Technical References:

References to be provided to applicants during exam:

None

Learning Objective: To be determined

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:	55.45.5	

Examination Outline Cross-Reference	Level	RO
Knowledge of UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.) design feature(s) and/or interlocks which provide for the following: transfer from preferred power to alternate power supplies (CFR: 41.7) (K4.01)	Tier #	2
	Group #	1
	K/A #	262002 K4.01
	Rating	3.1

Question 47

An equipment operator has been directed to manually transfer a BOP static inverter from the normal to the alternate source of power.

When the manual static transfer switch is depressed, the inverter will transfer to the alternate power source only if the:

A. The INVERTER FAILURE indicator is lit.

B. The REVERSE POLARITY indicator is not lit.

C. The IN SYNC indicator is lit.

D. The MANUAL BYPASS SWITCH is in the BYPASS position.

Answer: D

Explanation:

A. This is incorrect because this indicator indicates a transfer should have occurred but does not have any input to a manual transfer.
B. This is incorrect because this indicator indicates the power inputs to the inverter are reversed but this will not prevent a transfer.
C. This is correct because if the two power sources are not in synch the transfer will not occur.
D. This is incorrect because taking this switch to bypass will immediately transfer the power source regardless of the static transfer switch.

Technical References: Lesson Plan L62, page 14

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	
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.5	

Examination Outline Cross-Reference	Level	RO
Ability to manually operate and/or monitor [UPS] in the control room: transfer from alternative source to preferred source (CFR: 41.7 / 45.5 to 45.8) (A4.01)	Tier #	2
	Group #	1
	K/A #	262002 A4.01
	Rating	2.8

Question 48

An automatic transfer of a BOP inverter from the alternative source to the normal source would be indicated by a(n):

A. SPDS Computer Trouble Alarm.

B. RCIS Trouble Alarm.

C. Static Inverter Trouble Alarm.

D. RCIC Trouble Alarm.

Answer: C

Explanation:

A. This is incorrect because this alarm will not annunciate unless there is a power loss which would not be the case in this scenario.
B. This is incorrect because this alarm will not annunciate unless there is a power loss which would not be the case in this scenario.
C. This is correct because any static transfer will activate the Static Inverter Trouble Alarm.
D. This is incorrect because this alarm will not annunciate unless there is a power loss which would not be the case in this scenario.

Technical References: Lesson Plan L62, page 16/17

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	
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.5	

Examination Outline Cross-Reference	Level	RO
Knowledge of the physical connections and/or cause-effect relationships between D.C. ELECTRICAL DISTRIBUTION and Battery charger and battery: (CFR: 41.2 to 41.9 / 45.7 to 45.8).	Tier #	2
	Group #	1
	K/A #	263000 K1.02
	Rating	3.2

Question 49

Losing power to ESF panel PP 15P42 would result in loss of charging capability for battery bank(s):

A. 1A3

B. 1K3 and 1L3

C. 1J2

D. 1H1 and 1H2

Answer: D

Explanation:

A. This is incorrect because of the Explanation for Answer D. Battery bank 1A3 is associated with ESF DC battery charger 1A4.

B. This is incorrect because of the Explanation for Answer D. Battery banks 1K3 and 1L3 are associated with 125 VDC battery chargers 1K4 and 1L4, respectively (BOP).

C. This is incorrect because of the Explanation for Answer D. Battery bank 1J2 is associated with 24 VDC battery charger 1J5.

D. This is correct because PP 15P42 is the power supply for 24 VDC battery chargers 1H4 and 1H5 per GLP-OPS-L1100. These battery chargers provide charge to the battery banks indicated.

Technical References:

GLP-OPS-L1100, GFIG-OPS-L1100

References to be provided to applicants during exam:

None (To be determined – check system books)

Learning Objective: To be determined

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:	55.41.2 to 55.41.9; 55.45.7 to 55.45.8	

Examination Outline Cross-Reference	Level	RO
Knowledge of EMERGENCY GENERATORS (DIESEL/JET) design feature(s) and/or interlocks which provide for speed droop control (CFR: 41.7).	Tier #	2
	Group #	1
	K/A #	264000 K4.03
	Rating	2.5

Question 50

Diesel Generator 11 (DG 11) is running in parallel to the grid during a quarterly surveillance. During the surveillance, the operator inadvertently places the Unit Parallel Switch in RESET.

In response to this, the DG 11 output breaker would;

A. Remain closed and the voltage regulator would shift to the isochronous mode.

B. Immediately trip open.

C. Immediately trip open ONLY if there is a LOCA signal present.

D. Remain closed and the voltage regulator would remain in the droop mode.

Answer: B

Explanation:

A. This is incorrect because placing the switch in the reset position will immediately trip the generator output breaker.

B. This is correct because placing the switch in the reset position will immediately trip the generator output breaker.

C. This is incorrect because placing the switch in the reset position will immediately trip the generator output breaker.

D. This is incorrect because placing the switch in the reset position will immediately trip the generator output breaker.

Technical References:
GLP-OPS-P7500.16, Page 45 of 73

References to be provided to applicants during exam:

None

Learning Objective: To be determined

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:	55.41.7	

Examination Outline Cross-Reference	Level	RO
Knowledge of the connections and / or cause- effect relationships between INSTRUMENT AIR SYSTEM and the cooling water to compressor (CFR: 41.2 to 41.9 / 45.7 to 45.8).	Tier #	2
	Group #	1
	K/A #	300000 K1.04
	Rating	2.8

Question 51

On a loss of off-site power, the alignment of cooling water for the Instrument Air System compressors shifts as follows:

A. TBCW cooling is lost, and SSW Loop B is automatically aligned to provide cooling to the Unit 1 Instrument Air Compressor

B. CCW cooling is lost, and Plant Service Water is automatically aligned to provide cooling to both Instrument Air Compressors

C. TBCW cooling is lost, and Plant Service Water is automatically aligned to provide cooling to the Unit 2 Instrument Air Compressor

D. CCW cooling is lost, and SSW Loop A is automatically aligned to provide cooling to the Unit 1 Instrument Air Compressor

Answer: A

Explanation:

A. This is correct because this is the automatic realignment for Loss of Power in GLP-OPS-P5300.

B. This is incorrect because the cooling water supplies stated have no connection to the Instrument Air System, so they do not provide alignment changes on Loss of Power.

C. This is incorrect because Plant Service Water is not the backup cooling water supply, and the Loss of Power realignment goes to the Unit 1 Instrument Air Compressor.

D. This is incorrect because CCW is not the primary source of cooling water to the Instrument Air System. Also, the Loss of Power alignment is associated with SSQ Loop B vice A.

Technical References:

GLP-OPS-P5300, GFIG-OPS-P5300

References to be provided to applicants during exam:

None

Learning Objective: To be determined

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:	55.41.2 to 55.41.9; 55.45.7 to 55.45.8	

Examination Outline Cross-Reference	Level	RO
Ability to monitor automatic operations of the INSTRUMENT AIR SYSTEM including air temperature (CFR: 41.7 / 45.7).	Tier #	2
	Group #	1
	K/A #	300000 A3.02
	Rating	2.9

Question 52

The Unit 2 Instrument Air System compressor trips with the Unit 1 compressor starting automatically. Indications at the Instrument Air Compressor Local Panel show "High Discharge Air Temperature" and "High Condensate Level" alarm lights lit.

With regard to the Unit 2 Instrument Air Compressor:

A. The compressor tripped based on compressor 3rd stage inlet temperature being greater than 125 F

B. The compressor tripped on high intercooler/aftercooler drain trap levels greater than 6 inches

C. Both alarm lights indicate undesirable compressor parameters, but they do not result in an auto trip of the compressor. The compressor trip was the result of another fault.

D. The compressor tripped on high intercooler/aftercooler drain trap levels greater than 10 inches

Answer: A

Explanation:

A. This is correct because the high air discharge temperature alarm is associated with an automatic compressor trip function at the stated set point per GLP-OPS-P5300.

B. This is incorrect because although there is an alarm condition on the condensate levels in the compressor, there is no automatic compressor trip function associated with this alarm.

C. This is incorrect because the high air discharge temperature alarm does trigger an automatic trip of the compressor.

D. See the Explanation for Answer B.

Technical References:

GLP-OPS-P5300, GFIG-OPS-P5300

References to be provided to applicants during exam:

None

Learning Objective: To be determined

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:	55.41.7, 55.45.7	

Examination Outline Cross-Reference	Level	RO
Knowledge of tagging and clearance procedures (CFR: 41.10 / 45.13).	Tier #	2
	Group #	1
	K/A #	400000 General – 2.2.13
	Rating	4.1

Question 53

Repairs are planned on the Component Cooling Water Post-Accident Sample Cooler inlet isolation valve that involves grinding on the valve as part of the repair. One of the requirements for this grinding activity is that:

A. A vent and drain path must be established from the heat exchanger

B. The welding or grinding is on the side of the valve isolated by the tagout

C. The repair activity will allow for the valve to be returned to service in a short time deemed appropriate by the Operations Supervisor

D. An additional upstream isolation valve must be tagged closed in addition to the valve(s) already identified

Answer: B

Explanation:

A. This is incorrect because although it is a standard practice for working on heat exchangers, it is not one of the criteria stated for welding/grinding activities for tagged valves.

B. This is correct because the conditions for welding or grinding on a tagged valve are stated in Section 2.13 of Attachment 9.2, EN-OP-102. This is one of the explicit conditions stated.

C. This is incorrect because Section 2.13 states that the nature of the activity will not prevent the valve from functioning as required by the tagout. There is no allowance for management judgment as to whether it can be reassembled and usable in a short period of time.

D. This is incorrect because it is not one of the criteria for welding/grinding on tagged valves as stated in Section 2.13.

Technical References:

EN-OP-102, Attachment 9.2

References to be provided to applicants during exam:

None

Learning Objective: To be determined

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:	55.41.10, 55.45.13	

Examination Outline Cross-Reference	**Level**	RO
Ability to manually operate and/or monitor the SDV isolation valve test switch in the control room (CFR: 41.7 / 45.5 to 45.8).	**Tier #**	2
	Group #	2
	K/A #	201001 A4.06
	Rating	2.8

Question 54

The operator is testing the operation of the CRD SDV isolation valves. When the operator depresses pushbutton C11-F009A, CRD DISCH VOL VENT/DR SOV TEST, on Control Room Panel P680, the operator should see:

A. The vent valve pair close, as indicated by one green (CLOSED) light

B. The red (OPEN) lights are illuminated for the vent and drain valve pairs

C. The drain valve pair close, as indicated by one green (CLOSED) light

D. The vent and drain valve pairs close, as indicated by two green (CLOSED) lights

Answer: B

Explanation:

A. This is incorrect because both test pushbuttons need to be depressed to deenergize the SDV pilot solenoid valves, which operate both sets of vent and drain valves. Pressing one pushbutton will not operate a subset of the vent and drain valves.

B. This is correct because both test pushbuttons, C11-F009A and C11-F009B, need to be depressed to deenergize the SDV pilot solenoid valves. Therefore, pressing one pushbutton will not operate either the vent or drain valve pairs, and the red (OPEN) lights should remain illuminated.

C. See explanation for answer A.

D. See explanation for answer A.

Technical References:

GLP-OPS-C111A, GFIG-OPS-C111A

References to be provided to applicants during exam:

None

Learning Objective: To be determined

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:	55.41.7, 55.45.5, 55.45.6, 55.45.7, 55.45.8	

Examination Outline Cross-Reference	Level	RO
Knowledge of the physical connections and/or cause-effect relationships between CONTROL ROD AND DRIVE MECHANISM and the control rod drive hydraulic system: (CFR: 41.2 to 41.9 / 45.7 to 45.8).	**Tier #**	2
	Group #	2
	K/A #	201003 K1.01
	Rating	3.2

Question 55

Control rod withdrawal speed would increase if the:

A. Flow control valve fails open

B. Pressure control valve fails open

C. Associated stabilizer valve fails closed

D. Cooling/exhaust water pressure equalizing valve fails closed

Answer: A

Explanation:

A. This is the correct it would provide for increased drive water pressure to the CRD mechanisms. The Pressure Control Station would respond to help compensate, but the increased flow/pressure capacity would have the dominant effect on the control rod speed.

B. This is incorrect because CRD drive pressure would increase, while upstream flow would be maintained constant by the flow control valve. Control rod speed would decrease.

C. This is incorrect because the stabilizer valves failing closed would provide the normal drive pressure expected, since they are closed when the control rods are withdrawn or inserted.

D. This is incorrect because these valves are normally closed at power operations.

Technical References:

GLP-OPS-C111B, GFIG-OPS-C111A, GLP-OPS-C111A, GFIG-OPS-B1300

References to be provided to applicants during exam:

None

Learning Objective: To be determined

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:	55.41.2 through 55.41.9; 55.45.7 and 55.45.8	

Examination Outline Cross-Reference	Level	RO
RCIS - 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. (CFR: 41.5 / 43.5 / 45.12)	Tier #	3
	Group #	
	K/A #	201005 - 2.2.44
	Rating	4.2

Question 56

Following a malfunction with the rod position indicator for control rod 28-33, it is bypassed in the Rod Action Control System (RACS).

Verification that the correct control rod has been bypassed could be performed by:

A. Ensuring rod 28-33 will not move when selected for individual rod movement and given a withdraw command.

B. Ensuring the POS BYP pushbutton on P-680 is lit.

C. Depressing the POS BYP pushbutton on P-680 and ensuring rod 28-33 has a green indicator lit on the full core display.

D. Depressing the SUBST POS pushbutton on P-680 and ensuring rod 28-33 has a red indicator lit on the full core display.

Answer: C

Explanation:

A. This is incorrect for bypassing the rod in the RACS but would be correct if the rod is bypassed in the RGDS.

B. This incorrect because any control rod being bypassed in the RACS will illuminate the POS BYP pushbutton.

C. This is correct because depressing the POS BYP pushbutton will illuminate the green indicating light adjacent to all the control rods bypassed in the RACS.

D. This is incorrect because depressing the SUBST POS pushbutton on P-680 will only indicate those rods that have substitute data entered, and not those that are bypassed.

Technical References:
GLP-OPS-C1102.03, page 29 of 50.

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	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.7	

Examination Outline Cross-Reference	Level	RO
Knowledge of the effect that a loss or malfunction of the Control rod drive system (plant-specific) will have on the RECIRCULATION SYSTEM (CFR: 41.7 / 45.7).	Tier #	2
	Group #	2
	K/A #	202001 K6.05
	Rating	2.7

Question 57

While operating at 100% power, the alarm CRD PMP SUCT BKW FLTR DP HI is received. Sustained operations with this condition can affect the Recirculation System by:

A. Limiting the recirculation Flow Control Valve operation due to high HPU oil temperature

B. Decreasing the cooling capacity to the recirculation pump motor oil heat exchanger

C. Degrading the integrity of the recirculation pump shaft seal heat exchanger

D. Degrading the integrity of the recirculation pump shaft seals

Answer: D

Explanation:

A. This is incorrect because air provides cooling to HPU oil coolers.

B. This is incorrect because CCW provides cooling to this component.

C. This is incorrect based on the Explanation for Answer B.

D. This is correct because the CRD Hydraulic system provides cooling to the shaft seals in the recirculation pumps.

Technical References:

GLP-OPS-B3300, GLP-OPS-C111A, GFIG-OPS-B3300, GFIG-OPS-C111A

References to be provided to applicants during exam:

None

Learning Objective: To be determined

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:	55.41.7, 55.45.7	

Examination Outline Cross-Reference	Level	RO
Ability to predict and/or monitor changes in parameters associated with operating the NUCLEAR BOILER INSTRUMENTATION controls including recorders and meters (CFR: 41.5 / 45.5).	Tier #	2
	Group #	2
	K/A #	216000 A1.01
	Rating	3.4

Question 58

The plant indications are as follows:

- Reactor coolant temperature is 120^0F
- Actual reactor vessel level is +35"
- Recirculation pump A and B are running in slow speed

Given these conditions, the most accurate level indication would be the:

A. Narrow Range

B. Wide Range

C. Shutdown Range

D. Upset Range

Answer: C

Explanation:

A. This is incorrect because Narrow Range is calibrated for RPV pressure at 1025 psig, which result in error in the given indication.

B. This is incorrect because wide Range is calibrated for RPV pressure at 1025 psig with no recirculation flow, which results in error in the given indication.

C. This is correct because the conditions most closely match the calibrated conditions for this level instrumentation. The reactor coolant temperature at less than 200^0F implies that reactor pressure is 0 psig, which is one of the calibration conditions. With the available range being +0" to +400", this is a level that would indicate on scale.

D. This is incorrect because the Upset Range is calibrated for RPV pressure at 1025 psig, which result in error in the given indication.

Technical References:

GLP-OPS-B2101

References to be provided to applicants during exam:

None

Learning Objective: To be determined

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:	55.41.5, 55.45.5	

Examination Outline Cross-Reference	Level	RO
Knowledge of RHR/LPCI: CONTAINMENT SPRAY SYSTEM MODE design feature(s) and/or interlocks which provide for pump minimum flow protection (CFR: 41.7).	Tier #	2
	Group #	2
	K/A #	226001 K4.05
	Rating	2.5

Question 59

Residual Heat Removal (RHR) A is operating in Containment Spray (CS) Mode following an automatic initiation of the CS sub-system of RHR. A control room operator inadvertently places the control switch for E12-F004A, RHR PMP A SUCT FM SUPP POOL, in the "CLOSE" position.

As a result of this action, E12-F004A will:

A. Close and RHR Pump A will trip.

B. Close and then reopen with RHR Pump A continuing to run.

C. Not close because of the CS signal.

D. Not close because of the LOCA signal.

Answer: A.

Explanation:

A. This is correct because E12-F004A does not have any interlocks associated with it and the RHR pump will automatically trip if E12-F004A is not fully open.

B. This is incorrect because E12-F004A does not have any interlocks associated with it to reopen the valve.

C. This is incorrect because E12-F004A does not have any interlocks associated with containment sprays.

D. This is incorrect because E12-F004A has no LOCA signals associated with it.

Technical References:

GLP-OPS-E1200, GFIG-OPS-E1200

References to be provided to applicants during exam:

None

Learning Objective: To be determined

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:	55.41.7	

Examination Outline Cross-Reference	Level	RO
Knowledge of the operational implications of turbine inlet pressure vs. reactor pressure as they apply to REACTOR/TURBINE PRESSURE REGULATING SYSTEM: (CFR: 41.5 / 45.3).	Tier #	2
	Group #	2
	K/A #	241000 K5.04
	Rating	3.3

Question 60

Reactor power is at 90% when the second stage reheat steam is lost to Main Steam Reheater (MSR) A. Following this failure the operator would expect to see steady state reactor power:

A. Decrease, main turbine control valves throttle closed, and generator output decreases

B. Stay the same, main turbine control valves stay in the same position, and generator output decreases

C. Reactor power increases, main turbine control valves open, and generator output increases

D. Reactor power stays the same, main turbine control valves throttle closed, and generator output decreases

Answer: D

Explanation:

A. This is incorrect because the reactor power would not decrease. The situation caused no changes in recirculation pump speed or rod position, so no appreciable changes would be seen.

B. This is incorrect because the main turbine control valves would close to adjust for the decrease in reactor pressure.

C. This is incorrect because the transient would not result in a change reactor power, let alone result in an increase. If the reactor power increased, the other indications would be consistent with this, but the effect on the reactor assumed is incorrect.

D. This is correct because turbine and cycle efficiency decrease. With the loss of steam reheating, the turbine produces less torque (work) to the generator and the condensate temperature will decrease. This will result in less generator output and a lower feedwater temperature to the reactor. With no changes in reactor power (no change in rod position or recirculation pump speed), additional reactor power will be needed to maintain main turbine throttle pressure. This will result in less steam flow and consequently a decrease in reactor pressure. The main turbine control valves would throttle closed to maintain pressure at setpoint. With the main turbine control valves closed more than before the event, the output of the generator decreases further.

Technical References:

GFIG-OPS-N3202

References to be provided to applicants during exam:

None

Learning Objective: To be determined

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:	55.41.5, 55.45.3	

Examination Outline Cross-Reference	Level	RO
Knowledge of the physical connections and/or cause-effect relationships between MAIN TURBINE GENERATOR AND AUXILIARY SYSTEMS and the D. C. electrical distribution: (CFR: 41.2 to 41.9 / 45.7 to 45.8).	Tier #	2
	Group #	2
	K/A #	245000 K1.09
	Rating	2.7

Question 61

Backup power is supplied to the Main Turbine EHC Control System by bus(es):

A. 11DH and 11DJ

B. 11DD

C. 11DK and 11DL

D. 11DF

Answer: A

Explanation:

A. This is correct because the backup power supplies to the various EHC Control System subsystems are these DC buses, per GLP-OPS-N3202.

B. This is incorrect because of the Explanation for Answer A.

C. This is incorrect because of the Explanation for Answer A.

D. This is incorrect because of the Explanation for Answer A.

Technical References:

GLP-OPS-N3202, GFIG-OPS-L1100

References to be provided to applicants during exam:

None

Learning Objective: To be determined

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:	55.41.2 to 55.41.9; 55.45.7 to 55.45.8	

Examination Outline Cross-Reference	Level	RO
Ability to monitor automatic operations of the REACTOR CONDENSATE SYSTEM including pump starts (CFR: 41.7 / 45.7).	Tier #	2
	Group #	2
	K/A #	256000 A3.02
	Rating	3.0

Question 62

The reactor plant is in Mode 2 and:

- Condensate booster pumps A and B were recently started
- Condensate booster pump suction pressure is 200 psig
- Condensate booster pump discharge pressure is 575 psig
- Condensate booster pump A discharge valve is partially open and the discharge valve for pump B full open
- Condensate booster pump A lube oil low pressure indication is lit

Based on these conditions, the operator would expect Condensate Booster Pump A to trip due to:

A. Low pump suction pressure for 15 seconds

B. the pump's discharge valve being partially open for 15 seconds

C. low lube oil pressure condition for 15 seconds

D. Low oil pressure and the suction valve not fully open

Answer: D

Explanation:

A. This is incorrect because the low pump suction pressure automatic trip set point is 44 psig with a 15 second time delay. Current pump suction pressure is well above this set point.

B. This is incorrect because the automatic pump trip associated with the discharge valve is activated if the valve is fully closed for 15 seconds.

C. This answer is incorrect because although the low lube oil pressure (<3 psig) results in an automatic action after 15 seconds, it does not directly secure the pump. It closes the pump's suction valve.

D. This answer is correct because the sustained low lube oil pressure (< 3 psig) causes an automatic closure of the pump suction valve. If the pump's suction valve is not fully open, the associated condensate booster pump trips automatically.

Technical References:

GLP-OPS-N1900, GFIG-OPS-N1900 (NEED TO CHECK VERSUS CONTROL ROOM INDICATIONS)

References to be provided to applicants during exam:

None

Learning Objective: To be determined

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:	55.41.7, 55.45.7	

Examination Outline Cross-Reference	Level	RO
Ability to (a) predict the impacts of following on the REACTOR FEEDWATER SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Loss of Extraction Steam (CFR: 41.5 / 45.6) (A2.04)	Tier #	2
	Group #	2
	K/A #	259001 A2.04
	Rating	3.3

Question 63

Following a refueling outage the control room crew is in the process of returning the reactor to 100 percent power. Currently;

- Reactor power is 30 percent
- Extraction Steam to Feedwater Heaters 5A and 6A is isolated for completion of outage work

Following a transient involving the loss of extraction steam to Feedwater Heater 6B, the crew notes feedwater inlet temperature to the reactor has stabilized at 260 degrees and reactor power is stable at 32 percent.

According to Procedure 05-1-02-v-5, Loss of Feedwater Heating, the:

A. MCPR limits are no longer valid and the crew should trip the main turbine.

B. LHGR limits are no longer valid and the crew should scram the reactor.

C. MCPR limits are no longer valid and the crew should scram the reactor.

D. LHGR limits are no longer valid and the crew should trip the main turbine.

Answer: A

Explanation:

A. This is correct because with feedwater temperature 260 degrees and power 32 percent, the reactor is in Region III of the Feedwater Temperature versus Power map. This requires the crew to trip the main turbine per step 3.2.3 of the referenced procedure. According to step 1.4 of the same procedure, the MCPR limits are no longer valid.
B. This is incorrect because scramming the reactor is not required unless Region IV of the Feedwater Temperature versus Power map is entered.
C. This is incorrect because scramming the reactor is not required unless Region IV of the Feedwater Temperature versus Power map is entered.
D. According to step 1.4 of the same procedure, the MCPR limits are no longer valid.

Technical References: Procedure 05-1-02-v-5, Loss of Feedwater Heating

References to be provided to applicants during exam:

Attachment I of Procedure 05-1-02-v-5, Loss of Feedwater Heating

Learning Objective:

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.10	

Examination Outline Cross-Reference	Level	RO
Knowledge of RADIATION MONITORING System design feature(s) and/or interlocks which provide for redundancy (CFR: 41.7).	Tier #	2
	Group #	2
	K/A #	295010 K4.01
	Rating	2.7

Question 64

During normal full power operations, the OG POST-TREAT RAD HI-HI-HI/INOP alarm comes in. The alarm is due to high voltage Post Treatment LCRM Channel A. What is the response of the Offgas System?

A. The system auto isolates

B. The system shifts to bypass mode

C. The system shifts out of bypass mode if it was in bypass mode at the time

D. The system continues to operate in its normal lineup

Answer: D

Explanation:

A. This is incorrect based on the Explanation for Answer D.

B. This is incorrect because the bypass of the Charcoal Adsorber is only used during pre-operation and startup, when gas activity is very low and a significant amount of moisture is in the process stream (protects charcoal from freezing).

C. This is incorrect because if the Offgas System was in bypass mode, the lineup would be changed to normal lineup with the receipt of a OG POST-TREAT RAD HIGH alarm only.

D. This is correct because a redundant indication of a high radiation condition is needed to initiate system isolation. There are two channels for the LCRMs that must both initiate a signal to result in auto isolation. With one channel providing a fault condition, the alarm comes in, but the system alignment is unaffected.

Technical References:

GLP-OPS-D1721, GLP-OPS-N6465

References to be provided to applicants during exam:

None

Learning Objective: To be determined

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:	55.41.7	

Examination Outline Cross-Reference	Level	RO
Fire Protection: Knowledge of the effect that a loss or malfunction of the FIRE PROTECTION SYSTEM will have on personnel protection: (CFR: 41.7 / 45.4) (K3.02)	Tier #	1
	Group #	2
	K/A #	286000 AK3.02
	Rating	3.2

Question 65

An operator observes a flashing red LED on a ionization detector associated with the Halon Control Room System.

This flashing red LED indicates:

A. The detector is operating normally.

B. The detector has an alarm condition.

C. The detector has a trouble condition.

D. The detector has a low battery condition.

Answer:

Explanation:

A. This answer is correct because a flashing red LED indicates the detector is operating normally.

B. This answer is incorrect because an alarm condition is indicated by a steady red LED on the detector.

C. This answer is incorrect because a trouble condition is indicated by the red LED on the detector being off.

D. This answer is incorrect because while many detectors have battery backup, the Halon detectors do not.

Technical References:

GLP-OPS-P6400, Page 32 of 48.

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:	55.41.7	

Examination Outline Cross-Reference	Level	RO
Knowledge of facility requirements for controlling vital/controlled access. (CFR: 41.10 / 43.5 / 45.9 / 45.10)	**Tier #**	Administrative
	Group #	
	K/A #	Generic - 2.1.13
	Rating	2.5

Question 66

While transiting between spaces in the plant, an operator uses his/her card key to attempt access into a vital area. He/she swipes the card key, then a red light is displayed with audible beeps heard. The next action of the operator is to:

A. Swipe the card key again and open the door when the audible beeps are heard.

B. Open the door and enter the vital area.

C. Contact security for assistance.

D. Swipe the card key again and open the door when a green light is displayed.

Answer: D

Explanation:

A. The audible beeps, along with the red light, indicate that access has been rejected. The vital area door will not allow access.

B. With the first swiping of the card key, the indications are that access has been rejected. Therefore, the system will not allow one to open the door.

C. Per procedure 01-S11-10, the card key needs to be swiped a second time upon initial access rejection prior to contacting security for assistance.

D. With the initial access rejection, the individual needs to swipe the card key another time. If a green light is displayed, the door to the vital area can be opened.

Technical References:

Procedure 01-S-11-10, Sections 6.11.2 and 6.12

References to be provided to applicants during exam:

None

Learning Objective: To be determined

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:	55.41.10, 55.43.5, 55.45.9, 55.45.10	

Examination Outline Cross-Reference	Level	RO
Ability to make accurate, clear, and concise verbal reports. (CFR: 41.10 / 45.12 / 45.13)	**Tier #**	Admin
	Group #	
	K/A #	Generic - 2.1.17
	Rating	3.9

Question 67

A valve lineup on RHR System Cooling A Loop is in progress prior to starting shutdown cooling. As part of this, the Control Room Operator wants the Roving Watch to verify the position of valve F029A. The basic instruction to the Roving Watch is as follows:

A. "Roving Watch, verify the position of RHR A Pump valve F029A."

B. "Roving Watch, this is the Control Room Operator. Verify the position of RHR Alpha Pump valve F029A, that is, Foxtrot zero two nine Alpha."

C. "This is the Control Room Operator. Verify the position of RHR Alpha Pump valve F029A, that is, Foxtrot zero two nine Alpha."

D. "Roving Watch, this is the Control Room Operator. Verify the position of RHR Alpha Pump valve F029A."

Answer: B

Explanation:

A. This is incorrect because the title/work station of the calling individual is not identified. Also, use of the phonetic alphabet is not employed for the valve identifier (F029A).

B. This is correct because it contains all of the necessary message format pieces. The three parts of the basic message are provided: name/work station of called individual, title/work station of the calling individual, and the message (01-S-06-14, Section 6.3.11). Also, if the communication contains alpha-numeric information, the sender and receiver should use the phonetic alphabet. Exceptions to this are approved standard abbreviations, like system names. "RHR" is an example of this exception (for Residual Heat Removal) 01-S-06-14, Section 6.3.3).

C. This is incorrect because the name/work station of the called individual is not identified.

D. This is incorrect because the phonetic alphabet is not employed for the valve identifier

(F029A).

Technical References:

Procedure 01-S-06-14, GFIG-OPS-E1200, GLP-OPS-E1200

References to be provided to applicants during exam:

None (To be determined – check system books)

Learning Objective: To be determined

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:	55.41.10	

Examination Outline Cross-Reference	Level	RO
Knowledge of conservative decision making practices. (CFR: 41.10 / 43.5 / 45.12)	**Tier #**	Administrative
	Group #	
	K/A #	Generic - 2.1.39
	Rating	3.6

Question 68

During shift turnover, it is communicated to the oncoming Reactor Operator that the speeds on the Recirculation Pumps need to be adjusted to counter xenon effects. The outgoing Shift Supervisor has directed the Reactor Operator being relieved to adjust the Recirculation Pump speeds to maintain power at 70%. The following actions at shift turnover should transpire:

A. The oncoming Reactor Operator should assume the duties at the controls and make the Recirculation Pump speed adjustments as needed.

B. The current Reactor Operator should make the adjustments to the Recirculation Pump speeds prior to turning over the duties to the oncoming Reactor Operator.

C. The oncoming Reactor Operator should assume the duties at the controls and make the Recirculation Pump speed adjustments, with the outgoing Reactor Operator remaining nearby to verify his/her actions are correct.

D. The oncoming Reactor Operator should assume the duties at the controls and allow the outgoing Reactor Operator to make the Recirculation Pump speed adjustments.

Answer: B

Explanation:

A. In the Shift Turnover practices stated in the Conduct of Operations procedure (Section 5.16) does allow for turning over the shift to the on-going operator if changes in the equipment are understood by the oncoming operator. However, in cases when reactivity changes (Section 5.4) are directed, they are to be avoided at shift turnover. Taking controls of a directed reactivity change at shift turnover is not the most conservative choice.

B. Since a directed reactivity change needs to be taken care of near shift turnover time, it is best to allow the current Reactor Operator to make the change prior to relieving his/her position. The current Reactor Operator has the knowledge of the situation from his/her previous shift, and would be most capable of completing the change.

C. While this could be conducted this way, the outgoing Reactor Operator may have other duties to attend to that may limit their ability to assist in the reactivity change. Also, related to Answer B., the current Reactor Operator is most capable of making the change based on knowledge of the situation during the past shift.

D. The oncoming Reactor Operator, when he/she assumes the controls, is responsible for changes in reactivity at the controls. Allowing someone else to make the changes is not consistent with responsibility of the on-shift Reactor Operator.

Technical References:

Procedure EN-OP-115, Sections 5.4 and 5.16

References to be provided to applicants during exam:

None

Learning Objective: To be determined

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:	55.41.10	

Examination Outline Cross-Reference	Level	RO
Knowledge of the process for managing maintenance activities during shutdown operations, such as risk assessments, work prioritization, etc. (CFR: 41.10 / 43.5 / 45.13)	Tier #	
	Group #	
	K/A #	Generic – 2.2.18
	Rating	2.6

Question 69

The plant is performing a risk management review of tasks planned for an upcoming refueling outage. A review of the requirements at all given times during an outage is required for emergency core cooling systems. Additional systems that require this type of review are:

A. Plant service water and fuel pool cooling systems

B. Shutdown cooling and fuel handling area ventilation systems

C. Fuel transfer and fuel pool cooling systems

D. Fuel pool cooling and shutdown cooling systems

Answer: D

Explanation:

A. This is incorrect because one of the systems listed is not part of those stated in Procedure 01-S-18-6, Section 6.4.2.

B. See Explanation for Answer A.

C. See Explanation for Answer A.

D. This is correct based on additional risk management requirements stated in Procedure 01-S-18-6, Section 6.4.2. Three system classifications must be include a review of requirements at any given time during the outage [Sub-item b)].

Technical References:

GLP-SM-MRP, Procedure 01-S-18-6

References to be provided to applicants during exam:

None

Learning Objective: To be determined

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:	55.41.10	

Examination Outline Cross-Reference	Level	RO
Ability to determine operability and/or availability of safety related equipment. (CFR: 41.7 / 43.5 / 45.12)	**Tier #**	Admin
	Group #	
	K/A #	Generic - 2.2.37
	Rating	3.6

Question 70

In accordance with the Technical Specifications, the minimum ADS accumulator supply pressure for an operable ADS is:

A. 140 psig

B. 150 psig

C. 160 psig

D. 170 psig

Answer: A

Explanation:

A. There is one correct value for the ADS accumulator pressure settings, which is stated in Answer B.

B. This is correct because the applicable SR associated with ADS (LCO 3.5.1, SR 3.5.1.3) directs the plant to verify that the ADS accumulator supply pressure is equal to or greater than 150 psig.

C. See Explanation to Answer B.

D. See Explanation to Answer B.

Technical References:

Technical Specifications, LCO 3.5.1; Bases Discussion for LCO 3.5.1; GLP-OPS-E2202

References to be provided to applicants during exam:

None

Learning Objective: To be determined

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:	55.41.7, 55.43.5, 55.45.12	

Examination Outline Cross-Reference	Level	RO
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. (CFR: 41.5 / 43.5 / 45.12)	Tier #	Admin
	Group #	
	K/A #	Generic - 2.2.44
	Rating	3.6

Question 71

The reactor is in MODE 1 with both Reactor Feedpumps (RFPTs) in operation. One of the control room operators has the TEST LOCKOUT pushbutton on P-680 depressed as part of an over-speed test of RFPT A. During this time the RFPT A experiences an actual over-speed condition.

In this situation, RFPT A would:

A. Be tripped by the electrical trip solenoid ONLY.

B. Be tripped by the mechanical trip device ONLY.

C. Be tripped by both the electrical trip solenoid and the mechanical trip device.

D. Not trip until the TEST LOCKOUT pushbutton is released.

Answer: A

Explanation:

A. This answer is correct because the test blocks the mechanical device trip but leaves the electrical trip system active and there for only the electrical trip is active during the test.

B. This answer is incorrect because the test blocks the mechanical device trip but leaves the electrical trip system active and there for only the electrical trip is active during the test.

C. This answer is incorrect because the test blocks the mechanical device trip but leaves the electrical trip system active and there for only the electrical trip is active during the test.

D. This answer is incorrect because the test blocks the mechanical device trip but leaves the electrical trip system active and there for only the electrical trip is active during the test.

Technical References:

GLP-OPS-N2100.09, Page 24 of 58.

References to be provided to applicants during exam:

None

Learning Objective: To be determined

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:	55.41.4	

Examination Outline Cross-Reference	Level	RO
Knowledge of radiation exposure limits under normal or emergency conditions. (CFR: 41.12 / 43.4 / 45.10)	**Tier #**	Admin
	Group #	
	K/A #	Generic - 2.3.4
	Rating	3.2

Question 72

Preparations are being made for valve maintenance in a High Radiation Area (HRA). In evaluating the radiological considerations, the following assumptions are made.

- The dose rate near the valve is estimated to be 100 mRem/hour
- The individual proposed to complete the task has negligible dose received in the given year

For scoping purposes, what is the maximum time the individual work on the valve without needing approval of the associated RWP by the ALARA Manager's Committee?

A. 29 hours

B. 49 hours

C. 59 hours

D. 99 hours

Answer: B

Explanation:

A. This is incorrect based on the guidelines stated in the Explanation for Answer B.

B. This is correct because it is the maximum number of hours that could be worked on the proposed job without exceeding an individual's maximum annual dose of 5 Rem. Section 5.3 of procedure EN-RP-105 requires that an estimate of the Person-Rem expected for a given job be calculated (Step [8]). To address ALARA concerns, the proposed job has to be approved by the ALARA Manager's Committee if the estimate equals to or exceeds 5 Person-Rem (Step [12]). Dividing 5 Person-Rem by an estimated dose rate of 0.1 Rem/Hour yields a maximum amount of time of 50 hours.

C. This is incorrect based on the guidelines stated in the Explanation for Answer B.
D. This is incorrect based on the guidelines stated in the Explanation for Answer B.

Technical References:

Procedure EN-RP-105

References to be provided to applicants during exam:

None

Learning Objective: To be determined

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:	55.41.12, 55.43.4, 55.45.10	

Examination Outline Cross-Reference	Level	RO
Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities. (CFR: 41.12 / 43.4 / 45.10)	Tier #	Admin
	Group #	
	K/A #	Generic - 2.3.14
	Rating	3.4

Question 73

Compared to a typical sample, a coolant sample indicative of a fuel leak would contain higher than normal amounts of:

A. Cl-38

B. Pu-239

C. Mn-56

D. Co-60

Answer: B

Explanation:

A. This is incorrect because this is a coolant activation product with a half-life of 37 minutes. Finding coolant activation products in a coolant sample is expected during normal plant operations.

B. This is correct because the isotope is a transuranic. Except in trace quantities, these should not be found in the reactor coolant unless fuel failures are present.

C. This is incorrect because it is an activated corrosion product that is typically found in a coolant sample during normal plant operations.

D. See Explanation for Answer C.

Technical References:

GLP-OPS-MCD06

References to be provided to applicants during exam:

None

Learning Objective: To be determined

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:	55.41.12, 55.43.4, 55.45.10	

Examination Outline Cross-Reference	Level	RO
Knowledge of crew roles and responsibilities during EOP usage. (CFR: 41.10 / 45.12)	**Tier #**	Admin
	Group #	
	K/A #	Generic - 2.4.13
	Rating	4.0

Question 74

The Reactor Operator has the following indications:

- Drywell Pressure 1.25 psig
- Reactor Vessel Level at 11"
- No Full In position indications lit on the Rod Display Module

The actions the Reactor Operator will take will be:

A. Notify the Control Room Supervisor of the indications and await direction

B. Notify the Control Room Supervisor of the indications and recommend to insert a manual scram

C. Initiate a manual scram and announce action to Control Room Supervisor

D. Initiate emergency troubleshooting to determine any problems with the Rod Display Module and announce action to Control Room Supervisor

Answer: C

Explanation:

A. This is incorrect because this situation fits the conditions requiring a manual reactor scram by licensed operators. The Reactor Operator needs to take action to scram the reactor.

B. This is incorrect because this situation fits the conditions requiring a manual reactor scram by licensed operators. The Reactor Operator needs to take action to scram the reactor. Making recommendations will not suffice in the situation.

C. This is correct because the situation involves a situation that requires a manual scram of the reactor immediately. Procedure EN-OP-115, Section 5.4, Step [24] says that emergency reactivity changes require authorization of the SRO (Control Room Supervisor) with Control Room Command and Control, unless an immediate manual scram is required by Subsection 5.1. One of the conditions in Subsection 5.1 that requires an immediate manual reactor scram is when operating parameters exceed any of the reactor protection set points and an automatic shutdown does not occur. With the given indications, two RPS scram set points are exceeded (Reactor Low Level of 11.4" and High Drywell Pressure of 1.23 psig).

D. This is incorrect because the immediate action delays the most conservative action, which is to ensure that the rods are inserted in the reactor.

Technical References:

GFIG-OPS-C1102, GLP-OPS-C1102, GLP-OPS-C7100, Procedure EN-OP-115

References to be provided to applicants during exam:

None

Learning Objective: To be determined

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:	55.41.10, 55.45.12	

Examination Outline Cross-Reference	Level	RO
Knowledge of the parameters and logic used to assess the status of safety functions, such as reactivity control, core cooling and heat removal, reactor coolant system integrity, containment conditions, radioactivity release control, etc. (CFR: 41.7 / 43.5 / 45.12)	**Tier #**	3
	Group #	
	K/A #	2.4.21
	Rating	4.0

Question 75

The reactor is operating in MODE 1 when a RECIRC PMP B SEAL STG FLO HI/LO alarm is received. The reactor operator notes the following indications for Reactor Recirc Pump B:

- #1 Seal cavity pressure is 500 psig
- #2 Seal cavity pressure is 970 psig

Based on these indications, Reactor Recirc Pump B has a degraded;

A. #1 seal.

B. #2 seal.

C. Seal staging flow orifice.

D. Break down bushing.

Answer: A

Explanation:

A. This is the correct answer because as the seal degrades, it's pressure drop goes down. This pressure drop is picked up by the other seal and the staging flow increases.

B. This answer is incorrect because as an RCP seal degrades, it's pressure drop goes down and in this case it has gone up indicating the other seal is degrading.

C. A degraded seal orifice may result in changing seal flow rates but will not affect the distribution of pressure drops across the pump seals.

D. A degraded break down bushing may result in changing seal flow rates but will not affect the distribution of pressure drops across the pump seals.

Technical References:

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	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.10, 55.43.5, 55.45.13	

Examination Outline Cross-Reference	Level	SRO
Ability to determine and/or interpret the jet pump operability as it applies to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION (CFR: 41.10 / 43.5 / 45.13).	Tier #	1
	Group #	1
	K/A #	295001 AA2.05
	Rating	3.4

Question 76

With the plant operating at 70% power, the on-coming CRS reviews the operating logs taken at 0700 and notes the following:

- Recirculation Loop A total jet pump flow is 36 Mlb/hr
- Recirculation Loop B total jet pump flow is 36 Mlb/hr
- Recirculation Pump Loop A flow is 14 Mlb/hr
- Recirculation Pump Loop B flow is 13 Mlb/hr
- Recirculation loop Flow Control Valves are in their normal position for 70% power operation and the expected Recirculation Loop flow for these conditions is 12 Mlb/hr

Based on this information, the on-coming CRS should ensure:

A. One recirculation loop is shutdown before 0900.

B. The plant is in Mode 3 no later than 1900.

C. The plant is in Mode 2 no later than 1400.per LCO 3.0.3.

D. A special test be performed per LCO 3.0.7.

Answer: B

Explanation:

A. This answer is incorrect because this LCO is only applicable if there is a jet pump flow mismatch per TS 3.4.1. In this case the flows are matched so there is no entry into this LCO.

B. This answer is correct because per TS Surveillance 3.4.3.1, since criteria a and b are not met, TS LCO 3.4.3 Condition A would be entered. This LCO Condition requires the unit to be in Mode 3 within 12 hours.

C. This answer is incorrect because there is no condition that requires entering LCO 3.0.3.

D. This answer is incorrect because it only applies to LCOs that are associated with special operations in section 3.10 of TS.

Technical References:

LGGNS Technical Specification LCOs 3.4.1 and 3.4.3

References to be provided to applicants during exam:

GGNS Technical Specifications 3.4.1 and 3.4.3

Learning Objective: GLP-OPS-

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:	55.41.10, 55.43.2, 55.45.13	

Examination Outline Cross-Reference	Level	SRO
Control Room Abandonment - Knowledge of abnormal condition procedures (CFR: 41.10 / 43.5 / 45.13).	Tier #	1
	Group #	1
	K/A #	295016 G – 2.4.11
	Rating	4.2

Question 77

An armed attack resulted in abandoning the control room at 0930 and implementing procedure ONEP 05-1-02-II-1, Shutdown from the Remote Shutdown Panel. It is now 0950 and the crew has been unable to get to the Remote Shutdown Panel.

Based on this sequence of events, the CRS should declare a(n):

A. Unusual Event

B. Alert

C. Site Area Emergency

D. General Emergency

Answer: C

Explanation:

A. This answer is incorrect based on the Explanation for Answer C.

B. This answer is incorrect based on the Explanation for Answer C.

C. This answer is correct because a Control Room evacuation with no plant control established within 15 minutes results in a Site Area Emergency per Emergency Plan Procedure 10-S-01-1, Attachment I, Emergency Classification HS3. There is a declaration of an Alert for the Security Event (Procedure 10-S-01-1, Attachment I, Emergency Classification HA1b), but the higher of the two classification levels is dominant.

D. This answer is incorrect based on the Explanation for Answer C.

Technical References:

Lesson Plan GLP-OPS-C6100, Procedure 05-1-02-II-1, ONEP 10-S-01-1

References to be provided to applicants during exam:

Procedure 10-S-01-1, Attachment I

Learning Objective: GLP-OPS-

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:	55.41.10, 55.43.5, 55.45.13	

Examination Outline Cross-Reference	Level	SRO
High Reactor Pressure - Ability to interpret and execute procedure steps (CFR: 41.10 / 43.5 / 45.12).	Tier #	1
	Group #	1
	K/A #	295025 G – 2.1.20
	Rating	4.6

Question 78

The reactor plant was operating at 100% power when a plant transient occurred. Currently;

- The MSIVs are closed
- All control rods are fully inserted
- RPV water level cannot be determined
- RCIC and HPCS are unavailable
- All radiation levels are normal
- Suppression Pool level is 10 feet and stable
- RPV Pressure is 1070 psig with one SRV cycling
- Containment pressure is 2.0 psig and stable

Per EP-2, RPV Control, the CRS should:

A. Transition to Steam Cooling

B. Control RPV pressure with SRVs, RCIC or RWCU

C. Transition to Emergency RPV Depressurization

D. Implement Attachment 9, Defeating All MSIVs and MSL Drains Isolation Interlocks, and depressurize using the main turbine bypass valves

Answer: D

Explanation:

A. This answer is incorrect because it is required for pressure control if RPV level drops below the top of active fuel and no injection source is available (see Page 28, Attachment IV, Emergency Procedure 02-S-01-40). Injection sources are available so these conditions are not met.

B. This answer is incorrect because SRVs can not be used for RPV pressure control due to low SP level.

C. This answer is incorrect because there is no ATWS, injection sources are available, and there is a low SP level.

D. This answer is correct because RPV pressure can be controlled based on the indication of a full reactor scram. With level unknown and SP level below 10.5 feet, the RPV must be depressurized using systems other than the SRVs. To do this Attachment 9 must be implemented to reopen the MSIVs.

Technical References:

Emergency Procedure 05-S-01-EP-2, Emergency Procedure 02-S-01-40

References to be provided to applicants during exam:

None

Learning Objective: GLP-OPS-

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:	55.43.5	

Examination Outline Cross-Reference	Level	SRO
Ability to determine and/or interpret reactor pressure as it applies to SUPPRESSION POOL HIGH WATER TEMPERATURE (CFR: 41.10 / 43.5 / 45.13).	Tier #	1
	Group #	1
	K/A #	295026 EA2.03
	Rating	4.0

Question 79

The plant is operating at 100 percent power with a stuck open SRV. Reactor pressure is 1020 psig and suppression pool temperature is now 112 degrees and rising.

Assuming all systems with the exception of the stuck open SRV operate as designed, which of the following is the appropriate procedure path used in this situation?

Enter Procedure:

A. EP-3, Primary Containment Control, which requires entry into EP-2, RPV Control, which requires entry into 05-1-02-I-1, Reactor Scram

B. EP-3, Primary Containment Control, which requires entry into EP-2, RPV Control, which requires an RPV Emergency Depressurization.

C. EP-2, RPV Control, which requires entry into EP-3, Primary Containment Control, which requires an RPV Emergency Depressurization.

D. EP-2, RPV Control, which requires entry into EP-3, Primary Containment Control, which requires depressurizing within cooldown limits.

Answer: A

Explanation:

A. This answer is correct because a suppression pool temperature above 95 degrees requires entering EP-3. EP-3 directs entering EP-2 when SP temperature cannot be maintained below 110 degrees. EP-2 then directs entry into the Scram ONEP.

B. This answer is incorrect because EP-3 does not require an ED unless the HCTL cannot be maintained. In this situation the HCTL will not be exceeded due to the stuck open SRV depressurizing the RPV following the reactor scram.

C. This answer is incorrect because there is no direct entry condition to EP-2 based on SP temperature.

D. This answer is incorrect because there is no direct entry condition to EP-2 based on SP temperature.

Technical References:

Procedure 05-S-01-SAP-01, Procedure 05-S-01-EP-01, Procedure 05-S-01-EP-02, Procedure 05-S-01-EP-03

References to be provided to applicants during exam:

EOP Figure 1, Heat Capacity Temperature Limit

Learning Objective: GLP-OPS-

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:	55.41.10, 55.43.5, 55.45.13	

Examination Outline Cross-Reference	Level	SRO
High Drywell Temperature - Knowledge of EOP mitigation strategies (CFR: 41.10 / 43.5 / 45.13).	**Tier #**	1
	Group #	1
	K/A #	295028 G – 2.4.6
	Rating	4.7

Question 80

Following a transient the control room crew is implementing EP-2, RPV Control, and EP-3, Containment Control. Currently;

- Drywell pressure is 1.40 psig
- Drywell temperature is 320^{0}F and rising
- Containment pressure is 1.0 psig and stable
- Reactor power is 6 percent and slowly trending down
- Actions have been taken to maximize drywell cooling
- Suppression pool temperature is 93^{0}F and stable
- Suppression pool level is 19' 8" and slowly rising
- MSIV's are open

Based on these conditions, the Control Room Supervisor should rapidly depressurize the RPV by implementing:

A. EP-2 and opening 8 SRVs.

B. EP-2 and opening the main turbine bypass valves

C. EP-2A and opening 8 SRVs

D. EP-2A and opening the main turbine bypass valves

Answer: C

Explanation:

A. This answer is incorrect because EP-2 is for a non-ATWS transient

B. This answer is incorrect because EP-2 is for a non-ATWS transient

C. This answer is correct because with drywell temperature approaching the 330 degree limit, an emergency depressurization is required. EP-3 then implements, EP-2, and then EP-2A. EP-2A requires an ED using the 8 ADS/SRVs.

D. This answer is incorrect because EP-2A requires 8 ADS/SRVs be opened.

Technical References:

Emergency Procedures 05-S-01-EP-1, 05-S-01-EP-2, and 05-S-01-EP-3

References to be provided to applicants during exam:

None

Learning Objective: GLP-OPS-

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:	55.41.10, 55.43.5, 55.45.13	

Examination Outline Cross-Reference	Level	SRO
Ability to determine and/or interpret adequate core cooling as it applies to REACTOR LOW WATER LEVEL (CFR: 41.10 / 43.5 / 45.13).	Tier #	1
	Group #	1
	K/A #	295031 EA2.04
	Rating	4.8

Question 81

Following a LOCA, all high pressure injection capability has been lost and RPV level cannot be determined. Currently;

- RPV pressure is 750 psig
- All control rods are fully inserted
- Off-Site power has been lost
- There are no indications of fuel damage

Based on these conditions, the CRS should be implementing the:

A. Level/Pressure Control leg of EP-2

B. RPV Flooding leg of EP-2.

C. Alternate Level Control leg of EP-2.

D. Steam Cooling leg of EP-2.

Answer: B

Explanation:

A. This answer is incorrect. With RPV level unknown, RPV Flooding is required. This answer would be correct if RPV level were known.

B. This answer is correct. With RPV level unknown, RPV Flooding is required.

C. This answer is incorrect. With RPV level unknown, RPV Flooding is required. This answer would be correct if RPV level were known and level could not be restored maintained greater than -180 inches.

D. This answer is incorrect. With RPV level unknown, RPV Flooding is required.

Technical References:

Procedure 05-S-1-EP-1, Procedure 05-S-1-EP-2

References to be provided to applicants during exam:

None

Learning Objective: GLP-OPS-

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:	55.41.10, 55.43.5, 55.45.13	

Examination Outline Cross-Reference	Level	SRO
Plant Fire On Site - Knowledge of the emergency plan. (CFR: 43.5)	Tier #	1
	Group #	1
	K/A #	600000 G – 2.4.29
	Rating	3.9

Question 82

With the reactor in MODE 1, an Alert would have to be declared if:

A. A fire in Reactor Protection System Motor Generator A took 30 minutes to extinguish and resulted in a loss of RPS A.

B. A generator hydrogen fire burned for 60 minutes and required the unit be taken off line.

C. A barrel of oily rags burned for 25 minutes in the Division I Emergency Diesel Generator room and the fire damaged the EDG control panel.

D. A fire in the maintenance welding shop burned for 20 minutes and resulted in 4 injured mechanics being transported to the hospital.

Answer: C

Explanation:

A. This answer is incorrect because RPS is not a safety system required for safe shutdown or maintenance of a safe shutdown. Therefore this would be an unusual event.

B. This answer is incorrect because the generator is not a safety system required for safe shutdown or maintenance of a safe shutdown. Therefore this would be an unusual event.

C. This answer is correct because there was a fire that lasted greater than 15 minutes and damaged a safety system required for safe shutdown of the reactor.

D. This answer is incorrect because of the maintenance shop is not a structure identified as applicable for an Alert classification and the 4 injuries do not affect the emergency plan declaration. Therefore, this would be an unusual event.

Technical References:

Emergency Plan Procedure 10-S-01-1

References to be provided to applicants during exam:

EPP 01-02, EP Flowchart

Learning Objective: GLP-OPS-

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:	55.41.6, 55.45.4	

Examination Outline Cross-Reference	Level	SRO
Ability to determine and/or interpret the drywell temperature as it applies to HIGH DRYWELL TEMPERATURE (CFR: 41.10 / 43.5 / 45.13).	Tier #	1
	Group #	2
	K/A #	295012 AA2.01
	Rating	3.9

Question 83

The plant is in MODE 1 and the reactor operator just discovered the average drywell temperature is 140 degrees and slowly rising. According to the Technical Specification Bases document, if this condition is not corrected then a design basis LOCA could result in the design:

A. heat capacity of the suppression pool being exceeded.

B. drywell temperature being exceeded.

C. fuel zone level instrumentation temperature being exceeded.

D. limit of 2 percent hydrogen in the drywell being exceeded.

Answer: B

Explanation:

A. This answer is incorrect because the Technical Specification Bases document for high drywell temperature states the design drywell temperature of 330 degrees cannot be ensured following a design basis LOCA if drywell temperature is not maintained below 135 degrees.

B. This answer is correct because the Technical Specification Bases document for high drywell temperature states the design drywell temperature of 330 degrees cannot be ensured following a design basis LOCA if drywell temperature is not maintained below 135 degrees.

C. This answer is incorrect because the Technical Specification Bases document for high drywell temperature states the design drywell temperature of 330 degrees cannot be ensured following a design basis LOCA if drywell temperature is not maintained below 135 degrees.

D. This answer is incorrect because the Technical Specification Bases document for high drywell temperature states the design drywell temperature of 330 degrees cannot be ensured following a design basis LOCA if drywell temperature is not maintained below 135 degrees.

Technical References:

GGNS Technical Specifications Bases for technical specification 3.6.5.5.

References to be provided to applicants during exam:

None

Learning Objective: GLP-OPS-

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:	55.43.2	

Examination Outline Cross-Reference	Level	SRO
Ability to determine and/or interpret reactor pressure as it applies to HIGH SUPPRESSION POOL WATER LEVEL (CFR: 41.10 / 43.5 / 45.13).	Tier #	1
	Group #	2
	K/A #	295029 EA2.02
	Rating	3.6

Question 84

The plant is in MODE 1 and reactor pressure is 1020 psig. A non-isolatable leak has resulted in suppression pool level being above the upper limit established by the Technical Specifications and is still slowly increasing.

According to the Technical Specification Bases document for Technical Specification 3.6.2.2, Suppression Pool Water Level, with reactor pressure 1020 psig and suppression pool level above the limit:

A. The peak drywell design pressure may be exceeded during a design basis LOCA.

B. An inadvertent upper pool dump could overflow the weir wall into the drywell.

C. RCIC may trip on high exhaust back-pressure

D. The peak containment design pressure may be exceeded during a design basis LOCA.

Answer: B

Explanation:

A. This answer is incorrect because the bases document does not state the drywell design pressure could be exceeded with a high suppression pool level but is credible because drywell pressure would be higher given a DBA LOCA and a higher SP water level.

B. This answer is correct because the referenced bases document states the upper limit is based on not overflowing the weir wall given an inadvertent upper pool to lower pool dump.

C. This answer is incorrect because the bases document does not state the RCIC turbine may trip with a high suppression pool level but is credible because with a higher SP water level, RCIC back pressure would be higher.

D. This answer is incorrect because the bases document states the containment design pressure would not be exceeded with a high suppression pool level.

Technical References:

Technical Specification Bases for TS 3.6.2.2

References to be provided to applicants during exam:

None

Learning Objective: GLP-OPS-

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:	55.43.2	

Examination Outline Cross-Reference	Level	SRO
High Secondary Containment Area Radiation Levels - Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc. (CFR: 41.12 / 43.4 / 45.9).	Tier #	1
	Group #	2
	K/A #	295033 G 2.3.15
	Rating	3.1

Question 85

The plant is in MODE 1 and while the Control Room Supervisor is reviewing the logs, it is discovered at 0700 on 3/1/2009 the Fuel Handling Area Ventilation radiation monitor D17-K617A has not been channel checked for 24 hours.

Based on this discovery and Technical Specification 3.3.6.2, the CRS is required to:

A. Immediately declare D17-K617A INOPERABLE and enter LCO 3.3.6.2

B. Perform the surveillance no later than 1900 on 3/1/2009

C. Perform the surveillance no later than 0700 on 3/2/2009.

D. Immediately declare D17-K617A INOPERABLE and enter LCO 3.3.6.2 and Perform the surveillance no later than 1900 on 3/1/2009

Answer: C

Explanation:

A. This answer is incorrect because TS 3.0.3 states if a surveillance is missed then immediate entry into the LCO is not required.

B. This answer is incorrect because TS 3.0.3 states the surveillance must be performed within the periodicity or 24 hours; whichever is longer.

C. This answer is correct because TS allows up to 24 hours to complete a missed surveillance.

D. This answer is incorrect because TS 3.0.3 states if a surveillance is missed then immediate entry into the LCO is not required.

Technical References:

Lesson Plan GLP-OPS-D1721 (lesson plan and figures), Lesson Plan GLP-OPS-T4801, GGNS Technical Specifications Section 3.3.6.2 (LCO and Bases)

References to be provided to applicants during exam:

GGNS Technical Specifications

Learning Objective: GLP-OPS-

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:	55.41.12, 55.43.4, 55.45.9	

Examination Outline Cross-Reference	Level	SRO
Shutdown Cooling - Knowledge of procedures and limitations involved in core alterations. (CFR: 41.10 / 43.6 / 45.7).	**Tier #**	2
	Group #	1
	K/A #	205000 G 2.1.36
	Rating	4.1

Question 86

The plant is in the following conditions:

- The reactor is shut down, with the vessel head removed
- The water level is 25 feet above the RPV flange
- EDG 11 is tagged out for repairs
- Irradiated fuel is being moved within the reactor vessel
- One qualified circuit between the off-site transmission network and Divisions 1, 2, and 3 electrical power distribution systems are operable
- RHR B is in operation
- ADHRS is OPERABLE

If the RHR B pump trips, then the immediate (≤ 1 hour) required actions per Technical Specification 3.9.8 are:

A. 3.9.8.A.1 ONLY.

B. 3.9.8.C.1 ONLY.

C. 3.9.8.C.1 and 3.9.8.C.2 ONLY.

D. 3.9.8.A.1 and 3.9.8.C.1 and 3.9.8.C.2.

Answer: D

Explanation:

A. This answer is incorrect because it list only part of the required actions, implying that one RHR cooling subsystem is in operation when it is not.

B. This answer is incorrect because it assumes RHR A is OPERABLE and it is not (per the bases document). Also, there are two actions for this condition that are tied by an AND logical connector making both actions required.

C. This answer is incorrect because it assumes RHR A is OPERABLE and it is not (per the bases document).

D. This answer is correct because it addresses the conditions of no operable RHR pumps and no decay heat removal subsystem in operation.

Technical References:

Procedure 05-1-02-III-1; Lesson Plans GLP-OPS-P7500, GLP-OPS-R2100; GGNS Technical Specifications LCOs 3.9.8 and 3.9.9

References to be provided to applicants during exam:

GGNS Technical Specifications 3.9.8

Learning Objective: GLP-OPS-

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:	55.43.2	

Examination Outline Cross-Reference	Level	SRO
Source Range Monitor – Ability to diagnose and recognize trends in a timely manner utilizing the appropriate control room reference material. (CFR: 41.10 / 43.5 / 45.12).	Tier #	2
	Group #	1
	K/A #	215004 G 2.4.47
	Rating	4.0

Question 87

The reactor is in MODE 2, a reactor startup is in progress, and reactor power is currently 2×10^2 CPS. At 0800, SRM A was declared inoperable and at 0900, all the SRMs were declared inoperable based on a common mode failure.

The required Technical Specification action(s) for this sequence of events is to:

A. Immediately place the Mode Switch in Shutdown.

B. Immediately suspend rod withdrawal.

C. Fully insert all control rods within one hour AND then place the Mode Switch in Shutdown.

D. enter LCO 3.0.3 and be in MODE 3 not later than 2200.

Answer: B

Explanation:

A. This answer is incorrect because TS 3.3.1.2 states all control rod withdrawal is to be suspended and to be in MODE 3 within 12 hours. This is a partially correct answer for MODES 3 or 4.

B. This answer is correct because TS 3.3.1.2 states all control rod withdrawal is to be suspended and to be in MODE 3 within 12 hours.

C. This answer is incorrect because TS 3.3.1.2 states all control rod withdrawal is to be suspended and to be in MODE 3 within 12 hours. This is the correct answer for MODES 3 or 4.

D. This answer is correct because TS 3.3.1.2 states all control rod withdrawal is to be suspended and to be in MODE 3 within 12 hours. This answer is credible because if the TS did not have an action for the inoperability of all 4 SRMs the correct action would be to enter 3.0.3.

Technical References:

GGNS Technical Specification 3.3.1.2

References to be provided to applicants during exam:

None

Learning Objective: GLP-OPS-

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:	55.43.2	

<table>
<tr><td>Examination Outline Cross-Reference</td><td>Level</td><td>SRO</td></tr>
<tr><td rowspan="5">PCIS/Nuclear Steam Supply Shutoff - Knowledge of less than or equal to one hour Technical Specification action statements for systems. (CFR: 41.7 / 41.10 / 43.2 / 45.13).</td><td>Tier #</td><td>2</td></tr>
<tr><td>Group #</td><td>1</td></tr>
<tr><td>K/A #</td><td>223002 G - 2.2.39</td></tr>
<tr><td>Rating</td><td>4.5</td></tr>
<tr><td></td><td></td></tr>
</table>

Question 88

While operating at 100% power, at 0700 on 8/1 the CRS is informed Containment/ Drywell ventilation exhaust radiation monitors D17-K609A and D17-K609B have failed Surveillance Requirement SR 3.3.6.1.1, Channel Check.

Assume no operator action is taken. Based on Technical Specification 3.3.6.1, Primary Containment and Drywell Isolation Instrumentation, the CRS is required to:

A. Isolate the affected penetration flowpath(s) no later than 0900 on 8/1.

B. Place both channels in trip no later than 0800 on 8/2.

C. Isolate the affected penetration flow path(s) no later than 0800 on 8/2.

D. Place one channel in trip no later than 0800 on 8/1.

Answer: A

Explanation:

A. This answer is correct because the isolation logic is 2-out-of-2 taken once logic and therefore, the isolation capability is lost. The CRS will have to restore isolation capability within 1 hour per TS 3.3.6.1.B or enter TS 3.3.6.1.C. TS 3.3.6.1.C requires the CRS immediately enter table 3.3.6.1-1 and per Condition 2.g isolate the affected penetrations within one hour. Therefore the penetrations must be isolated no later than 0900 on 8/1.

B. This answer is incorrect because condition A requires the affected channels be placed in trip within 24 hours, or no later than 0700 on 8/2.

C. This answer is incorrect because the penetrations need to isolated no later than 0900 on 8/1 per the discussion in A above..

D. This answer is incorrect because the applicable LCO (A) provides 24 hours to place the channel in a tripped condition.

Technical References:

Lesson Plan GLP-OPS-M7101 (and associated Figures), Lesson Plan GLP-OPS-D1721 (and associated Figures), GGNS Technical Specifications (LCO 3.3.6.1)

References to be provided to applicants during exam:

GGNS Technical Specification 3.3.6.1, Primary Containment and Drywell Isolation Instrumentation including applicable SRs and Table 3.3.6.1-1.

Learning Objective:

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:	55.41.7, 55.41.10, 55.43.2, 55.45.13	

Examination Outline Cross-Reference	Level	SRO
Ability to (a) predict the impacts of the following on the STANDBY GAS TREATMENT SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: High system pressure (plant-specific) (CFR: 41.5 / 45.6).	Tier #	2
	Group #	1
	K/A #	261000 A2.14
	Rating	3.2

Question 89

At 1200 on 6/1, the plant is operating at 100% power and the Standby Gas Treatment System (SGTS) Train A monthly surveillance 3.4.6.1 is started and at 1500, the SGTS FLTR TR A HEPA/CHAR ΔP HI alarm annunciates. While reviewing the surveillance closeout paperwork at 2300, the CRS discovers an error and then determines the SGTS flow rate does not meet the acceptance criteria contained in SR 3.4.6.1.

According to the Technical Specifications, the required action is to:

A. Declare SGTS A inoperable effective at 1500 and perform SR 3.4.6.1 on SGTS B no later than 0300 on 6/2.

B. Declare SGTS A inoperable effective at 2300 and restore SGTS subsystem operability by 2300 on 6/8.

C. Declare SGTS A inoperable effective at 1200 and restore SGTS subsystem operability by 1200 on 6/8.

D. Declare SGTS A inoperable effective at 2300 and restore SGTS subsystem operability by 1500 on 6/8.

Answer: B

Explanation:

A. While it may be prudent to test the other train, this answer is incorrect because there is no TS requirement to perform an SR on the other train.

B. This answer is correct because the entry time into an LCO is based on the time of discovery. Since this was at 2300, the inoperablility time would be 2300 and the TS requires it be returned to an operable status within 7 days.

C. This answer is incorrect because the entry time into an LCO is based on the time of discovery. While the surveillance was started at 1200, the inoperability was not discovered until 2300 by the CRS. Therefore, the LCO entry time would be 2300.

D. This answer is incorrect because the entry time into an LCO is based on the time of discovery and not the time the condition actually occured. While the condition of inoperability probably occurred around 1500, the inoperability was not discovered until 2300 by the CRS. Therefore, the LCO entry time would be 2300.

Technical References:

Lesson Plan GLP-OPS-T4801, GGNS Technical Specifications Section 3.6.4.3

References to be provided to applicants during exam:

GGNS Technical Specification 3.6.4.3

Learning Objective: GLP-OPS-

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:	55.41.5, 55.45.6	

Examination Outline Cross-Reference	Level	SRO
Ability to (a) predict the impacts of Grounds on the D.C. ELECTRICAL DISTRIBUTION; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations (CFR: 41.5 / 45.6).	Tier #	2
	Group #	1
	K/A #	263000 A2.01
	Rating	3.2

Question 90

The unit is in MODE 1 when it is discovered there are ground faults on the Division 1 battery due to faulty parts installed during a recent modification. Subsequently, at 0800 on 3/1/2009, the Division 1 DC power subsystem was declared inoperable. At 1200 on 3/1/2009, the Division 2 DC power subsystem was also declared inoperable for the same reason.

Then at 1600 on 3/1/2009, the Division 1 DC power subsystem ground faults are corrected and it is declared operable. However there are no remaining replacement parts in stock for Division 2.

Based on the current status, if the Division 2 DC power subsystem is not returned to an operable status the reactor must be in MODE 3 no later than:

A. 2200 on 3/1/2009.

B. 0100 on 3/2/2009.

C. 0200 on 3/2/2009

D. 0400 on 3/2/2009.

Answer: A

Explanation:

A. This answer is correct because with one division inoperable condition 3.8.4.C is entered. When the second division is declared inoperable at 1200, 3.0.3 is entered. When Division 1 is declared operable, 3.0.3 is exited leaving 3.8.4.C in effect. At 1000 3.8.4.E is entered requiring the unit to be in MODE 3 in 12 hours or at 2200. (See example 1.3-2).

B. This answer is incorrect based on the discussion for distractor A. It would be correct if the time was based on entering 3.0.3 at 1000.

C. This answer is incorrect based on the discussion for distractor A. This would be correct if based on the second train being inoperable.

D. This answer is incorrect based on the discussion for distractor A. This would be correct if based on entering 3.0.3 at 1200.

Technical References:

Lesson Plan GLP-OPS-L1100, GFIG-OPS-L1100, GGNS Technical Specifications LCO 3.8.4, 3.8.6, 3.8.7, and 3.5.1

References to be provided to applicants during exam:

GGNS Technical Specification 3.8.4

Learning Objective: GLP-OPS-

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:	55.41.5, 55.45.6	

Examination Outline Cross-Reference	Level	SRO
Ability to (a) predict the impacts of the following on the RECIRCULATION FLOW CONTROL SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Recirculation pump speed mismatch between loops. (CFR 41.5 / 45.6) (A2.04)	Tier #	2
	Group #	2
	K/A #	202002 A2.04
	Rating	3.2

Question 91

The plant was at 80 percent power when Reactor Recirc Pump A unexpectedly downshifted to slow speed. The cause of the downshift was an I&C technician improperly installing a jumper and the error has been corrected. The reactor is now operating in the Restricted Region of the Power-to-Flow Map.

Based on this, the CRS should enter Procedure:

A. 05-1-02-III-3, Reduction in Recirc System Flow Rate, and upshift Reactor Recirc Pump A to exit the restricted region of the power to flow map.

B. 05-1-02-III-3, Reduction in Recirc System Flow Rate, and monitor for power oscillations.

C. 04-1-01-b33-1, Reactor Recirc System, and upshift Reactor Recirc Pump A to exit the restricted region of the power to flow map

D. EP-2, RPV Control, and place the Mode switch in SHUTDOWN to preclude excessive power oscillations.

Answer: B

Explanation:

A. This is incorrect because the referenced procedure contains a caution not to upshift the reactor recirc pump.

B. This is correct because the downshift of the reactor recirc pump requires entering this procedure. The procedure states to monitor power for power oscillations and place the MODE switch in SHUTDOWN if oscillations are observed.

C. This is incorrect because the crew should enter the abnormal procedure and the slow running recirc pump should not be upshifted.

D. This is incorrect unless there are power oscillations observed.

Technical References:

Procedure 05-1-02-III-3, Reduction in Recirc System Flow Rate
References to be provided to applicants during exam:

None

Learning Objective: GLP-OPS-

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:	55.43.5	

Examination Outline Cross-Reference	Level	SRO
Ability to (a) predict the impacts of the Compressor trips (loss of air) (plant-specific) on the PRIMARY CONTAINMENT SYSTEM AND AUXILIARIES ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5 / 45.6) (A2.08)	**Tier #**	2
	Group #	2
	K/A #	223001 K3.02
	Rating	3.1

Question 92

The running instrument air compressor has tripped and the standby instrument and service air compressors failed to start. While implementing Off-Normal Event Procedure (ONEP) 05-1-02-V-9, Loss of Instrument Air, one control rod begins drifting into the core and it is now apparent the main turbine will soon trip on low condenser vacuum.

Based on these conditions, the Control Room Supervisor should now enter;

A. ONEP 05-1-02-IV-1, Control Rod/Drive Malfunctions, ONLY, and scram the reactor based on the drifting control rod.

B. ONEP 05-1-02-V-8, Loss of Condenser Vacuum, ONLY, and perform a fast shutdown based on the imminent turbine trip.

C. ONEP 05-1-02-IV-1 AND ONEP 05-1-02-V-8, then perform a fast shutdown based on the drifting control rod.

D. ONEP 05-1-02-IV-1 AND ONEP 05-1-02-V-8, then scram the reactor based on the imminent turbine trip.

Answer: D

Explanation:

A. This incorrect because the SRO must also enter the loss of vacuum ONEP. Additionally, one drifting rod does not require a scram.

B. This incorrect because the SRO must also enter the control rod/drive ONEP. Additionally, an imminent turbine trip requires a reactor scram, not a fast shutdown.

C. This is incorrect – both procedures would be entered, however the drifting control rod does not required a scram.

D. This is correct – both procedures would be entered and the imminent turbine trip requires a scram.

Technical References:

Off-Normal Event Procedure (ONEP) 05-1-02-V-9, Loss of Instrument Air , ONEP 05-1-02-IV-1, Control Rod/Drive Malfunctions, and ONEP 05-1-02-V-8, Loss of Condenser Vacuum

References to be provided to applicants during exam:

None

Learning Objective: GLP-OPS-

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:	55.43.5	

Examination Outline Cross-Reference	Level	SRO
Knowledge of new and spent fuel movement procedures. (CFR: 41.10 / 43.7 / 45.13) (2.1.42)	**Tier #**	2
	Group #	2
	K/A #	233000 G2.1.42
	Rating	3.4

Question 93

A Caution/Limitation in Integrated Operating Instruction 01-S-02-3, Refueling, states the maximum allowed weight of a load traveling over Fuel Assemblies in the Spent Fuel or Upper Containment Storage Pool Racks is _______ pounds.

A. 1000

B. 1140

C. 1350

D. 1430

Answer:

Explanation:

A. This is incorrect; the referenced procedure states in caution/limitation 2.8 the maximum weight traveling over the Spent Fuel or Upper Containment Storage Pool is 1140 pounds.

B. This is correct per the discussion above.

C. This is incorrect; the referenced procedure states in caution/limitation 2.8 the maximum weight traveling over the Spent Fuel or Upper Containment Storage Pool is 1140 pounds.

D. This is incorrect; the referenced procedure states in caution/limitation 2.8 the maximum weight traveling over the Spent Fuel or Upper Containment Storage Pool is 1140 pounds.

Technical References:

Integrated Operating Instruction 01-S-02-3, Refueling

References to be provided to applicants during exam:

None

Learning Objective: GLP-OPS-

Question Source: (note changes; attach parent)	Bank #	
	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.43.7	

Examination Outline Cross-Reference	Level	SRO
Ability to use procedures related to shift staffing, such as minimum crew complement, overtime limitations, etc. (CFR: 41.10 / 43.5 / 45.12)	Tier #	3
	Group #	
	K/A #	G 2.1.5
	Rating	3.9

Question 94

With the reactor in Mode 4, Technical Specification 5.2.2, Unit Staff, requires which of the following, by position, as the minimum for unit staffing;

A. 2 non-licensed operators, 2 licensed reactor operators, 1 licensed senior reactor operators, and no health physicists.

B. 1 non-licensed operator, 1 licensed reactor operator, 1 licensed senior reactor operator, and 1 health physicists.

C. No non-licensed operators, 1 licensed reactor operator, 1 licensed senior reactor operator, and no health physicists.

D. No non-licensed operators, 1 licensed reactor operator, 1 licensed senior reactor operator, and 1 health physicists.

Answer: B

Explanation:

A. This is incorrect because with the reactor in Mode 4, there is fuel in the RPV. According to the referenced TS, a health physicist is required to be on-site.

B. According to the referenced TS, these are the minimum staffing requirements.

C. This is incorrect because with the reactor in Mode 4, there is fuel in the RPV. According to the referenced TS, a health physicist and one NLO are required to be on-site.

D. This is incorrect because with the reactor in Mode 4, there is fuel in the RPV. According to the referenced TS, an NLO is required to be on-site.

Technical References:

TS Technical Specification 5.2.1, Unit Staff

References to be provided to applicants during exam:

None

Learning Objective: GLP-OPS-

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:	55.43.2	

Examination Outline Cross-Reference	Level	SRO
Ability to perform specific system and integrated plant procedures during all modes of plant operation. (CFR: 41.10 / 43.5 / 45.2 / 45.6)	Tier #	3
	Group #	
	K/A #	G 2.1.23
	Rating	4.4

Question 95

Following a LOCA with limited fuel damage, the Shift Technical Advisor reports containment hydrogen concentration has reached 3%. Based on this;

A. EP-2, RPV Control, is exited and EP-3, Containment Control is continued.

B. EP-2, RPV Control, is exited, EP-5, RPV Flooding is entered, and EP-3, Containment Control, is continued.

C. EP-2, RPV Control, is continued, EP-3, Containment Control is exited, and the SAPs are entered.

D. All EPs are exited and all SAPs are entered.

Answer: D

Explanation:

A. This is incorrect because the EPs state if containment or drywell hydrogen concentration exceeds 2.9%, all the EPs are exited and all the SAGs are entered.

B. This is incorrect because the EPs state if containment or drywell hydrogen concentration exceeds 2.9%, all the EPs are exited and all the SAGs are entered

C. This is incorrect because the EPs state if containment or drywell hydrogen concentration exceeds 2.9%, all the EPs are exited and all the SAGs are entered

D. This is correct based on the discussion above.

Technical References:

EP-3, Containment Control, step H-1.

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:	55.43.5	

Examination Outline Cross-Reference	Level	SRO
Ability to explain and apply system limits are precautions. (CFR: 41.10 / 43.2 / 45.12)	Tier #	3
	Group #	
	K/A #	G 2.1.32
	Rating	4.0

Question 96

According to Technical Specification 3.6.4.3, two Standby Gas Treatment trains are required to be operable. One of the technical specification bases for this limit is to mitigate the radiological consequences of a:

A. Rod drop accident low in the power range.

B. Fuel handling accident involving recently irradiated fuel.

C. Stuck open SRV while at power.

D. Power excursion resulting in greater than 3 percent fuel damage.

Answer: B

Explanation:

A. This is incorrect because the event would result in fuel damage and the release of fission products SGT would not mitigate the radiological release unless there is leakage of fission products into the secondary containment.

B. This is correct per TS Basis for TS 3.6.4.3. The basis is for a DBA LOCA and a Fuel handling accident involving recently irradiated fuel.

C. This is incorrect because the event would result in the release of reactor coolant into the suppression pool, SGT would not mitigate the radiological release.

D. This is incorrect because even though the event did result in fuel damage and the release of fission products, SGT would not mitigate the radiological release unless there is leakage of the fission products into the secondary containment.

Technical References:

Technical Specification Basis for TS 3.6.4.3, Standby Gas Treatment

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:	55.43.2	

Examination Outline Cross-Reference	Level	SRO
Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits. (CFR: 41.5 / 41.7 / 43.2)	Tier #	3
	Group #	
	K/A #	G 2.2.25
	Rating	4.2

Question 97

One of the Technical Specification safety limits is the requirement to maintain the Minimum Critical Power Ratio (MCPR) greater than or equal to 1.08 for two loop operation when greater than 25% power. The bases for this limit is to ensure:

A. 99.9% of the fuel does not experience transition boiling.

B. The peak cladding temperature limit per 10 CFR 50.46 is not exceeded after a LOCA.

C. The Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) is not exceeded.

D. That following a LOCA, decay heat does not exceed the design of the containment systems.

Answer: A

Explanation:

A. According to the referenced TS limit bases, the limit on MCPR is to prevent transition boiling for 99.9% of the fuel rods making A correct.

B. This is the bases for the MAPLHGR limit. While MAPLHGR and MCPR are related, the bases for each is different.

C. See A and B above.

D. This is one of the bases for the fuel design and operating temperatures and sizing of the containment systems.

Technical References:

Technical Specification bases document for LCO 2.1.1, Reactor Core Safety Limits

References to be provided to applicants during exam:

None

Learning Objective: GLP-OPS-

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:	55.43.	

Examination Outline Cross-Reference	Level	SRO
Ability to recognize system parameters that are entry-level conditions for Technical Specifications. (CFR: 41.7 / 41.10 / 43.2 / 43.3 / 45.3)	Tier #	3
	Group #	
	K/A #	G 2.2.42
	Rating	4.6

Question 98

The on-coming Shift Manager is performing a control room walkdown prior to taking the shift and notes the following;

- The reactor is at 80% power
- Reactor Recirc loop A flow is 35,000 gpm
- Reactor Recirc loop B flow is 33,700 gpm
- One ADS valve is tagged out with a red DO NOT OPERATE tag
- Suppression pool level is 18’ 5”
- Primary Containment air temperature is 92 degrees

Based on these observations the on-coming Shift Manager should verify an entry has been made into Technical Specification:

A. 3.6.1.5, Primary Containment Air Temperature

B. 3.4.1, Recirculation Loops Operating

C. 3.5.1, ECCS Operating

D. 3.6.2.2, Suppression Pool Water Level

Answer: C

Explanation:

A. This is incorrect because 92 degrees is within the TS limit of 95 degrees.

B. This is incorrect because both loops are operating and they are within 5% of each other as required by this TS.

C. This is correct because all 8 ADS valves are required to be operable in Mode 1.

D. This is incorrect because suppression pool level is within the TS required band.

Technical References:

3.6.1.5, Primary Containment Air Temperature, 3.4.1, Recirculation Loops Operating, 3.5.1, ECCS Operating, 3.6.2.2, Suppression Pool Water Level

References to be provided to applicants during exam:

None

Learning Objective: GLP-OPS-

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:	55.43.2	

Examination Outline Cross-Reference	Level	SRO
Ability to control radiation releases. (CFR: 41.11 / 43.4 / 45.10)	Tier #	3
	Group #	
	K/A #	G 2.3.11
	Rating	4.3

Question 99

According to procedure 01-S-08-11, Radioactive Discharge Controls, final release authorization of a Batch Liquid Radwaste Discharge is the responsibility of the:

A. Shift Foreman ONLY.

B. Shift Manager ONLY.

C. Shift Foreman and Chemistry Supervisor/Coordinator.

D. Shift Manager and Chemistry Supervisor/Coordinator.

Answer: B

Explanation:

A. Incorrect – The referenced procedure states this is the responsibility of the SM.

B. Correct – The referenced procedure states in section 2.1, step 6.4.2.d, that this is a SM responsibility. The final release signature is by the SM in section 4 of the batch release form.

C. Incorrect – The Chemistry Supervisor does have responsibilities for a batch release, but they do not include final release approval.

D. Incorrect – The Chemistry Supervisor does have responsibilities for a batch release, but they do not include final release approval.

Technical References:

Procedure 01-s-08-11, Radioactive Discharge Controls

References to be provided to applicants during exam:

None

Learning Objective: GLP-OPS-

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:	55.43.4	

Examination Outline Cross-Reference	Level	SRO
Knowledge of the lines of authority during implementation of the emergency plan. (CFR: 41.10 / 45.13)	Tier #	3
	Group #	
	K/A #	G 2.4.37
	Rating	4.1

Question 100

The site has entered a general emergency and consideration is being given to evacuating 5 sectors of the 10 mile Emergency Planning Zone (EPZ). The decision to evacuate these sectors will be made by the:

A. Emergency Director

B. Off-Site Emergency Coordinator

C. State Radiological Assessment Officer (CONFIRM TITLE)

D. Radiation Emergency Manager

Answer: C

Explanation:

A. This is incorrect because the ED is responsible for actions taken on site but only makes recommendations for off-site actions such as PARs.

B. This is incorrect because the ED is responsible for actions taken on site but only makes recommendations for off-site actions such as PARs.

C. This is correct because the responsibility for deciding to evacuate any EPZ is made by the state radiological assessment office.

D. This is incorrect because the REM is responsible for making recommendations to the ED.

Technical References:

Lesson Plan GLP-OPS-

References to be provided to applicants during exam:

None

Learning Objective: GLP-OPS-

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:	55.43.5	