RO 1

The reactor was operating at rated power and the Fraction of Core Boiling Boundary (FCBB) equal to 2.0.

Reactor Feed Pump A spuriously trips.

When reactor water level stabilizes from this transient, FCBB will be approximately:

a. 0
b. 1
c. 2
d. 4

ans: c

FCBB limits guard against Thermal Hydraulic Instability that could cause violation of the MCPR safety limit. FCBB is the power distribution limit which is the ratio of power being produced in the bottom 4 feet of the core to the power in the bottom 4 feet necessary to produce bulk boiling at 4 feet. FCBB is primarily affected by axial power shaping using control rods. If a reactor feed pump trips at 100% power, level will lower to less than 32 inches, so a reactor recirc flow control valve runback will occur, reducing core flow to about 50%. Power will stabilize around 70%. Answer **c** is correct since power in the lower 4 feet of the core lowers in approximately the same proportion as does core flow. Answer **d** is used as a distracter because 4 is a familiar number associated with FCBB, 4 ft being the GGNS boiling boundary above which THI should not occur. Answer **b** is plausible if only the reduction in core flow is considered. Answer **A** is plausible if a reactor scram is assumed to occur.

KA 295001 K1.03
Knowledge of operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE CIRCULATION: **thermal limits**

10CFR55.41(b)(5)

TS B3.2.4
05-1-02-III-3

GLP-OPS-ONEP Obj. 37.0
GLP-OPS-C5106 Obj. 27
GSMS-RO-EP001

Difficulty 2

RO 2

The plant is in Mode 1 with all electrical busses aligned to their normal supplies.
How will loss of Service Transformer 21 initially affect components associated with Drywell Cooling?

a. A drywell cooler fans and A drywell chillers never lose power

b. A drywell cooler fans and A drywell chillers will de-energize

c. B drywell cooler fans and B drywell chillers never lose power

d. B drywell cooler fans and B drywell chillers will de-energize

Ans: d

A drywell cooler fans are supplied from bus 15AA. A drywell chillers are supplied from bus 14AE. B drywell cooler fans are supplied from bus 16AB. B drywell chillers are supplied from bus 16AB. Service Transformer 21 normally supplies busses 11HD, 14AE, 16AB, and 17AC. Loss of Service Transformer 21 would cause loss of all listed drywell cooling components except A drywell cooling fans. Answer **d** is the only correct combination listed. Answer **a** incorrectly states A chillers would remain energized. Answer **b** incorrectly states A fans would lose power. Answer **c** incorrectly states B fans and chillers would remain energized. All incorrect answers are plausible since loss of either Service Transformer will affect some combination of drywell cooling fans and chillers.

KA 295003 K2.04
Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF AC POWER and the following: **AC electrical loads**

10CFR55.41(b)(4)

GLP-OPS-R2100 Obj. 4.4, 4d
GLP-OPS-M5100 Obj. 7.1, 10.1, 10.2

04-1-01-R21-14
04-1-01-R21-15
04-1-01-R21-16
04-1-01-M51-1
04-1-01-P72-1

Difficulty 2

RO 3

The plant is at 100% power.

DC bus 11DB de-energizes due to a major fault on the DC bus.

Then, a transient causes reactor water level to fall to -50 inches wide range.

Under these conditions, the scram air header is:

a. vented by the backup scram valves but not by ARI valves.

b. vented by the ARI valves but not backup scram valves.

c. vented by backup scram valves and ARI valves.

d. neither vented by backup scram valves nor ARI valves.

Ans: b

Backup scram valves and ARI valves are DC powered and energize to vent. Two solenoids, supplied from separate DC supplies, for both backup scram and ARI valves must energize to open a vent path. Backup scram valves are powered by ESF busses 11DA and 11DB. ARI valves are powered from BOP busses 11DE and 11DK. ARI initiates at <-41.6 inches wide range reactor water level. All answers are plausible because they refer to DC powered valves. The student must know which solenoid valves must energize to vent, their power supplies, and scram and ARI initiation conditions for low reactor water level. Answer **b** is correct since the RPS B backup scram solenoids would not have power to reposition to vent, but ARI solenoids would energize. Answer **a** is incorrect in both regards. Answer **c** is incorrect regarding backup scram valves. Answer **d** is incorrect regarding ARI valves.

KA 295004 K2.03
Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF DC POWER and the following: **D.C. bus loads**

10CFR55.41(b)(6)

GLP-OPS-C7100 Obj. 5.5, 20
GLP-OPS-C111A Obj. 5.5, 8.7

04-1-01-C71-1
04-1-01-L11-1
04-1-01-C11-1
E1173-14,21

Difficulty 2

RO 4

The plant is at 100% power when the Main Generator trips due to vibration.

Reactor water level will:

a. go below Level 2, and the level band should be +30 inches to -30 inches.

b. stay above Level 2, and the level band should be +30 inches to -30 inches.

c. go below Level 2, and the level band should be +11.4 inches to +53.5 inches.

d. stay above Level 2, and the level band should be +11.4 inches to +53.5 inches.

Ans: d

This question tests knowledge of the effect on reactor level of a turbine/generator trip and knowledge of level control expectations. Nominal level bands are prescribed based on whether SRVs are being cycled for pressure control and/or whether ECCS is injecting for level maintenance. Based on plant and simulator data, reactor water level will initially shrink but will reach a minimum of about -20 inches wide range, remaining above level 2, -41.6 inches, following a Turbine/Generator trip from 100% power and normal reactor water level. Therefore, HPCS should not inject. Though Low-Low Set will initiate, SRVs will not repeatedly cycle due to main bypass valve operation. Feedwater will be available for level control, and bypass valves will automatically take control of reactor pressure. Therefore, answer **d** is correct. The other answers are incorrect because they state either the wrong level response or the wrong nominal level band. The wrong answers are plausible because they involve some combination of outcomes that would be based on varying assumptions of plant response and application of Operations Philosophy.

KA 295005 K1.03
Knowledge of operational implications of the following concepts as they apply to MAIN TURBINE GENERATOR TRIP: **pressure effects on reactor level**

10CFR55.41(b)(5),(7),(10)

02-S-01-27, Operations Philosophy steps 6.6.8b, 6.6.8d

GLP-OPS-N3201 Obj. 18
GLP-OPS-PROC Obj. 59.3
GLP-OPS-MCD12 Obj. 3

Difficulty 2

RO 5

The plant was at 95% power during start up from a refueling outage when a manual scram was inserted.

Reactor Feed Pump (RFP) B has been tripped by the operator.

Why will reactor water level subsequently rise above the RFP trip setpoint if the operator does not remove RFP A from Feedwater (FW) AUTO mode and reduce its speed?

a. Digital FW Control System is tuned for 100% power and RFP speed will not lower quickly enough in FW AUTO mode to prevent over-feeding.

b. Both Reactor Recirc Pumps automatically will shift to slow speed causing a step rise in downcomer level.

c. RFP A discharge pressure will not automatically remain below reactor pressure as reactor pressure decays.

d. Feedwater injected due to initiation of Setpoint Setdown will inevitably swell above the RFP trip setpoint.

Ans: c

The low speed stop for RFPs in FW AUTO control mode is 2100 rpm, which produces approximately 830 psig RFP discharge pressure. Following an extended shut down for a refueling outage, decay heat is small. After a scram with low decay heat, reactor pressure will fall below the discharge pressure of RFPs at their low speed stop, and feed flow will occur. Step 3.2.2 of 05-1-02-I-1, Reactor Scram ONEP, directs the operator to reduce RFP speed in SPEED AUTO or SPEED MANUAL mode to control level below 56 inches for this low decay heat condition. Therefore, answer c is correct. Answer a is plausible because DFCS is tuned to be stable at full power, however, controller output can still change much faster in FW AUTO than it can be changed in SPEED AUTO or SPEED MANUAL modes. Answer b is plausible since a scram results in immediate level shrink to below Level 3, where an automatic Recirc Pump downshift occurs, which results in a higher downcomer water level. Answer d is plausible because it is a potential effect if a scram occurs from low power, when less level shrink occurs, and the additional water injected due to Setpoint Setdown is enough to cause swell above Level 9.

KA295006 K3.01
Knowledge for the reasons for the following responses as they apply to SCRAM: **reactor water level response**

10CFR55.41(b)(5),(7)

05-1-02-I-1 step 3.2.2

GLP-OPS-C3400 Obj. 22.0
GLP-OPS-ONEP Obj. 19

Difficulty 1

RO 6

The control room has been abandoned due to a fire.

A reactor operator sent to the Remote Shutdown Panel places the Transfer Switch for Lockout Transfer Relay C61-HSS-M150 at 1H22-P152 to ON.

Which of the following components is electrically separated from its control room indications and controls by this action?

a. Remote position indication for Low Pressure Core Spray pump breaker

b. Reactor Core Isolation Cooling flow controller

c. Control Rod Drive pump A

d. Division 1 wide range Suppression Pool level indication

Ans: c

This question tests knowledge of equipment affected by Lockout Transfer Relay C61-HSS-M150, not its specific effects on components. The 36 Lockout Transfer Relays on H22-P152 that are repositioned by the referenced operator action only affect indication associated with some equipment controlled from Division 1 Remote Shutdown Panel (RSP) H22-P150. Answer a is plausible because LPCS is major Div 1 equipment, but it is incorrect because LPCS is not safe shutdown equipment, cannot be controlled from the RSP, and is not affected by the transfer switch. Answer b is plausible since RCIC can be controlled from the Div 1RSP, but it is incorrect because the flow controller has a separate handswitch located on the Div 1 RSP that provides electrical separation. Answer c is correct since relay C61-R28 disables control room indication and control for CRD A pump. Answer d is plausible since the Div 1 RSP does have suppression pool level indication, but it is incorrect since the instrumentation loops are already separate by design.

KA 295016 A1.07
Ability to operate and/or monitor the following as they apply to CONTROL ROOM ABANDONMENT: **Control room/local transfer mechanisms**

10CFR55.41(b)(7)

05-1-02-II-1 Att. XVIII pg 3

GLP-OPS-C6100 Obj. 6, 7.1.4

Difficulty 1

RO 7

Plant Service Water radial well flow is degrading at 100% power.

Listed below are possible effects of this condition for which procedures require operator action.

Assuming the prescribed action is taken in response to the listed effect, which one would result in thermal stratification within the RPV?

a. CRD A and B pump lube oil temperatures have risen above 135°F.

b. EHC fluid temperature has risen above 150°F.

c. CCW inlet temperatures to Reactor Recirc pumps A and B have risen above their alarm setpoints.

d. RWCU filter demineralizer inlet temperature has risen above 130°F.

Ans: a

Per 02-S-01-27, Operations Philosophy, thermal stratification will occur anytime forced circulation is lost (e.g. when Reactor Recirc pumps go to off). Loss of Plant Service Water flow will cause Component Cooling Water (CCW) temperature and Turbine Building Cooling Water (TBCW) temperature to rise. CCW supplies cooling to CRD pumps, Recirc Pumps, RWCU heat exchangers, and other loads. TBCW supplies cooling to EHC heat exchangers and other loads. Answer a is correct because per 05-1-02-V-1, Loss of CCW, CRD pumps must be secured if their oil temperatures cannot be maintained below 135°F, and a prerequisite for securing CRD pumps is to secure both Recirc pumps. Answer **b** is plausible but incorrect because EHC pumps are secured at 150°F which causes loss of automatic pressure control, but not to the extent that cooldown limits are challenged, since the pressure band for manual control, 800-1060 psig is within cooldown limits. Answer **c** is plausible because it does deal with Recirc pump parameters, but it is incorrect since no action is required solely based on Recirc pump cooling water temperature alarms. Answer **d** is plausible because this is the point at which RWCU is secured, and loss of RWCU flow accentuates cooldown effects when forced circulation is lost, but it is incorrect because loss of RWCU flow alone does not cause thermal stratification within the RPV.

KA 295018 K1.01
Knowledge of operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER: **effects on component/system operation**

10CFR55.41(b)(5),(10)

05-1-02-V-1 step 3.2 note and 3.2.1
04-1-01-B33-1 step 3.4
02-S-01-27 step 6.6.10 note
GLP-OPS-PROC Obj. 59.3
GLP-OPS-B3300 Obj. 36.6, 38.1, 42, 47

Difficulty 2

RO 8

The plant was at rated conditions when a loss of offsite power coincident with a drywell LOCA occurred.

The reason a backup supply of nitrogen is required to be connected to replace instrument air is to ensure:

a. SRV B21F051D can actuate 100 times in Low-Low Set mode for up to 6 hours at the maximum decay heat load.

b. functionality of four Automatic Depressurization System (ADS) valves and SRV B21F051D will be preserved for 5 days.

c. Automatic Depressurization System (ADS) valves can be maintained open for 100 days.

d. emergency depressurization can be conducted at the maximum drywell pressure.

Ans: c

This memory item comes straight from a learning objective and 05-1-02-V-9, Loss of Instrument Air, step 3.12 and is a design basis for the ADS air supply. All answers are plausible since they are ADS system design bases. Answer c is correct since the related design basis states the ADS air system requires makeup air to hold the valves open for 100 days. All other answers are incorrect since the ADS air system is designed to provide these functions without alternate makeup.

KA 295019 K3.01
Knowledge for the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR: **backup air system supply:** plant specific

10CFR55.41(b)(8),(10)

05-1-02-V-9 step 3.12

GLP-OPS-E2202 obj 3.0

Difficulty 1

RO 9

The plant is in Mode 3.

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

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

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

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

b. raise circulation through the idle recirc loops.

c. provide additional coolant mass to absorb decay heat.

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

Ans: a

With no Recirc pumps operating, raising reactor water level to above 82 inches allows natural circulation from the core up through the steam separator to the feedwater annulus for core cooling. Therefore, answer **a** is correct. Answer **b** is plausible because a historically common error is to imagine natural circulation to be through recirc loop piping, but it is actually from core to downcomer to core. Answer **c** is plausible because adding water would provide more heat sink, but that is not why the procedure requires raising level. Answer **d** is plausible because the procedure does require raising level to the main steam lines to establish flow through the SRVs for Alternate Shutdown Cooling, but this is not warranted since RHR B is available to be started for shutdown cooling.

KA 295021 K3.05
Knowledge for the reasons for the following responses as they apply to LOSS OF SHUTDOWN COOLING: **establishing alternate heat removal flow paths**

10CFR55.41(b)(2),(10)

05-1-02-III-1, Inadequate Decay Heat Removal, step 3.4.2 note

GLP-OPS-ONEP Obj. 17, 20

Difficulty 1

Modified question GGNS-NRC-00078a

RO 10

The plant is in Mode 5.

The fuel transfer canal is open.

During control rod blade replacement, a blade is dropped which punctures the refueling bellows in the reactor cavity causing a 100 gpm leak.

Which of the following would be the first to indicate a loss of fuel pool inventory?

a. Area Radiation Monitors for elevation 208 ft containment

b. Fuel Pool Drain Tank level PDS computer trend

c. Fuel Pool Leak Detection Standpipe system high level alarm

d. Computer trend for re-spanned wide range reactor water level

Ans: b

A leak in the refueling bellows would drain to the drywell. The reduction in pool level would immediately cause a reduction in flow through the pool scuppers which would be manifested by a reduction in fuel pool drain tank level. The volume of the FP drain tank is many times smaller than the reactor cavity volume, so a loss of system inventory would be sensed there first as FP drain tank level would fall much more quickly than reactor cavity level. Even Horizontal Fuel Transfer operation perturbates FP drain tank level as noted in 04-1-01-G41-1 step 3.33. Operators monitor this level on a PDS computer trend to assess overall operation of the FPCCU system, therefore, answer **b** is correct. Answer **a** is plausible because lowering pool level would eventually cause a rise in area radiation levels due to loss of water shielding. However, this would not manifest itself until well after the FP drain tank level low alarm was reached. Answer **c** is plausible since there is a Fuel Pool Leak Detection system, but it only senses leakage related to filter/demineralizers, outside primary containment. Leakage from the bellows would not drain to the tank. Answer **d** is plausible since the reactor level would be the same as cavity level, but although the level instrumentation is re-spanned to a higher level for Mode 5, its upper limit of +410 inches above instrument zero is about 7 feet below the normal pool level at the scuppers.

KA 295023 A2.02
Ability to determine and/or interpret the following as they apply to REFUELING ACCIDENTS**: fuel pool level**

10CFR55.41(b)(4)

GLP-OPS-G4146 Obj. 4.5, 22, 25

Difficulty 2

RO 11

The plant is in Mode 3 at rated pressure following a small break LOCA in the drywell. Drywell pressure is 3 psig and rising slowly.

High Pressure Core Spray automatically initiated at normal level and raised level to +50 inches wide range, where the handswitch for HPCS pump on H13-P601 was momentarily placed in STOP.

Reactor level swelled to +60 inches wide range and has now lowered to +40 inches.

What, if any, is the minimum operator action required to enable HPCS automatic injection if the HPCS low reactor water level setpoint is subsequently reached?

a. Depress and release HPCS High Level Reset pushbutton.

b. Depress and release HPCS High Level Reset and HPCS Initiation Reset pushbuttons.

c. HPCS will automatically inject with NO operator action required.

d. Depress and release HPCS Initiation Reset pushbutton.

Ans: d

For the stated conditions, HPCS would have initiated on 1.39 psig drywell pressure, and the HPCS injection valve would have automatically closed at 53.5 inches reactor water level. Since HPCS pump was manually stopped with an initiation signal present, the manual override signal is sealed in and will not automatically reset. The high drywell pressure initiation can be reset with the high drywell pressure signal still present. Resetting the initiation resets the manual override on HPCS pump. All answers are plausible since they involve resetting some HPCS logic that would be actuated under the current conditions. Answer **a** is incorrect since this does not clear the manual override on the pump. Answer **b** is incorrect because the stem asks for the minimum action required, and high level logic automatically resets at the HPCS low reactor water level setpoint, -41.6 inches, so depressing the high level reset pushbutton is not required. Answer **c** is plausible because no action is required to reset HPCS Level 8, but here HPCS pump was manually overridden, which must be manually cleared. Answer c would be correct if HPCS pump had not been stopped with an initiation signal sealed in. Answer **d** is all that is required for HPCS to automatically inject at -41.6 inches reactor water level.

KA 295024 A1.02
Ability to operate and/or monitor the following as they apply to HIGH DRYWELL PRESSURE: **HPCS**: plant specific

10CFR55.41(b)(7)

04-1-01-E22-1 step 3.10
E1183-03, 23
GLP-OPS-E2201 Obj. 9.3, 9.4

Difficulty 2

RO 12

The reactor scrammed from rated conditions following a trip of all EHC pumps.

Reactor pressure is 1090 psig.

Turbine pressure regulating system indications the operator would expect to be present on the apron section of H13-P680 for this condition are:

a. Pressure controller fault light off
Bypass valves A, B, C lift fault lights off

b. Pressure controller fault light on
Bypass valves A, B, C lift fault lights off

c. Pressure controller fault light off
Bypass valves A, B, C lift fault lights on

d. Pressure controller fault light on
Bypass valves A, B, C lift fault lights on

Ans: c

The turbine integrated pressure controller (IPC) setpoint is approximately 950 psig. A pressure controller fault signal would be generated when pressure controller output was not consistent based on pressure feedback. The pressure sensed by IPC is turbine inlet pressure, which would be slightly lower than the 1090 psig reactor pressure but above the 950 psig setpoint. IPC would be sending a signal to the bypass valves to open, which is proper, and therefore, no pressure controller fault signal would be generated. Bypass valve lift fault signals are generated when the position of the respective bypass valve does not match the position demand from IPC. With no EHC pumps running, there would be no motive force to open bypass valves. So, even with the open demand from IPC, bypass valves would remain closed, hence bypass valve lift fault signal for each valve would be generated. This corresponds to answer **c** only. All other answers either represent at least one wrong light on or one wrong light off.

KA 295025 A1.02
Ability to operate and/or monitor the following as they apply to HIGH REACTOR PRESSURE: **reactor/turbine pressure regulating system**

10CFR55.41(b)(7)

04-1-02-1H13-P680-9A-E2

GLP-OPS-N3202 Obj. 4.0, 9.0, 19

Difficulty 2

RO 13

(refer to attached HCTL curve)

The following conditions are present:

Reactor pressure	1100 psig and stable
Containment pressure	1 psig and stable
Suppression pool temperature	132°F and stable

Suppression pool level is falling.

Of the following suppression pool levels, which is the <u>lowest</u> level at which it is safe to emergency depressurize for the stated conditions?

a. 18.5 ft

b. 16.5 ft

c. 15.5 ft

d. 14.5 ft

Ans: b

The Heat Capacity Temperature Limit (HCTL) curve describes when it is safe to emergency depressurize based on RPV pressure, suppression pool temperature, and suppression pool level. The stem states RPV pressure and suppression pool temperature are constant. Only from the SAFE zone of the HCTL is it known to be safe to emergency depressurize. The user should default to the closest suppression pool level line above the value of actual suppression pool level of the HCTL curve to determine what the zone of operation is. He should not interpolate between two suppression pool level lines. For this question, RPV pressure and suppression pool temperature should be plotted on HCTL, and then, choose the suppression pool level line above the plotted point as the limit. Answers **a**, **b**, and **d** are plausible since they are all lines on the HCTL curve. Answer **c** is plausible because it corresponds to an interpolated value between two lines for the given RPV pressure and suppression pool level. Only answer **b** is correct since it is the first level line encountered above the plot, making it the lowest level at which operation is still in the SAFE zone of the HCTL.

KA 295026 A2.02
Ability to determine and/or interpret the following as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE**: suppression pool level**

10CFR55.41(b)(10)

05-S-01-EP-1 HCTL curve, 05-S-01-EP-3
GLP-OPS-EP03 Obj. 6
Difficulty 2

RO 14

A small break LOCA has occurred.

Containment temperature is 96°F and slowly rising.

Drywell pressure is 1.6 psig.

Based on this transient, the operator would expect to see the containment cooling fans are:

a. running without chilled water flow.

b. running and have chilled water flow.

c. stopped without chilled water flow.

d. stopped and have chilled water flow.

Ans: a

All answers are plausible because a LOCA signal causes many isolations and load sheds. Drywell pressure 1.23 psig will cause isolation of Plant Chilled Water to containment. The location of containment coolers is such that no chilled water flow to containment coolers will exist. However, containment cooler fans are BOP powered, do not shed, and will continue to operate. Answer **a** is the only answer reflecting the correct combination for fan and chilled water flow status.

KA 295027 A1.02
Ability to operate and/or monitor the following as they apply to HIGH CONTAINMENT TEMPERATURE (MARK III): **containment ventilation/cooling**

10CFR55.41(b)(4)

GLP-OPS-M4100 Obj. 7, 13.5
GLP-OPS-M7101 Obj. 7.6

04-1-01-M41-1
05-1-02-III-5

Difficulty 2

RO 15

(refer to attached EP Caution 1)

LOCA conditions exist:

Reactor water level (wide range)	-135 inches, slowly trending down
Reactor water level (compensated fuel zone)	-135 inches, slowly trending down
Drywell temperature (166 ft elev.)	250°F
Containment temperature (139 ft elev.)	140°F
Reactor pressure	0 psig

Regarding wide range and compensated fuel zone level instrumentation, which, if any, may be used to determine reactor water level?

a. Wide range, only
b. Compensated fuel zone, only
c. Wide range and compensated fuel zone
d. Neither wide range nor compensated fuel zone

Ans: c

All answers are plausible since they are dependent on indicated level and temperature of air surrounding their instrument tubing runs in either drywell or containment. EP Caution 1 provides limitations for instrument usage based on the probability of saturated conditions occurring in the instruments' reference legs which would render the instrument non-functional. For the conditions given, operation is in the Possible Boiling region of the RPVST curve, which is based on drywell temperature. However, this does not in itself render level instrumentation useless, and with the parity between instrument ranges and the trends stated, indication should be considered valid. Part 2 of Caution 1 provides additional limitations for wide range usage based on containment temperature. However, for the conditions given, containment temperature is below the limit of Caution 1, so wide range is not disqualified. Therefore, both compensated fuel zone and wide range may be considered valid and used, as depicted by answer **c**. Answers **a** and **b** are incorrect since they only list one instrument range and specify only that one may be used. Answer **d** is incorrect since it disqualifies both ranges.

KA 295028 K1.01
Knowledge of operational implications of the following concepts as they apply to HIGH DRYWELL TEMPERATURE: **reactor water level measurement**

10CFR55.41(b)(2),(7),(10)

05-S-01-EP-1 Caution 1 and related GGNS EP Technical Bases
02-S-01-27

GG-1-LP-RO-EP02 Obj. 11, 17

Difficulty 2

RO 16

The plant is in Mode 1.

Suppression pool level cannot be maintained above 14.5 feet due to a leak.

Why is EP-2 entered for this condition?

a. The reactor should be shut down and depressurized to preclude containment over-pressurization in the event of a drywell LOCA.

b. Technical Specifications requires an immediate plant shut down due to inoperability of all low pressure ECCS systems.

c. Emergency depressurization should be conducted while there is still adequate submersion of the Automatic Depressurization Valve tailpipes.

d. Containment flooding is accessed via EP-2 to use injection systems from sources external to primary containment.

Ans: a

Answers **b** and **c** are credible because they are genuine concerns regarding low suppression pool level. Answer **d** is credible since successful containment flooding would remedy low suppression pool level. Answer **a** is the correct answer as it reflects the basis for the associated EP-3 step. EP-3 step SPL-8 requires entering EP-2 before suppression pool level drops below 14.5 feet. EP-2 step 1 requires the Reactor Mode Switch be placed in Shutdown which causes a reactor scram. If suppression pool level cannot be maintained above 14.5 ft, thus guaranteeing adequate (2 ft) submergence for the Mark III horizontal vents, all steam from a LOCA in the drywell might not be condensed in the suppression pool and result in containment over-pressurization.

KA 295030 K3.06
Knowledge for the reasons for the following responses as they apply to LOW SUPPRESSION POOL WATER LEVEL: **reactor SCRAM**

10CFR55.41(b)(7),(10)

05-S-01-EP-3 step SPL-8 bases (05-S-01-PSTG)
GGNS EP Technical Bases pg 7-33

GLP-OPS-EP03 Obj. 6

Difficulty 1

RO 17

A LOCA has occurred.

Reactor level is stable -215 inches and cannot be raised.

Which of the following conditions would ensure adequate core cooling for this condition?

a. Reactor pressure 150 psig with High Pressure Core Spray injecting 5000 gpm and Low Pressure Core Spray injecting 3000 gpm.

b. Reactor pressure 0 psig with RHR B and C injecting 7200 gpm each.

c. Reactor pressure 200 psig with 9 Safety Relief Valves open.

d. Reactor pressure 20 psig with Low Pressure Core Spray injecting 7200 gpm.

Ans: d

Adequate core cooling is defined as level above TAF (-167 inches), or above the minimum steam cooling reactor water level (-191), or above the minimum zero injection water level (-212) with no injection, or above the Minimum Steam Cooling Pressure (≥219 psig for ≥8 SRVs open), or above -217 inches and being sprayed with HPCS or LPCS at ≥ 7000 gpm. Combinations of systems or components that do not meet one of these requirements are not considered as adequate core cooling methods. All answers are credible since they are indicative of some measure of cooling, but only answer **d** satisfies the prescribed definition for adequate core cooling.

KA 295031 K1.01
Knowledge of the operational implications of the following concepts as they apply to REACTOR LOW WATER LEVEL: **adequate core cooling**

10CFR55.41(b)(8),(10)

02-S-01-27 step 5.1

GLP-OPS-PROC Obj. 59.3

Difficulty 1

RO 18

EP-2A has been entered.

Reactor power is unknown.

For which of the following conditions will shutdown margin be sufficient to allow EP-2A to be exited without obtaining concurrence from Reactor Engineering?

a. Cold Shutdown Boron weight has been injected with five control rods at position 48.

b. Only one control rod is withdrawn, and it is at position 48.

c. Only two control rods are withdrawn, one is at position 02 and one is at position 24.

d. Three control rods are withdrawn, and they are all at position 04.

Ans: b

Shutdown margin ensures the reactor will remain subcritical under all conditions with the highest worth control rod fully withdrawn. Answer **b** reflects this assumption for shutdown margin. The other answers are all credible since they represent different amounts of negative reactivity insertion. Answer **a** is incorrect because reliance on boron to remain shutdown negates exiting EP-2A. Answers **c** and **d** are incorrect because even as inconsequential as the rod densities appear, they would still have to be analyzed by reactor engineering.

KA 295037 K1.07
Knowledge of operational implications of the following concepts as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN: **shutdown margin**

10CFR55.41(b)(5),(6),(10)

05-S-01-PSTG
GGNS EP Technical Bases pg 6-7
TS Bases B3.1.1

GLP-OPS-EP02A Obj. 2, 5

Difficulty 1

RO 19

An event causing high offsite release rates from the Turbine Building Ventilation (TBV) exhaust stack is in progress.

Which of the following describes operation of the TBV Accident Range Monitor (AXM) as the TBV exhaust effluent radiation level rises?

a. The AXM should automatically begin operating when the TBV Noble Gas (GE) Monitor reaches its HIGH alarm setpoint.

b. The AXM should automatically begin operating when the TBV Noble Gas (GE) Monitor reaches its HIGH-HIGH alarm setpoint.

c. The AXM should automatically begin operating when the TBV SPING Mid Range Noble Gas Monitor reaches its alarm setpoint.

d. The AXM can only be started manually and will not automatically start.

Ans: c

The TBV AXM can be started by three normal means: if the TBV SPING Mid Range Noble Gas Monitor reaches its alarm setpoint, as stated in correct answer c, or if the SPING is placed in FLUSH mode, or manually from the control terminal. The SPING and GE monitors are separate subsystems. Answers a and b are plausible since the TBV Noble Gas (GE) Monitors monitor the same ductwork as does the SPING and are required by TS/TRM and provide high effluent radiation alarms on 1H13-P601. Answer d is plausible since the AXM can be started manually, but it is wrong because it states that it will not automatically start.

KA 295038 - High Offsite Release Rate
Generic 2.1.27 – Knowledge of system purpose and/or function

10CFR55.41(b)(11)

08-S-04-220 step 3.3
04-1-01-D17-1 step 4.8.1

GLP-OPS-D1721 Obj. 16

Difficulty 1

RO 20

In which of the following areas could a fire lasting 8 hours affect redundant Safe Shut Down (SSD) equipment?

a. Upper Cable Spreading Room 189 ft. Control Building

b. Standby Service Water A Pump Room

c. Fire Water Pump House

d. Diesel Generator Fresh Air Corridor (Breezeway)

Ans: d

Of the areas listed, only the Diesel Generator Fresh Air Corridor (Breezeway) contains Division 1 and Division 2 components, cable trays and conduit. The fire rating for the fire wrap on the cabling is less than 8 hours. Therefore, answer d is correct. The Upper Cable Spreading Room does not contain Div 2 SSD components. SSW A pump room also contains Division 3 SSW pump, but it is not Safe Shut Down. Fire water components are not SSD.

KA 600000 A2.04
Ability to determine and/or interpret the following as they apply to PLANT FIRE ON SITE**: the fire's extent of potential operational damage to plant equipment**

10CFR55.41(b)(10)

10-S-03-2 step 6.2.3d, Att. II

GLP-OPS-PROC Obj. 67.3

Difficulty 1

RO 21

The plant was at rated power when a loss of condenser vacuum occurred.

Vacuum lowered to 14 inches Hg, where it has been stable for 5 minutes.

Reactor water level is 45 inches and rising.

Which of the following describes operation of Reactor Feed Pump A (RFP A) discharge valve N21-F014A under these conditions?

a. RFP A trip can be reset by depressing the RFPT A TRIP RESET pushbutton on 1H13-P680. Only after that can N21-F014A be fully opened.

b. Low vacuum must be bypassed by depressing RFPT A VAC TRIP RESET pushbutton on 1H13-P680. Only after that can N21-F014A be fully opened.

c. N21-F014A can be fully opened with no preceding action. Then, N21-F014A will remain open even if narrow range reactor water level continues to rise offscale.

d. N21-F014A can be fully opened with no preceding action. Then, if narrow range reactor water level continues to rise offscale, N21-F014A will automatically close.

Ans: c

Any RFPT trip will cause its associated discharge valve to close. The automatic close signal to the discharge valve is removed when the trip signal clears, or the auto closure can be bypassed by depressing the open pushbutton for the discharge valve 2 minutes after receipt of the trip signal if the trip signal is still present. The stem states condenser vacuum has been 14 in. Hg for 5 minutes, indicating more than 2 minutes has elapsed since the RFP trip signal at 16 in. Hg; therefore, N21F014A can be reopened with no additional action by the operator. Once the auto closure to the discharge valve is bypassed, unless the initial trip signal clears, which would reset the bypass logic, subsequent trip signals are also bypassed so that N21F014A would not close on high reactor water level at 56 in. These factors make answer c correct. Answer a and b are plausible since vacuum is below the trip setpoint of 16 in. Hg and the referenced actions would be necessary to reset the RFP. However, they are incorrect because neither the vacuum trip nor the RFPT have to be reset to open the discharge valve since more than 2 minutes have elapsed. Answer d is plausible since it correctly states N21F014A can be opened with no additional action, but it is wrong because it states a subsequent high reactor water level would result in valve closure.

KA 295008 K2.02
Knowledge of the interrelations between HIGH REACTOR WATER LEVEL and the following: **reactor feedwater system**

10CFR55.41(b)(4)

ARI 04-1-02-1H13-P680 2A-A2, 4A2-D1

GLP-OPS-N2100 Obj. 12, 16, 36
Difficulty 2

RO 22

Which of the following conditions would result in an automatic Recirc System drive flow reduction?

a. During plant shutdown at 60% core flow, reactor water level drops to +30 inches when the first Reactor Feed Pump is secured.

b. At 30% power during plant start up, the main turbine has to be manually tripped.

c. At 25% power during plant start up, reactor water level falls to +5 inches due to a level controller failure.

d. At 100% power, a Feedwater leak in the Turbine Building causes reactor water level to fall to +15 inches.

Ans: a

A flow control valve runback to 15-20% valve position, 52% core flow, will occur if the associated Recirc Pump is in fast speed, less than two RFPs are running, and reactor water level falls below 32 inches. For answer a, both recirc pumps must be in fast speed for core flow to be 60%, and the other requirements for a FCV runback are stated to be met; therefore, answer a is correct. Answers b and c are plausible since a turbine trip or Low Level 3 would at higher powers cause a recirc pump downshift; however, these answers are incorrect since at 30% power, recirc pumps are in slow speed already. Answer d is plausible since at 100% power Recirc Pumps are in fast speed and Low Level 4, 32 inches, would cause a FCV runback for loss of one or both RFPs. But answer d is incorrect since loss of feed flow caused the level reduction, not loss of a feed pump.

KA 295009 A1.03
Ability to operate and/or monitor the following as they apply to LOW REACTOR WATER LEVEL**: recirculation system (plant specific)**

10CFR55.41(b)(6)

GLP-OPS-B3300 Obj. 24.2, 24.3, 47

04-1-02-1H13-P680 3A-D1
04-1-01-B33-1
03-1-01-2

Difficulty 2

RO 23

A LOCA is occurring in the drywell.

Drywell pressure peaked one minute ago at 7 psig and is falling.

Containment pressure is 1.5 psig and rising.

Drywell Purge Compressors are designed to start when drywell to containment differential pressure reaches:

a. +0.88 psid

b. +0.87 psid

c. -0.88 psid

d. -0.86 psid

Ans: b

Drywell Purge Compressors are designed to start if a LOCA signal, 1.39 psig in the drywell, is received, with a 30 second time delay, when drywell to containment differential pressure falls to +0.87 psid as steam in the drywell is condensed. Answer b reflects this setpoint. Other answers are plausible because they relate to Combustible Gas Control System set points. Answer a is the Post-LOCA Vacuum Relief opening setpoint, only with the sign convention changed from negative to positive. Answer c is the Post-LOCA Vacuum Relief opening setpoint. Answer c is the Post-LOCA Vacuum Relief closing setpoint.

KA 295010 K2.03
Knowledge of the interrelations between HIGH DRYWELL PRESSURE and the following:
drywell/containment differential pressure

1010CFR55.41(7)

04-1-01-E61-1 step 3.9

GLP-OPS-E6100 Obj. 6.7

Difficulty 1

RO 24

Containment Spray initiation is prohibited if in the unsafe zone of the Containment Spray Initiation Pressure Limit (CSIPL) curve because:

a. Containment air temperature is too high with respect to the existing containment pressure, and the negative pressure limit could be exceeded.

b. Containment air temperature is too low with respect to the existing containment pressure, and the negative pressure limit could be exceeded.

c. Suppression Pool water temperature is too high with respect to the existing containment pressure, and the positive pressure limit could be exceeded.

d. Suppression Pool water temperature is too low with respect to the existing containment pressure, and the negative pressure limit could be exceeded.

Ans: a

Containment Spray Initiation Pressure Limit (CSIPL) is based on containment pressure and containment air temperature. Initiation of containment spray when containment pressure is low and air temperature is too high could result in a prompt reduction in containment pressure below the design limit, causing containment liner separation, before containment spray could be secured. Thus answer **a** is correct, and others are wrong. All answers are plausible because they deal with factors that influence containment pressure. Knowledge of the CSIPL in necessary to make sense of the parameters involved in each answer.

KA 295011 A2.02
Ability to determine and/or interpret the following as they apply to HIGH CONTAINMENT TEMPERATURE (MARK III)**: containment pressure**

10CFR55.41(b)(9),(10)

GLP-OPS-EP03 Obj. 6

GGNS EP Technical Bases Section 11.3

Difficulty 1

RO 25

A small break LOCA has occurred. All ECCS are injecting and reactor water level is -100 inches and rising.

What is the lowest Suppression Pool temperature at which two loops of suppression pool cooling are **required** to be placed into operation under these circumstances?

a. 85°F

b. 95°F

c. 105°F.

d. 110°F.

Ans: b

This is a small LOCA. Three ECCS are capable of adequate core cooling by design for a DBA. All ECCS are available. EP-3 requires suppression pool temperature be maximized when SP temperature reaches 95°F, answer **b**, if adequate core cooling is assured by other means. All ECCS are available, and from the accident analysis, only 3 ECCS are required to maintain adequate core cooling for the worst case LOCA. This leaves RHR A and B available for suppression pool cooling. All other answers are plausible because they are setpoints or limits associated with high suppression pool temperature. Answer **a** is one of two temperature alarm setpoints. Answer **c** is the Tech Spec limit at which testing which adds heat to the suppression pool must be stopped. Answer **d** is the Tech Spec scram criteria and the limit for EP-2A actions.

KA 295013 K2.01
Knowledge of the interrelations between HIGH SUPPRESSION POOL TEMPERATURE and the following: **suppression pool cooling**

10CFR55.41(b)(10)

GGNS EP Technical Bases Section for steps SPT-1, SPT-2

GLP-OPS-EP03 Obj. 6

Difficulty 2

RO26

Plant start up is in progress at the point of placing the first reactor feed pump in operation on the start up level controller.

The running Control Rod Drive (CRD) pump trips, and the standby CRD trips when it starts. CRD temperature alarms and accumulator fault alarms are received.

An operator responding to a control rod accumulator fault alarm locally at the CRD HCU reports accumulator pressure for control rod 32-33, which is at position 48, is 1500 psig, and the Shift Supervisor declares the control rod inoperable.

The Reactor Mode Switch must be placed in SHUTDOWN because:

a. CRD mechanisms are overheating.

b. Reactor Recirc pump seals are overheating.

c. Control rod 32-33 may fail to scram.

d. Starting limitations for the CRD pump prevent further start attempts.

Ans: c

The stem states plant conditions as at the point of placing the first reactor feed pump in operation, about 450 psig. Below 600 psig, Technical Specifications bases B3.1.3 C states control rods with accumulator pressure below 1520 psig may fail to scram. Technical Specification 3.1.3 action D.1 requires a manual scram when a withdrawn control rod's accumulator is inoperable with low charging water header pressure, i.e. no CRD pumps running. Therefore, answer c is correct. Answers a and b are plausible since they are effects of Losing CRD flow. Answer d is plausible because Technical Specifications allows 20 minutes to restore charging water pressure if reactor pressure is above 600 psig. Because CRD pump is 4160V, starting limitations could be assumed, however, none are listed in the SOI. Answers a and b are somewhat true but are not the driving reason the Reactor Mode Switch must be immediately placed to SHUTDOWN. This is also an immediate operator action for ONEP 05-1-02-IV-1, CRD Malfunctions, step 2.1.1.

KA 295022 K3.01
Knowledge of the reasons for the following responses as they apply to LOSS OF CRD PUMPS: **Reactor SCRAM**

10CFR55.41(b)(6),(10)
10CFR55.43(b)(2)

TS 3.1.3 D
05-1-02-IV-1 step 2.1.1
GLP-OPS-ONEP Obj. 1.0
GLP-OPSC111A Obj.23
Difficulty 2

RO27

The plant was at 100% power when a steam line break in the Reactor Core Isolation Cooling room occurred.

MCCs supplying RCIC steam supply isolation valves lost power when RCIC was isolating.

RCIC room temperature is 200°F.

Which of the following describes the use of plant procedures in this situation?

a. Emergency Procedures, Loss of AC Power ONEP, and Emergency Plan procedures are used concurrently.

b. Emergency Procedures are entered and all other procedures are exited.

c. Loss of AC Power ONEP is completed, then Emergency Procedures are entered.

d. Emergency procedures and Emergency Plan procedures are entered, and all other procedures are exited.

Ans: a

EP-2, 05-S-01-EP-4 step 2.5 allows use of any procedures that do not contradict or hinder use of emergency procedures when emergency conditions exist. All answers are plausible since they involve procedures pertaining to stated conditions. Answers b and d are incorrect since actions from Loss of AC power ONEP are required to attempt to re-energize the isolation valves to stop an offsite release. Answer c is incorrect since EP-4 must be entered when RCIC room temperature exceeds the operating limit, well below the RCIC isolation setpoint and Maximum Safe limit of 185°F. Answer a is correct since the listed procedures would have to be used simultaneously to mitigate the event, and since their requirements do no conflict. An operator is required to know the requirements for restoring power as an immediate action for Loss of offsite Power, entry conditions for EP-4, and that an un-isolatable steam line break in the auxiliary building constitutes entry into the Emergency Plan.

KA 295032 - High Secondary Containment Area Temperature
Generic 2.4.5 - Knowledge of the organization of the operating procedures network for normal / abnormal / and emergency evolutions.

10CFR55.41(b)(10)
10CFR55.43(b)(5)

05-S-01-EP-4 step 2.5

GLP-OPS-EP04 Obj. 2

Difficulty 1

RO 28

Residual Heat Removal (RHR) A was operating in Suppression Pool Cooling when a Loss of Offsite Power and LOCA occurred.

Division 1 Diesel Generator failed to start and was placed in MAINTENANCE mode.

Ten minutes later, Division 1 Diesel Generator has been repaired and is ready to be returned to operational status.

RCIC is controlling reactor water level at -25 inches.

How do these conditions affect operation of RHR A?

a. Automatic RHR A pump start should be prevented before bus power is restored, then plant procedures require a system fill and vent.

b. Automatic RHR A pump start should be prevented before bus power is restored. System fill and vent is not required by plant procedures.

c. RHR A pump will remain off when bus power is restored. System fill and vent is required by plant procedures before manually starting the system.

d. RHR A TEST RTN TO SUPP POOL VLV E12F024A will require manual action to close when bus power is restored. System fill and vent is not required by plant procedures

Ans: a

Loss of AC Power ONEP requires ECCS systems that were in operation to be filled and vented if without power for more than 2 minutes unless required for adequate core cooling. With E12F024A open and RHR A pump stopped, the system will quickly drain. Starting RHR A pump in this configuration could result in significant piping damage due to water hammer effects. With level stable above top of active fuel and all other ECCS available, adequate core cooling is not challenged. Therefore, a fill and vent of RHR A is required. The LOP does not defuse the LOCA sequence. When the Division 1 Diesel Generator is returned to operational, it will start and re-energize bus 15AA, and load sequencing will occur. RHR A pump will start, unless prevented, and E12-F024A will automatically close due to the LOCA signal when power is available. Answer **a** is the only answer that is correct regarding both fill and vent and E12F024A actuation. Though wrong, all other answers are plausible because they involve the same two concerns as the correct answer.

KA 203000 A2.16
Ability to **(a)** predict the impacts of the following on the RHR/LPCI: INJECTION MODE; <u>and</u> **(b)** based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: **LOCA**

10CFR55.41(b)(7),(10)
10CFR55.43(b)(5)

05-1-02-I-4 step 3.1 caution
GLP-OPS-ONEP Obj. 10

Difficulty 2

RO 29

Five minutes ago, a small steam leak caused drywell pressure to rise to 1.5 psig.

Which of the following describes status and operation of RHR HX B BYP VLV E12-F048B under this condition?

a. E12-F048B will be closed and can be opened using its H13-P601 handswitch, and it will remain open.

b. E12-F048B will be closed and can be opened using its H13-P601 handswitch, but it will automatically re-close.

c. E12-F048B will be open and can be closed using its H13-P601 handswitch, and it will remain closed.

d. E12-F048B will be open and can be closed using its H13-P601 handswitch, but it will automatically re-open.

Ans: d

E12F048B is the RHR heat exchanger bypass valve, which controls cooling flow.
All answers are plausible since they contain a variation of the same two attributes, as-found position and stroke capability, for E12-F048B. E12F048B is interlocked open on a LOCA for 10.85 minutes for maximum injection flow. It can be stroked closed before 10.85 minutes but will reverse when either the handswitch is release or full open is reached. Answer **d** is the only answer that correctly states both of the two attributes.

KA 203000 A4.04
Ability to manually operate and/or monitor in the control room: **heat exchanger cooling flow**

10CFR55.41(b)(7)

E1181-27

GLP-OPS-E1200 Obj 8.7

Difficulty 2

RO 30

Residual Heat Removal (RHR) A was operating in Shutdown Cooling when a tube leak developed in the RHR A heat exchangers.

Standby Service Water (SSW) A had been initiated using the SSW A manual initiation pushbuttons on 1H13-P870 to support RHR A operation.

In order to manually close SSW INL TO RHR HX A P41F014A and SSW OUTL FM RHR A HX P41F068A using their 1H13-P870 hand switches under these conditions,

a. no additional action is necessary.

b. RHR A pump must be stopped.

c. SSW A initiation logic must be manually reset on 1H13-P870.

d. RHR A pump must be stopped, and SSW A initiation logic must be manually reset on 1H13-P870.

Ans: a

This questions tests understanding of operation of SSW A valves supplying RHR A heat exchangers during shutdown cooling operation. Distractors were chosen based on interlocks with SSW components that exist relative to other system or plant conditions. For example, if there is an SSW (LOCA) initiation present, SSW to RHR Hx valves cannot be closed until the LOCA initiation signal is reset. Also, if RHR pump is running, the SSW pump discharge and basin return valves cannot be closed until RHR pump is stopped. For this situation, nothing prevents closing the SSW inlet and outlet valves to RHR HX, therefore answer a is correct. Answer b is plausible if one assumes RHR pump must be stopped to isolate the required SSW valves, but this is not necessary. Answer c is plausible since SSW A was manually initiated, however the logic does not seal in and the initiation signal resets when the initiation push buttons are released. Answer d is plausible if one assumes RHR pump must be stopped and initiation logic requires manually restting.

KA 205000 A4.04
Ability to manually operate and/or monitor in the control room: **heat exchanger cooling water valves**

10CFR55.41(b)(8)

04-1-01-P41-1
M1061C
E1225-01, 07
GLP-OPS-P4100 Obj. 5.10, 6.10, 10.2

Difficulty 2

RO 31

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

a. LPCS flow meter on H13-P601 stuck downscale

b. red bulb socket above the handswitch for LPCS INJ MOV E21-F005 on H13-P601 non-functional

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

d. LPCS MOV Test Switch in TEST for 4 hours

Ans: c

All answers are plausible since they deal with LPCS components. Answer c is correct since per SOI 04-1-01-E21-1 step 3.16, the minimum flow valve must be capable of performing its intended function, otherwise, LPCS must be declared inoperable. With the flow switch failed upscale, min flow valve E21F011 would automatically close, which is not its standby line up, and would not automatically open when required. Answer a and b are incorrect, because these components are not required for LPCS to perform its safety function. Answer d is incorrect, because TRM 6.8.2 allows the MOV Test switches to be in test for up to 8 hours before valve operability is affected.

KA 209001 - LPCS
Generic 2.1.33 - Ability to recognize indications for system operating parameters which are entry-level conditions for Technical specifications.

10CFR55.41(b)(7)
10CFR55.43(b)(2)

04-1-01-E21-1 step 3.17
17-S-06-5 Att. II p15
GLP-OPS-E2100 Obj. 8.2, 11

Difficulty 2

RO 32

High Pressure Core Spray (HPCS) is being started for its quarterly surveillance test.

When HPCS Pump is started, HPCS MIN FLO TO SUPP POOL valve E22-F012 would be expected to automatically open if:

a. The HPCS pump breaker is closed and system flow is 1000 gpm.

b. The HPCS pump breaker is closed and discharge pressure is 135 psig.

c. System flow is 1000 gpm and discharge pressure is 135 psig.

d. System flow is 1000 gpm and E22-F012 is closed.

Ans: c

All answers are plausible because they are common combinations of parameters used in minimum flow circuits for emergency injection systems. E22-F012 opens if discharge pressure is above 130 psig and flow is below 1206 gpm, as stated in answer c. RHR min flow valves look at breaker position and have setpoints of 1154 gpm. RCIC min flow valve looks for 125 psig discharge pressure.

KA 209002 A1.02
Ability to predict and/or monitor changes in parameters associated with operating HPCS controls including: **HPCS pressure**

10CFR55.41(b)(7)

17-S-06-5 Att. II p20
04-1-01-E22-1 steps 5.2.2d(3), 5.2.2f

GLP-OPS-E2201 Obj 9.5

Difficulty 1

RO 33

The reactor is in Startup at 300 psig.

If Standby Liquid Control (SLC) B is initiated, what specifically will cause the Reactor Water Clean Up pump(s) to stop?

a. RWCU SPLY TO RWCU HXS G33-F251 not fully open

b. SLC B key locked switch on H13-P601 in START

c. RWCU PMP SUCT DRWL INBD ISOL G33-F001 not fully open

d. RWCU flow less than 70 gpm

Ans: a

At 300 psig RWCU is aligned in post-pump mode with suction through G33-F251 and G33-F004. G33F251 and G33F001, which is closed for post-pump alignment, automatically close when SLC B is started. G33F004 isolates from SLC A. RWCU does not trip directly from the SLC handswitch contacts, but from valve not full open limit switches, so answer **b** is wrong. It is credible since the SLC handswitch does cause automatic action within RWCU. Answers **c** is credible since it is a valid pump trip in pre-pump mode when G33F001 is open and F251 is closed, but is incorrect for post-pump mode. Answer **d** is credible since it is a pump trip, but it would only be achieved when G33-F251 was very nearly fully closed, well after not fully open. For these reasons, only answer **a** is correct.

KA 211000 A4.06
(pertaining to Standby Liquid Control) Ability to manually operate and/or monitor in the control room: **RWCU system isolation**

10CFR55.41(b)(7)

03-1-01-1 step 6.2.11
04-1-01-C41-1 step 5.3.2b(4)
E1203-02
E1204-12

GLP-OPS-G3336 Obj. 3.2, 8.1, 9.1

Difficulty 2

RO 34

All Reactor Protection System Turbine First Stage Pressure Transmitters, C71-N052A-D, instrument sensing lines were left isolated during Mode 4 due to human error.

Now, startup is in progress, and reactor power is 42%.

Reactor Recirc pump A is in fast speed. Reactor Recirc pump B is in slow speed.

Which Reactor Recirc Pump breakers would trip open from the closed position if the main generator tripped now?

Assume for the transient the maximum reactor pressure is 1140 psig, and the lowest reactor level is +5 inches.

a. CB-5A and CB-1B, 2B

b. CB-4A, 5A and CB-1B, 2B

c. CB-5A only

d. CB-4 A, 5A and CB-1B, 2B, 4B

Ans: d

Initial conditions have the following CB breakers closed: 3A, 4A, 5A, 1B, 2B, 3B, 4B. With the sensing lines isolated, EOC-RPT function would remain bypassed, therefore CB-3s and CB-4s would not trip as designed on a Turbine Trip. However, any closed CB-1s, 2s, 4s, and 5s would trip due to ARI-RPT at 1126 psig. Answer a is credible if assumed that neither CB-3s nor CB-4s trip due to EOC-RPT being bypassed. Answer b, like d, requires knowing which breakers are associated with ARI-RPT and that CB-3B and CB-4B would be closed even in slow speed. Answer c is credible if one does not know the ARI-RPT pressure setpoint and assumes only a downshift of pump A at 11.4 inches. Only d has the correct combination of breakers.

KA 212000 K3.11
Knowledge of the effect that a loss or malfunction of the REACTOR PROTECTION SYSTEM will have on the following: **recirculation system**

10CFR55.41(b)(7)

04-1-01-B33-1
17-S-06-5 Att. II p12,13

GLP-OPS-B3300 Obj. 15.1, 15.2, 27.5, 28.2, 28.3,

Difficulty 2

RO 35

The plant is in Mode 1 at 7% power.

A half scram is received, and two amber alarms received on H13-P680 are:

RPS CH B IRM UPSC TRIP/INOP and

APRM CH B/F UPSC TRIP/INOP

Which of the following events would cause these alarms under the stated conditions?

a. Start Up Level Control valve failure open

b. RPS Motor Generator set B trip

c. Inverter 1Y88 trip

d. Inverter 1Y79 trip

Ans: c

The question requires knowledge of APRM and IRM power supplies and effects of their loss. Answer a is credible since cold water injection would make neutron flux rise, however IRM upscale trips are bypassed IN Mode 1 with the Reactor Mode Switch in Run and would not alarm; therefore, answer a is wrong. Normally, an INOP trip would also be bypassed, but power is required to enforce the bypass in Mode 1. With the loss of supply power, the INOP will not be bypassed, thus accounting for the IRM alarm. The other answers are credible since they are instrument quality power supplies. Answer **b** would cause a half scram but not the NI alarms. IRM Channel B and APRM B/F are powered from 1Y88, not 1Y79; therefore, **c** is correct and **b** and **d** are wrong.

KA 215003 K2.01
Knowledge of electrical power supplies to the following: **IRM channels/detectors**

10CFR55.41(b)(6)

04-1-01-C51-1 Att III
04-1-02-1H13-P680-7A-A8, 7A-A11

GLP-OPS-C5102 Obj. 8.1, 8.3, 11.1, 18

Difficulty 2

RO 36

The plant is in Mode 2 during a start up at 1% power.

Intermediate Range Monitor (IRM) A fails downscale due to a faulty detector.

Which of the following is an administrative requirement for IRM A detector being inoperable under these conditions?

a. The detector drive feeder breaker and push button must be Danger tagged.

b. The detector must be fully withdrawn and its drive push button Caution tagged.

c. The detector must be fully withdrawn and its drive push button Danger tagged.

d. The IRM must be bypassed and a Danger tag placed on its Bypass joystick.

Ans: c

04-1-01-C51-1 step 3.6 states anytime an IRM detector is declared inoperable, ensure the detector is withdrawn and a danger tag is placed on the detector drive handswitch (push button). This corresponds to answer c. Answer a is plausible because it deals with danger tagging the detector drive, but tagging the drive feeder breaker is unnecessary. Answer b is plausible because it is similar to the actual requirement and a caution tag would be appropriate for the bypass joystick, but a caution tag is inappropriate for the detector drive push button. Answer d is plausible since the IRM would have to be bypassed to continue start up, but a caution tag is appropriate in lieu of a danger tag.

KA 215003 - IRMS
Generic 2.2.11 - Knowledge of the process for controlling temporary changes

10CFR55.41(b)(10)

04-1-01-C51-1 step 3.6

GLP-OPS-C5102 Obj. 12.1

Difficulty 1

RO37

The power supply for the drive motor for Source Range Monitor (SRM) F detector is:

a. MCC 11B12

b. MCC 14B11

c. Inverter 1Y88

d. Inverter 1Y95

Ans: b

The power supply for all SRM detector drives is MCC 14B11, answer b. Answer a is plausible because 11B12 powers TIP detector drives. Answers c and d are plausible because 1Y88 and 1Y95 power Division 2 SRM drawers and detectors for channels B/F and D, respectively.

KA 215004 K2.01
Knowledge of electrical power supplies to the following: **SRM channels/detectors**

10CFR55.41(b)(6)

GLP-OPS-C5101 Obj. 10.3

04-1-01-C51-1 Att. III

Difficulty 1

RO 38

The plant is at 100% power on the 100% load line.

Average Power Range Monitors (APRM) C and F gain adjustment factors (GAF) are 0.96.

APRM A and B GAFs are 1.02.

All other APRMs have GAFs of 1.00.

A reactivity transient causes power to rise over a period of 5 minutes to the APRM trip setpoint.

The reactor will scram when actual reactor power is approximately:

a. 107%

b. 111%

c. 113%

d. 118%

Ans: b

APRM reading = actual power ÷ GAF

The thermal neutron scram is clamped at 111%. When an APRM in each division of RPS reads 111%, the reactor will scram. APRMs C and F indicate higher than actual power because their GAFs are 0.96. The actual power existing when APRMs C and F indicate 111%, the scram setpoint, can be found by:

111 % (APRM scram)= actual power at scram ÷ 0.96

actual power at scram = (111%)(0.96) = 106.56% ≈ answer **a**

Answer b was chosen since it is the thermal upscale setpoint. Answer c was chosen because it will be the result if the GAF given for APRM A and B, 1.02, is used. Answer d was chosen since it is the high flux trip setpoint, but it is incorrect since power is stated as slowly rising; therefore, thermal upscale would trip first at 111%.

KA 215005 K4.02
Knowledge of AVERAGE POWER RANGE MONITOR / LOCAL POWER RANGE MONITOR SYSTEM design features and/or interlocks which provide for the following: **reactor SCRAM signals**

10CFR55.41(b)(6)

GLP-OPS-C5104 Obj. 4.1, 4.2, 4.3, 7.2

Difficulty 2

RO 39

Reactor Core Isolation Cooling (RCIC) was operating for a surveillance but tripped due to low pump suction pressure when flow reached 750 gpm.

Suction pressure then returned to normal.

Which of the following would you expect regarding the RCIC Trip/Throttle valve for this condition?

a. Both the RCIC TURB TRIP/THROT SUPV light and RCIC TURB TRIP/THROT VALVE light would be green.

b. RCIC TURB TRIP/THROT SUPV light would be red. RCIC TURB TRIP/THROT VALVE light would be green.

c. RCIC TURB TRIP/THROT SUPV light would be green. RCIC TURB TRIP/THROT VALVE light would be red.

d. Both the RCIC TURB TRIP/THROT SUPV light and RCIC TURB TRIP/THROT VALVE light would be red.

Ans: c

Low pump suction pressure is an electrical trip. On an electrical trip, RCIC TURB TRIP/THROT SUPV, which indicates valve stem position, would turn green. The motor operator position is indicated by RCIC TURB TRIP/THROT VALVE, which does not drive closed on a trip; therefore, answer **a** is wrong. Since high exhaust pressure only actuated when RCIC flow reached 700 gpm, when RCIC trips this signal will clear, allowing reopening of the Trip/Throttle valve. If an electrical trip is still present, the valve will not reopen as the motor actuator is stroked open, but it will if the condition clears (the logic auto resets). Answer **b** reflects that the motor can be closed, and when reopened, will latch and reopen the valve, so it is correct. Answer **c** reflects an electrical trip condition that is not clear. Answer **d** reflects a mechanical overspeed trip that requires local manual action to relatch the valve.

KA 217000 A4.10
RCIC - Ability to manually operate and/or monitor in the control room: **RCIC lights and alarms**

10CFR55.41(b)(7)

E1185-15, 39

GLP-OPS-E5100 Obj. 9.2, 9.3

Difficulty 2

RO 40

Which of the following conditions would directly cause alarm ADS / SRV LEAK, 1H13-P601-18A-G2 to annunciate?

a. ADS valve tailpipe pressure high.

b. Low ADS /SRV accumulator pressure.

c. SRV tailpipe temperature high.

d. RPV flooded to the main steam lines.

Ans: c

This alrm is generated from a recorder that monitors all SRV tailpipe temperatures. Answer a is plausible because an open SRV will cause high tailpipe pressure and another alarm, ADS/SRV VALVE OPEN on P601 also. Answer b is plausible because one may assume LEAK refers to accumulator air leakage, and low ADS accumulator pressure does bring in another alarm on P601. Answer c is correct since SRV operation from rated conditions would produce 220°F in the tailpipe, the setpoint f or the subject alarm. Answer d is plausible because Table 14 of EP-5 lists using SRV tailpipe temperature to indicate when the RPV is flooded, however tailpipe temperature would fall and cause the alarm to clear, not annunciate.

KA 218000 A3.02
Ability to monitor automatic operations of the AUTOMATIC DEPRESSURIZATION SYSTEM including: **ADS valve tailpipe temperatures**

10CFR55.41(b)(7)

04-1-02-1H13-P601-18A-G2
GLP-OPS-E2202 Obj. 18

Difficulty 1

RO 41

A large break LOCA has occurred in the drywell.

Reactor water level is -140 inches and slowly falling.

Which of the following equipment failures would prevent automatic initiation of Automatic Depressurization System (ADS) B if water level continues to fall to -200 inches?

a. Wide Range Reactor Water Level (ECCS) transmitter B21-N091F fails upscale

b. Drywell Pressure (ECCS) transmitter B21-N094B fails downscale

c. Residual Heat Removal (RHR) B pump breaker faulted and tripped upon starting

d. Loss of electrical power to DC distribution panel 1DA1

Ans: a

Both Wide Range Reactor Water Level (ECCS) transmitters B21-N091B and F must deliver low water level signals, -150.3 inches, to their respective trip units to make up the logic for ADS B to initiate. Therefore the correct answer is a. Answer b is plausible because high drywell pressure can also initiate ADS, but it is wrong because the requirement for high drywell pressure to exist to initiate ADS is bypassed after 9.2 minutes if a low water level condition persists. Answer c is plausible because RHR pump B discharge pressure high is a permissive for ADS B logic, but it is wrong because RHR pump C discharge pressure high will serve the same purpose. Answer d is plausible because ADS B logic is DC powered, but it is wrong because ADS B gets DC logic power from bus 11DB.

KA 218000 K1.03
Knowledge of the physical connections and/or cause-effect relationships between AUTOMATIC DEPRESSURIZATION SYSTEM and the following: **Nuclear boiler instrumentation system**

10CFR55.41(b)(7)

E1161-05, 06
17-S-06-5 Att. II p19

GLP-OPS-E2202 Obj. 9.2, 12.3, 19.3

Difficulty 2

RO 42

The plant is at rated power.

A false high Main Steam Line (MSL) Radiation signal had caused Group 10 valves to isolate.

The condition has been corrected, and you have been directed to restore the isolation.

What must be done <u>before</u> the Group 10 valves are re-opened?

a. Plant Chemistry must be notified to isolate certain manual valves to prevent damage to low pressure piping.

b. Plant Chemistry must be notified to take grab samples of the Reactor Coolant System because continuous conductivity monitoring has been lost.

c. Reactor Engineering must be notified to perform 06-RE-1B33-D-0001, Jet Pump Operability Surveillance, due to affects on Jet Pump flow instrumentation.

d. Main Steam Line Radiation Monitors must be manually reset at each monitor to prevent subsequent re-isolation when the affected valves reach full open.

Ans: a

Group 10 includes reactor sample line drywell isolation valves B33-F019 and B33-F020 which supply the reactor water sample station inside containment. Low pressure sample lines could be damaged by the pressure spike which occurs when the automatic isolation MOVs are opened at rated pressure. Therefore, Chemistry must be notified to close certain manual sample isolation valves that are under Chemistry's control, not Operations' control, so answer a is correct. Answer b is plausible since the Reactor Recirc conductiviy sample line is isolated by this condition. However, RWCU influent conductivity monitoring remains in service to satisfy the requirement for continuous conductivity monitoring, so grab samples are not required. Answer c is plausible since Post Accident Samples are taken from Jet Pump flow sensing lines. However, the Jet Pump flow sensing lines are not affected by a Group 10 isolation, so neither Jet Pump nor core flow instrumentation operability would be ini question. Answer d is plausible since the MSL Radiation circuit caused the event, however the MSL Radiation Monitors have no manual reset feature. The logic involved is reset using NSSSS isolation reset pushbuttons on H13-P601. Group 10 valves would not open until NSSSS was reset, and NSSSS could not be reset unless sufficient MSL Radiation Monitors were below the isolation setpoint.

KA 223002 A2.04
Ability to **(a)** predict the impacts of the following on the PRIMARY CONTAINMENT ISOLATION SYSTEM / NUCLEAR STEAM SUPPLY SHUT-OFF; <u>and</u> **(b)** based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: **process radiation monitoring system failures**

10CFR55.41(b)(4),(10)

04-1-01-B33-1 step 5.3

M1078A

GLP-OPS-B3300 Obj. 38

Difficulty 2

RO 43

(Refer to Loop Logic 17-S-06-5 Att 2 pg 24 attached)

The plant is in Mode 3 cooling down following a scram.

Shutdown Cooling B is in service. Reactor Pressure High trip unit B21-PS-N679A fails downscale.

What effect does this have regarding the Group 3 isolation functions for Shutdown Cooling common suction line valves E12-F008 and E12-F009?

a. E12-F008 will isolate because of the trip unit failure. All safety functions for Group 3 isolation are maintained.

b. E12-F008 High Reactor Pressure isolation is defeated. However, all safety functions for Group 3 isolation are maintained by E12-F009 and its trip system.

c. E12-F008 High Reactor Pressure isolation is defeated. All safety functions for Group 3 isolation are maintained since E12-F008 will still isolate on low reactor water level.

d. E12-F008 will still isolate on High Reactor Pressure. All safety functions for Group 3 isolation are maintained.

Ans: d

The safety function involved here is isolation of Group 3 containment/drywell penetrations on high reactor pressure. Two channels of high reactor pressure is available in each of two trip systems in a one-out-of-two once logic. Downscale failure of one channel does not defeat a trip system. Therefore, answer d is correct. All incorrect answers are plausible since they involve combinations of effects on the associated valve and effects on safety function. Answer a is incorrect since it states E12F008 will be isolated due to the trip unit failure, however, the trip unit failed downscale, below the high pressure setpoint of 135 psig. Answers b and c are incorrect because they state E12F008 will not isolate on high reactor pressure. Answer c is also incorrect because it attributes maintenance of the subject safety function to E12-F008 isolation on low reactor level, which is an entirely different safety function.

KA 223002 K6.06
Knowledge of the effect that a loss or malfunction of the following will have on the PRIMARY CONTAINMENT ISOLATION SYSTEM / NUCLEAR STEAM SUPPLY SHUT-OFF: **various process instrumentation**

10CFR55.41(b)(7),(10)
10CFR55.43(b)(2)

17-S-06-5 Att. II pg 24
GLP-OPS-E1200 Obj. 12
GLP-OPS-TS001 Obj. 34, 35

Difficulty 2

RO 44

The plant is at 100% power when Safety Relief Valve B21-F051A fails to the open position.

Which of the following describes the response of the Main Turbine Control Valves (TCVs) for this condition?

a. TCVs would throttle open to attempt to maintain high pressure turbine first stage pressure constant.

b. TCVs would throttle open to attempt to maintain generator load and speed constant.

c. TCVs would throttle closed to attempt to maintain reactor pressure constant.

d. TCVs would throttle closed to attempt to maintain pressure downstream of the equalizing header constant.

Ans: d

Integrated Pressure Control (IPC) receives pressure feedback from between the main steam equalizing header and the HP Turbine Stop and Control Valves. The variance between the pressure control demand setting and this steam line pressure is summed into the position control circuit for the TCVs. The position control circuit for the TCVs is pressure dominant at 100% power. If an SRV, upstream of the equalizing header, were to open, sensed pressure would decrease, and the TCVs would throttle closed to stabilize pressure as specified in answer d. Answers a and b are plausible if one assumes TCV attempt to maintain turbine first stage pressure or generator load constant, since turbine inlet pressure and generator load would decrease due to the reduction in steam flow. Answer c is plausible if one assumes IPC receives pressure feedback from reactor pressure.

KA 239002 K3.01
Knowledge of the effect that a loss or malfunction of the RELIEF/SAFETY VALVES will have on the following: **reactor pressure control**

10CFR55.41(b)(7)

GLP-OPS-N3202 Obj. 4.1, 7.1, 16

Difficulty 2

RO 45

The plant has been in a Station Blackout for three hours.

All Safety Relief Valve (SRV) accumulator air pressures are zero psig.

How will SRVs respond to rising reactor pressure under these conditions?

a. When pressure reaches the relief setpoint, a solenoid valve will open to admit steam pressure to a spring-loaded disk to pop open the respective SRV.

b. At the safety setpoint, the actuator spring force begins to be overcome. The valve gradually opens, linearly with respect to pressure. At 110% of the setpoint, the valve will be fully open.

c. At the safety setpoint, valve inlet pressure exceeds the spring force. Steam pressure is then admitted to a spring-loaded disk which will cause the valve to pop fully open.

d. Ten Safety Relief Valves will open in the safety mode at 1165 psig. The remaining Safety Relief Valves will open in the safety mode at 1180 psig.

Ans: c

In safety mode, SRV operation is as described in answer **c**. No air pressure or DC is required. DC power is available Answer **a** is plausible since the Relief logic would actuate and the SRV solenoids would energize, but with no air pressure for motive force, the SRV would not open. Answer **b** is plausible since it involves the safety mode, but it incorrectly states SRV opening would be gradual when it would actually pop open. Answer **d** is plausible because in safety mode, 8 SRVs open at 1165 psig, 6 open at 1180 psig, and 6 open at 1190 psig.

KA 239002 K5.02
Knowledge of implications of the following concepts as they apply to RELIEF/SAFETY VALVES: **safety function of SRV operation**

10CFR55.41(b)(3)

GLP-OPS-E2202 Obj. 7

Difficulty 1

RO 46

As reactor power is raised from 70% to 100%, sensed narrow range reactor water level and the VARIABLE value displayed on Master Level Controller:

a. converge but never equal one another.

b. diverge linearly until 100% power is reached.

c. equal one another over the entire range.

d. converge until they equal one another.

Ans: d

Steam Programming maintains constant in-shroud water level by accommodating the rise in differential pressure across the steam dryer and separator as reactor power and steam flow rise. A bias of 5 inches is added to sensed level. This sum acts as feedback to the Master Level Controller at powers up to 45%, and the bias is removed linearly from 45% power to 100% power. The variable display on the Master Level Controller is actual level plus this bias. All answers are plausible and include the possible permutations for equality, convergence, and divergence, however, only answer d is correct for the stated power ascension.

KA 259002 A1.01
Ability to predict and/or monitor changes in parameters associated with operating REACTOR WATER LEVEL CONTROL SYSTEM controls including: **reactor water level**

10CFR55.41(b)(5),(7)

04-1-01-N21-1 step 4.6.2j

GLP-OPS-C3400 Obj. 3.2, 11

Difficulty 1

RO 47

Standby Gas Treatment System (SGTS) A was manually initiated.

How did this affect Fan Coil Units (FCUs) in the Auxiliary Building?

a. General area FCUs stopped as a direct result of auxiliary building inlet damper T41-F007 closing. The Steam Tunnel Cooler remained running.

b. General area FCUs and the Steam Tunnel Cooler stopped as a direct result of Auxiliary Building inlet damper T41-F007 closing.

c. General area FCUs stopped as a direct result of a SGTS A initiation logic relay contacts changing state. The Steam Tunnel Cooler remained running.

d. General area FCUs and the Steam Tunnel Cooler stopped as a direct result of a SGTS A initiation logic relay contacts changing state.

Ans: a

SGTS A logic sealing in, relays de-energizing, causes inlet damper T41-F007 to close. A limit switch on the damper opens when the damper is not fully open, which causes all Auxiliary Building FCUs to stop. All answers are plausible, but getting the correct answer, answer **a**, requires knowing which FCUs are affected and that simply closing T41-F007, by whatever means, will cause the general area FCUs to stop. The Steam Tunnel Cooler remains running to circulate air in the steam tunnel for cooling. SGTS system number is T48. Auxiliary Building Ventilation system number is T41.

KA 261000 K1.01
Knowledge of physical connections and/or cause-effect relationships between STANDBY GAS TREATMENT SYSTEM and the following: **reactor building ventilation system**

10CFR55.41(b)(7)

E1257-01
E1253-06

GLP-OPS-T4800 Obj. 8.7
GLP-OPS-T4100 Obj. 9a

Difficulty 2

RO 48

Which of the following conditions would cause the amber indicating light above the handswitch for Low Pressure Core Spray (LPCS) pump A to illuminate?

a. loss of DC power to the LPCS pump breaker while LPCS pump is running

b. Load Shedding and Sequencing system has sequenced due to a LOCA and LPCS pump breaker is open

c. placing the pump handswitch on H13-P601 to STOP with LPCS pump running and an automatic start signal present

d. bus 15AA lockout while LPCS pump is stopped

Ans: b

The amber light above large pump handswitches, including the LPCS pump handswitch, provides indication of an automatic pump trip while the pump is running, whether from manual or automatic start. Answer **b** is correct because the pump has an automatic start signal but the breaker is open. Answer **a** and **d** are plausible since amber lights are usually indicative of trouble condition, which could just as well be one of those listed in these answers. In fact, if answer d were changed to state the pump was running, it would also be correct. Answer **c** is plausible because manual override of an automatic feature is important enough to warrant a warning indication. However, this condition actually causes an annunciator to alarm, but since the pump stop was intentional and the amber light is meant to indicate an automatic stop, the amber light does not illuminate.

KA 262001 A3.01
Ability to monitor automatic operations of the AC ELECTRICAL DISTRIBUTION including: **breaker tripping**

10CFR55.41(b)(7)

GLP-OPS-E2100 Obj. 8.1

E1182-06, 26

Difficulty 1

RO 49

Uninterruptible Power Supply (UPS) Inverters will transfer from preferred power to alternate power supplies in the event of:

a. over-current, over-voltage, or under-frequency.

b. over-current, over-voltage, or under-voltage.

c. under-voltage, over-frequency, or under-frequency.

d. over-current, under-voltage, or under-frequency.

Ans: d

All answers are plausible since they involve frequency, current, and/or voltage, the three variables that determine power switching for the static inverters. Over-voltage and over-frequency are used to make the distractors wrong. Over-voltage and over-frequency are viable because other systems exist that do employ monitoring for over-voltage and over-frequency and trigger actions based on those conditions. Answer **a** and **b** use over-voltage, and answer **c** uses over-frequency. That makes them incorrect. The static inverters at GGNS automatically swap to the alternate AC source in the event of over-current, under-voltage, or under-frequency, as stated in answer **d**.

KA 262002 K4.01
Knowledge of UNINTERRUPTIBLE POWER SUPPLY (AC/DC) design features and/or interlocks which provide for the following: **transfer from preferred power to alternate power supplies**

10CFR55.41(b)(4)

GLP-OPS-L6200 Obj 4.1

Difficulty 1

Exam bank question GGNS-OPS-00303

RO 50

Turbine Building Cooling Water is circulated to generator Seal Oil Coolers during a loss of offsite power using electrical power from which of the following sources?

a. D and E batteries

b. A battery

c. K and L batteries

d. B battery

Ans: c

This requires knowledge that the TBCW DC Emergency Cooling Water Pump is powered from the F DC bus and that the F DC bus derives its power from batteries K and L., consistent with answer **c**. All other answers are plausible because they would be available during a loss of offsite power event.

KA 263000 K2.01
Knowledge of electrical power supplies to the following: **major DC loads**

10CFR55.41(b)(4)

04-1-01-L11-1 att. 1F pg 1
04-1-01-P43-1 att. III pg1

GLP-OPS-L1100 obj. 6.2, 8.3

Difficulty 2

Modified bank question GGNS-OPS-03463

RO 51

Which of the following conditions is required for initiation of a Loss of Offsite Power (LOP) sequence by Load Shedding and Sequencing (LSS) system, but may not be necessary for the other kinds of LSS load sequences?

a. The respective diesel generator output breaker must be closed.

b. No bus under-voltage (BUV) signal is present on the respective ESF bus.

c. No LOCA condition exists.

d. The ESF transformer which normally supplies the respective ESF bus is de-energized.

Ans: a

All answers are plausible because they list conditions which relate to LSS sequence permissives. A LOP is sensed by LSS as no offsite power available upstream of ESF transformers 11, 21, and 12. This means that bus power can only be restored by the respective DG. LSS will initiate a LOP sequence if no offsite power is sensed, if no LOCA is sensed, if the respective DG output breaker is closed, and when bus power is restored (no BUV exists). BUV sequence permissives also include absence of a LOCA signal present and bus power restored, answers b and c. A LOCA sequence can be initiated if either the respective DG output breaker or offsite power feeder to the bus is closed. Answer d describes a condition which may exist during a BUV or LOCA sequence. Therefore, the only correct answer is a.

KA 264000 K5.06
Knowledge of implications of the following concepts as they apply to EMERGENCY GENERATORS (DIESEL): **load sequencing**

10CFR55.41(b)(7)

E1039

GLP-OPS-R2100 Obj. 15, 16, 17

Difficulty 1

RO 52

Drywell pressure is 2 psig.

All air compressors were shut down due to a rupture of the instrument air header in the Water Treatment Building.

All air receivers have been depleted.

Which of the following describes availability of cooling water flow to Plant Air Compressor B at this time?

a. Standby Service Water B flow can be aligned from the control room to supply cooling water.

b. Turbine Building Cooling Water flow can be aligned from the control room to supply cooling water.

c. Standby Service Water B flow is automatically supplying cooling water.

d. Cooling water flow is unavailable under these conditions.

Ans: d

SSW B flow is locked out to air compressors if a LOCA signal (>1.39 psig drywell pressure) is present. The TBCW supply to the air compressors is via air operated valve P43-F289 in the water treatment building. It would fail closed if air receiver pressure was depleted. With no air compressor running, P43-F289 would remain closed. Therefore, answer d is correct. Answers a and c are plausible since SSW B can supply cooling water under all but LOCA conditions and automatically aligns to supply sooling water to air compressors during a LOSP. Answer b is plausible since TBCW normally supplies cooling water to air compressors, even during a LOCA.

KA 300000 K1.04
Knowledge of physical connections and/or cause-effect relationships between INSTRUMENT AIR SYSTEM and the following: **cooling water to compressor**

10CFR55.41(b)(4),(7)

E1225-32

GLP-OPS-P5300 Obj. 15.7, 18, 30

Difficulty 2

RO 53

The plant is at 50% power.

Component Cooling Water (CCW) temperature control valve P44-F501 fails closed.

How will this affect Plant Service Water (PSW) supply header pressure to Turbine Building Cooling Water (TBCW) and TBCW temperature?

a. PSW pressure to TBCW will rise; TBCW temperature will stabilize at its original value.

b. PSW pressure to TBCW will fall; TBCW temperature will stabilize at a higher value.

c. PSW pressure to TBCW will rise; TBCW temperature will stabilize at a lower value.

d. PSW pressure to TBCW will fall; TBCW temperature will stabilize at its original value.

Ans: a

P44-F501 and TBCW temperature control valve P44-F513 are essentially in parallel to one another, so if F501 closes, pressure and flow are boosted through F513. F513 will automatically throttle to maintain TBCW temperature constant. Therefore, answer a is correct. All other answers are plausible since they depend on knowledge of the PSW flow path arrangement to eliminate. With different piping configurations, they could each be correct.

KA 400000 K6.01
Knowledge of the effect that a loss or malfunction of the following will have on the CCWS: **valves**

10CFR55.41(b)(4)

M1072A

GLP-OPS-P4447 Obj. 29
GLP-OPS-P4300 Obj. 8.2, 17

Difficulty 2

RO 54

A broken control rod lock plug leads to inadvertent uncoupling of control rod 16-33 during reactor start up at 10% power.

In the event of the worst case accident postulated for this condition, what design feature is assumed restrict the rate of rise of reactor power?

a. Rod Withdrawal Limiter

b. Intermediate Range Monitor High Flux trip

c. Average Power Range Monitor High Flux trip

d. Control Rod Velocity Limiter

Ans: d

Answer **a** is plausible even though is specifically prevents violation of the MCPR safety limit, since it limits control rod movement above the Low Power Setpoint of 35% power, which is above the power level stated in the stem. Answers **b**, and **c** are plausible because they would work to mitigate effects of the control rod drop accident. However, they act to limit the peak power and fuel enthalpy during the event. Only the control rod velocity limiter, answer **d**, limits the rate of descent of the control rod, and hence, with Doppler effects, limits the reactivity addition rate.

KA 201003 K3.01
Knowledge of the effect that a loss or malfunction of the CONTROL ROD AND DRIVE MECHANISM will have on the following: **reactor power**

10CFR55.41(b)(2)
10CFR55.43(b)(2)

UFSAR 15.4.9.3.1, Table 15.4-9, Figure 15A.6-35
TS Bases 3.3.1.1 2.b, 3.3.2.1 1.a
GLP-OPS-C111B obj 3.9, 6.5

Difficulty 1

RO 55

Concerning Reactor Recirc flow control, a flow control valve Motion Inhibit is generated when:

a. the Rotary Variable Differential Transformer (RVDT) senses flow control valve movement greater than 11% with no change in position demand.

b. the Linear Velocity Transducer (LVT) senses flow control valve movement greater than 11% per second.

c. the Function Generator output changes greater than 11% with no change in flow control valve position.

d. the Flow Limiter senses flow control valve movement greater than 11% per second.

Ans: b

All answers are plausible because they are components within the Recirc flow control circuit. To answer correctly, one must know the rate of movement produces the Motion Inhibit and that the LVT senses the rate. Only answer b gives the correct rate and sensor.

KA 202002 K5.02
Knowledge of implications of the following concepts as they apply to RECIRCULATION FLOW CONTROL SYSTEM: **feedback signals**

10CFR55.41(b)(6)

04-1-02-1H13-P680-3A-C1

GLP-OPS-B3300 Obj. 23.2

Difficulty 1

RO 56

At rated power, Leading Edge Flow Monitor (LEFM) goes out of service.

Where is the operator required to enter substitute values for Feedwater Flow Venturi Fouling and Temperature Correction Factors?

a. LEFM control terminal, turbine bldg. elev. 133 ft.

b. Either Plant Data System console on H13-P680

c. Control room Powerplex CYCLOPS console

d. Shift Manager's Plant Data System console

Ans: b

04-1-01-N21-1 section 6.9 specifies that the operator must perform the substitution as given in answer b. Other answers are plausible because of their relative significance. Though substitute values can be entered from the LEFM control terminal, operators are not qualified to perform manipulations there. The STA Powerplex console is actively manipulated daily to perform core monitoring functions. And specifying the Shift Manager's PDS console connotes importance due to his title.

KA 216000 A4.03
Ability to manually operate and/or monitor in the control room: **process computer**

10CFR55.41(b)(7),(10)

GLP-OPS-C3400 Obj. 7.2
GLP-OPS-N2100 Obj. 27, 29

04-1-01-N21-1 steps 3.19, 6.9

Difficulty 1

RO 57

The plant was at 100% power when the MSIVs closed.

Residual Heat Removal System (RHR) pump B is red tagged out of service.

The breaker for RHR A TEST RETURN TO SUPP POOL E12-F024A tripped with the valve closed.

Instrument Air to containment has been restored.

You have been directed to control reactor pressure in the normal band using SRVs.

What SRVs should be used to control reactor pressure?

a. Use only those SRVs that are cycling to minimize the number of SRVs that may subsequently weep.

b. Rotate use of SRVs to SRVs at different azimuths around the pool to prevent localized heating of the pool.

c. Rotate use of only SRVs designated by a colored key in the handswitch since those are known to weep.

d. Rotate use of only Automatic Depressurization System (ADS) valves to equalize stress on the Main Steam Lines.

Ans: b

Failure of E12-F024A to open with RHR B being out of service defeats Suppression Pool Cooling and pool circulation. Flow through minimum flow valve E12-F064A does not provide adequate circulation of pool contents. SOI 04-1-01-B21-1 section 4.2.2c provides instructions for SRV use with or without pool circulation and with or without instrument air. Answer a is plausible since a standing concern for SRV use is wear and required rebuild. Answers c and d are plausible because they are listed in the SOI at options but are for different conditions. Answer c is preferred only if suppression pool cooling is circulating the suppression pool. Answer d is preferred only when there is no continuous supply of instrument air to the SRVs. Answer b is correct since it reflects SOI step 4.2.2 c (2) for instrument air being available but pool cooling not in service.

KA 219000 A2.11
Ability to **(a)** predict the impacts of the following on the RHR/LPCI SUPPRESSION POOL COOLING MODE; <u>and</u> **(b)** based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: **motor operated valve failures**

10CFR55.41(b)(8),(10)

04-1-01-B21-1 step 4.2.2c

GLP-OPS-E2202 Obj. 20.1

Difficulty 2

RO 58

Which of the following is considered an Operation with a Potential for Draining the Reactor Vessel OPDRV in Mode 5 when reactor cavity water level is low?

a. Removal of all stem packing from RECIRC PMP A SUCT VALVE B33F023A

b. Removing the valve bonnet for RPV DR VLV TO RWCU G33F101 with a freeze seal in place upstream of the valve.

c. Opening manual vent and drain valves on main steam lines, resulting in 15 square inches of drain opening, with main steam line plugs not installed.

d. Replacement of the disc for HPCS TESTABLE CHK VLV E22F005 with HPCS INJECTION TO RX ISOL E22F036 red tagged closed

Ans: b

To mitigate the consequences of an accident, administrative controls are required to be established during a major breach of a primary containment penetration during Operations with a Potential for Draining the Reactor Vessel (OPDRV) during shutdown conditions. An OPDRV in Mode 5 when the reactor cavity is not flooded is defined as opening of a reactor penetration greater than 3.62 inches in diameter and located below 30 inches above Level 1 (-120.3 inches) that is not protected by an automatic isolation valve capable of closing on low water level or by a closed manual valve, deactivated closed automatic valve, or blind disc. Temporary plugs such as freeze seals, plumbers plugs, and inflatable bladders do not meet the isolation requirement. Answer b is correct since it qualifies as an OPDRV. G33F101 is on the reactor bottom head drain line below -120.3 inches and is 4 inches in diameter. Removing the bonnet would create a drain path and a freeze seal is an unacceptable isolation means. All other answers are plausible since they are RPV penetrations. Answer a is wrong because removal of valve packing does not constitute a major breach per 01-S-06-50 step 6.6.24. Answer c is incorrect because the main steam lines are at RPV elevation +101 inches. Answer d is incorrect because the penetration is protected by a closed manual valve.

KA 223001 - Primary CTMT and Auxiliaries
Generic 2.4.9 - Knowledge of low power / shutdown implications in accident (LOCA or loss of RHR) mitigation strategies.

10CFR55.41(b)(10)

01-S-06-50 step 6.6.24
TRM Section 1.1 definition for OPDRV

GLP-OPS-TS001 Obj. 13

RO 59

Which of the following is required if Standby Service Water (SSW) A is used to cool the Fuel Pool Cooling and Clean Up (FPCCU) heat exchangers?

a. the FPCCU heat exchangers must first be flushed

b. the plant must not be in either Mode 4 or Mode 5

c. Division 1 Load Shedding and Sequencing must be shut down

d. SSW A basin temperature must be above 50°F

Ans: c

It is the operators responsibility to ensure precautions and limitations are observed for all equipment operation. Aligning SSW to FPCCU is a non-automatic function that is performed by the operator. Due to possible water hammer that could damage pipe hangers if SSW where to shed and sequence, SOI 04-1-01-G41-1 step 3.32 and 04-1-01-P41-1 step 3.22 require the SSW be declared inoperable and the associated LSS panel be shut down to prevent automatic SSW stop and start. The operator is responsible to ensure LSS is shutdown before manually aligning SSW to FPCCU heat exchangers. Therefore, answer c is correct. Answer a is plausible because it is the reverse of the requirement to flush the heat exchangers after SSW has be used to cool them. Answer b is plausible because of the various FPCCU requirements associated with Modes 4 and 5, and the plant would normally be in shutdown when there would be allowance for one SSW and LSS to be inoperable, but this is not required. Answer d is plausible because 50°F is the basin temperature at which there begin to be restrictions on how SSW is operated, but operation is permitted if SSW temperature is above 45°F.

KA 233000 - Fuel Pool Cooling and Cleanup
Generic 2.1.2 - Knowledge of operator responsibilities during all modes of operation.

10CFR55.41(b)(7),(10)

04-1-01-G41-1 step 3.32
04-1-01-P41-1 step 3.22
GLP-OPS-G4146 Obj. 14.1
GLP-OPS-P4100 Obj. 13.1

Difficulty 1

RO 60

The plant is at rated power with EHC pumps B and C operating.

What would be the direct effect of loss of bus 13AD on the Turbine Pressure Control System?

a. All turbine control valves would fail open

b. All turbine control valves would fail closed

c. All turbine control valves would remain operational

d. All turbine control valves would fail as-is

Ans: c

Bus 13AD supplies normal power to the turbine pressure control system and EHC pump A. Loss of bus 13AD would not result in loss of a running EHC pump. The Turbine Pressure Control System would automatically swap to the backup channel powered by DC. Therefore, only answer c is correct. The other answers are plausible failures that can occur due to control system failures.

KA 241000 K1.14
Knowledge of physical connections and/or cause-effect relationships between REACTOR/TURBINE PRESSURE REGULATING SYSTEM and the following: **AC electrical power**

10CFR55.41(b)(7)

04-1-01-N32-1 Att. III

GLP-OPS-N3201 Obj. 2.1, 2.4, 15

Difficulty 2

RO 61

What indication is used by the operator in determining reactor coolant system Identified Leakage rate as the primary method to verify Identified Leakage limits of Technical Specifications are met?

a. Drywell Floor Drain Sump recorder E31R618 flow rate indication (red pen)

b. Drywell Floor Drain Sump recorder E31R618 sump level indication (blue pen)

c. Drywell Equipment Drain recorder E31R185 flow rate indication (red pen)

d. Drywell Equipment Drain Sump recorder E31R185 sump level indication (blue pen)

Ans: d

All answers are plausible because they list various indications available to monitor overall leakage within the drywell. Identified leakage is that which is routed to the Drywell Equipment Drain Sump. Leakage collected by the Drywell Floor Drain Sump is considered to be unidentified leakage. This makes answers a and b wrong. The primary method for verifying TS leakage limits are met involves a manual calculation of sump fill rate by summing the total sump level rise over time. This is indicated on recorder E31R185 blue pen. Therefore, answer d is correct. Flow rate indication from the sump integrator function of E31R185 is recorded but is not used as the primary method to meet TS. This makes answer c wrong.

KA 268000 A4.01
Ability to manually operate and/or monitor in the control room: **sump integrators**

10CFR55.41(b)(10)

06-OP-1000-D-0001 Att. I, Data Sheet II, item 26

GLP-OPS-P4500 Obj. 5.1, 8.1
GQQC-OPS-LOQC: Administration, item 4

Difficulty 1

RO 62

The control room must be evacuated due to toxic fumes, and actions are being taken in accordance with Shutdown from the Remote Shutdown Panel ONEP.

What areas will be more vulnerable to effects from a plant fire due to defeating CARDOX subsystems as required by that ONEP?

a. Remote Shutdown Panel rooms and Division 1, 2, and 3 Switchgear rooms

b. Remote Shutdown Panel rooms and Division 1 and 2 Switchgear rooms only

c. Remote Shutdown Panel rooms and Division 1 Switchgear rooms only

d. Remote Shutdown Panel rooms only

Ans: a

The Shutdown from the Remote Shutdown Panel ONEP would be entered in the event the control room had to be abandoned. It requires aborting the CARDOX (CO2) systems for Remote Shutdown Panel rooms and Division 1, 2, and 3 Switchgear rooms before manning the Remote Shutdown Panels. This makes those areas more susceptible to fire damage, as stated in answer **a**. The incorrect answers are all plausible since they are subsets of the correct answer and represent incremental expansions of the areas surrounding the Remote Shutdown Panels and all contain the Remote Shutdown Panels themselves.

KA 286000 K3.03
Knowledge of the effect that a loss or malfunction of the FIRE PROTECTION SYSTEM will have on the following: **plant protection**

10CFR55.41(b)(4),(10)

05-1-02-II-1 step 3.4
04-1-01-P64-3 Att. III pg 1
GLP-OPS-C6100 Obj. 4, 5, 7

Difficulty 1

RO 63

The plant is in Mode 5 with core loading in progress.

Which of the following combinations of signals would terminate a release from Containment Ventilation extended outage mode and start Standby Gas Treatment System A?

a. Fuel Handling Area Pool Sweep Exh Radiation channel A 32 mr/hr and
Fuel Handling Area Pool Sweep Exh Radiation channel C 45 mr/hr

b. Fuel Handling Area Vent Exh Radiation channel A 45 mr/hr and
Fuel Handling Area Vent Exh Radiation channel D 5 mr/hr

c. Containment and Drywell Vent Exh Radiation channel A 6 mr/hr and
Containment and Drywell Vent Exh Radiation channel D 32 mr/hr

d. Fuel Handling Area Pool Sweep Exh Radiation channel A 35 mr/hr and
Fuel Handling Area Vent Exh Radiation channel A 45 mr/hr

Ans: b

All answers are plausible since the listed instrumentation and readings are associated with automatic isolations required operable under some conditions in Mode 5, and all readings are in excess of associated instrumentation trip setpoints. However, instrumentation in answer **c**, though the accident occurs inside containment, has nothing to do with starting SGTS. Instrumentation listed in the other answers is not associated with containment isolation, but effects isolation of the exhaust stack used for extended outage mode containment ventilation, which is actually an auxiliary building boundary. Answer **b** is correct and answers **a** and **d** are incorrect because the logic to initiate SGTS A is channels A and D of either FHAV or FPS exhaust radiation.

KA 288000 K4.02
Knowledge of PLANT VENTILATION SYSTEMS design features and/or interlocks which provide for the following: **secondary containment isolation**

10CFR55.41(b)(9),(11)

05-1-02-III-5

GLP-OPS-T4801 Obj. 8.6, 8.7

Difficulty 2

RO 64

A loss of off site power occurred one hour ago.

Control Room Air Conditioning Unit B is running.

How would Control Room Air Conditioning Unit B be affected if Standby Service Water pump B were to trip?

a. the compressor would trip on high evaporator pressure and the fan would continue to run

b. the compressor would trip on high evaporator pressure and the fan would trip

c. the compressor would trip on high condenser pressure and the fan would continue to run

d. the compressor would trip on high condenser pressure and the fan would trip

Ans: c

SSW provides cooling water flow to control room AC condensers during a LOSP. Without cooling water flow, condenser pressure would rise and cause a trip of the compressor. The air handling unit fan would not be affected and would continue to run to provide necessary flow to Control Room Standby Fresh Air system. This is reflected by answer **c**. Although evaporator pressure would rise, the is not a trip associated with high evaporator pressure. There is a trip for low evaporator pressure, which shares a reset switch common to the high condenser pressure trip. This makes answers **a** and **b** plausible. The incorrect answers either mention an evaporator trip or a fan trip which do not occur.

KA 290003 K6.02
Knowledge of the effect that a loss or malfunction of the following will have on the CONTROL ROOM HVAC: **component cooling water systems**

10CFR55.41(b)(4),(7)

04-S-02-SH13-P855-2A-B3

GLP-OPS-Z5100 Obj. 3.1, 7, 13.3, 19, 22

Difficulty 2

RO 65

Which of the following could cause unacceptable cyclic stress to Reactor Pressure Vessel internals in Mode 1 when starting Reactor Recirc Pump A?

a. Bottom head coolant temperature minus Recirc Loop A coolant temperature equals 55°F

b. RPV saturation temperature minus bottom head coolant temperature equals 55°F

c. RPV saturation temperature minus Recirc Loop A coolant temperature equals 55°F

d. Recirc Loop A coolant temperature minus RPV saturation temperature equals -8°F

Ans: c

This question asks for the answer that would equate to unacceptable operation. Technical Specifications 3.4.11 specifies pressure and temperature limits for RCS components for all modes to ensure cyclic stresses on RCS components remain within design assumptions. TS SR 3.4.11.3 requires the difference between RPV coolant temperature and bottom head coolant temperature to be ≤100°F. SR 3.4.11.4 requires the difference between RPV coolant temperature and the temperature of the coolant in the idle recirc loop to be started to be ≤50°F. Answer c exceeds this limit. Since it represents unacceptable operation, it is the correct answer. Answer a is plausible because it involves two of the three parameters addressed in SR 3.4.11.3 and SR 3.4.11.4. However, no limitation is established in TS or plant procedures based on the relationship of bottom head coolant temperature and idle recirc loop temperature. This answer cannot be limiting and is, therefore, incorrect. Answer b is plausible since it addresses the requirement specified in SR 3.4.11.3, but it meets the 100°F limit. Since it represents acceptable operation, it is incorrect. Answer d is plausible because it is a familiar limitation associated with starting recirc pump. However, it pertains to recirc pump cavitation, not thermal shock on RCS components and is, therefore, incorrect.

KA290002 - Reactor Vessel Internals
Generic 2.2.22 - Knowledge of limiting conditions for operations and safety limits

10CFR55.41(b)(2),(5)
10CFR55.43(b)(2)

GLP-OPS-B3300 Obj 38.1, 44

04-1-01-B33-1 step 3.3.1, 3.3.2
TS SR3.4.11.3, SR3.4.11.4, TS Bases 3.4.11

Difficulty 1

RO 66

Intermediate Range Monitors (IRMs) are on range 6 during plant start up.

A 20 minute delay in withdrawing control rod 32-17 from position 00 has occurred due to air in the control rod drive.

Control rod 32-17 is now free to move at position 02.

In the meantime, the reactor has gone subcritical due to cooldown.

For the current step of the control rod sequence, allowance for continuous gang mode withdrawal is denoted.

How should control rod removal proceed and why?

a. continuously withdraw the current control rod gang because that is all the control rod sequence allows.

b. continuously withdraw control rod 32-17 to ensure remaining air in the control rod is properly vented.

c. Individually notch each control rod in the gang to provide the maximum control rod drive water pressure for each drive in case other rods are air bound.

d. Individually notch each control rod in the gang to re-achieve criticality, because void feedback is not available to turn power.

Ans: d

This is a combination of IOI 03-1-01-1 precautions 2.1.6, 2.1.16 and 2.1.17. The reactor going subcritical due to cooldown implies void feedback is not available. With the reactor subcritical and cooling down, care should be used to bring the reactor back critical, attempting to minimize control rod movement worth. Gang mode is not allowed. Notch withdrawal should be used to allow proper monitoring of neutron feedback. This is why answer **d** is correct. Answer **a** is plausible because operators are trained not to deviate from the approved control rod sequence. However, an operator can always be more conservative in control rod movement (i.e. move rods fewer notches per unit time) as he deems suitable. Answer **b** is plausible because air in the CRD was the initiator of the delay and is a problem experienced in every start up. However, once a CRD is sufficiently vented to move it off of position 00, air is no longer a concern. Answer **c** is plausible because it might seem moving one control rod would concentrate drive water flow and pressure, however, stabilizing valve are designed to normal drive water flow and pressure whether for individual rod movement or gang movement. The real concern is avoiding a short reactor period and an uncontrolled high flux condition.

KA Generic 2.1.32
Conduct of Operations: Ability to explain and apply system limits and precautions

10CFR55.41(b)(1),(10)

03-1-01-1 precautions 2.1.6, 2.1.16, 2.1.17

GLP-OPS-IOI01 Obj. 3.5

Difficulty 2

RO 67

Which of the following would be a violation of Technical Specification work hour limitations?

a. An individual works 6 days in a row. Each day he works 9 hours, which includes 1 hour of turnover time. Between completing end-of-shift turnover on day 5 and starting beginning-of-shift turnover on day 6, he has a 7 hour break.

b. An individual works 12 hours, which includes 1 hour turnover time. He returns to work after a 9 hour break and works 12 hours with no turnover time.

c. An individual works 18 hours straight, 2 hours of which is turnover time. He returns to work after a 9 hour break and works 8 hours with no turnover time.

d. An individual works 18 hours straight, 2.5 hours of which involves conducting turnover at his work station.

Ans: a

ROs are responsible for knowing work hour limitations. All answers are plausible because they represent potential work schedules. Answer a would violate TS since the break between the last two work days is only seven hours. TS 5.2.2e.3 requires a break of at least 8 hours between work periods, including shift turnover time. Answers b and c relate to the TS 5.2.2e.2 limitation for no more than 24 hours work in a 48 hour period, excluding turnover time, and the TS 5.2.2e.3 requirement for a 8 hour break between work periods. Answer c additionally tests the TS 5.2.2e.1 limitation of no more than 16 hours work in a 24 hour period, excluding turnover time. In answers b and c the limits are met. Answer c just meets the limits. Answer d meets the limit of TS 5.2.2e.1 since work time excluding turnover time is less than 16 hours.

KA Generic 2.1.10
Conduct of Operations: Knowledge of conditions and limitations in the facility license

10CFR55.41(b)(10)

TS 5.2.2e

GLP-OPS-ADMIN Obj. 8.26

Difficulty 2

RO 68

You have been directed to have the safe shutdown operator install the following Emergency Procedure attachment:

Att. 18 – Defeating ARI/RPT Logic trips

Where will you send the operator to perform this attachment?

a. Main control room back panels and upper control room

b. Reactor Protection System motor generator rooms

c. ARI/RPT test panel on elevation 139 ft in the Auxiliary Building

d. Turbine Building elevations 166 ft and 113 ft

Ans: d

EP Att. 18 is performed by opening one breaker on each of DC panels 1DE2 (Area 3, 113 ft) and 1KE1 Area 4, 166 ft), as in answer **d**. Answer **a** is plausible since so many logic trips are defeated by EP Attachments by jumper and/or relay manipulations in the listed panels. Answer **b** is plausible since ARI/RPT is related in function to RPS and the listed locations house the RPS power panels. Answer **c** is plausible since the ARI/RPT test panel could be used to manipulate ARI/RPT system.

KA Generic 2.1.8
Conduct of Operations: Ability to coordinate personnel activities outside the control room

10CFR55.41(b)(6),(10)

05-S-01-EP-1 Att. 18

GLP-OPS-EP07 Obj. 10

Difficulty 1

RO 69

According to the Cautions and Limitations in SOI 04-1-01-N21-1, Reactor Feedwater, do NOT maintain or set Reactor Feed Pump Turbine speed:

a. 2750 rpm to 2850 rpm.

b. 2850 rpm to 2950 rpm.

c. 2950 rpm to 3050 rpm.

d. 3050 rpm to 3150 rpm.

Ans: c

SOI 04-1-01-N21-1 precaution and limitation 3.4 states that reflected in answer c. All answers are plausible since they are of nearly the same magnitude and range.

KA Generic 2.2.2
Equipment Control: Ability to manipulate the console controls as required to operate the facility between shutdown and designated power levels

10CFR55.41(b)(4)

04-1-01-N21-1 step 3.4

GLP-OPS-N2100 Obj. 29.1

Difficulty 1

RO 70

For which of the following conditions could Residual Heat Removal (RHR) B be considered an OPERABLE ECCS system in MODE 4?

a. Injection valve E12-F042B breaker is red tagged out of service and undergoing megger testing by Electrical Maintenance.

b. Suction valve E12-F004B is closed and de-energized due to temporary power installation for its Motor Control Center.

c. RHR B Minimum Flow transmitter instrument lines are isolated for transmitter replacement.

d. RHR B is operating in Shutdown Cooling Mode, with suction valve E12-F004B closed, returning to the RPV via feedwater line.

Ans: d

Technical Specification 3.5.2 allows one of two required ECCS systems to be OPERABLE via manual realignment for the express consideration of an otherwise OPERABLE RHR operating in shutdown cooling mode. Therefore, answer **d** is correct. Answer **a** is plausible since local manipulations are allowed as part of manual realignment, but this does not allow inoperability due to maintenance activities. Answer **b** is plausible since the suction valve could be opened manually. However, here again the inoperability is due to maintenance on a support system for the suction valve. This situation was listed because of plant experience for RF13. Answer **c** is plausible because LCO 3.3.5.1 for the inoperable instrumentation allows some amount of time (7 days) for the associated instrumentation to be inoperable before the affected system has to be declared inoperable, as do most instrumentation LCOs. However, SOI 04-1-01-E12-1 states that the minimum flow valve must operate as designed for LPCI B to be considered operable.

KA Generic 2.2.22
Equipment Control: Knowledge of limiting conditions for operations and safety limits

10CFR55.41(b)(8)
10CFR55.43(b)(2)

TS 3.5.2 LCO Note, TS B3.5.2
04-1-01-E12-1 step 3.2.5

GLP-OPS-E1200 Obj. 17

Difficulty 2

RO 71

The plant is in Mode 5.

New fuel loading into the core is in progress.

In which of the following cases is control rod movement allowed while loading new fuel?

a. withdrawal of any control rod for replacement after its control cell has been defueled

b. withdrawal of any control rod for the sole purpose of demonstrating shutdown margin

c. withdrawal of any control rod as long as a spiral reload sequence is being used for core loading

d. withdrawal of any control rod as long as the surrounding 5 X 5 array of control rods are fully inserted and disarmed

Ans: a

Basic to the refueling process, 03-1-01-5, Refueling, steps 2.2, 2.17, and the caution at step 5.36 says once in-vessel fuel shuffling or loading begins, no control rod movement is to be performed except in those cells where all four fuel bundles have been removed until a complete core verification has been performed. This is consistent with TS 3.9.3, 3.10.5, and 3.10.6 and corresponds to answer a. Answer b is plausible because it relates to a requirement of TS 3.10.5, however, TS 3.10.5 also requires <u>no</u> fuel loading to be ongoing and provides no exception to TS 3.9.3. Answer c is plausible because fuel loading is only permitted with a control rod withdrawn per TS 3.10.6, but it is wrong because TS 3.10.6 also requires the control cell to be defueled before withdrawing the control rod. Answer d is plausible because it is a requirement of TS 3.10.5, however, TS 3.10.5 also requires no fuel loading to be ongoing.

KA Generic 2.2.27
Equipment Control: Knowledge of the refueling process

10CFR55.41(b)(10)
10CFR55.43(b)(2),(6),(7)

03-1-01-05 step 2.2, 2.17, caution at step 5.36
TS 3.9.3, TS 3.10.5, TS 3.10.6

GLP-OPS-IOI05 Obj. 2.6

Difficulty 1

RO 72

In which of the following situations should independent verification be waived?

a. checking position of a G36 valve in a 100 mr/hr field, where 10 minutes is estimated to perform valve position check.

b. Opening air isolation valve for Condensate long-cycle clean up valve N19-F510 which is posted as a high contamination area.

c. Opening the breaker for 10 meter tower electrical power at the MET shack with severe weather in the area.

d. Closing the power panel breaker for RWCU filter/demineralizer outlet valve G36-F004A, which still has its air supply isolated under red tag.

Ans: a

Due to ALARA considerations, independent verification may be waived if the component is located in a radiation area that may result in a dose of 10 mrem whole body per 01-S-06-29 step 6.5.1. The conditions stated in answer a would result in whole body exposure of 17 mrem (100 mr/hr X 10 min)/60 min/hr = 17 mrem). Therefore, answer a is correct. The incorrect answers are all systems that require independent verification, but the situations listed are not exempted by procedure, though the verification might be delayed.

KA Generic 2.3.2
Knowledge of the facility ALARA program

10CFR55.41(b)(10)

01-S-06-29 step 6.5.1

GLP-OPS-PROC Obj. 20.7

Difficulty 2

RO 73

Which of the following is necessary when performing a Low Volume Purge of the containment in Mode 1?

a. Containment temperature is less than 90℉.

b. Containment pressure is less than 1.23 psig.

c. 20 inch containment purge valves are open.

d. Drywell purge valves are closed.

Ans: d

To open the 6 inch containment purge valves, Technical Specifications SR3.6.5.3.1 requires the 20 inch containment purge valves and the drywell purge valves to be closed. That is why answer **d** is correct and answer **c** is wrong. High Volume Purge EPI 04-1-03-M41-3 also cautions against this at step 4.3.1. Low Volume Purge EPI 04-1-03-M41-2 prerequisites require M41 be in its normal configuration (i.e. 20 inch purge valves closed). Containment low volume purge (using 20 inch valves) is allowed for containment pressure and temperature control. The prerequisites for low volume purge allows it to be used to control containment temperature if above 90℉ or co ntainment pressure, therefore, answers **a** and **b** are incorrect. They are plausible since these conditions would warrant EP-3 entry, which might not be associated with allowing breaching containment.

KA Generic 2.3.9
Knowledge of the process for performing a containment purge

10CFR55.41(b)(4),(10)
10CFR55.43(b)(2)

04-1-03-M41-2 step 4.2, 4.3
04-1-03-M41-3 step 4.3.1 caution
04-1-01-M41-1 step 3.10

GLP-OPS-M4100 Obj. 14

Difficulty 1

RO 74

(refer to attached Power-Flow map)

Period Based Detection System (PBDS) B Decay Ratio Hi-Hi is in alarm on H13-P680.

In which of the following conditions must this alarm be considered valid?

a. Reactor power 75%
Core flow 70 mlbm/hr
PBDS B Count Data Trend on PDS indicates 15 counts

b. Reactor power 75%
Core flow 55 mlbm/hr
PBDS A Count Data Trend on PDS indicates 15 counts

c. Reactor power 40%
Core flow 40 mlbm/hr
PBDS A Decay Ratio Hi-Hi in alarm

d. Reactor power 100%
Core flow 85 mlbm/hr
PBDS B Decay Ratio Hi in alarm

Ans: a

A valid PBDS HI-Hi alarm is a symptom of thermal hydraulic instability which requires an immediate manual reactor scram, so operators must be able to quickly interpret a PBDS HI-Hi alarm as either valid or invalid. Alarm Response Instruction 04-1-02-1H13-P680-7A-C6 describes validation of the Hi-Hi alarm as being in the Monitored or Restricted Region of the power-flow map in conjunction with a high decay ratio alarm on the same channel or PBDS counts for the same channel on the PDS computer indicating 11 or more counts. Answer **a** is plotted in the Monitored Region of the PF map, and computer point counts for the same channel, B, is >11, thus validating the Hi-Hi alarm. So, answer **a** is correct. All other answers are plausible since they reference the same variables and considerations for alarm verification. Answers **b** and **c** are wrong since they refer to indications on a different PBDS channel. Answer **d** is wrong since operation would be plotted to the right of the Monitored Region.

KA Generic 2.4.46
Ability to verify that alarms are consistent with plant conditions

10CFR55.41(b)(6),(10)

Power-Flow map
ARI 04-1-02-1H13-P680-7A-C6

GLP-OPS-C5106 Obj. 22.2

Difficulty 2

RO 75

Which of the following unexplained sets of conditions could warrant entry into Jet Pump Anomalies ONEP?

a. Reactor power drop
Recirc A drive flow rise
Core plate differential pressure drop

b. Reactor power rise
Recirc A drive flow rise
Core plate differential pressure rise

c. Reactor power rise
Recirc A drive flow drop
Core plate differential pressure rise

d. Reactor power drop
Recirc A drive flow drop
Core plate differential pressure drop

Ans: a

The symptoms as listed in correct answer **a** are from Jet Pump Anomalies ONEP 05-1-02-III-6 section 4.0 and would be indicative of jet pump failure (e.g. jet pump beam failure and rams head ejection). Power falls because core flow is reduced. Recirc drive flow in the affected loop rises because of less flow resistance since drive flow is bypassing the constricted jet pump nozzle. Core plate dp decreases because total jet pump flow decreases. All other answers are plausible since they list the same parameters, but they are wrong because one or more parameters in each answer is listed as moving in the wrong direction.

KA Generic 2.4.4
Ability to recognize abnormal indications for system operating parameters which are entry-level conditions for emergency and abnormal operating procedures

10CFR55.41(b)(2),(10)

05-1-02-III-6 section 4.0

GLP-OPS-ONEP Obj. 43

Difficulty 1

SRO 1

The plant was at 100% power when a Station Blackout occurred, resulting in loss of power to busses 15AA, 16AB, and 17AC.

The reactor remains at rated pressure.

Reactor water level is -150 inches and stable.

Reactor Core Isolation Cooling (RCIC) is injecting 800 gpm.

A steam leak in the RCIC room occurs, and RCIC room temperature is 140°F and rising slowly.

Regarding RCIC, the CRS should direct appropriate personnel to:

a. immediately attempt to isolate the steam leak in accordance with 02-S-01-27, Operations Philosophy, by closing RCIC steam supply valves to protect plant equipment and personnel.

b. defeat RCIC isolations and trips per the Loss of AC Power ONEP and EP-2 attachments to allow and preserve RCIC operation.

c. isolate steam to RCIC when the Max Safe temperature for the RCIC room is reached per EP-4, and Emergency Depressurize per EP-2 when RPV level drops below -191 inches.

d. defeat RCIC isolations and trips per the Loss of AC Power ONEP and EP-2, and isolate steam to RCIC when the Operating Limit is reached for RCIC room temperature per EP-4.

Ans: b

This question tests understanding of seemingly conflicting instructions between EP-4 and EP-2 and requires SRO level judgment to determine whether EP-2 or EP-4 actions regarding RCIC take precedence. EP-2 instructions regarding injection system operation are of higher priority than EP-4 or Operations Philosophy instructions, if necessary to ensure adequate core cooling. For the situation given, EP-4 step 5 requires isolating primary systems discharging outside of primary containment through a leak, unless needed for EP actions. RCIC is the only injection system available during a Station Blackout. For the conditions stated, level stable at -150 inches with RCIC injecting, tripping RCIC would result in level falling and loss of adequate core cooling. Therefore, RCIC operation is needed per EP-2, regardless of RCIC room temperature. This makes answers a, c, and d incorrect. Answer d is incorrect because EP-2 directs. All incorrect answers are plausible since they are procedural actions that would be proper under different circumstances, when adequate core cooling could be accomplished by another means.

KA 295006 - SCRAM
Generic 2.4.16 - Knowledge of EOP implementation hierarchy and coordination with other support procedures.

10CFR55.43(b)(5)

GGNS EP Technical Bases for EP-4 step 4 and EP-2 step L-3 and Table 1
05-1-02-I-4 step 3.2.7

02-S-01-27 step 6.4.1
GLP-OPS-EP04 Obj. 7
Difficulty 2

SRO 2

(Attachments I and II of Shutdown from the Remote Shutdown Panel ONEP are attached)

The plant had been at 90% power when the control room had to be abandoned.

Manual scram pushbuttons were used to shut down the reactor, and Main Steam Isolation Valves (MSIVs) have closed.

When operators reach the Remote Shutdown Panel, indicated reactor pressure is 800 psig and slowly rising.

Remote Shutdown Panel wide range water level indicates +10 inches.

High Pressure Core Spray (HPCS) and Condensate System are unavailable.

The SRO at the Remote Shutdown Panel should:

a. Enter EP-2, and establish an initial level band +30 inches to -30 inches on indicated wide range level. Then, exit EP-2, and enter Shutdown from the Remote Shutdown Panel ONEP.

b. Enter Shutdown from the Remote Shutdown Panel ONEP, and establish an initial level band +11.4 inches to +53.5 inches on indicated wide range level.

c. Enter EP-2 and Shutdown from the Remote Shutdown Panel ONEP concurrently. Establish an initial level band +30 inches to -30 inches on indicated wide range level.

d. Enter EP-2 and Shutdown from the Remote Shutdown Panel ONEP concurrently. Establish an initial level band +36 inches to -24 inches on indicated wide range level.

Ans: c

This question requires knowledge of procedure hierarchy, level band control philosophy, and understanding of using level indication nomographs for the Remote Shutdown Panels (RSP). With indicated level below the EP-2 entry condition of +11.4 inches, EP-2 should be entered. The Shutdown from the Remote Shutdown Panel ONEP would also be entered since it gives explicit instructions for manipulating RSP controls. With MSIVs closed, SRVs would be the only available means of pressure control. Operations Philosophy 02-S-01-27 step 6.6.8d requires a level band of +30 to -30 inches when SRVs are being used. With HPCS and Condensate unavailable, 05-1-02-II-1 step 3.7.1 requires a level band of +30 to -30 inches. These factors make answer **c** correct. Other answers are plausible because they involve procedures in use and relate to familiar nominal level bands. Answer **a** is wrong because it states only the EP-2 or the ONEP should be entered at time when they should be executed concurrently. Answer **b** is wrong because it states an improper level control band for SRV use. Answer **d** is wrong because of the wrong level band. The limits used in answer **d** were derived based on what level band would have to be controlled on wide range to result in a narrow range band of +30 to -30 inches, if narrow range went that low. The intent of the band is not dependent on any single range of instrumentation. Operations Philosophy requires controlling the prescribed band, in this case +30 to -30 inches as indicated on the panel from which level is being controlled, in this case, the RSP. If controlling RCIC from H13-P601, a wide range band of +30 to -30 inches would be used, regardless of pressure. With the upper limit held at +30 inches wide range, as reactor pressure lowers, margin to RCIC Level 8 (narrow range) is gained.

Maintaining -30 inches wide range as the lower limit preserves margin to Level 2 (wide range).

KA 295016 A2.02
Ability to determine and/or interpret the following as they apply to CONTROL ROOM ABANDONMENT: **reactor water level**

10CFR55.43(b)(5)

05-1-02-II-1 step 3.7.1
02-S-01-27 step 6.6.8d

Difficulty 2

SRO 3

(refer to attached TS/TR 3.4.10)

The plant is in Mode 4 after an unplanned scram.

Two hours ago, a leak from the seal of Residual Heat Removal (RHR) pump A, while it was operating in shutdown cooling, caused reactor level to fall to 0 inches narrow range.

Reactor water level has been restored to +5 inches narrow range.

Regarding Shutdown Cooling only, which of the following should be performed in this condition?

a. Direct starting RHR B in Shutdown Cooling.

b. Direct starting Alternate Decay Heat Removal System (ADHRS).

c. Enter Technical Specification 3.0.3.

d. Enter Technical Requirements Manual 6.0.1.

Ans: d

All answers are plausible since they are contingencies for loss of shutdown cooling (SDC). TS LCO 3.4.10 requires one SDC loop be in operation in Mode 4, although neither SDC loop may be in operation for up to 2 hours per 8 hour period. With reactor level below Level 3, +11.4 inches, the common suction line for SDC would be automatically isolated and neither SDC nor ADHR would be available to be started, making answers **a** and **b** wrong. Answer **c** is wrong because TS 3.0.3 only applies to Modes 1, 2, and 3. TRM 6.0.1 is the correct action if a TS Action cannot be met in Mode 4, answer **d**.

KA 295021 A2.03
Ability to determine and/or interpret the following as they apply to LOSS OF SHUTDOWN COOLING: **reactor water level**

10CFR55.43(b)(2)

TS 3.4.10, TS 3.0.3, TR 6.0.1

GLP-OPS-TS001 Obj. 14, 34
05-1-02-III-5

Difficulty 2

SRO 4

An ATWS is in progress following a LOCA in the drywell.

Only Low Pressure Core Spray (LPCS) and Residual Heat Removal (RHR) A are available for injection but are overridden off.

Reactor power is 6%.

Reactor water level is -195 inches and slowly falling.

Reactor pressure is 350 psig and stable.

Suppression Pool temperature is 120°F and rising.

Containment pressure is 9 psig and is in the Unsafe Zone of the Pressure Suppression Pressure Curve.

The best use of LPCS and RHR A to minimize offsite radiation releases in this situation is to:

a. Execute EP-3 and place RHR A in Suppression Pool Cooling. Execute EP-2A and line up LPCS for injection to restore reactor level above -191 inches and avoid emergency depressurization.

b. Execute EP-2A and emergency depressurize the reactor. Then, inject with LPCS and with RHR A through RHR A SHUTDN CLG RTN TO FW E12F053A to restore reactor level above -191 inches.

c. Execute EP-2A and emergency depressurize the reactor. Slowly inject with RHR A, only, through RHR A SHUTDN CLG RTN TO FW E12F053A to restore reactor level above -191 inches. Leave LPCS overridden.

d. Execute EP-3 and place RHR A in Containment Spray per EP-3. Execute EP-2A and line up LPCS for injection into the reactor to restore reactor level above -191 inches.

Ans: c

All answers are plausible since they involve EP contingency actions that ultimately function to limit offsite releases to protect the public. The foremost goal of the EPs is to protect the fuel clad. Protecting containment is secondary. Therefore, if needed for adequate core cooling, RHR should be used for reactor level control regardless of suppression pool temperature or PSP concerns while executing the EPs. During ATWS conditions, emergency depressurization in order to use systems which inject outside of the shroud is preferred over use of systems which inject inside the shroud due the potential fuel damage which could occur. RHR A via E12F053A injects outside of the shroud. LPCS injects directly into the shroud. Therefore, emergency depressurization and injection with RHR A via E12F053A is preferred over injection with LPCS, even though LPCS could maintain level at the stated reactor pressure without emergency depressurization. Answers a and d are wrong because EP-3 only allows use of RHR in suppression pool cooling and/or containment spray if adequate core cooling can be assured by some other injection system. In this question, only RHR A and LPCS are available to ensure adequate core cooling, and LPCS should not be used if RHR A via E12F053A can restore and maintain level above -191 inches. Answer b is wrong because LPCS should not be used if RHR A via E12F053A can restore and maintain level above -191 inches.

KA 295024 - High Drywell Pressure
Generic 2.3.11 - Ability to control radiation releases

10CFR55.43(b)(5)

05-S-01-EP-2A
05-S-01-EP-3
GGNS EP Technical Bases for EP-3 steps SPT-2, CNT-7 and EP-2A step L-6, Table 4

GG-1-LP-RO-EP02A Obj. 2

Difficulty 2

Modified bank question GGNS-NRC-00918

SRO 5

(refer to HCTL curve)

The plant was at 100% power when two Safety Relief Valves failed open and unable to be closed.

The Reactor Mode Switch was placed in SHUTDOWN.

The following conditions exist:

Reactor power	8% and stable
Reactor pressure	900 psig and stable
Suppression pool temperature	150°F and rising
Suppression pool level	18.9 ft and slowly rising

For the stated conditions, the Control Room Supervisor should first direct:

a. lowering suppression pool level per EP-3.

b. lowering reactor pressure per EP-2, exceeding cool down rate limitations only if necessary.

c. lowering reactor pressure per EP-2, by fully opening Main Bypass Control Valves.

d. immediate Emergency Depressurization of the reactor per EP-3.

Ans: b

This question discerns proper understanding of EP wording and strategies for high reactor pressure with respect to suppression pool Heat Capacity Temperature Limit (HCTL). All answers are plausible since they involve pressure control strategies employed by the EPs. For these conditions, containment is being jeopardized due to diminishing margin to HCTL. The HCTL limit for this reactor pressure and suppression pool level is approximately 158°F. Margin to HCTL is approximately 8°F. EP-2A step P-1 allows lowering reactor pressure to provide additional operating margin to HCTL if *SP temperature cannot be maintained in the HCTL safe zone.* This is an SRO decision step. EP-3 step SPT-5 requires emergency depressurization if *SP temperature and RPV pressure cannot be maintained in the HCTL safe zone.* The technical basis for this step states that emergency depressurization should be performed if SP temperature *still* cannot be maintained below HCTL, after lowering reactor pressure. That is why answer **b** is correct and answer **d** is incorrect. Emergency depressurization would result in a more rapid cooldown than a more controlled pressure reduction, which is undesirable. Answer **a** is plausible because 18.9 feet is above the EP-3 entry condition, but it is wrong because HCTL is more limiting for the given condition than high SP level, and lowering SP level would cause a reduction in margin to HCTL. Answer **c** is incorrect because EP-2A step P-1 allows exceeding the 100°F/hr cool down rate limit, but the bases state only to the extent necessary to control RPV pressure within the HCTL safe zone. Fully opening main bypass valves at 8% power would quickly result in exceeding the100°F/hr cool down rate limit.

KA 295025 A2.03

Ability to determine and/or interpret the following as they apply to HIGH REACTOR PRESSURE: **suppression pool temperature**

10CFR55.43(b)(5)

05-S-01-EP-2A
05-S-01-EP-3
GGNS EP Technical Bases for EP-3 step SPT-5, EP-2A step P-1

GG-1-LP-RO-EP02A Obj. 2

Difficulty 2

Modified bank question GGNS-NRC-00910

SRO 6

The plant is in Mode 1.

The actuator diaphragm for PCW RTN FM SMPL WTR CLRS/CTMT CLRS P71F148 has failed, causing the valve to close.

Average containment air temperature is 100°F and rising.

To control containment temperature, the Control Room Supervisor should:

a. Enter EP-2 and immediately direct a manual reactor scram, then enter EP-3.

b. Enter EP-3 and direct a manual reactor scram only when temperature rises above 185°F.

c. Enter EP-3 and direct containment high volume purge be placed in operation per 04-1-03-M41-3.

d. Enter EP-3 and direct containment low volume purge be placed in operation per 04-1-03-M41-2.

Ans: d

EP-3 must be entered when containment temperature reaches 95°F. Then, if containment temperature cannot be controlled below 185°F, EP-2 should be entered to shut down the reactor and containment sprays initiated for cooling before the containment temperature design limit, 185°F is reached. Low Volume Purge is an alternate method for cooling the containment atmosphere. Answer **a** is plausible since a scram is conducted per EP-2 and in a trainee's simulator experience, EP-2 usually takes priority, but it is wrong since the entry condition met is for EP-3. Answer **b** is plausible since it reflects actions listed in the containment temperature control leg of EP-3, but it are wrong because there exists ample margin to attempt other methods for controlling temperature before creating unnecessary transients or challenges and because the listed contingency should be attempted before reaching the design limit if time permits. TS 3.6.1.5 allows containment temperature above 95°F to exist for 8 hours before a 12 hour shutdown statement would be entered, allowing plenty of time for problem correction or a controlled plant shut down. Answer c is plausible because venting of containment is proper, but not with the 20 inch purge valves for temperature control. 04-1-03-M41-3 step 4.3.2 only allows their use for controlling explosive gases, low oxygen concentrations, radiological concerns, toxic atmospheres, pressure control, or required testing. Answer **d** is correct because ample margin to attempt other methods for controlling temperature and to restore the failed valve or otherwise open it exists. Prerequisite 4.2 of EPI 04-1-03-M41-2, Containment Low Volume Purge, specifically states one occasion for use is when an SRO has determined it is required to mitigate elevated containment temperature.

KA 295027 A2.01
Ability to determine and/or interpret the following as they apply to HIGH CONTAINMENT TEMPERATURE (MARK III): **containment temperature: Mark III**

10CFR55.43(b)(5)
EP-3 step CNT-3
04-1-03-M41-2 step 4.2.2
04-1-03-M41-3 step 4.3.2
GLP-OPS-EP03 Obj. 5, 6

GLP-OPS-M4100 Obj. 14.3
Difficulty 2

SRO 7

(TS 3.9.8 and TS 3.9.9 are attached)

Core fuel shuffle is being performed in Mode 5.

Standby Gas Treatment System B is in standby.

Reactor Recirc Pumps are stopped.

Residual Heat Removal (RHR) A system is tagged out of service for major maintenance.

Alternate Decay Heat Removal (ADHR) is in standby.

RHR B had been operating in Shutdown Cooling for 12 continuous hours when a large leak developed from the RHR B pump seal, and RHR B was shutdown and RHR PMP B SUCT FM SHUTDN CLG E12F006B closed.

Reactor water level is 6 inches below the top of the adjustable skimmers in the upper containment pool due to the leak.

The decay heat load is beyond the capabilities of Fuel Pool Cooling and Clean Up and Reactor Water Clean Up.

Which one of the following would satisfy the minimum action required by Technical Specifications for this event?

a. Place ADHR in operation within 2 hours

b. Start at least one Reactor Recirc pump in slow speed within 1 hour and monitor reactor coolant temperature once per hour

c. Within 1 hour place ADHR in operation and initiate action to restore Secondary Containment functions operable.

d. Enter TRM 6.0.1 and develop a plan to restore RHR B to operable status.

Ans: c

This question involves an operational condition change from High Water Level to Low Water Level during refueling, with High Water Level defined by level being even with the top of the adjustable skimmers in the upper containment pool. This means going from a condition where only one decay heat removal system is required operable to a condition where two are required operable, from TS 3.9.8 requirements to TS 3.9.9 requirements. For the stated conditions, TS 3.9.9 is not met. Neither can TS 3.9.9 action A be met, since RHR A and B are unavailable, and decay heat is too much for FPCCU and RWCU. Therefore, TS 3.9.9 conditions B and C apply. With SGTS B already operable, this requires initiating action to restore secondary containment penetration operability and placing ADHR in operation for core circulation within 1 hour, as reflected by answer c. Answer a is plausible if one misapplies the LCO Note allowing for no decay heat removal system to be in operation for up to 2 hours per 8 hour period. Answer b might be chosen if one does not recognize the condition change from high water level to low water level. Answer d might b chosen if one misapplies TRM 6.0.1, which is to be entered only if there is no TS/TR action that can be performed in the given condition. In this case, establishing secondary containment is provided as an alternate action when only one decay heat removal system is functional and is not optional.

KA 295031 A2.01
Ability to determine and/or interpret the following as they apply to REACTOR LOW WATER LEVEL: **reactor water level**
10CFR55.43(b)(2)

TS 3.9.8
TS 3.9.9
06-OP-1000-D-0001 Att III, Data Sheet 1, item 19
GLP-OPS-TS001 Obj. 34

Difficulty 2

SRO 8

(portions of Loss of Feedwater Heating ONEP 05-1-02-V-5 and TS 3.2.2 are attached)

The plant was at 100% power when extraction steam to High Pressure Feedwater Heaters 5A and 6A isolated and Heater Drain Pumps tripped.

It is estimated to take 12 hours to restore these features.

Reactor power is 68%.

Average feedwater temperature is 320°F.

MFLCPR indicated on the CYCLOPS computer display for a current MON edit is 0.6781.

The minimum required action for this condition is to:

a. reduce power to ≤ 25% within 6 hours.

b. reduce power to ≤ 25% within 4 hours.

c. initiate a potential LCO for MCPR monitoring.

d. declare all FCTR channels inoperable and enter TS 3.0.3.

Ans: a

Operation is in Region II of the FEEDWATER TEMPERATURE VS. CORE POWER curve of the Loss of Feedwater Heating (LOFWH) ONEP. MCPR limits programmed in Powerplex are not valid if feedwater temperature is not in Region I of the FEEDWATER TEMPERATURE VS. CORE POWER curve of the LOFWH ONEP (i.e. for a feedwater temperature reduction of > 50° F). Therefore, if operation is determined to be <u>not</u> in Region I of the FEEDWATER TEMPERATURE VS. CORE POWER curve of this ONEP, MCPR must be declared to be <u>not</u> met per TS 3.2.2 irrespective of the MFLCPR value indicated by MON/Powerplex. Also, with core power > 30% rated power, Stability region boundaries of the Power-to-Flow Map are only valid for operation in Region I of the FEEDWATER TEMPERATURE VS. CORE POWER curve of this ONEP. If operation is determined to be not in Region I of the FEEDWATER TEMPERATURE VS. CORE POWER curve of the LOFWH ONEP, FCTR card algorithms (setpoints) no longer meet the Tech Spec allowable values specified in the COLR (see TS 3.3.1.1-1 note (b) and TS Bases 3.3.1.1 function 2.d).

Actions to be taken for MCPR per TS 3.2.2 (i.e. restoring feedwater temperature to return operation to Region I or reducing core power to <25% within 6 hours) bound the 12 hour action required by TS 3.3.1.1 (i.e to restore FCTR allowable values or reduce power to <30%).

Answer a is correct because it reflects TS required action 3.2.2 A and B and timing. Answer b is plausible if one assumes the foreknowledge that normal conditions cannot be restored within the 2 hour time frame of TS 3.2.2 A, then action B is required to be entered. Answer c is plausible since the stated value for MCPR with respect to its limit is alright, but the limit itself is fraudulent. Answer d, TS 3.0.3, is plausible ` all FCTR card setpoints are affected, but TS 3.3.1.1 allows actions for inoperable channels to be delayed for 12 hours in this situation.

KA 295014 A2.04
Ability to determine and/or interpret the following as they apply to INADVERTENT REACTIVITY ADDITION: **violation of fuel thermal limits**

10CFR55.43(b)(2)

05-1-02-V-5
TS 3.2.2, TS 3.3.1.1-1 function 2.d note b

GLP-OPS-ONEP Obj. 36.0
GLP-OPS-TS001 Obj. 34

Difficulty 2

SRO 9

(a portion of 10-S-01-11 and EPP 10 mile sector map are attached)

A General Emergency due to failure of fission product barriers has been declared, and a release is in progress.

PDS indicates wind direction at elevation 162 feet is from 220° at a speed of 10 mph.

Plant personnel should be directed to evacuate to the ____________ reception center via exiting the _____________ gate.

a. Hazlehurst High School; north

b. Hinds Community College Utica Campus; south

c. Warren Central High School; north

d. Natchez High School; south

Ans: d

This function is performed by an SRO. This question tests knowledge of the orientation of wind direction channels and evacuation routes to minimize exposure to the public during radioactive releases. All answers are plausible because personnel could be routed to the proposed reception centers via the listed gates given different release conditions. For the wind direction given, sectors affected by the radioactive plume would be BCD. The route listed that would minimize or eliminate exposure to the plume is reflected by answer **d**. All of the other answers list paths that would cause travel within the plume which is to be avoided.

KA 295017 A2.05
Ability to determine and/or interpret the following as they apply to HIGH OFFSITE RELEASE RATE: **meteorological data**

10CFR55.43(b)(4),(5)

10-S-01-11 step 6.1.2.b(2)

GLP-EP-EPTS6 Obj. 4

Difficulty 2

SRO 10

(portions of 10-S-01-1 and 10-S-01-11 are attached)

The plant was operating at rated power on a clear summer day when an event occurred.

The reactor is at normal water level and rated pressure.

The following conditions exist in the auxiliary building:

RCIC Room Area Radiation Monitor	85,000 mr/hr
Drywell and Containment Radiation	500 mr/hr
Site Boundary Dose Rate	90 mr/hr TEDE

As a general guideline, the minimum areas recommended for evacuation, excluding facilities required for emergency response, based only on these conditions are:

a. Auxiliary Building and Containment

b. Power Block and Protected Area

c. Owner Controlled Area and Protected Area

d. Owner Controlled Area and Protected Area and 2 mile radius and 5 miles down wind

Ans: c

The given RCIC Room radiation exceeds the SAE/GE limit of Table F1 of Att. I of 10-S-01-1. This condition exceeds EAL RC3 and PC3 as potential loss of the RCS and actual loss of Primary Containment fission product barriers. Therefore, EAL FS1 requires classification of a Site Area Emergency. A General Emergency is not warranted because no fuel cladding EAL has been exceeded. The drywell radiation level threshold for an actual loss of the fuel clad is 3000 mr/hr, whereas in the stem the drywell radiation level is stated as only 500 mr/hr. 10-S-01-11, Evacuation of Onsite Personnel, step 6.1.1 states as a general guideline for a Site Area Emergency, the Owner Controlled Area and Protected Area should be evacuated. Answers a and b are plausible because they are most affected by the steam line break and might be considered for Limited Evacuations, but they wrong because the listed areas are only portions of that which is required. Answer d is plausible because it would be required for a General Emergency, but it is wrong because conditions do not meet the GE EAL.

KA 295033 - High Secondary Containment Area Radiation Levels
Generic 2.1.25 - Ability to obtain and interpret station reference material such as graphs / monographs / and tables which contain performance data

10CFR55.43(b)(4),(5)

10-S-01-1 Att. 1 pgs 10, 11
10-S-01-11 step 6.1.1
GLP-EP-EPTS6 Obj. 1, 2

Difficulty 2

SRO 11

(TS 3.4.9 and 3.4.10 are attached)

The plant is in Mode 4 preparing for start up.

Alternate Decay Heat Removal System (ADHRS) is in operation.

Plant management desires securing ADHRS to allow the reactor to heat up and pressurize to 60 psig using decay heat to establish operation on bypass valves to warm the steam lines.

RHR PMP A SUCT FM SHUTDN CLG E12F006A is tagged out of service, de-energized in the closed position due to actuator gear failure.

Which of the following describes Technical Specifications allowances and/or restrictions regarding entry into Mode 3?

a. Mode 3 may be entered while RHR A remains tagged out of service.

b. Mode 3 may not be entered until RHR A pump is restored OPERABLE.

c. Mode 3 may be entered by declaring RHR B the alternate method of decay heat removal system for RHR A.

d. Mode 3 may be entered by declaring ADHRS the alternate method of decay heat removal system for RHR A.

Ans: a

The answer to this question can be found in the Bases for TS 3.4.9. With E12F006A non-functional and closed, Shutdown Cooling A is inoperable; however, RHR A is still operable for ECCS and all other required TS functions. TS 3.4.9 ACTIONS note says that TS3.0.4 is not applicable. TS B3.4.9 (bases) says that mode change is allowed due to system redundancy, low consequences at low reactor pressure, and low probability of occurrence of an event. Therefore, mode change may be made with no reliance on any factor even when the LCO or its actions are not met, so answer **a** is correct. Answer **b** is plausible since RHR A operability is required to meet LCO 3.4.9. Answer **b** is wrong because it states a condition that is not necessary, RHR A operability. Answer **c** is credible since it could be assumed that TS3.4.9 action A would have to be met to change modes, but this is wrong, and RHR B cannot serve as an alternate for RHR A while RHR B is required as a separate operable system itself. This was included because this concept has historically proven a point of confusion among operators. Answer **d** is plausible for the same reason as **c**, but ADHRS operation is prohibited in Mode 3.

KA 205000 - Shutdown Cooling
Generic 2.2.25 - Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits

10CFR55.43(b)(2)

TS 3.4.9, B3.4.9, TS 3.4.10, TS 3.0.4

GLP-OPS-TS001 Obj. 17, 34

Difficulty 2

SRO 12

The plant was at 65% power when a fault in the Baxter Wilson switchyard caused breakers J5236 and J5228 in the GGNS switchyard to open and the EHC Load Reject Relay to actuate.

Breaker J5232 has not yet tripped.

Which of the following describes the effect this will have on the Reactor Protection System and proper response by the Control Room Supervisor?

a. The reactor will scram directly from low Turbine Control Valve secondary fluid pressure. Enter the Loss of AC Power ONEP, and direct transferring busses 12HE and 13AD to Service Transformer 21 **before** the generator trips.

b. The reactor will scram directly from high reactor pressure. Enter the Loss of AC Power ONEP, and direct transferring busses 12HE and 13AD to Service Transformer 21 **before** the generator trips.

c. The reactor will scram directly from low Turbine Control Valve secondary fluid pressure. **When** the generator trips, enter Loss of AC Power ONEP and direct transferring busses 12HE and 13AD to Service Transformer 21.

d. The reactor will scram directly from high reactor pressure. **When** the generator trips, enter Loss of AC Power ONEP and direct transferring busses 11HD and 14AE to Service Transformer 11.

Ans: c

This question requires knowledge of switchyard electrical distribution, turbine control system load reject circuitry, and <u>timing</u> for implementing Loss of AC Power ONEP actions for imminent loss of service transformer power.

For the Load Reject Relay to actuate, generator output must decrease by 1123 MWe. The initial condition of 65% power equates to only about 892 MWe; therefore, the Load Reject Relay will not actuate. However, with instantaneous loss of all load, turbine control valve secondary fluid pressure will rapidly drop to maintain reactor pressure. Fluid pressure will drop to less than the RPS trip setpoint of 46 psig TCV fluid pressure. With reactor power above the trip bypass setpoint of approximately 27%, the reactor will scram directly due to turbine control valve fast closure. Also, a generator trip does not occur directly from load rejection. In this question, the initiating event is opening of the two breakers that connect the main generator to the east and west switchyard busses. The generator remains connected to Service Transformer 11, which normally supplies busses 12 and 13, via breaker J5232. Even after the scram, the generator would continue to rotate to supply power to in-house loads until speed decreased, due to loss of steam and inertia, to the underfrequency trip setpoint. Some amount of time would be available to swap busses 12 and 13 to ST21 before power to ST11 was lost, which would be tempting due to the complications, such as loss of reactor feed pumps, that could be averted by this action. However, this could result in equipment damage to rotating equipment supplied from busses 12 and 13 due to the potential for an out of phase power supply transfer, which is make before break, and is therefore improper. The transfer should only be performed after the generator has tripped and busses 12 and 13 are de-energized. This makes answer c correct.

All other answers are plausible because they consider the same attributes pertaining to the correct answer, but each reflects faulty knowledge or reasoning. Answer a is incorrect because it states the Load Reject Relay would trip and it states power supplies should be swapped before the generator trips. Answer b is incorrect because it reflects the scram would be due to high reactor pressure, a plausible consequence of not receiving a scram from TCV fast closure, and it states power supplies should be swapped before the generator trips. Answer d is incorrect because it reflects busses 11 and 14 are normally fed by ST11, but they are normally fed from ST21.

KA 212000 A2.15
Ability to **(a)** predict the impacts of the following on the REACTOR PROTECTION SYSTEM; <u>and</u> **(b)** based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: **load rejection**

10CFR55.43(b)(5)

TS B3.3.1.1 function 10
GGNS Root Cause Evaluation Report 00-32 for Scram #99
E0001

GLP-OPS-N3202 Obj. 3.3
GLP-OPS-N3202 Obj. 3.1, 3.4, 3.5, 3.6

Difficulty 2

SRO 13

(TS 3.8.4, TS 3.8.7, TRM 6.7.3, and 02-S-01-17 Att. II are attached)

The plant is in Mode 1.

Battery charger 1A4 is in equalize.

Standby Service Water pump A was tagged out of service for planned maintenance at 0600.

At 1000, breaker 52-164101 for Safeguard Switchgear and Battery Room air handling unit B, 1Z77B001B, trips due to a fault in the fan motor windings.

Engineering is unable to assure operability of affected equipment under all required conditions with the current outside air temperature.

Which of the following actions is required for this condition?

a. AC and DC electrical distribution systems are operable. When switchgear and/or battery room temperatures exceed 104°F, then declare requirements for AC and/or DC distribution not met. Then, enter TS 3.0.3, and begin a controlled plant shutdown per 03-1-01-2 within 1 hour.

b. AC and DC electrical distribution systems are inoperable. Enter TS 3.0.3 and commence a controlled plant shutdown in accordance with 03-1-01-2 within 1 hour.

c. AC and DC electrical distribution systems are inoperable. Restore either SSW A or 1Z77B001B OPERABLE within the next 2 hours; otherwise, begin a controlled plant shutdown in accordance with 03-1-01-2.

d. AC and DC electrical distribution systems are operable. Enter TRM 6.7.3 and when switchgear and/or battery room temperatures exceed 134°F, then declare all required equipment in the affected area inoperable within 4 hours.

Ans: b

Battery charger 1A4 being in equalize indicates charging is in progress. SOI 04-1-01-Z77-1 step 3.2 states ventilation must be in operation during charging, due to potential build up of hydrogen gas, a combustion hazard. When 1Z77B001B trips, 1Z77B001A would auto start to maintain air circulation; therefore, charging operations could continue. This question involves a safety function determination using 02-S-01-17. With 1Z77B001B, division 2, inoperable, only 1Z77B001A would remain available to cool the switchgear and battery rooms. 04-1-01-Z77-1 step 3.7c states Z77 should be treated as a supported system feature when implementing TS 3.0.6. SSW A supports operation of 1Z77B001A, so it would be considered inoperable per TS 3.0.6 with SSW pump A inoperable. Step 3.7b requires entering LCOs TS 3.8.4, DC Sources - Operating, and TS 3.8.7 Distribution Systems – Operating when both divisions of Z77 are inoperable. So for the stated conditions, both divisions of ESF batteries, AC distribution, and DC distribution would be declared inoperable. DC sources and DC distribution would be the most limiting. TS 3.8.4 for DC Sources does not include a condition for both division 1 and 2 batteries inoperable, therefore TS 3.0.3 would apply. This is consistent with TS 3.8.7 Condition E for division 1 and 2 DC Distribution inoperable. Therefore, answer **b** is correct.

Answer **a** is plausible since temperature of the affected rooms is restricted by TRM 6.7.3, and it would be reasonable to base the effect on operability on environmental qualification. However, this could only be accomplished through the condition reporting process and analysis for GL 91-18 would be done by engineering. Answer **a** is incorrect because it states TS 3.0.3 would be entered only after exceeding the room temperature limit and because it states charging should be secured. Answer **c** is plausible since it reflects TS 3.8.7 Action B and could be reached if the wording of TS 3.8.7 Condition B is misunderstood as affecting divisions 1 and 2. Answer **c** is wrong since TS 3.8.7 Condition B is applicable if only one DC division is affected. Answer **d** is plausible since it reflects the requirements of TR 6.7.3 Condition C for high area temperatures, but it is wrong because it states the electrical distribution systems are operable.

KA 263000 A2.02
Ability to **(a)** predict the impacts of the following on the DC ELECTRICAL DISTRIBUTION; and **(b)** based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: **loss of ventilation during charging**

10CFR55.43(b)(2)

04-1-01-Z77-1 step
TS 3.0.6
TS 3.8.4
TS 3.8.7
TRM 6.7.3
TS 3.0.3
02-S-01-17 Att. II

GLP-OPS-TS001 Obj. 19, 34, 35

Difficulty 2

SRO 14

Division 2 Diesel Generator is paralleled to bus 16AB for monthly operability testing when regional damage to the 500KV distribution system causes grid voltage to begin swinging from 501 KV to 490 KV.

Grid frequency begins swinging from 59.7 to 61 Hz and does not cease.

Considering only the effect on bus 16AB voltage and frequency, which of the following describes the effect this will have on Division 2 Diesel Generator and required response by the Control Room Supervisor?

a. Division 2 Diesel Generator will remain paralleled to bus 16AB. Enter the Loss of AC Power ONEP, and direct Division 1 and 3 Diesel Generators be synchronized to their respective busses and all ESF busses separated from offsite power.

b. Bus 16AB offsite feeder will trip and Division 2 Diesel Generator output breaker will remain closed. Enter the Loss of AC Power ONEP, and verify Division 1 and 3 Diesel Generators have automatically started and are carrying their respective busses.

c. Bus 16AB offsite feeder will trip and Division 2 Diesel Generator output breaker will trip and then re-close. Direct Division 1 and 3 Diesel Generators be synchronized to their respective busses and separated from offsite power per 04-1-01-P75-1.

d. Division 2 Diesel Generator output breaker will trip. Enter 04-1-01-P75-1 and direct a normal shutdown of Division 2 Diesel Generator. Enter TS 3.8.1 for Division 2 Diesel Generator inoperable.

Ans: a

Offsite power feeder breakers to division 1 and 2 ESF busses trip on 90% BUV for 9 seconds. This is 3744 VAC, which equates to a grid voltage of 450 KV, well below the stated grid voltage; therefore, no BUV would be sensed. Div 2 DG output breaker trip setpoints for the stated conditions, grid voltage swinging 98% to 100.2%, are not reached. Therefore, DG12 would remain connected to bus 16AB in parallel with offsite power. ONEP 05-1-02-I-4, Loss of AC Power, defines grid instability as voltage <491 kV and voltage and frequency swinging. Step 3.4.6 requires supplying all 3 ESF busses from their DGs alone. This is reflected in the correct answer, **a**. Answer **b** is plausible since the offsite feeder could trip at a higher corresponding grid voltage than would the DG output breaker, which trips at 70% BUV only. It is wrong because it states or implies the offsite feeders to ESF busses would trip. Answer **c** is plausible since the subject breakers do trip on undervoltage, but is wrong since the setpoints are not reached. Answer **d** is plausible since DG12 output breaker does trip on underfrequency and undervoltage. It is incorrect since the trip setpoints are not reached and because this would not make the DG inoperable.

KA 264000 A2.05
Ability to **(a)** predict the impacts of the following on the EMERGENCY GENERATORS (DIESEL); <u>and</u> **(b)** based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: **synchronization of the emergency generator with other electrical supplies**

10CFR55.43(b)(5)

05-1-02-I-4 step 3.4
GLP-OPS-ONEP Obj. 2
Difficulty 2

SRO 15

(01-S-06-3 is attached)

For which of the following Temporary Alterations to the Instrument Air System may the requirement to temporarily revise controlled drawings and/or procedures be conditionally waived?

a. installing a bypass around instrument air branch filter station P53D057 in the Turbine Building during Mode 4

b. raising the setpoint for ADS Booster Compressor supply air regulators beyond that specified by procedure in Mode 3

c. routing Construction Water to cool the Instrument and Plant Air Compressors during Mode 5

d. supplying temporary power to Unit 1 Instrument Air Compressor from bus 14AE during Mode 4

Ans: d

SROs are responsible for the review of Temporary Alteration installation packages to ensure required preliminary changes have been prepared and are ready for issue before authorizing Temporary Alteration installation. 01-S-06-3, Control of Temporary Alterations, step 6.1.2 states that Temporary Alterations associated with bus outages do not require temporary changes to drawings or procedures provided certain conditions are met. Therefore, answer d is correct. All other answers are plausible because they describe changes for which Temporary Alterations would be required, but they are wrong since they list alterations that are not bus outage related.

KA 300000 - Instrument Air System
Generic 2.2.14 - Knowledge of the process for making configuration changes

10CFR55.43(b)(3)

01-S-06-3 step 2.5

GLP-OPS-PROC Obj. 9.9

Difficulty 1

SRO 16

(TS 3.9.1 and TR 6.9.3 are attached)

The plant is in Mode 5.

All control rods are inserted except for four control rods that are removed per Technical Specification 3.10.6.

While new fuel loading is in progress, the Main Hoist Fuel Loaded interlock fails to produce its safety function due to load cell failure just as a fuel bundle begins lowering toward the core.

Which of the following describes what, at minimum, is required with respect to Refueling Platform operation?

a. Refuel Platform lateral travel will be inhibited. Fuel handling using the Refueling Platform must be suspended, and it must not be used to handle any fuel or control rods for any reason until the problem is repaired and SR 3.9.1.1 performed.

b. The associated control rod withdrawal block will clear. In-vessel fuel movement must be suspended per TS requirements. The Refueling Platform may be used to load new fuel after a control rod withdrawal block is inserted and all control rods are verified to be inserted.

c. The Refuel Platform can exit the core zone but will be unable to re-enter it. The Refueling Platform may only be used to place the loaded bundle in a safe location in the containment fuel storage racks. Then fuel loading may resume when the problem is repaired and SR 6.9.3.5 performed.

d. Refuel Platform travel is now unimpeded with respect to any control rod position. In-vessel fuel movement must be suspended, except to place the new fuel bundle in a safe location. In-vessel fuel movement may only begin after the problem is repaired and SR 3.9.1.1 and SR 6.9.3.5 are performed.

Ans: b

This requires determination of the proper TS action. To determine the correct action, one must be able to predict the impact of the failure. In this case, a required control rod block governed by TS 3.9.1is disabled by the failure. TS 3.9.1 actions provide two alternative actions for this condition: A.1) suspend in-vessel fuel movement until OPERABILITY is restored, or A.2) insert all control rods associated with fueled cells and then verify all rods inserted. Answer b reflects the second option. The impact portions of all incorrect answers are plausible because the hoist loaded interlock is part of the platform travel interlock logic. Answer a contingency is plausible because the answer references associated TS 3.9.1 action A.1. But the answer is wrong because Platform motion would not be impeded and because it prevents using the platform for handling any fuel when ex-vessel fuel handling would be permitted by TS. Answers c and d contingencies are plausible since the interlock is related SR 3.9.1.1 and SR 6.9.3.5 which would need to be satisfied for retest. TRM 6.9.3.5 Answer c is wrong because the Refuel Platform would be able to re-enter the core zone and because it only lists SR 6.9.3.5 and not 3.9.1.1. Answer d is wrong because it states in-vessel fuel movement can begin only after the problem is repaired and retested, and it does not consider allowed in-vessel fuel movement under TS 3.9.1 action A.2.

KA 234000 A2.01
Ability to **(a)** predict the impacts of the following on the FUEL HANDLING EQUIPMENT; and **(b)** based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: **interlock failure**

10CFR55.43(b)(2)

TS 3.9.1
TR 6.9.3
06-OP-1C71-V-0002 Att. II

GLP-OPS-F1101 Obj. 7.1, 15.5, 15.6, 15.7, 29.2

Difficulty 2

SRO 17

The plant is in Mode 3 at rated pressure following a steam line break in the Reactor Core Isolation Cooling (RCIC) room which is unable to be isolated.

RCIC room temperature is above the maximum safe temperature.

Main Steam Isolation Valves (MSIVs) are open.

Steam from the leak is also causing Main Steam Tunnel temperature to rise, and the Control Room Supervisor anticipates Main Steam Tunnel temperature will exceed its maximum safe value within the next five minutes.

With respect to reactor pressure, the CRS should:

a. immediately execute Emergency Depressurization per EP-4 using Safety Relief Valves.

b. immediately direct fully opening main bypass valves to completely depressurize the reactor per EP-2.

c. direct lowering pressure to 450 psig to 600 psig in accordance with 02-S-01-27, Operations Philosophy, using main bypass valves per EP-2.

d. wait until Main Steam Tunnel temperature reaches maximum safe, then execute Emergency Depressurization per EP-4 using Safety Relief Valves.

Ans: b

EP-2 step F-1 requires rapid depressurization of the reactor using bypass valves if emergency depressurization is anticipated to minimize the amount of energy which will be ultimately introduced into containment, as reflected in answer b. All other answers are plausible because they are pressure control strategies for varying conditions which could be encountered during EOP execution. Answer a is wrong because emergency depressurization is not allowed by EP-4 until 2 max safe temperatures are exceeded, not just expected. Answer c is wrong because EP-2 step F-1 is a requirement to rapidly depressurize. EOP bases states this is accomplished by fully opening the main bypass valves to fully depressurize. Answer d is wrong since this would result in a higher energy deposit in containment which could be avoided.

KA 239001 - Main and Reheat Steam
Generic 2.4.6 - Knowledge of symptom based EOP mitigation strategies.

10CFR55.43(b)(5)

EP-4, GGNS EP Technical Bases for EP-4 step 10
EP-2, GGNS EP Technical Bases for step P-1

GG-1-LP-RO-EP02 Obj. 7
GLP-OPS-EP04 Obj. 3b, 7

Difficulty 2

SRO 18

An ATWS is in progress with power 5% at 500 psig.

Reactor water level is currently being controlled inside the band -167 inches to -191 inches using two Condensate Booster Pumps via Start Up Level Control valve N21F513.

Then, the instrument air header completely ruptures in the Water Treatment Building.

If reactor power remains constant, the CRS can accomplish long term water level control by directing:

a. installation of EP Attachments 1 and 2 and directing level be maintained with Reactor Core Isolation Cooling (RCIC) because N21F513 will eventually fail closed.

b. use of N21-F040 and/or N21-F009A/B to control reactor water level in band per the Feedwater System Malfunctions ONEP because N21F513 will fail as-is.

c. installation of EP Attachment 4 and directing level be maintained with High Pressure Core Spray (HPCS) because Condensate System pumps will trip due to low flow.

d. installation of EP Attachment 12, Emergency Depressurization, and then level control using Residual Heat Removal systems because Condensate System pumps will trip due to low flow.

Ans: d

This question involves prioritization of EP contingency attachments, which are procedures unto themselves, and, thus, repair teams which is an SRO function. Loss of air causes all Condensate and Feedwater system minimum flow valves to eventually close, as well as the Startup Level Control Valve. So the overriding effect on Condensate system for a complete loss of air is loss of pumps due to low minimum flow. To stabilize level with power only 5%, there is not enough forward flow to provide minimum flow requirements, 1.0 mlbm/hr. With no minimum flow path, Condensate system pumps will sequentially trip. Condensate would only be available by successively re-starting pumps to establish slugs of flow resulting in level control of an uncontrolled fashion. 5% power is beyond the makeup capacity of RCIC and CRD. HPCS is not allowed until after emergency depressurization and exhaustion of EP-2A Table 4 systems.
All answers are plausible depending on assumptions regarding effects of loss of air and on knowledge of ATWS level control strategies apart from use of Condensate. Answer a is wrong because RCIC is only capable of flow rates corresponding to about 2% power. Answer b describes proper direction for failure of the Startup Level Control Valve, but is wrong since it assumes Condensate pumps would remain available. Answer c is wrong since HPCS cannot be used at the stated pressure and before preferred systems are used. Answer d is correct since level cannot be restored and maintained above -191 inches for the stated conditions, emergency depressurization will be required. Att. 12 must be installed to enable use of RHR through the shutdown cooling return lines as preferred.

KA 256000 A2.13
Ability to **(a)** predict the impacts of the following on the REACTOR CONDENSATE SYSTEM; and **(b)** based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: **loss of applicable plant air systems**

10CFR55.43(b)(5)

05-1-02-V-9 step 5.20, 5.22, 5.23
04-1-01-N19-1 steps 3.4, 3.6
05-S-02-EP-2A
GG-1-LP-RO-EP02A Obj. 2, 3d, 10d

Difficulty 2

SRO 19

The plant is in Mode 1.

The Shift Supervisor has just initiated an LCOTR for the Component Cooling Water Radiation Monitor due to planned maintenance by I&C.

Which of the following are required to be verbally notified of this LCOTR by the Shift Supervisor?

a. Duty Manager and Plant Chemistry

b. Radiation Protection and Plant Chemistry

c. Control Room Operator and Plant Chemistry

d. Control Room Operator and Radiation Protection

Ans: c

The Shift Supervisor is responsible for all notifications resulting from LCOTR initiation. 02-S-01-17, Control of Limiting Conditions for Operations, step 6.1.6 states the Control Room Operator should be verbally notified of entry into any LCOTR. Step 6.2.6 further states if another department must be relied upon to meet the required action of an LCOTR, that department must also be notified. TRM 6.3.1 requires periodic sampling of CCW when the radiation monitor is inoperable. Plant Chemistry performs this sampling duty. Therefore, answer c is correct. The Duty Manager is only required to be informed of entry into unplanned entry into LCOs. Radiation Protection is not responsible for internal monitoring of in-service cooling systems. All wrong answers are plausible because they reference persons/disciplines that do warrant notification under varying circumstances regarding system status.

KA Generic 2.1.14
Knowledge of system status criteria which require the notification of plant personnel.

10CFR55.43(b)(5)

02-S-01-17 steps 6.1.6, 6.2.6

GLP-OPS-TSLCO Obj. 2, 6

Difficulty 1

SRO 20

For occurrence of which of the following events at rated power would **Technical Specifications** require the CRS to initiate a plant shutdown the soonest?

a. total plugging of Control Rod Drive (CRD) drive water filters

b. Low-Low Set spurious initiation that cannot be reset

c. simultaneous failure of Low Pressure Core Spray and High Pressure Core Spray jockey pumps

d. Reactor Feed Pump B trip with Period Based Detection System (PBDS) A inoperable

Ans: b

All answers are credible since they necessitate action in less than one hour as required by TS. For answer a, plugging of CRD drive water filters would result in a loss of CRD charging water pressure and HCU accumulator faults. However, at rated pressure, operation is permitted for up to 20 minutes from declaration of the second inoperable accumulator associated with a withdrawn control rod per TS 3.1.5. For answer b, suppression pool temperature would rise approximately 10°F per minute. From the minimum administrative temperature limit of 70°F (listed in 04-1-01-E12-1 step 3.7) this would result in reaching 110°F, the TS 3.6.2.1 limit at which the Reactor Mode Switch must be immediately placed in Shutdown, within 4 minutes. Plant OE from 6/6/96 shows a manual scram was inserted at 105°F, reached within 2.5 minutes of a spurious Low-Low Set initiation. For answer c, the stated condition results in inoperability of HPCS and LPCS which requires entry into LCO 3.0.3 immediately per TS 3.5.1 condition H. However, TS 3.0.3 only requires action to initiate plant shutdown within 1 hour of entry. For answer d, a RFP trip and resultant Recirc flow control valve runback initiated from the highest allowable load line would at worst result in Restricted Region entry on the power-flow map. As long as PBDS B remained operable, plant shutdown would not be required, only immediate action to insert control rods to exit the Restricted Region. Therefore, answer b is correct because it is the most time limiting.

KA Generic 2.1.11
Knowledge of less than one hour Technical Specification action statements for systems.

10CFR55.43(b)(2)

TS 3.6.2.1
Reference: NRC Information Notice 96-42

GLP-OPS-TS001 Obj. 36

Difficulty 2

SRO 21

The Shift Supervisor does **not** release major maintenance work on:

a. Radial well 3 ventilation louvers

b. Containment chemistry sample station components

c. Security computer MUX involving turbine building fire zone information

d. Protected area Land Vehicle Barrier system

Ans: d

01-S-07-1, Control of Work, step 2.3.2 lists the exceptions to Shift Supervisor authority for releasing plant equipment. Step 2.3.2b lists non-fire protection related elements of the security system as excepted, of which the LVB is one. Therefore, answer d is correct. All other answers are plausible in that they represent plant equipment of diverse function and locale. Yet answers a, b, and c are wrong due to their potential effect on plant operation or requirements.

KA Generic 2.2.19
Knowledge of maintenance work order requirements.

10CFR55.43(b)(3)

01-S-07-1 step 2.3.2

GLP-OPS-PROC Obj. 23

Difficulty 1

SRO 22

The plant is in Mode 1.

The motor for LPCS PMP SUCT FM SUPP POOL E21F001 is to be meggered from its supply breaker under a routine task by Electrical Maintenance.

Which of the following describes the containment operability requirements for this work?

a. Before beginning the work, the redundant valve for the penetration must be closed and deactivated. After the work, E21F001 must be stroke timed closed per its surveillance procedure.

b. Before beginning the work, LPCS system piping must be determined to be intact and closed. After the work, E21F001 must restored to standby configuration per its SOI. No additional tests for operability are required.

c. Before beginning the work, E21F001 must be closed and deactivated and leakage through E21F001 measured to be within limits. After the work, E21F001 must be stroke timed closed per its surveillance procedure.

d. Before beginning the work, E21F001 must be closed. After the work, E21F001 must restored to standby configuration per its SOI. No additional tests for operability are required.

Ans: d

The subject valve is a primary containment isolation valve. This question involves general SRO knowledge relating to LCOs and retest control. All answers are plausible since they could be true for other primary containment isolation valves, depending on their safety function. Answer a is incorrect because no series valve exists for the related penetration. Also, stroke timing is not required for the subject work since valve performance is not affected (ref. 01-S-07-2 Att V p11). Answer b is incorrect because TS 3.6.1.3 requirements to isolate the penetration cannot be met by simply verifying closed loop integrity, as stated in 02-S-01-17. This answer was included based on historical misunderstanding of closed loop considerations. Answer c is incorrect because LPCS piping would not be drained for the subject work; therefore, verifying allowable containment leakage per 02-S-01-17 is not an issue. Also, stroke timing is not required for the subject work. Answer d is correct since TS 3.6.1.3 requires isolation of the penetration within 4 hours for an inoperable primary containment isolation valve, and the work involved would not potentially affect valve stroke times.

KA Generic 2.2.21
Knowledge of pre and post maintenance operability requirements.

10CFR55.43(b)(2),(3)

02-S-01-17
01-S-07-2
FSAR Table 6.2-49 p4
GLP-OPS-TSLCO Obj. 14
GLP-OPS-PROC Obj. 27.2

Difficulty 2

SRO 23

(10-S-01-17 is attached)

A Site Area Emergency has just been declared.

The SRO acting as Emergency Director in the room must ensure emergency dosimetry issue requirements are met for the Operator-At-The-Controls if:

a. radiation levels in the Control Room reaches 5 mr/hr.

b. an evacuation of the Control Building is ordered.

c. offsite radiation levels are expected to exceed General Emergency levels.

d. the designated Radiation Protection technician assigned to the Control Room is not present.

Ans: a

Correct answer a is from 10-S-01-17, Emergency Personnel Exposure Control, step 6.2.2. The reading given is 10 times the alarm setpoint listed for the Control Room ARM in TR 6.3.1-1 function 9.b. All other answers are plausible since they represent conditions with potentially elevated plant radiation and/or contamination levels. However, they are all incorrect, because the only other condition, in addition to that stated in answer a, that requires the Shift Supervisor/Manager to ensure proper emergency dosimetry issuance is when personnel are leaving the control room envelope.

KA Generic 2.3.5
Knowledge of use and function of personnel monitoring equipment.

10CFR55.43(b)(4),(5)

10-S-01-17 step 6.2.2
10-S-01-1 step 2.1.1
TR 6.3.1-1 function 9.b

GLP-EP-EPTS6 Obj. 8

Difficulty 1

SRO 24

(01-S-06-5 is attached)

Which of the following events has the shortest time limit for notification of an offsite agency?

a. MCPR Safety Limit violation due to a pressure transient from 100% power

b. Spurious Low Pressure Core Spray injection due to human error in Mode 2

c. Stem leakage from Recirc Pump A suction valve 60 gpm in Mode 3

d. Violation of the License Condition limit for maximum core power

Ans: c

All answers are plausible since they are events which require notification of an offsite agency. Correct answer c constitutes entry into the Emergency Plan at the Unusual Event level and requires notification of state and local officials within 15 minutes of declaration of the Unusual Event. With declaration required within 15 minutes of the stated conditions, the maximum allowable time for notification of an offsite agency is 30 minutes. Although 01-S-06-5, Reportable Events or Conditions, step 6.2.3b directs notification to the NRC as soon as possible, answer a is incorrect because a one hour limit is granted. Answer b is incorrect because 01-S-06-5 lists this condition as a 60 day report pursuant to 10CFR50.73. Answer d is incorrect because this condition is only a 24 hour report per 01-S-06-5 step 6.2.2b. References are not provided for this question since SROs should know this constitutes loss of the primary containment barrier and that this meets EAL criteria, and further, the stringent notification requirements for state and local agencies for Emergency Plan entry.

KA Generic 2.4.30
Knowledge of which events related to system operations / status should be reported to outside agencies.

10CFR55.43(b)(5)

01-S-06-5
10-S-01-1

GLP-OPS-PROC Obj. 11.6, 11.8, 11.10, 11.13
GLP-EP-EPTS6 Obj. 3

Difficulty 2

SRO 25

The Shift Manager/Emergency Director should implement the line-of-sight (2-person) rule when:

a. a bomb is discovered inside the Owner Controlled Area.

b. an insider threat is determined to be credible.

c. notified by Security that Security Code Orange exists.

d. notification of a hostage situation in the Protected Area is received.

Ans: b

Consistent with answer **b**, PL-174, Selected Interim Security Policy, section 5.3 outlines use of the line-of-sight rule specifically for insider threats to restrict the ability of any one insider from potentially subversive activities. This is an SRO responsibility. All other answers are credible since they are security events for which various contingencies have been assigned, however, the line-of sight rule is not warranted until an insider threat has been established. For these situations, personnel movements are restricted for individual safety, not to prevent acts of sabotage.

KA Generic 2.4.28
Knowledge of procedures relating to emergency response to sabotage.

10CFR55.43(b)(5)

05-1-02-VI-4 step 3.6
Corporate Policy PL-174 step 5.3.3

GSMS-RO-ON022 Obj. C.1

Difficulty 1