1. The plant was at rated power with all systems normally aligned and no equipment out of service. An event occurred that resulted in a reactor scram. The following annunciators are noted as <u>in alarm</u>:

   - 4160 STATION POWER BUS 1B – MN BRKR 1B 86 LKOUT TRIP
   - 4160 STATION POWER BUS 1B – BUS 1B UV
   - 4160 STATION POWER BUS 1D – MN BRKR 1D TRIP
   - 4160 STATION POWER BUS 1D – MN BRKR 1D 86 LKOUT TRIP
   - 4160 STATION POWER BUS 1D – BUS 1D VOLTS LO

   Which of the following states the correct action for these conditions?

   a. Confirm DW Recirc. Fans 1-1, 1-2, and 1-3 are operating, IAW ABN-48, Loss of USS 1B2
   b. Confirm EDG2 is supplying Bus D, IAW ABN-36, Loss of Offsite Power
   c. Restart RPS MG #2, IAW ABN-51, Loss of VMCC 1B2
   d. Verify air compressor 1-2 in service IAW ABN-47, Loss of USS 1B1

Answer: a

Justification: The indications provided show a fault and loss of power to 4160 AC busses 1B and 1D. When 1D de-energizes, EDG2 would normally start and load onto the bus. But, since there is a fault on Bus 1D, EDG2 does not pickup Bus 1D and Bus 1D stays de-energized. With Bus 1D de-energized, Buses 1B1, 1B2 (and VMCC 1B2) and 1B3 become de-energized.

IAW ABN-48, DW Recirc. Fans 1-1, 1-2, and 1-3 should be confirmed running (powered from Bus 1A23). These fans are not effected by the loss of power to Bus 1D (DW recirc. fans 1-4 and 1-5 will loose power). Answer a is correct.

ABN-36 does say to start EDG2 if it is not running or supplying power to Bus 1D. But with a fault on Bus 1D, the EDG2 cannot close onto Bus 1D. Answer b is incorrect.

ABN-51 says that if VMCC 1B2 is de-energized (RPS MG 2 is de-energized), then to re-power RPS from transformer 1 (which is normally powered from VMCC 1A2, and is not effected by the power losses in the question). Since there is no power to VMCC 1B2, RPS MG 2 cannot be started. Answer c is incorrect.

USS 1B1 powers air compressors 1-2 and 1-3, and is not powered given the power losses in the question. With no power to USS 1B1, air compressor 1-2 will not be running. Answer d is incorrect.

295003 AA2.01
Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER : Cause of partial or complete loss of A.C. power (CFR: 43.5)

OC Learning Objective: 2621.828.0.0012 (262-10444: Describe the interlock signals and setpoints for the affected system components and expected system response including power loss or failed components.)

Cognitive Level: Comprehension or Analysis

Question Type: New

Changed answers from the original due to missing CFR as noted by NRC review.

9/8/06
Rossi reviewed. Changed ‘lists’ to ‘states’ in the stem. Added ‘for these conditions’ in the question.

2. Given the following plant conditions:

- The plant is at 80% power and is recovering to 100% after removing Recirculation Pump C from service
- The Feedwater and Condensate Systems are aligned for rated power

An event occurred resulting in several annunciators, and the following indication is noted:

- Bus 1A ammeter indicates 0 amps

Which of the following states the condition of the reactor following required manual operator actions?

a. The reactor was manually scrammed due to loss of feedwater pumps, IAW ABN-17, Feedwater System Abnormal Operation
b. Reactor recirculation flow was lowered to 8.5 E4 gpm, IAW ABN-2, Recirculation System Failures
c. The reactor was manually scrammed due to the loss of reactor recirculation pumps, IAW ABN-2, Recirculation System Failures
d. A manual rapid power reduction has occurred due to the loss of a feedwater pump, IAW ABN-17, Feedwater System Abnormal Operation

Answer: c

Handouts: None

Justification: The loss of Bus 1A results in the loss of feedwater pump 1A, condensate pump 1A, and reactor recirculation pumps A and E (recirculation pump C is also on this bus, but it was secured in the question stem). The loss of a single feedwater or condensate pump would require the crew to perform a rapid power reduction IAW ABN-17. Multiple feedwater or condensate pumps would require a manual scram. The loss of a single recirculation pump (in 4-loop or 5-loop operation) requires that recirculation flow either lower flow or to insert cam rods IAW ABN-2. For multiple pump trips (in 4/5 loop), a manual scram is required. Answer c is the correct answer. All other answers are incorrect since they give an inappropriate action for the given conditions. (Also see procedures 317 and 301.2)

295006 AA2.06
Ability to determine and/or interpret the following as they apply to SCRAM : Cause of reactor SCRAM (CFR: 43.5)

OC Learning Objective: 2621.828.0.0017 (10450: Describe and interpret procedure sections and steps for plant emergency or off-normal conditions that involve this system including personnel allocations and equipment operation in accordance with applicable ABN, EOP & EOP Support Procedures.)

Cognitive Level: Comprehension or Analysis

Question Type: Modified Bank

9/8/06
Rossi reviewed. Placed parts of the initial paragraph into bullets, with minor wording changes to make more clear. Moved part of first paragraph into a new paragraph.

9/20/06 NRC comments: Added ‘required’ in the question.

3. The plant was shutting down for an outage, with the following conditions:

- RPV coolant temperature is 320° F and is cooling down
- Shutdown Cooling pumps A and B are in service with a total SDC system flow of 4000 GPM
- RBCCW flow through each of the in-service SDC heat exchangers is 1000 GPM
- Shutdown Cooling pump C is tagged out for repair
- All Reactor Recirculation pumps are running
- RPV water level is being maintained at 155" TAF

The following annunciator <u>came into alarm</u>:

- SHUT DN CLG – PUMP B TRIP

The new plant conditions are as follows:

- Shutdown Cooling flow has been verified at 2000 GPM
- Investigation shows that the SDC pump B tripped on over-current

Which of the following lists the required action to increase RPV cooling?

a. Maximize RBCCW cooling water flow through the SDC loop A heat exchanger to no more than 2000 GPM IAW procedure 309.2, Reactor Building Closed Cooling Water System
b. Maximize SDC loop A flow to no more than 3400 GPM IAW procedure 305, Shutdown Cooling System Operation
c. Increase RPV water level to above 170" TAF IAW procedure 305, Shutdown Cooling System Operation
d. Initiate alternate RPV cooldown IAW ABN-3, Loss of Shutdown Cooling

Answer: b

Handouts: None

Justification: Initial conditions indicate a partial loss of shutdown cooling flow. The RAP for the tripped SDC pump directs another pump be started if possible (which is not possible).

In both procedures 309.2 and 305, it stipulates that RBCCW cannot exceed 1500 GPM through a SDC heat exchanger. Answer is a incorrect.

Procedure 305 explains how to place SDC in service. The SDC pump discharge valves are throttled to establish the desired cooldown rate. The same procedure sets a limit on SDC flow of 3400 GPM through a heat exchanger. Since current

SDC flow is only 2000 GPM, then SDC flow can be increased up to 3400 GPM. Answer b is correct.

IAW procedure 305, with reactor recirc pumps running, RPV water level should be maintained within the normal band. Raising water level is only required during a partial SDC flow loss when no reactor recirc pumps are running. Answer c is incorrect.

Initiating alternate cooling through the cleanup system is only performed when SDC flow is lost or cannot be established, IAW ABN-3. Answer d is incorrect.

295021 AA2.02
Ability to determine and/or interpret the following as they apply to LOSS OF SHUTDOWN COOLING: RHR/shutdown cooling system flow (CFR: 43.5)

OC Learning Objective: 2621.828.0.0045 (02602: Identify and interpret procedures for plant emergency or off-normal situations which involve the SDC System, including personnel and equipment allocations.)

Cognitive Level: Comprehension or Analysis

Question Type: New

4. The plant was at 90% power in preparation for recovering a control rod that was manually scrammed for testing purposes. An electrical grid disturbance occurred resulting in a turbine trip.

The following plant conditions exist:

- RPV water level is 150” TAF and steady (up from a low of 112” TAF)
- RPV pressure band is 900 – 1000 psig, being controlled with Isolation Condenser B
- Drywell pressure is 3.2 psig and rising slowly
- Drywell temperature is 180° F and rising slowly
- Torus water temperature is 91° F and steady
- All control rods indicate full-in

The following annunciator came into alarm:

- ISOL COND – SHELL B LVL HI/LO

The following conditions are noted:

- Isolation Condenser B shell water level is 8’ and rising slowly, and that makeup is secured
- Chemistry reports that their sampling of the shell water indicates greater than the expected radionuclide concentrations for the given plant conditions

Which of the following procedural actions is required?

a. Initiate containment spray in the torus cooling mode
b. Close the Isolation Condenser DC valves
c. Emergency Depressurize
d. Isolate Isolation Condenser B

Answer: d

HANDOUT: EOPs

Justification: Even though the Primary Containment Control EOP has been entered, torus water temperature is below the entry condition and is stable. The first action step in the EOP says to maintain torus water temperature less than 95° F and to initiate torus cooling. The EOP Users Guide says to maintain temperature and to initiate torus cooling as required. In this case, with temperature all ready less than 95° F and not rising, torus cooling is not required. Answer a is incorrect.

The RPV Control – No ATWS EOP directs closing of the isolation condenser DC valves when RPV water level reaches 180”. Since water level was given as 150” and steady, this action is not required. Answer b is incorrect.
An emergency depressurization IAW Radioactivity Release Control EOP is required when a General Emergency (from off-site dose) is declared. There is no indication that this has been reached. Therefore, answer c is incorrect.

One entry condition into the Radioactivity Release Control EOP (EMG-3200.12) is an isolation condenser tube leak. The indications of this have been provided in the question stem. The first action step in this EOP is to isolate primary systems discharging outside the primary and secondary containment (such as the isolation condenser vent lines), except for systems required by EOPs. Even though isolation condenser B is being used IAW EOPs, there are several other RPV pressure control methods available to take the place of isolation condenser B when it gets isolated (IC A, EMRVs). Therefore, IC B should be isolated IAW the Radioactivity Release Control EOP, and another pressure control method utilized. Answer d is correct.

295038 2.4.6 High Off-site Release Rate
Knowledge symptom based EOP mitigation strategies. (CFR: 43.5)

OC Learning Objective: 2621.845.0.0012 (02483: Using procedure EMG-3200.12, evaluate the technical basis for each step and apply this evaluation to determine the correct course of action under emergency conditions.)

Cognitive Level: Comprehension or Analysis

Question Type: New

9/8/06
Rossi reviewed. Added ‘procedural’ into the question. Placed ‘The following plant conditions exist:’ from the first paragraph into a new paragraph. Placed last stem paragraph into bullets.

5. The plant is at rated power.

   Which of the following (1) would <u>require</u> a notification, and (2) to whom this notification must be made?

   a. (1) An unexpected primary containment isolation
   (2) Duty Station Manager of the isolation IAW OP-OC-106-101, Significant Event Notification and Reporting

   b. (1) The failure of a non-APRM LPRM, with all other LPRMs operable
   (2) Duty Station Manager of the LPRM failure IAW OP-OC-106-101, Significant Event Notification and Reporting

   c. (1) A spurious trip of the running TBCCW and auto start of the standby TBCCW pump
   (2) PA announcement to plant personnel of the started pump IAW OP-AA-104-101, Communications

   d. (1) The plant scrammed with one control rod stuck full-out
   (2) State and local authorities IAW EP-OC-114-100, State/Local Notifications

Answer: a

Handouts: None

Justification: OP-OC-106-101 requires the SM notify the Duty Station Manager of any event that proceeds in a way significantly different than expected. The primary containment isolation is significant and was not expected. The isolation also affected DW sump and equipment leakage monitoring ability. The same requirement is also found in OP-AA-106-101 (other examples are unexpected ½ scram). Also, the unexpected isolation would require a prompt investigation (OP-AA-106-101-1001). A prompt investigation requires DSM notification also. Answer a is correct.

OP-AA-106-101, and OP-OC-106-101 requires the SM notify the Duty Station Manager of any forced entry into a 72-hour (or less) TS shutdown LCO. IAW TS 3.2.B.2, the loss of a single LPRM does not cause any APRM to be inoperable, and thus no TS entry. Answer b is incorrect.

OP-AA-104-101 requires a PA announcement prior to staring equipment where safety is a concern or when energizing/de-energizing major electrical switchgear and busses. In answer c, the component has already started and there would be no requirement no announce the start for safety concerns. Answer c is incorrect.

If two control rods had failed to insert to at least position 04, then an emergency plan entry is required. But with one control at any position failing to insert, the reactor is assured to be shutdown. Therefore, state/local authorities would not be notified. Answer d is incorrect.

295020 Inadvertent Containment Isolation
2.1.14 Knowledge of system status criteria which require the notification of plant personnel. (CFR: 43.5)

OC Learning Objective: 2621.830.0.0005 (01638: Given a description of an event, describe the following: 1) what category the event belongs in; 2) who must be notified; 3) time limit; 4) any follow-up reports.)

Cognitive Level: Memory Fundamental

Question Type: New

The handout were deleted. NRC comment.

9/20/06 NRC comment: Added ‘rod’ to answer selection d.

6. The plant is in SHUTDOWN due to a loss of condenser vacuum (which has NOT been restored).

   The following conditions currently exist:

   - Shutdown Cooling loop A is in service, with all reactor recirculation pumps in service
   - RPV water level is in the normal band
   - Unit Substation 1B2 is de-energized for emergent maintenance (it is expected the bus will be returned to service in 30 minutes)
   - RPV coolant temperature is 220° F and is trending down slowly

   The following component failure has just occurred:
   - TE-31J (Reactor Recirculation Pump E suction temperature element) has failed upscale

   Which of the following states (1) the effect on the plant and (2) the required actions?

   a. (1) Shutdown Cooling Pump A has tripped due to high SDC Pump A suction temperature
   (2) Start SDC Pump B or C to restore SDC System flow IAW procedure 305, Shutdown Cooling System Operation

   b. (1) Shutdown Cooling Pump A has tripped due to isolation valve closure
   (2) Bypass the failed temperature element and restore SDC Pump A IAW ABN-3, Loss of Shutdown Cooling

   c. (1) Shutdown Cooling Pump A has tripped due to isolation valve closure
   (2) Initiate Alternate Shutdown Cooling Using EMRVs and Core Spray IAW ABN-3, Loss of Shutdown Cooling

   d. (1) Shutdown Cooling Pump A has tripped due to high SDC Pump A suction temperature
   (2) Initiate Alternate RPV cooldown (cleanup system letdown) per procedure 303, Reactor Cleanup Demineralizer System

Answer: b

Handouts: None

Justification: A high temperature sensed on any reactor recirculation loop (350°) will isolate SDC. When SDC IV V-17-19 closes, all SDC pumps trip. Because it has been determined that the recirculation temperature sensor has failed, ABN-3 allows bypassing the sensor and restoring SDC flow. Answer b is correct.

It is true that 350° F SDC suction temperature will isolate SDC and trip the SDC pump, this is not what was given in the question. Also, SDC Pumps B and C are powered by USS 1B2, which is de-energized. Answer a is incorrect.

As stated, the SDC IVs close on high recirc. temperature. ABN-3 directs that the failed temperature sensor be bypassed and SDC restored in step 3.2.2. Later, in step 3.2.8, it directs alternate cooling with core spray and EMRVs. Since step 3.2.2 can be performed, there would be no reason to perform step 3.2.8. Answer c is incorrect.

As stated, SDC pump trips from IV position, not SDC loop temperature. Also, cleanup letdown as an alternate path is not available since the condenser is not available (given in the question stem). Answer d is incorrect.

205000 A2.05
Ability to (a) predict the impacts of the following on the SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE) ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: System isolation (CFR: 41.5 / 43.5)

OC Learning Objective: 2621.828.0.0045 (02602: Identify and interpret procedures for plant emergency or off-normal situations which involve the SDC System, including personnel and equipment allocations.)

Cognitive Level: Comprehension or Analysis

Question Type: New

A bullet in the question was deleted (which provided confirming information that SDC had isolated). This was a NRC comment.

9/8/06
Rossi reviewed. Changed ‘which is still being investigated’ to ‘which has NOT been restored’ to clarify that the condenser is not available in the stem. Placed ‘The following conditions currently exist:’ from the first paragraph into a new paragraph. Modified first paragraph to read better.

7. Tech Specs 3.4.B.3 states that if the EMRV operability for the ADS function cannot be met, then reactor pressure shall be reduced to 110 psig.

   Which of the following provides the basis for the 110 psig?

   a. A small break LOCA will NOT result in a primary containment temperature of 281° F, which would require emergency depressurization

   b. Core spray flow into the RPV during a small break LOCA is sufficient to ensure that peak fuel centerline temperature does not exceed 2200° F

   c. There is no credible event in which an RPV over-pressure condition would challenge the reactor coolant system pressure safety limit, requiring the use of the ADS valves

   d. Core spray flow into the RPV during a small break LOCA is sufficient to ensure that cladding oxidation will not exceed 0.17 times the total cladding thickness before oxidation

Answer: d

HANDOUT: None

Justification: The relief valves of the ADS System enable the core spray system to provide protection against the small break LOCA in the event feedwater system is not available. Under the conditions of a small break LOCA at high RPV pressures and no feedwater available, the ADS valves will open to depressurize the RPV to allow core spray to inject for core cooling. At an RPV pressure of 110 psig, core spray can provide the design flow necessary to maintain adequate core cooling. Thus if the small break LOCA occurred at or less than this pressure (110 psig), the ADS valves are not required to depressurize the RPV to allow core spray injection. The Emergency Core Cooling System design must meet the criteria of 10CFR50.46 (Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Plants). Two of these criteria is that ECCS will maintain peak cladding temperature (not fuel temperature) less than 2200° F, and maximum cladding oxidation of 0.17 times the total cladding thickness before oxidation. Answer d is correct, and answer b is incorrect.

All other answers are credible but are not correct ISW TS 3.4.B.3 basis.

218000 2.2.25 (ADS)
Knowledge of bases in technical specifications for limiting conditions for

operations and safety limits. (CFR: 43.2)

OC Learning Objective: 2621.850.0.0090 (01658: State requirements associated with given areas of Technical Specifications (safety limits, LSSS, etc.).)

Cognitive Level: Comprehension or Analysis

Question Type: New

The original question provided the TS 3.4 as a handout and the question simply asked for the basis of 110 psig in that TS. The handout was deleted and question stem re-written to be asked without the handout (NRC comment).

9/8/06
Changed ‘not’ to ‘NOT’ in answer a.

8. The plant was at rated power, with SERVICE AIR COMPRESSOR 2 tagged out for repair.

   The following annunciators came into alarm:

   - SERVICE AIR – RCVR 1 PRESS LO
   - SERVICE AIR – RCVR 2/INST AIR PRESS LO
   - SERVICE AIR – RCVR 3 PRESS LO
   - SERVICE AIR – COMPR 1 BREAKER TRIP

   The INSTR AIR SUPPLY PRESS indicator shows 82 psig and lowering. ABN-35, Loss of Instrument Air, has been entered. The SRO directed a manual reactor scram IAW the ABN.

   - All control rods indicate full-in

   Which of the following actions is correct for these conditions?

   a. Establish and maintain RPV water level 138” to 175” TAF with the LRFV in MANUAL IAW ABN-1, Reactor Scram
   b. Establish and maintain RPV pressure 800 – 1000 psig with the Main Turbine Bypass Valves IAW EMG-3200.01A, RPV Control – No ATWS
   c. Establish and maintain RPV water level 138” to 175” TAF using letdown from the RPV with the Cleanup System if required, IAW ABN-1, Reactor Scram
   d Establish and maintain RPV pressure 800 – 1000 psig with Isolation Condensers IAW EMG-3200.01A, RPV Control – No ATWS

Answer: d

Handouts: None

Justification: The first alarm (M3a) comes in at 95 psig in service air receiver 1-1. The second alarm (M3b) comes in at either 80 psig in service air receiver 1-2 or 85 psig in the air header. The third alarm (M3c) comes in at 85 psig in the service air receiver 1-3. This indicates a system-wide air loss. As provided in the question stem, ABN-35, Loss of Instrument Air, has been entered. It directs a manual reactor scram when instrument air pressure drops to 55 psig (or if control rods begin to drift into the core). It is apparent that the instrument air pressure continues to lower.

The feedwater flow regulating valves are air-operated and lock-up on loss of air (even though they might drift). With these valves locked-up, the LFRV controller in MANUAL will not function to operate the LFRV valves (also air opewrated). Answer a is incorrect.

With instrument air pressure lowering, the air-operated outside MSIVs will auto close. With these valves closed, the TBV are no longer available to control RPV pressure. Answer b is incorrect.

ABN-1 does direct to letdown through the Cleanup System if necessary, but with the air loss, the Cleanup System letdown path is not available (cleanup will isolate). Answer c is incorrect.

Isolation condensers are available for use (even though shell makeup must be performed locally). RPV pressure control below 1045 psig is directed by EMG-3200.01A. Answer d is correct.

300000 2.4.6 (Instrument Air)
Knowledge symptom based EOP mitigation strategies. (CFR: 43.5)

OC Learning Objective: 2621.828.0.0043 (10450: Describe and interpret procedure sections and steps for plant emergency or off-normal conditions that involve this system including personnel allocation and equipment operation in accordance with applicable ABN, SDRP, EOP and EOP support procedures and EPIPs.)

Cognitive Level: Comprehension or Analysis

Question Type: New

9/8/06
Rossi reviewed. Deleted: with all systems normally aligned in the stem, and re-worded the stem using panel nomenclature. Placed 'The following annunciators came into alarm:' from the first paragraph into a new second paragraph. Modified the question for focus from 'Which of the following is correct?' to 'Which of the following actions is correct for these conditions?'

9. The plant was starting up after a refuel outage. The Reactor Operator was withdrawing control rods, when the control rod position indication went dark for the control rod being withdrawn.

   Control rod position indication was NOT regained when they inserted the control rod one notch. The Operator then attempted to fully insert the control rod in preparation for isolating the control rod. Neutron monitoring showed no change in counts as the control rod was inserted.

   With the control rod valved out of service, IAW procedure 302.1 Control Rod Drive System, and control rod position not known, which of the following Technical Specifications actions is required?

   a. The SHUTDOWN MARGIN must be verified within 6 hours, including the effects of the unknown-positioned control rod
   b. The reactor must be placed in the SHUTDOWN CONDITION within 12 hours
   c. Immediately initiate action to fully insert all insertable control rods
   d. Verify there are no more than 8 inoperable control rods valved out of service, prior to continuing with control rod withdrawals

Answer: a

HANDOUT: None

Justification: Shutdown margin is determined with the strongest reactivity control rod assumed fully withdrawn and all other control rods fully inserted. But since the control rod in the question has no position indication, and they are unable to verify that the control rod is fully inserted, it’s position is unknown. Because of this, shutdown margin must be verified with this control not fully inserted. This is required IAW TS 3.2.A.2. Answer a is correct.

Only if the SDM cannot be verified within the time allowed (6 hours), the plant must be placed in the shutdown condition within the following 12 hours IAW TS 3.2.A.3. Answer b is incorrect.

If SDM cannot be met while in REFUEL mode, then TS 3.2.A.5 requires that all control rods be fully inserted. Answer c is incorrect.

TS 3.2.B.4 allows only 6 inoperable, valved out of service control rods. In any event, the startup cannot continue even if this verification was made. Answer d is incorrect.

214000 2.2.22
Knowledge of limiting conditions for operations and safety limits. (CFR: 43.2)

OC Learning Objective: 2621.828.0.0011 (10451: Given Tech Specs, identify and explain associated actions for each section of Tech Specs relating to this system including personnel allocations and equipment operation.)

Cognitive Level: Comprehension or Analysis

Question Type: New

The TS 3.2 handout was deleted. NRC comment.

9/8/06
Rossi reviewed. Re-wrote the stem for simplicity and brevity.

9/20/06 NRC comment: Added ‘IAW procedure 302.1 Control Rod Drive System’ in the question.

10. The plant is high on Range 9 of the Intermediate Range Monitors during a startup.

    Which of the following lists the on-duty shift requirements IAW Technical Specifications:

    a. 1 Shift Manager
       3 licensed Nuclear Plant Operators
       2 licensed or non-licensed Nuclear Plant Operators
       1 Shift Technical Advisor

    b. 1 Unit Supervisor
       3 licensed Nuclear Plant Operators
       2 licensed or non-licensed Nuclear Plant Operators
       1 Shift Technical Advisor

    c. 1 Shift Manager
       2 licensed Nuclear Plant Operators
       3 licensed or non-licensed Nuclear Plant Operators

    d. 1 Shift Manager
       2 licensed Nuclear Plant Operators
       3 licensed or non-licensed Nuclear Plant Operators
       1 Shift Technical Advisor

Answer: d

HANDOUT: None - TS 6.0 NOT provided.

Justification: TS 6.2.2.2.a requires the following: 1 SM, 2 licensed Nuclear Plant Operators, 3 licensed or non-licensed Nuclear Plant Operators, 1 Shift Technical Advisor (STA is not required in shutdown or refuel with the reactor < 212° F, according to TS 6.2.2.2.h.). Answer d is correct. All other answers are incorrect.

Conduct of Operations
2.1.4 Knowledge of Shift Staffing requirements.

OC Learning Objective: 2621.850.0.0090 (01658: State requirements associated with given areas of Technical Specifications (Safety Limits, LSSS, etc.)

Cognitive Level: Comprehension or Analysis

Question Type: Bank

11. Which of the following proposed plant changes would require a 10CFR50.59 Evaluation and NRC approval prior to procedural implementation? (neglect any effects of the proposed changes on the Updated Safety Analysis Report)

    a. Changing the Reactor Building Vent Radiation Monitors upscale trip setting to 20 mR/hr
    b. Changing the RPV heatup/cooldown rates to 16° F/10 minutes
    c. Changing the requirement to use the Rod Worth Minimizer in the Low Power Mode until 15% power on a startup
    d. Changing the Scram Discharge Volume Hi-Hi setpoint to 25 gallons

Answer: a

HANDOUT: None

Justification: IAW 10CRF50.59(c)(1)I: A licensee may make changes in the facility, without obtaining a license amendment, as long as a tech spec change is not required. A 50.59 review would need to be performed for a change that would require a change in Tech Specs. (See LS-AA-104). A change in Tech Specs must first be approved by the NRC.

TS Table 3.1.1.j requires the hi setpoint of RB vent radiation monitors be set at ≤ 17 mR/hr (currently set at 9 mR/hr (see RAP-10F1f)). Changing this setpoint above the TS value (to 20 mR/hr) would first need a Tech Spec change, preceded by a 50.59 review and evaluation. Answer a is correct.

TS 3.3.c.1 allows a 100° F/hr limit on heatup/cooldown rate. Startup and shutdown procedures limit this to 95° F/hr. Setting this limit to 96° F/hr (which equals 16° F/10 minites) is still less than the TS requirement and thus does not violate TS. Answer b is incorrect.

TS 3.2.B.2 requires the RWM be operable on a startup up t0 10% power (and this is reflected in procedure 409, Operation of the Rod Worth Minimizer). Requiring the RWM to be operable up to 15% power on a startup does not effect Tech Specs. No NRC notification would be required. Answer c is incorrect.

TS Table 3.1.1.a requires a scram signal from ≤ 29 gallons in the SDV, and is currently set at 26 gallons (se RAP-H1b). Setting this limit at 25 gallons is more conservative and does not violate TS. Answer d is incorrect.

Equipment Control
2.2.9 Knowledge of the process for determining if the proposed change / test or experiment increases the probability of occurrence or consequences of an accident during the change / test or experiment.
(CFR: 43.3)

OC Learning Objective: 2621.830.0.0005 (02618: State what actions an operator and supervisor make to initiate a procedure change.)

Cognitive Level: Comprehension or Analysis

Question Type: Bank

The handouts were deleted. NRC comment. Answer d was changed so that the new setpoint given was less that the expected known actual scram setpoint (not TS scram setpoint) for scram discharge volume.

12. The plant is at rated power. An NLO was required to manipulate a manual valve (located at floor level, and requires no tools to manipulate) in a Locked High Radiation Area (LHRA). This area has a peak dose rate of 1050 mr/hr, and is routinely surveyed by Radiation Protection.

    IAW RP-AA-460, Controls for High and Very High Radiation Areas, which of the following steps are required by the Operator (besides signing onto the appropriate RWP)?

    1. Review the currently available survey data for the area
    2. Receive a briefing from the RP Tech
    3. Ensure that the RP Tech accompanies you into the LHRA
    4. Verify the maximum dose rate with your electronic dosimetry
    5. When leaving the area, notify RP to second check you that the access is closed and locked

    a. 1, 2, and 3
    b. 2, 4, and 5
    c. 1, 3, 4, and 5
    d. 1, 2, and 5

Answer: d

Justification: RP-AA-460, Controls for High and Very High Radiation Areas (section 4.8), the following are required: 1) review survey data (it does allow RP Tech to accompany the worker into the area, if there is no current survey data); 2) review/sign RWP; 3) receive an RP brief; 4) When exiting, wait there and notify RP so that they can come and verify the gate/door is closed and locked. Answer d is correct and all other answers are incorrect.

Verifying maximum dose rates is not a responsibility of the worker.

Therefore, items 1, 2 and 5 are required. Answer d is correct.

Radiation Control
2.3.10 Ability to perform procedures to reduce excessive levels of radiation and guard against personnel exposure. (CFR: 43.4)

OC Learning Objective:

Cognitive Level: Comprehension or Analysis

Question Type: New

9/8/06

Rossi reviewed. Deleted: ‘with all systems normally aligned’ in the stem, and added ‘rated’. Simplified the remaining stem for brevity.

13. The reactor was at rated power when an event occurred. Present plant conditions are as follows:

- Annunciator SCRAM CONTACTOR OPEN is in alarm
- All red scram lights are ON
- Annunciator ARI INITIATED is in alarm
- RPV water level indicates 120” TAF and rising slowly
- Drywell pressure is 2.2 psig and rising very slowly
- Drywell temperature is 170° F and rising very slowly
- Torus water temperature is 100° F and rising
- All reactor Recirculation Pumps DRIVE MOTOR switches are green-flagged (switch semaphore indicates green)
- Annunciators EMRV OPEN and SV/EMRV NOT CLOSED are in alarm
- Annunciator APRM DNSCL is **NOT** in alarm
- Annunciator ROPS BYPASSED is in alarm

Which of the following states the next required operator action?

a. Initiate drywell sprays IAW EMG-3200-02, Primary Containment Control

b. Pull the open EMRV control fuses IAW ABN-40, Stuck Open EMRV

c. Perform scram reset and scram IAW EMG-3200-01B, RPV Control – With ATWS

d. Vent the scram air header IAW EMG-3200-01B, RPV Control – With ATWS

Answer: c

HANDOUT: EOPs

Justification: The indications provided show that an electromatic relief valve (EMRV) is open (EMRV open and not closed alarms) and that a reactor scrammed occurred (scram contactor open alarm and scram lights on). It also shows that the reactor is not shutdown and that power is greater than 4% (APRM downscale alarm not in), and alternate rod insertion (ARI initiated alarm) has been initiated.

Answer a is incorrect because even though drywell sprays could be initiated now in the drywell temperature leg, temperature is far away from 281° F, and other actions are of higher priority. Answer a is incorrect.

Actions to close the EMRV should be performed in conjunction with EOP actions. But pulling control fuses to close the EMRV is not an action in ABN-40. Answer b is incorrect.

The initial conditions show that reactor overfill protection (ROPS) is bypassed and that all reactor recirculation pumps have been manually tripped (green-flagged switches). The next action in RPV Control – With ATWS is to insert control rods given a hydraulic ATWS exists (since all red scram lights are on, then all scram valves have opened and the ATWS is not electric). A possible method to insert control rods is to reset the scram, allow the scram discharge volume time to drain, and to scram again. Answer c is correct.

Venting the scram air header (performed for an electric ATWS) will not help in inserting control rods. Answer d is incorrect.

Emergency Procedures/Plan
2.4.46 Ability to verify that the alarms are consistent with the plant conditions. (CFR: 43.5)

OC Learning Objective: 2621-845.0.0005 (03060: During a walkthrough on the BPT or BWR simulator, demonstrate the ability to shutdown the reactor during a failure to scram situation in a timely manner IAW EMG-3200.01B, Support Procedure 21.)

Cognitive Level: Comprehension or Analysis

Question Type: New

9/20/06 NRC comment: Changed ‘lists’ to ‘states’ in the question.

14. The reactor is at 23% power and the turbine generator has just been placed on-line. The Reactor Operator is raising reactor power by withdrawing control rods.

    Which one of the following is correct regarding turbine generator operation?

    a. A turbine vibration of 15 mils (and trending up) requires an immediate reactor scram and turbine trip IAW TURBINE MECH – VIBRATION HI annunciator response procedure

    b. A turbine vibration of 15 mils (and trending up) requires an immediate turbine trip ONLY IAW TURBINE MECH – VIBRATION HI annunciator response procedure

    c. A loss of both stator cooling water pumps requires an immediate reactor scram and turbine trip IAW ABN-11, Loss of Generator Stator Cooling

    d. A loss of both stator cooling water pumps requires an immediate turbine trip ONLY IAW ABN-11, Loss of Generator Stator Cooling

Answer: b

Handouts: None

Justification: IAW RAP-Q3b (vibration high), an immediate turbine trip is required if any turbine bearing reaches 12 mils or above (and continues to increase). Since reactor power is less than 30%, a reactor scram is not required. Answer a is incorrect and answer b is correct.

IAW ABN-11, if a turbine runback occurs (as a result of the loss of stator cooling) or stator temperatures are rising AND reactor power is less than 30%, then generator MVARs should be manually reduced to zero, or as low as allowed by grid conditions. At 24% power, a runback is not expected anyway because power is so low. No scram nor turbine trip are required. Answers c and d are incorrect.

295005 G2.4.49 Main Turbine Generator Trip / Emergency Procedures /Plan: Ability to perform without reference to procedures those actions that require immediate operation of system components and controls. (CFR: 43.5)

OC Learning Objective: <u>2621.828.0.0050, Objective S:</u>
Describe and interpret procedure sections and steps for plant emergency or off-normal conditions that involve this system including personnel allocation and equipment operation in accordance with applicable ABN, SDRP, EOP & EOP support procedures and EPIPs.

Cognitive Level: Comprehension or Analysis

Question Type: Modified Bank

References: RAP-Q3b, ABN-10

9/8/06
Rossi reviewed. Added ‘regarding turbine generator operation’ into the question for focus.

15. The Control Room has been evacuated due to Control Room fire. ABN-30, Control Room Evacuation, is being executed. Following the evacuation, the following conditions exist:

- The REACTOR MODE SELECTOR switch is in SHUTDOWN and all control rods verified full-in
- RPV water level is steady at 150" and adequate core cooling is assured
- RPV pressure is 900 psig and lowering
- All Core Spray Pumps and all EMRV's have been disabled IAW ABN-30

Based on the conditions given, which of the following actions must be met to comply with Technical Specifications?

1. Reduce RPV pressure to < 110 psig within 24 hours.
2. Place the reactor in COLD SHUTDOWN within 30 hours.

a. 1 ONLY
b. 2 ONLY
c. 1 and 2
d. Neither 1 or 2

Answer: c

Handouts: Tech Spec 3.4

Justification: A, B and D are incorrect – both 1 and 2 must be met.
C is correct – both Tech Spec action statements must be met…with the EMRV's disabled (IAW Attachment ABN-30-8) the ADS function is also disabled. Tech Spec 3.4.B (ADS) requires reactor pressure to be reduced to less than 110 psig within 24 hours if ADS operability requirements are not met. Table 3.4.1 (Core Spray) allows reduced Core Spray capability, provided several things are met: one is that the RPV be maintained < 212° F (currently at 900 psig). Since the requirements of the Table cannot be met, then 3.4.A.2 applies: place in Cold Shutdown within 30 hours. Answer c is correct.

The NRC first thought that this question was a K/A mismatch. After discussion, they realized that it was a match and the question remained as-is.

9/8/06
Rossi reviewed. He had a different answer than key. Spoke with Busk. He suggested changing the question since the Isolation Condenser Tech Specs, under the question conditions, are not clear. Question has been modified to that shown. Busk re-reviewed.

16. An ATWS has occurred and all control rods have been inserted IAW Support Procedure 21.

The following plant conditions currently exist:

- Reactor pressure is 900 psig
- Reactor water level is 10 inches
- Torus water level is 173 inches
- Torus water temperature is 160 °F

What action is required for these conditions?

a. Reduce RPV pressure to prevent exceeding TLL
b. Reduce RPV pressure to prevent exceeding HCTL
c. Emergency Depressurize due to exceeding HCTL
d. Initiate Standby Liquid Control due to exceeding BIIT

Answer: c

Handouts: EMG-3200.01A, EMG-3200.02 (Provide large figures for easier reading)

Justification: A is incorrect – this would be a viable answer if HCTL was not already exceeded.

B is incorrect – HCTL has already been exceeded…it's too late to lower pressure to "prevent exceeding HCTL."

C is correct – HCTL has been exceeded…Emergency Depressurization is required by EMG-3200.02, Primary Containment Control.

D is incorrect – all rods have been inserted…there is no requirement (or need) to initiate SLC relative to BIIT.

295026 EA2.01
Ability to determine and/or interpret the following as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE: Suppression pool water temperature (CFR: 43.5)

OC Learning Objective: 2621.828.0.0032, Objective J:
Identify and interpret normal, abnormal and Emergency Operating Procedures for the Primary Containment System.

Cognitive Level: Comprehension or Analysis

Question Type: Bank

9/8/06
Rossi reviewed. Changed the initial 2 sentences into one by adding ‘and’ between them. Placed the following into a new paragraph: ‘The following plant conditions currently exist:’

References: EMG-3200.01A, EMG-3200.02, EOP Users Guide

17. A turbine trip occurred while operating at rated power due to a loss of turbine operating oil. Current plant conditions are as follows:

- Reactor pressure is 1020 psig and rising slowly
- Reactor water level is 150 inches and rising slowly
- Aux flash tank pressure on Panel 7F indicates zero

Which of the following should be used to control reactor pressure?

a. EMRV's ONLY
b. Isolation Condensers ONLY
c. EMRV's <u>and/or</u> Isolation Condensers
d. Main Turbine Bypass Valves

Answer: c

Handouts: None

Justification: A is incorrect. EMRVs can be used under the given conditions, but it is not the only system available. The main turbine bypass valves are not available (due to loss of condenser vacuum as indicated by aux flash tank pressure). With RPV level below 160 inches, the Isolation Condensers are also available, for "stabilizing RPV pressure below 1045 psig" as directed by ABN-1. The other options given in ABN-1 (RWCU and IC tube side vents) are not practical for the given conditions.

B is incorrect. As stated above, the EMRVs are also available. There is nothing in the question stem that makes Isolation Condensers unavailable. An RPV water level above 160" would make the ICs unavailable.

C is correct since both the EMRVs and the ICs are available for pressure control and can be used.

D is incorrect – the main turbine bypass valves are not available without main condenser vacuum above 10" Hg (Vacuum Trip 2 trips the bypass valves at 10" Hg). Aux Flash Tank pressure at zero indicates main condenser vacuum is at zero.

295007 AA2.03 Ability to determine and/or interpret the following as they apply to HIGH REACTOR PRESSURE: reactor water level. (CFR: 43.5)

OC Learning Objective: 2621.828.0.0037, Objective N:
Describe and interpret procedure sections and steps for plant emergency or off-normal conditions that involve this system including personnel allocation and equipment operation IAW applicable ABN, EOP & EOP support procedures and EPIPs.

Cognitive Level: Comprehension or Analysis

Question Type: New

References: ABN-1, 307

18. Refueling is in progress when an accident occurs on the refuel floor. The following conditions exist five minutes later:

- Refueling has been suspended and the refuel floor has been evacuated
- ALL refuel floor radiation monitors indicate between 70 and 90 mR/hr on Panel 2R
- Reactor Building ventilation exhaust radiation monitors indicate 3 mR/hr on Panel 2R
- Reactor Building differential pressure is negative 0.25 inches WG

Which of the following describes how the Reactor Building Ventilation System (RBVS) and Standby Gas Treatment System (SGTS) should be operated during this event.

a. RBVS is in service and should remain in service
b. SGTS is in service and should remain in service
c. RBVS is in service; SGTS should be placed in service
d. SGTS is in service; RBVS should be placed in service

Answer: d

Handouts: EMG-3200.11

Justification: Under the given conditions, the high radiation levels on the refuel floor have auto started SGTS and isolated normal RB ventilation (see RAP-10F4m).

A is incorrect – RBVS is tripped and isolated; SGTS is in service.

B is incorrect – SGTS is in service but the conditions for placing RBVS back in service are met, as directed by the Secondary Containment Control EOP.

C is incorrect – RBVS is tripped and isolated; SGTS is in service.

D is correct – RBVS isolated on Hi Refuel Floor radiation (> 50 mR/hr w/ 2 minute time delay in either the spent fuel pool area or on the operating floor). This also caused SGTS to initiate, maintaining RB negative differential pressure. The Secondary Containment Control EOP directs placing the RBVS in service <u>IF</u> it has isolated or is shutdown, <u>AND</u> the drywell is not being vented through the RB supply fans (the drywell is open during refueling), <u>AND</u> RB ventilation exhaust radiation level is below 9 mR/hr, <u>OR</u> RB pressure is above 0 inches WG and a ground level release is imminent or in progress. Since all of these conditions are met, RBVS should be placed back in service. This will require overriding interlocks (Hi Refuel Floor radiation initiation signals), resetting and restarting RB Ventilation as directed by Support Procedure 50, which also causes SGTS to shutdown.

295034 EA2.01
Ability to determine and/or interpret the following as they apply to SECONDARY CONTAINMENT VENTILATION HIGH RADIATION: Ventilation radiation levels (CFR: 43.5)

OC Learning Objective: <u>2621.828.0.0042, Objective F:</u>
Given a set of system indications or data, evaluate and interpret them to determine limits, trends and system status.
<u>2621.828.0.0042, Objective M:</u>
Describe and interpret procedure sections and steps for plant emergency or off-normal conditions that involve this system including personnel allocation and equipment operations IAW applicable ABN, SDRP, EOP and EOP support procedures and EPIPs.

Cognitive Level: Comprehension or Analysis

Question Type: New

References: RAP-10F1f, RAP-10F3m, EMG-3200.11, EOP Users Guide

19. Given the following:

- A plant startup is in progress
- The reactor mode switch is in STARTUP
- SRM 22 is bypassed due to a failed detector
- IRM's 11-16 and 18 are indicating 10% on Range 8
- IRM 17 is indicating 75% on Range 7
- SRM 23 experiences an INOP condition due to a power supply failure

Which statement below describes how this impacts the reactor startup?

1. A withdraw rod block ____________.
2. The reactor startup ____________.

a. (1) is generated
(2) can continue because IRM 17 can be switched to Range 8

b. (1) is generated
(2) CANNOT continue because the rodblock cannot be bypassed

c. (1) is NOT generated
(2) can continue because only two SRM's are required to be operable during a reactor startup

d. (1) is NOT generated
(2) CANNOT continue because more than two SRM's are required to be operable during a reactor startup

Answer: a

Handouts: None

Justification: A is correct – a rod block is generated due to the SRM 23 INOP condition and not ALL of the correlating IRMs (15, 16, 17 and 18) are on or above Range 8. Since IRM 17 is at the top of the 25-75% band, it can be switched to Range 8, as directed by Procedures 201 and 402.3. This will bypass the SRM 23 rod block, allowing control rod withdrawal to continue.

B is incorrect – the first statement is correct. However, while it is true that one SRM is already bypassed, switching IRM 17 to range 8 automatically bypasses all SRM rod block functions, which will allow control rod withdrawal to continue. This is allowed (directed) by Procedures 201 and 402.3.

C is incorrect – the first statement is incorrect because a withdraw rod block IS generated. The second statement is correct, although insignificant for the given conditions.

D is incorrect – the first statement is incorrect because a withdraw rod block IS generated. The second statement is also incorrect in that Procedure 201 only requires two operable SRM's during a reactor startup (until all IRM's are on Range 8 or above).

215004 A2.02
Ability to (a) predict the impacts of the following on the SOURCE RANGE MONITOR (SRM) SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: SRM inop condition (CFR: 43.5)

OC Learning Objective: <u>2621.828.0.0029, Objective F:</u>
Describe the interlock signals and setpoints for the affected system components and expected system response including power loss of failed components.
<u>2621.828.0.0029, Objective G:</u>
Given a set of system indications or data, evaluate and interpret them to determine limits, trends and system status.
<u>2621.828.0.0029, Objective I:</u>
Given normal operating procedures and documents for the system, describe or interpret the procedural steps. [Describe and interpret procedure sections or steps and documents, under normal operating conditions, that involve this system.] [200s, 300s, 400s, 800s]

Cognitive Level: Comprehension or Analysis

Question Type: New

References: 201, 401.4, 402.3, RAP-G4d, GE 237E912

20. The reactor is operating at rated power when the SV/EMRV OPEN annunciator goes into alarm. Related indications are as follows:

- Generator MW is slightly lower than at shift turnover
- Drywell pressure is 1.2 psig and steady, which is the same as at shift turnover
- Drywell temperature is 132 °F and steady, which is 2 °F higher than at shift turnover
- Torus water temperature is 78 °F and steady, which is the same as at shift turnover
- The Acoustic Monitor for Safety Valve NR-28J is in the red "Valve Open Region"
- Tailpipe temperature for NR-28J is 245 °F; all others are reading approximately 130 °F

This indicates ____(1)____. The correct action to take for this is to ____(2)____.

a. (1) a Safety Valve is open
(2) enter ABN-1, Reactor Scram

b. (1) a Safety Valve is open
(2) enter ABN-40, Stuck Open EMRV

c. (1) a Safety Valve is leaking
(2) commence an immediate plant shutdown

d. (1) a Safety Valve is leaking
(2) write an IR for Engineering to evaluate

Answer: d

Handouts: None

Justification: A and B are incorrect – safety valve NR-28J is leaking. If it were open, tailpipe temperature would be higher than 275 °F and drywell pressure would be rising. Entering ABN-1, Reactor Scram, would be the correct action to take if this were the case; ABN-40, Stuck Open EMRV, does not contain any guidance for an open safety valve.

C is incorrect – safety valve NR-28J is leaking, but not enough to warrant an immediate plant shutdown.

D is correct – based on the given conditions, this is the appropriate action to take for the leaking safety valve.

239002 A2.02
Ability to (a) predict the impacts of the following on the RELIEF/SAFETY VALVES; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Leaky SRV (CFR: 43.5)

OC Learning Objective: 2621.828.0.0005, Objective K.2:
Describe and interpret procedure sections and steps for plant emergency or off-normal conditions that involve this system including personnel allocation and equipment operation IAW applicable ABN, SDRP, EOP & EOP support procedures and EPIPs.

Cognitive Level: Comprehension or Analysis

Question Type: Modified Bank

References: RAP-B4g, ABN-40

21. The reactor was operating at rated power when a loss of all off-site power occurred. The following conditions currently exist:

- All control rods are inserted to or beyond position '04'
- RPV water level is 105 inches and slowly lowering
- Reactor pressure is being controlled at 900-1000 psig
- Power was restored to Bus C two minutes ago
- Emergency Diesel Generator #2 failed to start
- It is necessary to maximize CRD flow to restore and maintain RPV water level

How can this be accomplished given the current plant conditions?

a. Control CRD flow using the in-service Flow Control Valve NC-30, IAW Support Procedure 3
b. Close charging header supply V-15-52, lineup and throttle CRD bypass flow at less than or equal to 150 gpm IAW Support Procedure 3
c. Cross-tie USS 1A2 to 1B2 and start a second CRD pump IAW ABN-36, Loss of Off-Site Power. Then control CRD flow using the in-service Flow Control Valve NC-30 IAW Support Procedure 3
d. Cross-tie USS 1A2 to 1B2 and start a second CRD pump IAW ABN-36, Loss of Off-Site Power. Then close charging header supply V-15-52, lineup and throttle CRD bypass flow as necessary to restore RPV water level IAW Support Procedure 3

Answer: b

Handouts: None

Justification: A is incorrect – the scram is not reset (since RPV level is 105 inches). Support Procedure 3 directs controlling CRD flow using the in-service Flow Control Valve NC-30, only if the scram is reset.

B is correct – in cases where the scram cannot be reset, Support Procedure 3 directs closing V-15-52, then opening V-15-237 (CRD bypass isolation) and throttling V-15-20 (CRD bypass) so as not to exceed 150 gpm (for one or two pump operation). A CAUTION in SP 3 states "Operating one CRD pump at greater than 150 gpm may result in a pump trip."

C and D are incorrect – USS 1A2 and 1B2 cannot be cross-tied when reactor temperature is above 212 °F, EXCEPT during station blackout conditions, as directed by ABN-37, Station Blackout.

201001 A2.03
Ability to (a) predict the impacts of the following on the CONTROL ROD DRIVE HYDRAULIC SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Power supply failures (CFR: 43.5)

OC Learning Objective: 2621.828.0.0011, Objective 16:
Describe and interpret procedure sections and steps for plant emergency or off-normal conditions that involve this system including personnel allocation and equipment operation IAW applicable ABN, EOP & EOP support procedures and EPIPs.

Cognitive Level: Comprehension or Analysis

Question Type: New

References: ABN-36, Support Procedure 3

22. Given the following:

- An ATWS is in progress with the MSIVs closed
- The conditions required by RPV Control – With ATWS for re-opening the MSIVs have been met
- Differential pressure across the MSIVs is 280 psid

Which one of the following describes the operating restrictions for opening the MSIVs under these conditions?

Shift Manager approval __(1)__ required. Damage to downstream piping __(2)__ occur.

a. (1) IS
(2) MAY

b. (1) IS NOT
(2) MAY

c. (1) IS
(2) WILL

d. (1) IS NOT
(2) WILL

Answer: b

Handouts: None

Justification: A is incorrect – SM approval is NOT required.

B is correct – Procedure 301.1 specifies the following restrictions for opening the MSIVs:

- d/p $\leq$ 100 psid – no damage to piping; no approvals required
- 100 < d/p $\leq$ 160 psid – approaching limit where damage may occur; SM approval required
- d/p $\geq$ 160 psid – repeated opening may damage piping; open IAW EOP's.
- d/p $\geq$ 360 psig – single opening will damage piping; open IAW EOP's.

C is incorrect – SM approval is not required; damage to downstream piping MAY occur.

D is incorrect – damage to downstream piping MAY occur.

239001 G2.1.32 Main and Reheat Steam System / Conduct of Operations: Ability to explain and apply system limits and precautions. (CFR: 43.5)

OC Learning Objective: 2621.828.0.0026, Objective S:
Identify and interpret normal, abnormal, and emergency operating procedures for the various steam systems and explain the reasons for the applicable precautions and limitations.

Cognitive Level: Memory of Fundamental

Question Type: New

References: 301.1, EMG-3200.01B

23. A small-break LOCA occurred while at rated power. Current plant conditions are as follows:

- Reactor water level is 92 inches
- Reactor pressure is 500 psig
- Drywell pressure is 13 psig
- Drywell bulk temperature is 450 °F
- TR-IA55 on Panel 8R readings are as follows:
    - Point 40 = 452 °F
    - Point 41 = 453 °F
    - Point 42 = 451 °F
    - Point 43 = 448 °F
    - Point 44 = 446 °F

Which of the following RPV water level instruments can be used to determine RPV water level?

a. WR GEMAC ONLY
b. YARWAY A & B ONLY
c. NR GEMAC A & B ONLY
d. NR GEMAC A & B <u>and</u> YARWAY A & B ONLY

Answer: b

Handouts: EMG-3200.02, Support Procedure 28

Justification: A is incorrect – at 451 °F and 92 inches, the WR GEMAC instrument is in the UNSAFE region of the graph in Support Procedure 28.

B is correct – at 448 °F (446 °F) and 92 inches, both Yarway instruments are in the SAFE region of the graph in Support Procedure 28. None of the other instruments are in the SAFE region of their respective graph.

C is incorrect – at 452 °F (453 °F) and 92 inches, both NR instruments are in the UNSAFE region of the graph in Support Procedure 28.

D is incorrect – both NR instruments are in the UNSAFE region. Only the Yarway instruments are in the SAFE region.

G2.1.25 Conduct of Operations: Ability to obtain and interpret station reference materials such as graphs / monographs / and tables which contain performance data. (CFR: 43.5)

OC Learning Objective: 2621.828.0.0032, Objective T:
Given a set of plant conditions, interpret Control Room and/or local Primary Containment System indications and evaluate them in terms of limits and trends using available data.

Cognitive Level: Comprehensive or Analysis

Question Type: New

References: EMG-3200.02, Support procedure 28

24. Which of the following is <u>NOT</u> considered a refueling error, as specified in Procedure 205.0, Reactor Refueling?

    When loading fuel into the RPV, a ____(1)____ fuel assembly, discovered upon unlatching and lifting the grapple, and ____(2)____ north/south movement of the bridge.

    a. (1) mislocated
       (2) after

    b. (1) mislocated
       (2) prior to

    c. (1) misoriented
       (2) after

    d. (1) misoriented
       (2) prior to

Answer: d

Handouts: None

Justification: A and B are incorrect – <u>any</u> mislocated fuel assembly is considered a refueling error.

C is incorrect – a misoriented fuel assembly that is not discovered until after north/south movement of the bridge is considered a refueling error.

D is correct – as stated in 205.0, "if a misoriented fuel assembly is discovered upon unlatching and lifting the grapple, and prior to north/south movement of the bridge, then reorient the fuel assembly per the Fuel Move Sheet immediately without declaring a refueling error."

G2.2.31 Equipment Control: Knowledge of procedures and limitations involved in initial core loading. (CFR: 43.7)

OC Learning Objective: <u>2621.812.0.0003, Objective K:</u>
Describe, in general, refueling and fuel handling procedures to include precautions and limitations per Procedure 205 series.

Cognitive Level: Memory of Fundamental

Question Type: New

References: 205.0

25. Given the following:

- The plant is operating at 25% power
- A fire is reported and confirmed in the Reactor Water Cleanup Cage area
- Several minutes later, the following parameters are noted:
  - RPV pressure has lowered
  - MWe has lowered
  - Torus temperature is rising

Which of the following is the correct action?

a. Initiate torus cooling while maintaining reactor power constant
b. Initiate torus cooling while reducing recirc. flow to minimum
c. Scram the reactor and execute ABN-1, Reactor Scram
d. Locally trip the recirc. pumps whose controls will be affected by the fire

Answer: c

Handouts: ABN-29, Attachment 29-1 in its entirety.

Justification: The indications provided show that an EMRV has opened (see ABN-40) during the fire event on RB 51' (the cleanup cage is located on RB 51'). IAW ABN-29, if there is a fire on RB 51' and spurious operation of an EMRV, then the action is to scram the reactor and to close the open EMRV. Answer c is correct.

Answer a and b are incorrect – it does not direct a scram, even though torus cooling may appropriate soon.

Answer d is incorrect – this would be an appropriate action if the fire was on elevation 23. Since the fire is on elevation 51, this would be an incorrect answer

G2.4.27 Emergency Procedures/Plan: Knowledge of fire in the plant procedure. (CFR: 43.5)

OC Learning Objective: <u>2621.828.0.0019, Objective E:</u>
Describe and interpret procedure sections and steps for plant emergency or off-normal conditions that involve this system including personnel allocation and equipment operation in accordance with applicable ABN, SDRP, EOP and EOP support procedures, and EPIPs.

Cognitive Level: Comprehensive or Analysis

Question Type: Bank

References: ABN-29, ABN-40

The NRC initially wanted the handout deleted. We felt the question was not appropriate without the handout. The entire attachment to the ABN will be provided.

1. The plant was at 65% power. A malfunction occurred in the master recirculation controller which caused recirculation flow and reactor power to lower. The Reactor Operator has taken all recirculation speed controllers to MANUAL and the flow/power reduction has ceased. The following conditions exist:

   - Reactor power is 42% and steady
   - Reactor recirculation flow is 6×10^4 GPM

   Which of the following actions are required?

   a. Manually scram the reactor
   b. Raise reactor recirculation flow or insert control rods
   c. Lower recirculation flow or insert control rods
   d. Raise recirculation flow or withdraw control rods

Answer: b

HANDOUT: OC POWER OPERATION CURVE (from 202.1)

Justification: The master controller malfunction has placed the plant in the Exclusion Zone of the Power Operation Curve. IAW 202.1, Power Operation, the Exclusion Zone is a region where reactor operation is not allowed due to stability concerns. If the zone is entered inadvertently, then exit the zone by using rods or flow (the same wording is also used in 301.2, Reactor Recirculation System). ABN-2, Recirculation System Failures, provides more detailed instruction if the exclusion zone is entered and how to exit the zone: exit the exclusion zone by raising pump speed and/or inserting the CRAM array (control rods). Therefore, answer b is correct.

Answer a is incorrect since scram is not the appropriate action. Answers c and d are also incorrect responses.

295001 AK1.04
Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION : †Limiting cycle oscillation (CFR: 41.8 to 41.10)

OC Learning Objective: 2621.828.0.0040 (00226: Identify and interpret procedures for plant emergency/off-normal situations which involve the Recirc. System, including personnel and equipment allocations.)

Cognitive Level: Comprehension or Analysis

Question Type: Bank

8/15/06: NRC Comments
Verified that the plant conditions did not impose a control rod block. We lowered the resultant power from from 45% to 42% since the power/flow conditions originally given was close to a rod block (but was not imposed).

9/8/06
Rossi reviewed. Deleted: following a short forced outage, and restoration of rated power is underway.

2. The plant was at rated power, when the following annunciator came into alarm:

- VITAL POWER DC PWR LOST – BUS A/B UV

With the affected DC Bus at 0 volts, and in accordance with the applicable RAP, the Unit Supervisor has declared the following valves INOPERABLE:

- V-16-2, Inlet Isolation Valve to Cleanup Auxiliary Pump
- V-16-14, Cleanup System Inlet Isolation Valve
- V-14-31, Steam Inlet Valve to "A" Emergency Condenser
- V-14-34, Emergency Condenser NE01A Condensate Return Valve

Which of the following automatic actions should have occurred as a result of this event?

a. 125 VDC DC-D transfers to 125 VDC DC A

b. 125 VDC DC-1 transfers to 125 VDC DC C

c. 125 VDC DC-E transfers to 125 VDC DC B

d. 125 VDC DC-2 transfers to 125 VDC DC A

answer: a

Justification: The given annunciator, along with the inoperable valves are enough to determine that 125 VDC Bus B is the effected DC bus. When voltage is lost to the bus, 125 VDC DC-D will automatically transfer from 125 VDC Bus B to 125 VDC Bus A. Answer a is correct.

125 VDC DC-1, also normally supplied by Bus B, also transfers to Bus A, not Bus C. Answer b is incorrect.

125 VDC DC-E is normally powered from 125 VDC Bus A and is unaffected by the event. DC-E would transfer to Bus B on a loss of volts to Bus A. Answer c is incorrect.

125 VDC DC-2, powered from 125 VDC Bus C is unaffected by the event. Answer d is incorrect.

References: RAP-9XF1d, revision 2; BR 3000, Electrical Power System Key One Line Diagram, revision 8; ABN-54, DC Bus B and Panel/MCC Failures, revision 1.

295004 AK1.02
Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER : Redundant D.C. power supplies: Plant-Specific (CFR: 41.8 to 41.10)

OC Learning Objective: 2621.828.0.0012 (01121: State potential consequences on plant operation, plant equipment and environment due to failure of DC Electrical System.)

Cognitive Level: Comprehensive or Analysis

Question Type: Bank

9/8/06
Rossi reviewed. Deleted: with all systems normally aligned in question stem.

3. The plant was at rated power when the following annunciators came into alarm over a short period of time:

- TURBINE VAC/SEALS - COND VAC LO 25 INCHES
- MAIN STEAM – COND VAC LO/TURB TRIP I and II
- TURBINE VAC/SEALS – COND VAC TRIP 1 22 INCHES
- TURBINE VAC/SEALS – COND VAC TRIP 2 10 INCHES

Condenser vacuum continues to degrade. The following conditions currently exist:

- RPV water level lowered to 130” and has recovered to 170”, and is stable
- All control rods indicate full-in

Which of the following systems will be used for RPV pressure control?

a. EMRVs

b. Isolation Condensers

c. Turbine Bypass Valves

d. Isolation Condenser Vent

answer: a

Justification: The turbine bypass valves are not available due to the loss of condenser vacuum (RAP-Q1c). Answer c is incorrect.

Isolation Condenser vents are unavailable since condenser vacuum is lost (Support Procedure 15 of EMG-3200.01A requires the condenser to be intact). Answer d is incorrect.

EMRVs are allowed IAW Support Procedure 12 of EMG-3200-01A. Answer a is correct. (There is no indication that torus water is too low that would preclude their use.)

The use of Isolation Condensers is prohibited due to RPV water level of 170”. Both EMG-3200.01A and ABN-1 require RPV water level less than 160”. Answer b is incorrect.

295005 AK2.07
Knowledge of the interrelations between MAIN TURBINE GENERATOR TRIP and the following: Reactor Pressure Control (CFR: 41.7)

OC Learning Objective: 2621.845.0.0004 (03012: Utilize appropriate EOP Support Procedures to determine various parameters required to support operation under the SBEOPs.)

Cognitive Level: Comprehension or Analysis

Question Type: New

8/15/06: NRC Comments
Added that condenser vacuum continued to degrade.

9/8/06
Rossi reviewed. Re-ordered answer selections from short-long. Changed the answer and justification to match new selections.

4. The reactor was at rated power when the Shift Manager declared the control room NOT habitable due to a toxic substance release, and that a control room evacuation is required.

Prior to leaving the control room, the following actions were taken:

- The reactor was scrammed and isolated
- The turbine was tripped
- Isolation Condenser B was placed into service.

While at the Remote Shutdown Panel, you have recorded the following RPV pressures:

Time (hhmm)	RPV Pressure (psig)
1100	1000
1110	895

Which of the following is correct regarding the RPV cooldown rate (assume the cooldown rate does not change)?

The RPV cooldown rate is...........

	Allowed by procedure 203, Plant Shutdown	Allowed by Tech Specs
a.	Less than allowed	Less than allowed
b.	Greater than allowed	Less than allowed
c.	Greater than allowed	Equal to allowed
d.	Greater than allowed	Greater than allowed

answer: a

HANDOUT: ATTACHMENT ABN-30-4

Justification: Procedure 203, Plant Shutdown (step 6.66) requires that normal cooldown rate be limited to < 15°/10 minute interval = 90° F/hour. TS 3.3.C.1 limits the cooldown rate to 100° F/hour.

From Attachment ABN 30-4, 1000 psig = 546.22° F, and 895 psig = 533.29° F (must interpolate). This gives a temperature change of 12.93° in a 10 minute period (which equals 77.58° F/hour). This cooldown rate is less than procedure 203, and less that TS 3.3.C.1. Answer a is correct. All other answers are incorrect.

295016 AA2.06
Ability to determine and/or interpret the following as they apply to CONTROL ROOM ABANDONMENT : Cooldown Rate (CFR: 41.10)

OC Learning Objective: 2621.828.0.0064 (10445: Given a set of system indications or data, evaluate and interpret them to determine limits, trends and system status.)

Cognitive Level: Comprehensive or Analysis

Question Type: New

8/15/06: NRC Comments
Added to assume that the cooldown rate does not change.

9/8/06
Rossi reviewed. Added “release” in stem, and changed answer selections to 3-column format.

5. The reactor is at rated power.

Which of the following would require an entry into Technical Specifications if instrument air were lost to the listed system/component?

a. Feedwater Control System

b. Scram Discharge Volume

c. Reactor Recirculation System

d. Shutdown Cooling System

Answer: b

HANDOUT: None

Justification: IAW ABN-35, a loss of air to the feedwater flow control system, the flow control valves will lockup, and may be manually controlled locally. There is no TS for these valves. Answer a is incorrect.

A loss of air to the scram discharge volume results in the vent/drain valves failing closed. TS 4.2.G requires that the SDV vent/drain valves verified open at least once per 31 days. Answer b is correct.

A loss of air to the reactor recirculation system results in the lockup of the fluid couplers, and can be manually controlled locally. A loss of air to the shutdown system results in the minimum flow valves failing open. There are no TS associated with recirculation pumps in manual control nor with an shutdown cooling loop. Answers c and d are incorrect.

295019 2.1.33
Ability to recognize indications for system operating parameters which are entry-level conditions for technical specifications. (Partial or complete loss of instrument air) (CFR: 43.2 / 43.3) (CFR 41.7 – this is my tie to RO CFR)

OC Learning Objective: 2621.850.0.0090 (01661: Using the Tech Specs, determine if the LCO requirements are/are not being met and determine the appropriate plant/operator response and state the basis for response.)

Cognitive Level: Comprehensive or Analysis

Question Type: New

8/15/06: NRC Comments
Deleted handout TS 4.2.

6. While at the controls during a fuel shuffle, you are notified that an irradiated fuel bundle was dropped while being moved over the core.

   Which of the following would be an expected radiation monitoring response from this event, if the design basis release were to occur?

   a. 119 elevation radiation monitor C10 will indicate elevated radiation levels, and when tripped high, will initiate the Standby Gas Treatment System (after a time delay)
   b. 119 elevation radiation monitor C5 will indicate elevated radiation levels, and when tripped high, will isolate the DW vent/purge valves (after a time delay)
   c. 119 elevation radiation monitor C9 will indicate elevated radiation levels, and when tripped high, will initiate the Standby Gas Treatment System (after a time delay)
   d. 119 elevation radiation monitor B9 will indicate elevated radiation levels, and when tripped high, will isolate the DW vent/purge valves (after a time delay)

Answer: c

Justification: All the listed radiation monitors measure radiation levels on the refuel floor (119'). Only rad monitors C9 and B9, when tripped high, will initiate SGT after a short time delay. Rad monitors C5 and C10 initiate no protective actions. The containment high range radiation monitors (CHRRM), when tripped high, will isolate DW/Torus vent and purge valves. Answer c is correct. All other answers are incorrect. (see RAP-10F1m, -10F2m, -10F3m, -10F4m, and –10F4k.)

295023 AA1.04
Ability to operate and/or monitor the following as they apply to REFUELING ACCIDENTS : Radiation Monitoring Equipment (CFR: 41.7)

OC Learning Objective: 2621.828.0.033A (00819: State any automatic actions initiated by the ARM System. State which monitors provide these actions and the setpoints.)

Cognitive Level: Comprehensive or Analysis

Question Type: Modified

8/15/06: NRC Comments
Added if the DB release were to occur.

7. The reactor was at rated power when an RPV over-pressure event occurred. One electromatic relief valve (EMRV) opened momentarily as designed.

While the EMRV was <u>open</u>, which of the following is correct? (select one from each part)

- Control Room panel *EMRV position indicating lights* are a(n) <u>1 (direct/indirect) </u>indication of EMRV position;
- *EMRV tailpipe temperature indication* is a(n) <u>   2 (direct/indirect)</u> indication of EMRV position.

a. (1) direct
(2) direct

b. (1) indirect
(2) direct

c. (1) direct
(2) indirect

d. (1) indirect
(2) indirect

Answer: b

Justification: The control room panel EMRV position indicating lights show the position of the EMRV pilot valve position – not the EMRV. This is an indirect indication of the actual EMRV position. The EMRV tailpiece temperature indicators indicate temperature in the EMRV tailpiece. Only when the EMRV is open, will there be elevated temperatures in the tailpiece. This provides direct indication of the EMRV position. (See RAP-B3g, -B4g, ABN-40)

295025 EK1.03
Knowledge of the operational implications of the following concepts as they apply to HIGH REACTOR PRESSURE : Safety/relief valve tailpipe temperature/pressure relationships (CFR: 41.8 to 41.10)

OC Learning Objective: 2621.828.0.0026 (00538: Describe Control Room and/or local steam system indications.)

Cognitive Level: Comprehensive or Analysis

Question Type: New

8/15/06: NRC Comments
Some question was raised over this question regarding the meaning of direct/indirect. Left as-is.

9/8/06
Rossi reviewed. Changed tailpiece to tailpipe.

8. In accordance with the Primary Containment Control EOP, EMG-3200.02, before bulk drywell temperature reaches 281° F, drywell sprays are lined-up and initiated.

   Which of the following states why containment sprays are initiated before the bulk drywell temperature reaches 281° F?

   Spraying the drywell will ensure that the……

   a. environmental qualification temperature of the EMRV solenoids is not exceeded
   b. environmental qualification temperature of the drywell/torus vent and purge valve solenoids is not exceeded
   c. drywell design temperature of 281° F at a design internal drywell pressure of 48 psig is not exceeded
   d. drywell design temperature of 281° F at a design internal drywell pressure of 35 psig is not exceeded

Answer: d

Justification: A drywell temperature of 281° F is the drywell design temperature at 35 psig (see USAR 6.2.1.3.5 and 3.8.2.3.b.2 and EOP Users Guide, 2000-BAS-3200.02). As stated on page 2-24, the EQ temperature of safety related equipment is only slightly above this temperature. The drywell design is 292° F at 44 psig and 281° F at 35 psig. Answer d is correct. All other answers are incorrect.

295028 EK3.03
Knowledge of the reasons for the following responses as they apply to HIGH DRYWELL TEMPERATURE : Drywell Spray Operation
(CFR: 41.5 / 45.6)

OC Learning Objective: 2621.845.0.0007 (03000: Using procedure EMG-3200.02, evaluate the technical bases foe each step in the procedure and apply this evaluation to determine correct courses of action under emergency conditions.)

Cognitive Level: Memory or Fundamental

Question Type: Bank

8/15/06: NRC Comments

They commented if answers c and d were RO knowledge components. Lesson plan on Primary Containment (2621.828.0.0032) did have an objective to state the PC temperature and pressure limits.

9/8/06
Rossi reviewed. Changed ‘lists’ to ‘states’ in the question.

9. The reactor was at rated power, when the following annunciators <u>came into alarm</u>:

- REACTOR LEVEL – RX LVL LO I
- REACTOR LEVEL – RX LVL LO II

Which of the following states (1) where the Feedwater Control System will control RPV water level in AUTO (prior to any Operator actions), and (2) the procedurally required manual operator actions to control RPV water level?

	Feedwater Control System	Action
a.	Will control RPV water level at the pre-scram level setpoint	Trip two feedwater pumps when RPV water level begins to rise
b.	Will control RPV water level at the post-scram level setdown level setpoint	Trip two feedwater pumps when RPV water level begins to rise
c.	Will control RPV water level at the post-scram level setdown level setpoint	Trip two feedwater pumps when RPV water level reaches 140”
d.	Will control RPV water level at the pre-scram level setpoint	Trip two feedwater pumps when RPV water level reaches 140”

Answer: b

Justification: RAP-H5e and –H6e (RX LVL LO) require if a scram occurs, to verify actuation of the post scram level setdown and to perform followup actions of ABN-1. (SP-2 of RPV Control – No ATWS also says the same correct answer.)

Following a scram and lowering RPV water level, feedwater level control will attempt to control RPV water level at the reactor level setdown setpoint (142”) (when feedwater level control is left in AUTO). ABN-1, Reactor Scram, requires that when RPV water level begins to rise, to trip two feedwater pumps. When RPV water level reaches 140”, to place the main feed regulating valves in manual and close. Answer b has both correct components and is correct. All other answers provide the incorrect setpoint after the scram or the incorrect operator actions. (See also MDD-OC-625-B.)

295031 (Reactor Low Water Level)
2.4.31 Knowledge of annunciators alarms and indications / and use of the response instructions.

OC Learning Objective: 2621.828.0.0018 (10446: Identify and explain system operating controls/indications under all plant operating conditions.)

Cognitive Level: Comprehensive or Analysis

Question Type: Modified

9/8/06
Rossi reviewed. Deleted: with all systems normally aligned from the stem. Placed answer selections in 3-column format. Added RPV Control – N0 ATWS Support Procedure 2 to justification. Changed feedwater control to Feedwater Control System in the question.

10. The plant was at rated power when an event occurred resulting in an airborne radiological release outside of the plant structures. The current conditions exist:

- All control rods indicate full-in
- A radiological release is in-progress

Which of the following states how and why the control room HVAC system should be aligned?

a. System A must be run in the PURGE Mode, to remove contaminated air from the Control Room, utilizing the fan <u>only</u>

b. System B must be run in the PURGE Mode, to remove contaminated air from the Control Room, utilizing the fan <u>only</u>

c. System A must be run in the PART RECIRC Mode to maintain a positive pressure in the Control Room

d. System B must be run in the FULL RECIRC Mode to minimize the use of outside air into the Control Room

Answer: c

Justification: There are no automatic actions of the control room ventilation system from any high radiation signal.

Procedure 331.1, Control Room and Old Cable Spreading Room Heating, Ventilation and Air Conditioning System, describes the partial recirculation mode: this mode of operation is provided to minimize contamination infiltration into the control room by maintaining a positive pressure in the control room using partial outside air.

Section 6.1.1 of 331.1, provides guidance for a radiological release with offsite power available. With offsite power available, System B or System A should be run in PART RECIRC mode. Only when there is a loss of offsite power, shall the System be run with the fan only (to limit EDG loading). Answer c is correct.

Running System A in the PURGE mode is incorrect. Purge mode is used to remove smoke, fumes, or other undesirable odors from the control room. Also, running the systems with fans only is required only when combined with a loss of off-site power to reduce EDG loading. Answer a is incorrect.

Running System B in the PURGE mode is incorrect. Purge mode is used to remove smoke, fumes, or other undesirable odors from the control room. Answer b is incorrect.

Running System A in the FULL RECIRC Mode is incorrect. Full Recirc mode is used to minimize the intrusion of toxic gases into the control room. Answer d is incorrect.

295038 EK3.03
Knowledge of the reasons for the following responses as they apply to HIGH OFF-SITE RELEASE RATE: Control Room Ventilation Isolation (CFR: 41.5)

OC Learning Objective: 2621.828.0.0054 (02324: Explain the basis, with use of the procedures, for the four different modes of control room ventilation damper alignment and the effects of the damper alignment modes on control room habitability.)

Cognitive Level: Comprehensive or Analysis

Question Type: Modified

9/8/06
Rossi reviewed. Deleted: The outside air temperature is 50° F and the control room air temperature is 74° F in the stem. The first draft of this question included a loss of off-site power. With the loss of power, air temperatures became a concern to make the question correct. In the final version of the question with power available, air temperatures are no longer a concern. Changed ‘lists’ to ‘states’ in the question. Deleted duplicate word in stem.

11. The reactor was at rated power when the following annunciator <u>came into alarm</u>:

- TURBINE VAC/SEALS – COND VAC LO 25 INCHES

The reactor operator lowered recirculation flow as directed by the associated RAP. Condenser vacuum has recovered to 25.8" and is steady. The Unit Supervisor directs you to restore RPV pressure to the pre-event value by adjusting the electronic pressure regulator (EPR), in accordance with 202.1, Power Operations.

Which of the following lists the required action and its effect?

Take the EPR RELAY POSITION control switch to ___1_____ position which will cause turbine control valves to _____2_____ .

a. (1) LOWER (↑%) (2) close further

b. (1) LOWER (↑%) (2) open further

c. (1) RAISE (↓%) (2) open further

d. (1) RAISE (↓%) (2) close further

Answer: d

Justification: As power is reduced, the EPR relay position also goes down (proportional to turbine load). To raise RPV pressure back up, the turbine control valves must close down some. Lowering the EPR relay position even further will do this (Raise (↓%)). As the TCV close down some, RPV pressure will rise. Therefore, the EPR relay position must be taken to the RAISE position, which will cause turbine control valves to close further, causing RPV pressure to rise. Answer d is correct. All other answers either manipulate the switch in the incorrect direction or the plant effect is incorrect. (See also procedure 315.4.)

RAP-Q3c directs a power reduction to maintain vacuum > 25".

295002 AA1.06
Ability to operate and/or monitor the following as they apply to LOSS OF MAIN CONDENSER VACUUM : Reactor/turbine pressure regulating system (CFR: 41.7)

OC Learning Objective: 2621.828.0.0051 (10446: Identify and explain system operating controls/indications under all plant operating conditions.)

Cognitive Level: Comprehensive or Analysis

Question Type: Modified

9/8/06
Rossi reviewed. Changed ‘switch’ to ‘control switch’ and added ‘position’ in stem. Corrected typo (valve to value). Underlined ‘into alarm’.

12. The plant was at rated power, when the following annunciators came into alarm:

- TORUS/DRYWELL – DW SUMP HI LEAK/PWR FAIL
- TORUS/DRYWELL – DW PRESS HI/LO

Drywell pressure peaked at, and currently indicates 1.4 psig, and the PWR FAIL has been ruled out as a cause.

Which of the following would NOT be used to determine the drywell unidentified leak rate?

a. Directly, by reading the Unidentified Drywell Leakage recorder (ULRM-1) on Panel 3F
b. Calculate, given the drywell sump flow integrator readings and times of the readings
c. Calculate, given the time between the drywell sump low and high level alarms
d. Estimate, given the drywell equipment drain tank leak rate and condenser hotwell makeup rate

answer: d

Justification: The given question stem identifies an increased DW pressure from an increase in DW unidentified leakage.

Procedure 312.9, Primary Containment Control, provides directions on how to calculate the DW unidentified leak rate: take the difference between the DW sump integrator readings and divide by the elapsed time of the readings (this is the method used for the daily surveillance). There also exists an unidentified DW leakage recorder on Panel 3F. If these are not functional, procedure 351.1, The Chemical Waste/Floor Drain Operating Procedure, provides a method to calculate DW unidentified leakage: measure the time between the low and high sump level alarms, and divide into the sump volume between these two alarms. Answers a, b, and c are all acceptable methods. Answers a, b, and c are incorrect.

Answer d is a mix of identified leak rates and possible sources of unidentified leak rates. Answer d is correct. (Refer to drawings 147474, RAP-C3h, and procedures 351.1, 351.2, and 312.9).

295010 AA2.01
Ability to determine and/or interpret the following as they apply to HIGH DRYWELL PRESSURE : Leak Rates (CFR: 41.10)

OC Learning Objective: 2621.828.0.0032 (00418: Given control panel indications, interpret the cause of Primary Containment System alarms, alone and in combination, as applicable.

Cognitive Level: Comprehensive or Analysis

Question Type: New

8/15/06: NRC Comments
Added that DW pressure peaked at 1.4 psig (to show that 2.9 psig was never reached).

9/8/06
Rossi reviewed. Deleted: with all systems normally aligned in stem.

13. The plant was at rated power when the Secondary Containment Control EOP, EMG-3200.11, was entered due to high area temperatures (not due to a fire).

    Which of the following area leak detection system annunciators will result in automatic isolation of the affected system?

    a. Cleanup System area leak detection: CLEANUP SYSTEM – RWCU HELB annunciators
    b. Shutdown Cooling System area leak detection: SD HX CLG – SD HX PUMP RM TEMP HI annunciators
    c. Isolation Condenser System area leak detection: ISOL COND – COND AREA TEMP HI annunciators
    d. Trunion Room area leak detection: MAIN STEAM – TRUNION RM TEMP HI annunciators

Answer: a

Justification: Cleanup system leaks will be annunciated by D1d and D2d (RWCU HELB at 160° F) and by D8d (CU ROOM TEMP HI). The HELB annunciators, when alarmed simultaneously, will isolate the cleanup system at 160° F area temperature. The other cleanup alarm does no auto action. Answer a is correct.

Shutdown cooling system leaks will be annunciated by C8d (SD HX PUMP RM TEMP HI) but provide no automatic actions. Answer b is incorrect.

Isolation condenser leaks will be annunciated by C8b (COND AREA TEMP HI) but provide no automatic actions. Answer c is incorrect.

Trunion room leaks will be annunciated by J8a (TRUNION RM TEMP HI) but provide no automatic actions. Answer d is incorrect. Main steam leaks into the steam tunnel (trunion room) are annunciated by J3a and J4a (FLOW HI/MN STM LINE AREA TEMP HI-HI 1 and II) but answer d is asking specifically for trunion room leak detection.

295032 EK1.03
Knowledge of the operational implications of the following concepts as they apply to HIGH SECONDARY CONTAINMENT AREA TEMPERATURE: Secondary containment leakage detection (CFR: 41.8 to 41.10)

OC Learning Objective: 2621.828.0.0039 (10449: State the function of system alarms, alone and in combination, as applicable in accordance with the system RAPs.)

Cognitive Level: Memory of Fundamental

Question Type: New

14. The plant is at rated power. While the NLO was investigating the Reactor Building Sump 1-7 high level alarm, he states that due to an apparent fault in the RB sump 1-7 control circuitry, neither sump pump will start.

    Which of the following actions is required?

    a. Manually isolate the inputs from RB Sump 1-6 into RB Sump 1-7
    b. Check that RB Sump 1-6 inputs into RB Sump 1-7 have automatically isolated
    c. Manually isolate Drywell Floor Drain Sump inputs into RB Sump 1-7
    d. Check that the Drywell Floor Drain Sump inputs into RB Sump 1-7 have automatically isolated

Answer: b

Justification: With the RB Sump 1-7 high alarm in (RAP-RB1C(1-7), the Secondary Containment EOP is entered (EMG-3200.11). All systems discharging into the sump should be isolated (except for systems required for EOPs or fire suppression – of which, neither currently apply). RB Sump 1-6 discharges into RB Sump 1-7, and sump 1-7 automatically isolates on a high water level alarm (which was given). According to the applicable RAP, the NLO should check for automatic valve closure of inputs from Sump 1-6 into Sump 1-7. Answer b is correct.

Since sump 1-6 output automatically isolated, no manual valve manipulations to isolate are required. Answer a is incorrect.

The drywell floor drain sump discharges to the chemical waste/floor drain collection tanks (the same place where RB sump 1-7 pumps discharge to (with a discharge check valve). Therefore, there should not be any input into RB sump 1-7 from the drywell floor drains. Answers c and d are incorrect. (See drawing 147434, sheet 3, rev. 58)

295036 EK 1.02
Knowledge of the operational implications of the following concepts as they apply to SECONDARY CONTAINMENT HIGH SUMP/AREA WATER LEVEL : Electrical ground/ circuit malfunction (CFR: 41.8 to 41.10)

OC Learning Objective: 2621.828.0.0015 (1414: Describe the operation of the pumps and level instrumentation associated with Reactor Building Sumps 1-6 and 107 including automatic isolations.)

Cognitive Level: Comprehensive or Analysis

Question Type: New

9/8/06
Rossi reviewed. Deleted: with all systems normally aligned in stem. Simplified the stem information.

15. The plant was at rated power with all systems normally aligned. The following annunciators came into alarm:

- ISOL COND – COND AREA TEMP HI
- RADIATION MONITORS AREA – AREA MON HI
- ISOL COND – COND A FLOW HI POSSIBLE RUPTURE

The Operator verifies the isolation condenser area rad monitor is above the high setpoint (Panel 2R) and area temperature has risen (Panel 10R).

Which of the following states the expected Isolation Condenser A lineup in this condition? (assume no operator actions)

a. The steam supply valves, condensate return valves and vent valves indicate OPEN
b. The steam supply valves, condensate return valves and vent valves indicate CLOSED
c. The steam supply valves and the condensate return valves indicate OPEN, and the vent valves indicate CLOSED
d. The steam supply valves and the condensate return valves indicate CLOSED and the vent valves indicate OPEN

answer: d

Justification: All three annunciators point to a rupture in Isolation Condenser A, in the vicinity of the IC (rad levels and temperatures: Rap-C8b, RAP-10F1k). These first annunciators have no automatic actions associated with them. The third annunciator (RAP-C3a), will isolate the steam and condensate return valves for the associated IC A from high Dp (high flow) (See also drawing 3029, sheet 2). The IC vent valves are unaffected by the isolation signal to the steam and condensate return valves. The vent valves, which are normally open while at power (auto close on system initiation), remain open following the isolation signal. The only answer which lists steam and condensate return valves closed and vent valves open, is answer d. Answer d is correct. All other answers are incorrect, since they provide the incorrect valve lineup. (The associated RAP does require the operator to close the vent valves, but no operator action is assumed in the question.)

207000 K4.01
Knowledge of ISOLATION (EMERGENCY) CONDENSER design feature(s) and/or interlocks which provide for the following: Isolation of the system in the event of a line break (CFR: 41.7)

OC Learning Objective: 2621.828.0.0023 (02030: Describe the Isolation Condenser design features and/or interlocks (including signals and setpoints) which provide for the following: 1) automatic system initiation; 2) automatic system isolation.)

Cognitive Level: Comprehensive or Analysis

Question Type: New

9/8/06
Rossi reviewed. Changed ‘lists’ to ‘states’ in the question.

16. Core Spray Main Pump NZ01A was being lined-up after completing pump maintenance and suction valve V-20-3 packing replacement. The surveillance test requires both flow measurements and valve timing to prove operability.

    If the System was not properly vented, which of the following states the impact during the surveillance?

    a. The suction valve would take less time to close
    b. The motors would run at a constant higher amperage value
    c. System discharge pressure would indicate a higher valve
    d. System flow would indicate a lower value

Answer: d

Justification: Running the pump with the system not fully vented would not impact the closure time of the discharge valve. Answer a is incorrect.

The pump would run with a reduced capacity. Answer d is correct. With a reduced capacity, motor amps would be less. Answer b is incorrect.

System discharge pressure would be less – not greater. Answer c is incorrect.

209001 K5.05
Knowledge of the operational implications of the following concepts as they apply to LOW PRESSURE CORE SPRAY SYSTEM : System Venting (CFR: 41.5)

OC Learning Objective: 2621.828.0.0010 (209-10445: Given a set of system indications or data, evaluate and interpret them to determine limits, trends and system status.)

Cognitive Level: Memory or Fundamental

Question Type: Modified

8/15/06: NRC Comments
The original question did not reflect the K/A. The question above is new.

9/8/06
Rossi reviewed. Changed stem to provide the necessary information while maintaining economy of wording.

17. The reactor was at rated power. An event occurred that caused the Reactor Operator to manually scram the plant. Reactor power was 22% following the scram. The following conditions exist:

- Reactor pressure is 1004 psig
- Standby Liquid Control System 1 was initiated.

Which of the following shows the expected indications for SLC System 1?

	PUMP ON Light	SQUIBS Light	Pump Discharge Pressure (psig)
a.	ON	ON	1085
b.	ON	OFF	1085
c.	OFF	ON	985
d.	ON	OFF	985

Answer: a

Justification: There is no automatic initiation of the SLC System. In a normal standby configuration, both the PUMPS ON and SQUIBS lights are OFF (pumps are off and squib valves are energized). When a system is manually initiated, both lights go ON (see procedure 304, and EMG-3200.01B, Support Procedure 22). SLC discharge pressure should be some valve greater than RPV pressure. Answer a is correct.

No other answer has this correct combination and are therefore incorrect.

211000 A3.02
Ability to monitor automatic operations of the STANDBY LIQUID CONTROL SYSTEM including: Explosive valves indicating lights: (CFR: 41.7)

OC Learning Objective: 2621.828.0.0046 (10446: Identify and explain system operating controls/indications under all plant operating conditions.)

Cognitive Level: Memory or Fundamental

Question Type: Modified

9/8/06
Rossi reviewed. Deleted: with all systems normally aligned.

18. A plant start-up is in-progress. The following plant conditions exist:

- Reactor power is on Range 8 of the Intermediate Range Monitors (IRM)
- IRM 11 is in BYPASS due to erratic detector output
- Control rods are being withdrawn to raise reactor power

The following annunciator came into alarm, followed by the listed automatic system initiation:

- VITAL POWER DC PWR LOST – 24 VDC PP-A PWR LOST
- Standby Gas Treatment System automatically initiated

Which of the following lists the effects on the IRM System from this event?

a. IRMs 11-14 meters indicate downscale on Panel 3R, and a rodblock and ½ scram exists
b. IRMs 11-14 meters indicate downscale on Panel 4F, and a rodblock only exists
c. IRMs 11-14 meters indicate upscale on Panel 3R, and a rodblock and ½ scram exists
d. Only IRMs 12-14 meters indicate upscale on Panel 4F and a rodblock and ½ scram exists

Answer: a

Justification: The indications in the stem are those of a loss of 24 VDC Panel A (RAP-9XF7d). Power is lost to 11, 12 SRMs and to 11-14 IRMs, and to the SGT trip relays (which causes SGT to auto start). The IRM trip auxiliary relays (and the IRM drawers on Panel 5R) (see drawings 706E812, sheet 9, 3 and 237E566, sheet 1) are powered from 24 VDC. The trip auxiliary relays are normally energized. When power is lost, all trips are instituted (upscale rodblock, upscale scram, inop. and rodblocks). The effected IRM drawers also show downscale on the meter from loss of power. Therefore, answer a is correct: the IRM meter shows downscale, and a rodblock and ½ scram exist. All other answers either provide incorrect meter indication or wrong trips.

295003 K6.05
Knowledge of the effect that a loss or malfunction of the following will have on the INTERMEDIATE RANGE MONITOR (IRM) SYSTEM : Trip Units (CFR: 41.7)

OC Learning Objective: 2621.828.0.0029 (10444: Describe the interlock signals and setpoints for the affected system components and expected response including power loss or failed components.)

Cognitive Level: Comprehensive or Analysis

Question Type: New

8/15/06 NRC Comments
They pointed out that RAP-9XF7d (24 Volt PP-A Pwr Lost) says that IRMs 15-18 fail downscale, but also says on the very same page that the plant will receive a ½ scram from IRMs 11-14. They wanted another correct reference. The companion rap for loss of 24 Volt PP-B mentions just IRMs 15-18 twice. See also drawing 3C-736-11-001. This shows 24 VDC Power Panel A feeding control room panel 3R (IRMs 11-14). See also drawing 706E812, sheet 9 for IRM 11.

9/8/06
Rossi reviewed. Changed ‘underway’ to ‘in-progress’ in the stem. Changed ‘lists’ to ‘states’ in stem. Corrected justification: power is lost to SRM 21, 22 (instead of incorrect designation SRM 11, 12).

19. A plant startup is in-progress. The following plant conditions exist:

- SRM 22 has failed and is in BYPASS
- The 8th control rod has just been fully withdrawn

An event occurs which results in the loss of instrument power to SRM drawer 24.

Which of the following lists the neutron monitoring indications from this event?

a. SRM recorder (Panel 4F) has lost power
b. Channel 24 period meter (Panel 4F) indicates infinity
c. SRM 24 meter (Panel 5R) indicates upscale
d. Channel 24 period meter (Panel 5R) indicates downscale

answer: d

Justification: 24 VDC powers the SRM drawer, including the trip relays. A loss of instrument power results in the downscale indication of the SRM meters and period meters, both on Panel 5R and 4F. Therefore, answer d is correct. (see drawings 706E812, sheets 4, 47, procedure 401.1)

The SRM recorder power comes from 120 VAC CIP Div 1 (see drawings 706E812, sheets 3 and 4). Therefore, the SRM recorder is still powered. Answer a is incorrect.

Answers b and c are incorrect since both fail downscale.

215004 K6.05
Knowledge of the effect that a loss or malfunction of the following will have on the SOURCE RANGE MONITOR (SRM) SYSTEM : Trip units (CFR: 41.7)

OC Learning Objective: 2621.828.0.0029 (10444: Describe the interlock signals and setpoints for the affected system components and expected response including power loss or failed components.)

Cognitive Level: Memory or Fundamental

Question Type: New

9/8/06
Rossi reviewed. Changed ‘underway’ to ‘in-progress’ in the stem. Changed ‘lists’ to ‘states’ in stem.

20. The reactor is at rated power. Below are the currently bypassed Local Power Range Monitors (LPRMs) into Average Power Range Monitors (APRMs):

APRM 1	APRM 5	APRM 6
28-33A	44-33D	04-33B
28-49C	36-41B	20-49D
36-41A		

Which of the following additional LPRM inputs to APRMs:

(1) CANNOT be bypassed (and maintain APRMs OPERABLE), as allowed by procedure 403, LPRM-APRM System Operations, and

(2) the effect if bypassed?

a. (1) 44-33A
(2) Too many LPRM inputs bypassed resulting in an automatic ½ scram

b. (1) 36-41D
(2) Too many LPRM inputs bypassed in one radial location resulting in an automatic ½ scram

c. (1) 28-49D
(2) Too many LPRM inputs bypassed resulting in an automatic ½ scram

d. (1) 20-49B
(2) Too many LPRM inputs bypassed in one radial location resulting in an automatic rodblock

Answer: a

HANDOUT: Attachment 202.1-1, Daily APRM Status Check

Justification: Procedure 403 has 2 precautions: 5.2.2.2 (and 5.3.2.3) says "Each APRM requires at least 5 LPRM signals. Inadvertently bypassing a 4th LPRM signal will initiate an INOP trip." 5.3.2.4 says "Failure of, or bypassing, two chambers from one radial location in any one APRM shall make that APRM channel inoperable."

Answer a bypasses the 4th LPRM from APRM 1 and this APRM will become inoperable, and an automatic ½ scram occurs. (APRM 1 and 5 are located in the same core quadrant. APRM 1 has a and c LPRM inputs. Even though the given table does not show any LPRMs into APRM 1 from 44-33, it can be seen that LPRM 44-33D inputs into APRM 5. Therefore, LPRM 44-33A must input into APRM 1.) Answer a is correct.

Answer b bypasses a second LPRM in the same radial location in APRM 15, which makes APRM 5 inoperable. There is no automatic function from this bypass. Answer b is incorrect.

Answer c bypasses a 3rd LPRM in APRM 5 and is allowed. No automatic function occurs from this bypass. Answer c is incorrect.

Answer d bypasses a second LPRM in the same radial location for APRM 6, which makes APRM 6 inoperable. There is no automatic action from this bypass. Answer d is incorrect.

215005 2.1.32
Ability to explain and apply system limits and precautions: APRM/LPRM (CFR: 41.10)

OC Learning Objective: 2621.828.0.0029 (10444: Describe the interlock signals and setpoints for the affected system components and expected response including power loss or failed components.)

Cognitive Level: Comprehensive or Analysis

Question Type: Modified

8/15/06: NRC Comments
Added “additional” LPRMS…

21. The reactor is at rated power. Which of the following would prevent the ability to determine reactor coolant system leak rate?

    a. Containment High Range Radiation Monitor indicates 45R/Hr
    b. Containment Airborne Particulate and Gaseous Radiation Monitoring System indicates $1x10^5$ CPM
    c. Drywell pressure at or below 2.0 psig
    d. Reactor water level at or below 86" TAF

Answer: d

Justification: Reactor coolant system leak rate into the containment is measured by how much water is pumped out of the primary containment over time. With the drywell equipment and floor sump isolation valves closed, the reactor coolant system leak rate cannot be determined. Anything that causes an isolation of these valves would prevent the ability to determine reactor coolant system leak rate.

RAP-C1g, CAPGRAMS Radiation High, has no automatic actions. Answer a is incorrect.

RAP-10F4k, Hi Range Rad. Monitor Abnorm., will close the torus/DW vent and purge valves at the high setpoint (not the DW floor and equipment drain valves). Answer b is incorrect.

Support procedure 1 of RPV Control – No ATWS (EMG-3200-01A) and RAP-C4h, shows that the drywell equipment and floor drain IVs isolate on a primary containment isolation signal (RPV water level at or below 86" TAF, or drywell pressure at or above 3.0 psig). Answer c is incorrect.

Answer d is correct since the drain valves close at or below 86" TAF (see Support procedure 1).

223002 K1.14
Knowledge of the physical connections and/or cause-effect relationships between PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF and the following: Containment drainage system
(CFR: 41.2 to 41.9)

OC Learning Objective: 2621.828.0.0037 (02456: Describe RPS isolation logic trip signals and functions, including the following: 1) purpose/design basis; 2) setpoints; 3) conditions that allow bypassing isolation signals; 4) how bypassing isolation signals is accomplished.)

Cognitive Level: Memory or Fundamental

Question Type: New

9/8/06
Rossi reviewed. Corrected the units in selections a and b. Changed selection c from ‘at or above 2 psig’ to ‘at or below 2 psig’.

9/22/06 Validator comments: Changed question from ‘With the reactor at power, ..’ to ‘The reactor is at rated power. Which of…’.

22. Which of the following would result in reactor water level being controlled in single-element control?

    a. The loss of a steam flow signal to the digital control computers while at rated power
    b. The loss of a feed flow signal to the digital control computers while at rated power
    c. The loss of both digital control computers while at rated power
    d. The loss of RPV water level inputs from LT-ID13A and LT-ID13B while at power

Answer: c

Justification: FW level control uses a steam flow signal from each of the two steam lines. When a steam flow input is lost, the system will double the good remaining steam flow input and will continue to use 3-element control. Answer a is incorrect.

The FW level control system uses feedwater flow from each of two feedwater lines. When one FW flow input is lost, the system will calculate the feed flow based upon valve position, number of running pumps and reactor pressure. The system will continue to use 3-element control. Answer b is incorrect.

When a single digital control computer is lost, the system will continue in 3-element control with the operable digital control computer. When both computers are lost, control is transferred to the Moore controllers, which will control in single-element control. Answer c is correct. (see RAP-J1c)

If RPV water level inputs ID13A and ID13B are bad, then the controller uses LT-0013C. The control system does not transfer to single element control. Answer d is incorrect. (See MDD-OC-625-B, FW and Recirc Control Systems Upgrade Modification)

259002 K4.09
Knowledge of REACTOR WATER LEVEL CONTROL SYSTEM design feature(s) and/or interlocks which provide for the following: Single element control (reactor water level provides the only input) (CFR: 41.7)

OC Learning Objective: 2621.828.0.0018 (10444: Describe the interlock signals and setpoints for the affected system components and expected response including power loss or failed components.)

Cognitive Level: Memory or Fundamental

Question Type: New

8/15/06: NRC Comments
The original answer d (Following a scram from rated power while controlling with the low flow regulating valve in MANUAL) was not a plausible distracter. Changed with what is shown.

23. The plant is at rated power. The following switch position is noted:

- STANDBY GAS SELECT is in position SYS 2

An event occurs which automatically initiates the Standby Gas Treatment System.

Five minutes after the initiation (with no operator action), which of the following is the correct fan/valve configuration if the lead system developed/maintained a low flow signal?

	System 1 Fan	System 2 Fan	System 2 Orifice Valve V-28-28
a.	ON	ON	OPEN
b.	ON	OFF	CLOSED
c.	OFF	ON	CLOSED
d.	ON	OFF	OPEN

Answer: a

Justification: On an automatic system initiation, both SGT fans start. If the lead fan develops adequate flow within the first 2-3 minutes, the lag fan will shutdown and the associated inlet/outlet valves close. If the lead fan does not develop adequate flow, the lag fan continues and the lead fan continues to run, but with the lead system inlet/outlet valves closed. The system orifice valves are normally closed (with the systems is standby) and stays closed when the lead system starts with proper flow. If the lead running system sees low flow, then besides what's already been said, the lead system orifice valve also opens (and inlet/outlet valves close and the redundant system assumes the SGT function). Therefore, 5 minutes after an auto initiation, system 2 fan (which was selected as lead) will be running with the loop inlet/outlet valves closed and loop orifice valve open. System 1 fan is also running performing the SGT function. Answer a is correct. (See also procedure 330)

All other answers are incorrect due to incorrect status of fan/valve. (See RAP-L5b).

261000 A3.01
Ability to monitor automatic operations of the STANDBY GAS TREATMENT SYSTEM including: System Flow (CFR: 41.7)

OC Learning Objective: 2621.828.0.0042 (10445: Given a set of system indications or data, evaluate and interpret them to determine limits, trends, and system status.)

Cognitive Level: Comprehensive or Analysis

Question Type: Modified

8/15/06: NRC Comments
Added with no operator action in the stem.

9/8/06
Rossi reviewed. Deleted: with all systems normally aligned from the stem.

24. The plant was at 50% power when a small LOCA into the primary containment occurred. Following the scram, all offsite power was lost and both emergency diesels could not be started. The following conditions exist:

- RPV water level is 113” TAF and lowering very slowly
- Drywell pressure is 5.1 psig and rising slowly
- RPV pressure is 820 psig and lowering slowly
- All control rods are fully inserted
- ABN-37, Station Blackout, has been entered
  - 4160 VAC Bus D has been powered from the combustion turbine, and critical loads have been restored in accordance with ABN-37-7
  - the SBO Transformer load is currently 2.4 MWe

Which of the following is the required action to restore RPV water level?

a. Lower RPV pressure as required and lineup and inject condensate transfer to core spray
b. Lower RPV pressure as required and lineup and inject fire water to core spray
c. Lineup and inject with Condensate Pump A / Feedwater Pump A
d. Lineup and inject with the maximum flow using both CRD pumps

Answer: d

HANDOUTS: ABN-37-7 and EMG-3200.01A

Justification: With the given conditions, only feedwater and CRD can inject into the RPV at 820 psig. There is ample room on the SBO transformer (8 MWe maximum allowable and currently at 2.4) to start one condensate pump and one feedwater pump (0.811 MWe for the CP, and 3.141 MWe for the FWP = 3.952 additional MWe for 1 condensate/feed pump; so 8 – 2.4 = 5.6 MWe room on the SBO transformer). But, condensate pump A and feedwater pump A are powered from 4160 bus 1A, which cannot get power from bus 1B (1B is powered from the combustion turbine). Answer c is incorrect.

To inject with condensate transfer or fire water, core spray system 1 or core spray system 2 must be unavailable. With the given conditions of power to Bus D, components on core spray system 1 and core spray system 2 are available. Therefore, conditions have not been met to inject with either condensate transfer of fire water. Answers a and b are incorrect.

Part of the lineup in ABN-37 is cross-tieing power to USS 1A2, which supplies CRD Pump 1A. CRD Pump 1B is also powered. Injection from these systems is directed. Answer d is correct. (Se ABN-37, EMG-3200-01A)

262001 (A.C. Electrical Distribution) 2.4.6
Knowledge symptom based EOP mitigation strategies.
(CFR: 41.10)

Cognitive Level: Comprehensive or Analysis

Question Type: New

9/8/06
Rossi reviewed. The correct answer was changed to answer d. ABN-37 cross-ties power to USS 1A2 (for CRD 1A) and CRD 1B already is aligned to receive power. Both can inject as required. Answer c (the original correct answer) was changed to say condensate pump A and feedwater pump A. Feedwater is required in the ABN, but bus 1A, which supplies these pumps, cannot be powered from the combustion turbine.

25. The plant was at rated power. An event occurred which resulted in a loss of 125 VDC Bus C.

    Which of the following components has lost DC power?

    1. Station Blackout Transformer Remote Monitoring Panel
    2. 4160V Switchgear 1A and 1B
    3. Turbine Generator Excitation Switchgear
    4. Remote Shutdown Panel Inverter
    5. Emergency Diesel Generator 1 Switchgear

    a. 1 and 2
    b. 2, 3, and 4
    c. 3 and 5
    d. 1 and 5

Answer: d

Justification: The following components receive DC power from that listed:

1. Station Blackout Transformer Remote Monitoring Panel – DC C (**correct**)
2. 4160V Switchgear 1A and 1B - 1A from DC C and 1B from DC B (incorrect)
3. Turbine Generator Excitation Switchgear – DC A (incorrect)
4. Remote Shutdown Panel Inverter – DC B (incorrect)
5. Emergency Diesel Generator 1 Switchgear – DC C (**correct**)

Therefore, only answer d has choices of 1 and 5. Answer d is correct and all other answers are incorrect. (See ABN-54, ABN-55, drawing BR 3028 and D-3033).

263000 K2.01
Knowledge of electrical power supplies to the following: Major DC Loads
(CFR: 41.7)

OC Learning Objective: 2621.828.0.0012 (01106: Draw a one-line diagram of the 125 VDC Distribution System including: major busses (A, B and C battery systems), battery charging power supplies, major breakers, automatic bus transfer switches, manual bus transfer switches, and major loads for each DC panel.)

Cognitive Level: Memory or Fundamental

Question Type: Bank

8/15/06: NRC Comments
Corrected the justification to show that answer d is correct (had said that answer a was correct, which was in error).

9/8/06
Rossi reviewed. Deleted: with all systems normally aligned in the stem.

26. The plant was shutdown with plant power supplied from the normal offsite source, when a total loss of offsite power occurred. The plant responded as designed.

Several hours later, all offsite power was restored and 4160 Bus 1A has been re-energized.

Which of the following states the correct method used to transfer Bus 1C from its EDG back to normal power?

Synchronize EDG1 with Bus 1A, reduce load on EDG1 and then…

a. place the local MODE SELECTOR switch in STOP
b. place the Control Room NORMAL START switch in STOP
c. open the EDG1 output breaker, then place the local MODE SELECTOR switch in STOP
d. open the EDG1 output breaker, then place the Control Room NORMAL START switch in STOP

Answer: b

Justification: When offsite power is lost, both EDG1 and 2 automatically start and load onto their respective emergency bus (EDG1 – Bus 1C; EDG2 – Bus 1D). There is a precaution in procedure 341, Emergency Diesel Generator Operation, which states that if the EDG1 (2) fast-started (ie, loss of Bus 1C voltage), then it can only be shutdown from the control room. Therefore, answers a and c are incorrect since the listed switches are local at the EDG.

341 directs the Bus 1C be paralleled, reduce load (KVAR and KW) on the EDG, then place the Normal Start switch in Stop. When placed in Stop, the EDG output breaker will open, the EDG slows to 400 RPM for 15 minutes, then shuts-down. There is no reason to first open the EDG output breaker, then shutdown the EDG since the switch in Stop performs this function automatically. Answer d (and c) are incorrect. Answer b is correct.

264000 A4.03
Ability to manually operate and/or monitor in the control room: Transfer of emergency control between manual and automatic (CFR: 41.7)

OC Learning Objective: 2621.828.0.0013 (264-10446: Identify and explain system operating controls/indications under all plant operating conditions.)

Cognitive Level: Memory or Fundamental

Question Type: New

8/15/06 NRC Comment
They felt that the references showed that 2 answers were plausible. This question has replaced the original question.

9/8/06
Rossi reviewed. Changed stem to provide the necessary information while maintaining economy of words.

27. The plant is at rated power.

If a total loss of offsite power, were to occur, which of the following Reactor Building Closed Cooling Water valves would still have electrical power to operate?

- V-5-147 CCW Inlet Isolation Valve
- V-5-166 RBCCW Outlet Isolation Valve
- V-5-167 RBCCW Outlet Isolation Valve

a. V-5-147 and V-5-166 only
b. V-5-147 and V-5-167 only
c. V-5-166 and V-5-167 only
d. V-5-147, V-5-166 and V-5-167

Answer: d

Justification: The valves are powered from the following busses:

- V-5-147 MCC 1B21A
- V-5-166 MCC 1B21B
- V-5-167 MCC 1A21

During a loss of offsite power, busses C and D (which will be re-powered from EDG1 and EDG2 respectively) load shed. Busses 1A2 and 1B2 always get power. These busses power 1B21A, B and 1A21. Therefore, all three RBCCW isolation valves have power. Answer d is correct. The other answers are incorrect since they do not list all valves that have electrical AC power. (see 309.2) (The valve noun names provided above are named as in the procedure.)

400000 K2.02
Knowledge of electrical power supplies to the following: CCW Valves (CFR: 41.7)

OC Learning Objective: 2621.828.0.0016 (262-10444: Describe the interlock signals and setpoints for the effected system components and expected system response including power loss or failed components) 2621.828.0.0035 (07261: State the location and explain how to manipulate all controls normally operated for the RBCCW System.)

Cognitive Level: Memory or Fundamental

Question Type: Bank

9/8/06
Rossi reviewed. Deleted: with all systems normally aligned in the stem.

28. Which of the following could be indicative of a Reactor Manual Control System control rod movement timer malfunction?

   a. The green INSERT light ON for 3.5 seconds during a control rod single notch ROD IN
   b. The red WITHDRAW light ON for 3 seconds during a control rod ROD OUT NOTCH
   c. The amber SETTLE light ON for 5 seconds following a control rod single notch ROD IN evolution
   d. The green INSERT light ON for 1 second during a control rod ROD OUT NOTCH

Answer: b

Justification: For a single notch-in, the green insert light should be on for 3.5 seconds (See ABN-6, Control Rod Drive System). Answer a is incorrect.

For a rod notch-out, the red withdraw light should be on for 1.5 seconds. The given answer is twice as long. Answer b is correct.

The amber settle light should be on for 5 seconds following the insert or withdraw evolution. Answer c is incorrect.
The green light is on for 1 second during a control rod out notch. Answer d is incorrect.

201002 A3.04
Ability to monitor automatic operations of the REACTOR MANUAL CONTROL SYSTEM including: Rod movement sequence timer malfunction alarm
(CFR: 41.7)

OC Learning Objective: 2621.828.0.036 (00726: Given a mode and direction for control rod movement, describe response of the timer, response of the CRD System, system indications and operation of controls.

Cognitive Level: Memory or Fundamental

Question Type: New

29. Which of the following would result in a Technical Specification violation?

a. While at power, a recirculation pump was started in an idle loop whose loop temperature was 40° F less than reactor coolant temperature
b. While at power (5-loop), over a one week period, two recirculation pumps tripped and the loops were placed in an IDLE condition
c. During a startup, while on Range 10 of the Intermediate Range Monitors, a malfunction in the Master Recirc Speed Controller lowered recirculation flow to $38x10^6$ lb/hr
d. While at power, an event occurred which required that a single recirculation scoop tube be placed into local manual control

Answer: c

HANDOUT: TECH SPECS 3.3

Justification: IAW TS 3.3.C2, an idle recirculation pump shall not be started unless the loop temperature is within 50° F of the reactor coolant temperature. Since this answer gave 40° F, the operation is allowed by TS and no TS violation is present. Answer a is incorrect.

IAW TS 3.3.F, 2 idle recirculation loops are allowed (except for starting) with no further actions. Since there is no TS entry violation, answer b is incorrect.

IAW TS 3.3.H.2, the minimum required recirculation flow rate while on IRM Range 10 is $39.65x10^6$ lb/hr. Since flow was below this minimum, a TS action would be required. Answer c is correct (this is also stated in procedure 301.2, Reactor Recirculation System).

Placing a recirculation MG set scoop tube in local manual control is not mentioned in TS, and therefore does not violate TS. Answer d is incorrect.

202002 2.1.33 (Recirculation Flow Control)
Ability to recognize indications for system operating parameters which are entry-level conditions for technical specifications. (CFR 41.10 / 43.2 / 43.3)

OC Learning Objective: 2621.850.0.0090 (016581: Identify whether or not a Tech Spec or License Limit has been exceeded.)

Cognitive Level: Comprehensive or Analysis

Question Type: Modified
8/15/06: NRC Comments

They thought that answers b and c were both correct. The question has been re-worded to ask which one would result in a TS violation.

9/8/06
Rossi reviewed. Corrected justification for 2 idle recirculation loops.

30. Which of the following power losses would cause the Rod Worth Minimizer to be INOPERABLE due to the loss of control rod position information?

a. Protection System Panel A
b. Instrument Panel 4A
c. Continuous Instrument Panel CIP-3
d. VACP-1

Answer: c

Justification: Of the listed power supplies, only continuous instrument panel 3 provides electrical power to the control rod position indications. All other listed power supplies do not provide this power. Answer c is correct. (See ABN-58, GU 3C-733-11-005)

214000 K1.01
Knowledge of the physical connections and/or cause/effect relationships between ROD POSITION INFORMATION SYSTEM and the following: RWM: Plant-Specific (CFR: 41.2 to 41.9)

OC Learning Objective: 2621.828.0.0041 (10444: Describe the interlock signals and setpoints for the effected system components and expected system response including power loss of failed components.)

Cognitive Level: Memory or Fundamental

Question Type: New

31. The plant is at 2% power during a startup, with drywell inerting in-progress. Drywell oxygen concentration is currently 9% and lowering slowly.

If a spurious high drywell pressure signal initiated Standby Gas Treatment system, which of the following is correct (assume no operator action)?

a. Drywell pressure will rise due to the nitrogen addition but venting at a slower rate through Standby Gas Treatment
b. Drywell pressure will lower due to nitrogen addition isolation and venting through Standby Gas Treatment system
c. The drywell oxygen indicator shows a stable valid indication due to the isolation of nitrogen and DW vent and purge valves
d. The drywell oxygen indicator no longer shows a valid indication due to the isolation of the drywell oxygen sampling system

Answer: d

With a high drywell pressure isolation signal, nitrogen into the drywell is isolated (V-23-14, -14), and the drywell atmosphere to RB HVAC isolate (V-27-1, -2). On the same isolation signal, the drywell oxygen sample primary containment isolation valves also close. Therefore, nitrogen inlet/outlets of the primary containment are isolated, and the oxygen sampling system is also isolated. Answer d is correct.

Answer a is incorrect since nitrogen addition is isolated. Answer b is incorrect since drywell venting is isolated. Answer c is incorrect since the oxygen reading is not valid since the oxygen sampling system is isolated. (see BR2011, 13432.19-1, M0012, 3E-666-21-1000, USAR Table 6.2-12; 312.9)

223001 A1.06
Ability to predict and/or monitor changes in parameters associated with operating the PRIMARY CONTAINMENT SYSTEM AND AUXILIARIES controls including: Oxygen Concentration CFR: 41.5)

OC Learning Objective: 2621.828.0.0032 (00394: Given auto isolation signals, list or identify causes, system responses, and affected primary containment system components.)

Cognitive Level: Comprehensive or Analysis

Question Type: New

8/15/06: NRC Comments
Added no operator actions.

32. A plant startup is in-progress. The startup is continuing with no noted problems, in accordance with procedure 201, Plant Startup. The current conditions exist:

- RPV pressure is 150 psig
- Turbine warming is in-progress, at the pre-warming mark
- The very next step is to open the turbine control valves and turbine stop valve #2
- The Mechanical Pressure Regulator is set at 250 psig
- Turbine Bypass Valves are CLOSED

Which of the following lists the method to open turbine control valves and turbine stop valve #2 for turbine warming?

Open the turbine control valves by using __(1)__, and open turbine stop valve #2 by using __(2)__.

a. (1) the BYPASS VALVE OPENING JACK switch
(2) the LOAD LIMIT CONTROL switch

b. (1) the MPR Control Switch
(2) the SPEED LOAD CHANGER switch

c. (1) the BYPASS VALVE OPENING JACK switch
(2) the MAIN STOP VALVE NO. 2 INTERNAL BYPASS switch

d. (1) the LOAD LIMIT CONTROL switch
(2) the MAIN STOP VALVE NO. 2 INTERNAL BYPASS switch

Answer: c

Justification: IAW 315.1, Turbine Generator Startup, given that the turbine bypass valves are closed, and the , placing the BYPASS VALVE OPENING JACK in RAISE will open the turbine control valves. Then the stop valve is opened to admit steam by placing the MAIN STOP VALVE NO. 2 INTERNAL BYPASS in RAISE position. If the turbine bypass valves were open, then the placing the load limit control switch to raise would open the turbine control valves; and stop valve number 2 bypass would open the stop valves. Answer c is correct.

245000 K5.02
Knowledge of the operational implications of the following concepts as they apply to MAIN TURBINE GENERATOR AND AUXILIARY SYSTEMS : Turbine operation and limitations

(CFR: 41.5)

OC Learning Objective: 2621.828.0.0050 (10446: Identify and explain system operating controls/indications under plant operating conditions.)

Cognitive Level: Memory or Fundamental

Question Type: Modified

9/8/06
Rossi reviewed. Changed ‘The plant is starting-up following a refuel outage’ to ‘a plant startup is in progress’.

9/22/06 Validator comment: Added ‘at the pre-warming mark’ in the second bullet.

33. The plant is at rated power. You have just received a report that fire detector R5D9 (Reactor Building 51’ North, Zone 1) failed its surveillance test to detect and alarm a fire. The SRO has declared this fire detector inoperable. (There are no alarms locked-in from this inoperable fire detector).

    Your initial investigation shows that there are 8 fire detectors on RB 51’ North, Zone 1, and 9 fire detectors on RB 51” North, Zone 2.

    Which of the following states how this inoperable fire detector effects the ability of the fire protection system to detect fires in RB 51’ North and to actuate Deluge System #5?

    a. The fire protection system can still detect fires and actuate the fire protection system for mitigation; no compensatory measures are required
    b. The fire protection system can still detect fires but CANNOT actuate the fire protection system for mitigation; no compensatory measures are required
    c. The fire protection system CANNOT detect fires nor actuate the fire protection system for mitigation; an hourly fire watch patrol must be established
    d. The fire protection system CANNOT detect fires nor actuate the fire protection system for mitigation; a continuous fire watch must be established

Answer: a

HANDOUT: PROCEDURE 333 (Plant Fire Protection System, Attachment 333-15) AND PROCEDURE 101.2 (OYSTER CREEK SITE FIRE PROTECTION PROGRAM, Attachment 101.2-3)

Justification: IAW procedure 645.6.031 (Attachment 645.6.031-2), there are 8 fire detectors in 51’ RB North Zone 1 (which includes the given inoperable detector). As stated in the stem, there are no other inoperable detectors. Therefore, all other fire detectors in Zone 1, and all those in Zone 2 of 51’ RB North will alarm at the fire panels when activated. Each fire detector is independent of the others to activate the alarms on the panels.

IAW procedure 333 (Attachment 333-15), actuation of Deluge System 5 (for 51’ RB North) requires only 1 detector from 51’ RB North Zone 1 to actuate, and 1 detector from 51’ RB North Zone 2 to actuate. Therefore, one inoperable detector will still allow the fire protection system to detect and mitigate a fire in 51’ RB North.

Procedure 101.2 tells us that 6 detectors are required for RB 51' North Zone 1, and 7 detectors are required for RB 51 North Zone 7. With the number of operable fire detectors less than required, a fire watch patrol must be established. Therefore, fire detection in RB 51' North is still operable and functional, and no compensatory actions are required. Answer a is correct.

Since the system is still able to mitigate a fire, answer b is incorrect. Since the system is still able to detect and mitigate a fire in the area, no compensatory measures are required. Answers c and d are incorrect.

286000 K3.01
Knowledge of the effect that a loss or malfunction of the FIRE PROTECTION SYSTEM will have on following: The ability to detect fires (CFR: 41.7)

OC Learning Objective: 2621.828.0.0019 (286-10445: Given a set of system indications or data, evaluate and interpret them to determine limits, trends, and system status.)

Cognitive Level: Comprehensive or Analysis

Question Type: New

8/15/06: NRC Comments
They want to see procedure 645.6.031. No comment regarding the question as written.

9/8/06
Rossi reviewed. Deleted: with all systems normally aligned and no equipment out of service from the stem.

34. The plant is at rated power.

Which of the following Emergency Operating Procedures has an entry condition that is available through the plant process computer and is **NOT** read on the control room panels?

a. EMG-3200.01A, RPV Control – NO ATWS
b. EMG-3200-01B, RPV Control – With ATWS
c. EMG-3200.02, Primary Containment Control
d. EMG-3200.11, Secondary Containment Control

Answer: c

Justification: Indications RPV water level, RPV pressure, drywell pressure, and reactor power (entry conditions into RPV Control – No ATWS, and RPV Control – With ATWS) can be found on the control room panels. Answers a and b are incorrect.

Indications for torus water temperature, drywell pressure, torus water level, and primary containment hydrogen concentration (entries for Primary Containment Control) can be found on control room panels. Bulk drywell temperature (also an entry into Primary Containment Control) is available only on the plant process computer (SPDS – Containment Conditions screen). (When the PPC is not operable, DWT can be calculated from plant indications, but the question stem states that no equipment is OOS.)

Indications of the Alert emergency classification for radioactivity release are found from control room panel indications, and from other data sources, including procedures. Indications of an isolation condenser tube leak are found from control room panel indications and alarms. Answer d is incorrect.

2.1.19 Ability to use plant computer to obtain and evaluate parametric information on system or component status. (CFR: 41.10)

OC Learning Objective: 2621.863.0.0007 (02233: Discuss the relevance of information shown on the PPC SPDS displays to the implementation of the SBEOPs.)

Cognitive Level: Memory or Fundamental

Question Type: New

8/15/06: NRC Comments: Bolded NOT.

9/8/06
Rossi reviewed. Deleted: with all systems normally aligned and no equipment out of service from the stem.

35. A plant startup is in-progress. The following conditions exist:

- The MODE Switch is in STARTUP, with control rod withdrawals in-progress
- IRMs 11, 12, 15, 16, 18 read 72-74 on Range 1
- IRMs 13, 14, and 17 read 9 - 10 on Range 2

A malfunction in the IRM drive circuitry caused IRM 13 to withdraw to the full-out position.

Which of the following states the effect on the plant and the required Operator actions to continue withdrawing control rods?

a. There are panel annunciators ONLY; withdrawing control rods may continue without any other control panel manipulations

b. There are panel annunciators and a rodblock from IRM downscale ONLY; bypassing the IRM is required to continue withdrawing control rods

c. There are panel annunciators and a rodblock from IRM downscale AND IRM detector position; bypassing the IRM is required to continue withdrawing control rods

d. There are panel annunciators, a rodblock and a ½ scram; bypassing the IRM and resetting the ½ scram is required to continue withdrawing control rods

Answer: c

Justification: The following IRM parameters provide roblocks only (no scram input): IRM downscale (in REFUAL and STARTUP; bypassed in Range 1 or in RUN), detector not fully inserted (bypassed in RUN), and IRM high (bypassed in RUN). When the IRM comes off the full-in position, a rodblock is instituted (plus panel annunciators). It is expected that the IRM will also go downscale as it drives to the fully withdrawn position (downscale also gives a rodblock except in Range 1). There are no ½ scrams from these conditions. Therefore, to continue to move control rods, IRM 13 (which is instituting a rodblock both from downscale and IRM position) must be bypassed. Answer c is correct.

Answer a is incorrect since it does not list rodblocks. Answer b is incorrect since it does not list all rodblocks. Answer d is incorrect since no ½ scram occurs. (See RAP-H7a, 237E912 and 796E212).

2.2.2

Ability to manipulate the console controls as required to operate the facility between shutdown and designated power levels. (CFR: 45.2)

OC Learning Objective: 2621.828.0.0029 (10449: State the function and interpretation of system alarms, alone and in combination, in accordance with system RAPs.)

Cognitive Level: Comprehensive or Analysis

Question Type: New

9/8/06
Rossi reviewed. Changed ‘The plant is starting-up after a 5-day forced outage’ to ‘a plant startup is in-progress’ in the stem.

36. The plant is shutdown for a refuel outage. A fuel shuffle is in-progress.

Which of the following, as stated by procedure 205.0, Reactor Refueling, states when a communication must be made between the Control Room Licensed Operator and the Refueling Senior Reactor Operator?

a. When a blade guide is vertically aligned and is being lowered into the core
b. When a new fuel bundle is vertically aligned and is being lowered into the core
c. When an irradiated fuel bundle is vertically aligned and is being lowered into the spent fuel racks
d. When a control rod is vertically aligned and is being lowered into the core

Answer: b

Justification: IAW procedure 205.0, (section 7.3.1) Reactor Refueling, a communication between the Refuel SRO and the CRO is required at the commencement and completion of each move, and whenever a bundle enters or exits the core. None of the answers provided are the commencement or completion of a step. Answer b does meet the procedural requirement in that a bundle is entering the core. Answer b is correct.

Procedure 205.0 does not require a communication regarding the blade guide into the core. Answer a is incorrect.

Procedure 205.0 does not require a communication regarding the fuel placement into the fuel pool racks. Answer c is incorrect.

Procedure 205.29, Control Rod Blade Removal and Replacement, does not require a communication as the blade enters the core. Answer d is incorrect.

2.2.28
Knowledge of new and spent fuel movement procedures. (CFR: 43.7)

OC Learning Objective: 2621.812.0.0003 (00323: State the responsibilities of the following personnel during refueling operations IAW procedure 205.0: 1) Reactor Engineer; 2) Shift Manager; 3) Control Room Licensed Operator; 4) Bridge Operator; 5) Fuel Move Checker; 6) Fuel Handling Director.)

Cognitive Level: Memory of Fundamental

Question Type: New

9/8/06
Rossi reviewed. Changed stem to provide the necessary information while maintaining economy of words. Changes ‘lists’ to ‘states’ in the stem.

37. The plant is at rated power with the following condition:

- The Reactor Water Cleanup System has been removed from service to support system repair (electrical)

The outside containment motor operated isolation valve is to be used as a clearance boundary. IAW OP-MA-109-101, Clearance and Tagging, this is allowed as long as the following conditions are met:

1. The valve control station is tagged
2. The electrical energy source is removed
3. The valve's handwheel is tagged

Given that the valve is located in a high radiation area, which of the following states who can waive the 3rd requirement above, due to ALARA considerations?

a. The Unit Supervisor
b. The Radiation Protection Technician
c. The Cleanup System Engineer
d. The Clearance Writer (Reactor Operator)

Answer: a

Justification: OP-MA-109-101, Clearance and Tagging, states that the above requirement shall only be waived by the clearance approver (who is a first line supervisor or above) or Shift Management when radiological or hazardous conditions exist. The same procedure defined clearance approver as an individual trained and qualified to approve clearances; a clearance approver should be a currently or previously licensed SRO. OP-OC-100, Oyster Creek Conduct of Operations provides the following for Shift Management: normally consists of an SRO licensed Shift Manager, an SRO licensed Unit Supervisor, and an SRO licensed Field Supervisor. Therefore, answer a is correct. All other answers are incorrect.

The RP Tech is not a licensed SRO. Answer b is incorrect.

The Cleanup System Engineer is not a licensed SRO. Answer c is incorrect.

The clearance writer is usually a RO licensed individual. Answer d is incorrect.

2.3.2
Knowledge of facility ALARA program. (CFR: 41.12)

OC Learning Objective: RWT (Objective 22: Describe the Station ALARA Program).

Cognitive Level: Memory or Fundamental

Question Type: New

9/8/06
Rossi reviewed. Changed stem to provide the necessary information while maintaining economy of words.

38. The plant was at rated power, when a loss of 125 VDC DC-E occurred, and cannot be restored.

- The applicable ABN has been entered

Which of the following states why available Non-Licensed Operators are directed to perform plant tours?

a. All fire protection mitigation systems have been disabled
b. Power has been lost to all control room annunciators
c. Automatic trip ability for feedwater pumps, condensate pumps and main turbine has been lost
d. Power is lost to all area radiation monitors and most process radiation monitors

Answer: b

Justification: The loss of DC-E results in the loss of all control room annunciators. ABN-53 requires that plant operators tour plant areas where the annunciators are lost (plant wide). Answer b is correct.

The ability of the fire protection system to mitigate fires is not effected, but fire annunciation is lost in the control room. Answer a is incorrect.

DC power for feedwater and condensate pumps (on 4160 Bus A and 4160 Bus B) are powered from DC Bus B and Bus C. Answer c is incorrect.

Radiation monitors are power by various AC power supplies (mostly vital AC). ARMs are powered by Continuous Instrument Panel-3 (CIP-3) (reference procedure 407.1) Answer d is incorrect.

2.4.32
Knowledge of operator response to loss of all annunciators. (CFR: 41.10)

OC Learning Objective: 2621.828.0.0012 (10450: Describe and interpret procedure sections and steps for plant emergency or off-normal conditions that involve this system including personnel allocation and equipment operation IAW applicable ABN, SDRP, EOP & EOP Support Procedures and EPIPs.)

Cognitive Level: Comprehensive or Analysis

Question Type: New

9/8/06
Rossi reviewed. Deleted: with all systems normally aligned in the stem.

39. The reactor was operating at rated power when a loss of all offsite power occurred. Plant conditions are as follows:

- Reactor water level dropped to 82 inches and is currently 110 inches and rising slowly
- Reactor pressure dropped to 890 psig and is currently 950 psig and rising slowly
- Drywell pressure has risen to 2.2 psig and is stable
- Both EDG output breakers have been closed for two minutes
- Restoration of power to plant buses IAW ABN-36-3, Plant Electrical Distribution Restoration, has NOT yet commenced.

Which one of the following statements is true for these conditions?

a. Drywell temperature is rising because there are no drywell recirc fans running
b. Instrument air pressure is lowering because there are no air compressors running
c. Reactor Building ∆P is zero because no reactor building ventilation fans are running
d. Service water temperatures are rising because there are no service water pumps running

Answer: b

Handouts: None

Justification: A is incorrect – since there is no LOCA signal (Hi DW pressure AND Lo-Lo RPV level), drywell recirc fans 1, 3 and 5 auto-started 2.5 seconds after power was restored to buses 1C and 1D (and USS 1A2 and 1B2). Although RPV level dropped below the Lo-Lo level setpoint (86 inches), a concurrent high drywell pressure must be received to prevent drywell recirc fans from starting when 1C/1D bus power is restored.

B is correct – on loss of all offsite power, all (4) 4160V buses de-energize. The UV logic for the 1C and 1D buses trips the load breakers on buses 1C and 1D as well as the feeder breakers to USS 1A1 and 1B1, which power Turbine Building loads. The feeder breakers for USS's 1A2/1B2 and 1A3/1B3 remain closed since UV trip devices on individual loads provide load shedding for these USS's. When power is restored to buses 1C and 1D by their respective EDG's, USS's 1A1 and 1B1 remain de-energized and USS's 1A2/1B2 and 1A3/1B3 automatically re-energize. Certain loads powered from these USS's are automatically sequenced on at various time intervals to prevent overloading the EDG's, including CRD pumps, RBCCW pumps and Service Water pumps. The service and instrument air compressors are powered from USS 1A1/1B1 and

therefore require manual restoration. This is directed by ABN-36 following restoration of power to USS 1A1/1B1 IAW Attachment ABN-36-3.
C is incorrect – reactor building ventilation fans tripped on the loss of offsite power, but the Standby Gas Treatment System started due to the Lo-Lo RPV level signal (SGTS does not require a concurrent high drywell pressure), restoring the Reactor Building negative ΔP.

D is incorrect – service water pumps automatically restart two minutes after power is restored to buses 1C and 1D with no LOCA signal present. A LOCA signal (preventing restart) requires Lo-Lo RPV level AND Hi DW pressure.

295003 AK1.02
Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER: Load Shedding (CFR 41.5)

OC Learning Objective:
2621.828.0.0016, Objective E:
Describe the interlocks signals and setpoints for the affected system components and expected system response including power loss or failed components.
2621.828.0.0016, Objective M:
Given a set of system indications or data, evaluate and interpret them to determine limits, trends and system status.
2621.828.0.0016, Objective O:
Describe and interpret procedure sections and steps for plant emergency or off-normal conditions that involve this system including personnel allocation and equipment operation IAW applicable ABN, EOP & EOP support procedures and EPIPs.

Cognitive Level: Comprehension or Analysis

Question Type: New

References: 338, ABN-36, GE 223R0173, sh. 1A

8/15/06: NRC Comments
Added that DW rose to 2.2 psig and stable.

40. Given the following:

- Reactor power is 28% with a power ascension in progress
- The main generator is loaded to 200 MW and 50 MVAR
- Stator cooling water pump 1A is tagged out for maintenance
- DC Bus A tripped due to a ground fault
- The electrical fault on DC-A causes a trip of breaker 1B1M

Which statement below describes the effect of this event on main generator voltage regulation?

Generator terminal voltage ____________

a. must be maintained manually
b. will be maintained automatically
c. will rise since there is a loss of automatic and manual voltage control
d. will lower since there is a loss of automatic and manual voltage control

Answer: c

Handouts: None

Justification: A and B are incorrect – a loss of DC-A, which powers the main generator excitation equipment, results in a loss of both automatic and manual voltage control. The CAUTION for Step 3.2 of ABN-53 states, in part, “Loss of Main Generator voltage control will result from a loss of power to DC Distribution Center A.” In addition, a NOTE for Step 3.3 of ABN-12 states, in part, “Loss of DC control power to the excitation switchgear will result in a loss of generator voltage control.”

C is correct – a trip of 1B1M causes a loss of USS 1B1, which results in a loss of the only available stator cooling water pump. When stator flow drops below 230 gpm, a generator runback occurs. This causes an automatic load reduction on the generator. Since reactor power is below 30%, the crew will not scram the reactor (ABN-11 directs a reactor scram if a generator runback occurs when reactor power is above 30%). If below 30% power, ABN-11 directs reducing MVARs to zero. Since there is a loss of both automatic and manual voltage control, the operator will be unable to reduce generator voltage/MVARs and as generator (real) load is reduced during the runback, generator terminal voltage will increase.

D is incorrect – a loss of stator cooling results in a generator runback. As the generator unloads with no method of controlling voltage, generator terminal voltage will increase.

295004 G2.1.28
Partial or Complete Loss of D.C. Power / Conduct of Operations: Knowledge of the purpose and function of major system components and controls. (CFR 41.5, 41.7)

OC Learning Objective:
<u>2621.828.0.0025, Objective A:</u>
Describe and interpret procedure sections and steps for plant emergency or off-normal conditions that involve this system including personnel allocation and equipment operation IAW applicable ABN, EOP & EOP support procedures and EPIPs
<u>2621.828.0.0025, Objective G:</u>
Given a set of system indications or data, evaluate and interpret them to determine limits, trends and system status.
<u>2621.828.0.0025, Objective H:</u>
Identify and explain system operating controls/indications under all plant operating conditions.

Cognitive Level: Comprehension or Analysis

Question Type: New

References: ABN-11, ABN-12, ABN-53

9/8/06
Rossi reviewed. Changed initial bullet from ‘A ground fault on DC Bus A causes it to de-energize’ to ‘DC BUS A has tripped due to a ground fault’.

9/22/06 Validator comments: ABN-11 says that a runback will occur at approximately 25% rated power (182 MWe). Changed the second bullet from 175 MW to 200 MW.

41. Initial plant conditions are as follows:

- A plant startup is in progress with reactor power at 12%
- The mode switch is in STARTUP
- Recirculation flow is 11 E4 gpm
- Reactor pressure is 1000 psig

A turbine bypass valve malfunction causes:

- A spike in reactor pressure to 1043 psig
- A spike in reactor power to 40%

What is the status of the reactor (assume no operator action)?

a. At power
b. Scrammed due to high reactor pressure
c. Scrammed due to high IRM neutron flux
d. Scrammed due to high APRM neutron flux

Answer: c

Handouts: Nome

Justification: A is incorrect – the reactor scrammed due to high IRM neutron flux.

B is incorrect – from RAP-H1f, the high reactor pressure scram setpoint is 1045 psig.
C is correct – based on the conditions given (STARTUP, at 12% power), the reactor is operating on IRM Range 10. The scram setpoint for IRM Range 10 is 38.4% (LSSS), which was exceeded.
D is incorrect – the APRM Hi-Hi scram setpoint with recirc flow at 11 E4 gpm would be greater than 60%.

295006 AK2.06
Knowledge of the interrelations between SCRAM and the following: Reactor power (CFR 41.5, 41.6)

OC Learning Objective:
<u>2621.828.0.0037, Objective C:</u>
Describe all RPS scram logic trip signals, including the following:
1. Purpose / Design Basis
2. Setpoints
3. Conditions that allow bypassing scram signals
4. How bypassing scram signals is accomplished

Cognitive Level: Comprehension or Analysis

Question Type: Modified Bank

References: 201, RAP-H1f, Tech Spec 2.3

8/15/06: NRC Comments
Added no operator action.

42. Given the following:

- The reactor is operating at rated power on a hot summer day
- RBCCW & TBCCW heat exchangers are being cooled by Service Water
- RBCCW pump 1-1 and heat exchanger 1-1 are in service
- RBCCW heat exchanger 1-2 is tagged out due to a tube leak
- RBCCW temperatures have been trending upward
- The crew has entered ABN-19, RBCCW Failure Response

Which one of the following actions can be utilized to reduce RBCCW system temperatures?

a. Place RBCCW pump 1-2 in service along with pump 1-1
b. Increase Reactor Water Cleanup regenerative heat exchanger flow
c. Lineup the A & B Fuel Pool Cooling heat exchangers to be cooled by TBCCW
d. Lineup the TBCCW heat exchangers to be cooled by the Circulating Water System

Answer: d

Handouts: None

Justification: A is incorrect – in order to prevent flow-induced vibration (due to excessive flow) in the RBCCW heat exchangers, Procedure 309.2 requires two RBCCW heat exchangers to be in service prior to placing a second RBCCW pump in service.

B is incorrect – this can't be done without increasing RWCU system flow rate, which would increase the heat load on the RBCCW system.

C is incorrect – the A & B FPC heat exchangers cannot be aligned to be cooled by TBCCW; only the augmented (C) FPC heat exchanger can.

D is correct – ABN-19 directs this action
Operating with 2 RBCCW pumps in service requires 2 heat exchangers to be in service.

295018 AA1.01
Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER: Backup systems
(CFR 41.4, 41.10)

OC Learning Objective: 2621.828.0.0035, Objective P:
Describe and interpret procedure sections and steps for plant emergency and off-normal conditions that involve this system including personnel allocation and equipment operation in accordance with plant procedures.

Cognitive Level: Comprehension or Analysis

Question Type: New

References: ABN-19, 309.2, 311.1, BR 2006

43. Given the following:

- The reactor is SHUTDOWN and a cooldown is in progress
- Reactor pressure is 15 psig; reactor water level is 160 inches
- Reactor recirculation system status is as follows:
  - Loops ‘A’ and ‘C’ are ISOLATED
  - Loops ‘B’ and ‘D’ are IDLE
  - Pump ‘E’ is in service
- Shutdown cooling pumps ‘A’ and ‘B’ are in service
- The auxiliary reactor water cleanup pump is in service
- Bus 1A de-energizes due to an electrical fault
- The crew places shutdown cooling pump ‘C’ in service

What action should be taken regarding the reactor recirculation system?

a. OPEN the ‘B’ pump discharge valve then CLOSE the ‘E’ pump discharge valve
b. CLOSE the ‘E’ pump discharge valve, then OPEN the ‘B’ pump discharge valve
c. OPEN the ‘D’ pump discharge valve then CLOSE the ‘E’ pump discharge valve
d. CLOSE the ‘E’ pump discharge valve, then OPEN the ‘D’ pump discharge valve

Answer: c

Handouts: None

Justification: A is incorrect – P&L 4.2.12 in Procedure 305, Shutdown Cooling System Operation, states “If the Cleanup System is in service, the B Recirc loop should not be the selected loop in those instances where one loop is required to be fully open.” <u>NOTE</u>: the auxiliary reactor water cleanup pump is powered from MCC 1B21 (Bus 1B) and therefore remains in service on loss of Bus 1A.

B is incorrect – for the reason stated above for choice A. In addition, and more importantly, closing the E pump discharge valve would violate Tech Spec 3.3.F.4, which states “With reactor coolant temperature greater than 212 °F and irradiated fuel in the reactor vessel, at least one recirculation loop discharge valve and its associated suction valve shall be in the open position.”

C is correct – a loss of Bus 1A causes a trip of recirc pump E. P&L 4.2.11 in Procedure 305 states: “ To prevent SDC System flow from short-cycling the core, the E Recirc Loop Discharge Valve must be CLOSED <u>or</u> the E Recirc Pump running.” This statement requires the operator to close the E recirc pump discharge valve due to the pump trip. However, Tech Spec 3.3.F.4 requires at least one recirc loop suction and associated discharge valve to be open. To meet both of these requirements, the correct action to take would be to open the D pump discharge valve, then close the E pump discharge valve.

D is incorrect – this violates Tech Spec 3.3.F.4…see explanation for choice B above.

295021 AA1.05
Ability to operate and/or monitor the following as they apply to LOSS OF SHUTDOWN COOLING: Reactor recirculation (CFR 41.10)

OC Learning Objective:
<u>2621.828.0.0038, Objective J:</u>
Given normal operating procedure and documents for the system, describe or interpret the procedural steps.
<u>2621.828.0.0038, Objective M:</u>
Given Technical Specifications, identify and explain associated actions for each section of the Technical Specifications relating to this system including personnel allocation and equipment operation.

Cognitive Level: Comprehension or Analysis

Question Type: New

References: ABN-2, ABN-3, 301.2, Tech Spec 3.3.F.4

44. The reactor was manually scrammed due to a steam leak into the Primary Containment. Current plant conditions are as follows:

- Reactor water level is 140 inches and steady
- Reactor pressure is 600 psig and lowering slowly
- Drywell pressure is 14 psig and rising slowly
- Drywell temperature is 210° F and rising slowly
- Torus pressure is 13 psig and rising slowly
- Torus water level is 153 inches and steady
- Torus water temperature is 110° F and rising slowly

Which of the following alarms would be shown on the PPC SPDS screens for the Primary Containment?

a. A RED Priority 1 alarm due to torus water temperature above 106° F
b. A RED Priority 1 alarm due to drywell temperature above 200 °F
c. A YELLOW Priority 2 alarm due to torus water level above 152”
d. A YELLOW Priority 2 alarm due to drywell or torus pressure above 12 psig

Answer: d

Justification (See OC-PPC-SRS-0001, System Requirements Specification for the Oyster Creek Safety Parameter Display System) IAW the reference, the following alarm priority 1 or 2 are activated when:

- Torus/DW pressure > that which exceeds PSP in EOPs: RED Priority 1
- Torus/DW pressure > 12 psig: YELLOW Priority 2
- Torus water level < 110”: RED Priority 1
- Torus water level exceeds torus load limit in EOPS: RED Priority 1
- Torus water level <143” or > 154” (TS limit): YELLOW Priority 2
- Torus water temperature exceeds HCTL in EOPS: RED Priority 1
- Torus water temperature > 96° F with power > 2%: RED Priority 1
- Torus water temperature > 95° F: YELLOW Priority 2
- DW temperature > 281° F: RED Priority 1
- DW temperature > 200° F: YELLOW Priority 2

With torus water temperature at 106° F, the alarm should be yellow priority 1 (also, since HCTL is not exceeded, and all control rods are inserted). Answer a is incorrect.

A DW temperature of 210° F would only be a yellow priority 2 alarm. Answer b is incorrect.

A torus water level of 153” would present no alarms. Answer c is incorrect.

A torus/DW pressure > 12 psig (and less that PSP) would only be a yellow priority 2 alarm. Answer d is correct.

295024 EK2.16
Knowledge of the interrelations between HIGH DRYWELL PRESSURE and the following: SPDE/ERIS/CRIDS (CFR 41.7, 41.10)

OC Learning Objective:
<u>2621.863.0.0007, Objective O:</u>
Discuss the relevance of information shown on the PPC SPDS displays to the implementation of SBEOPs.

Cognitive Level: Memory of Fundamental

Question Type: New

References: EMG-3200.02, OC-PPC-SRS-0001

8/15/06: NRC Comments
The original question did hit the K/A. The new question is as above.

9/8/06
Rossi reviewed. Replaced ‘(all rods in) from rated power due to a steam loss of coolant accident’ with ‘due to a steam leak into the Primary Containment’ in the stem.

45. The reactor was operating at rated power when a loss of all offsite power occurred. The transient resulted in EMRV actuation and one EMRV stuck open. Current plant conditions are as follows:

- Reactor water level is 80 inches and lowering slowly
- Reactor pressure is 350 psig and lowering slowly
- Torus water temperature is 145 °F
- EDG-1 is loaded to 1400 KW
- EDG-2 is loaded to 1800 KW

Which Containment Spray pumps, and associated ESW pumps, should be placed in the torus cooling mode?

a. Two Containment Spray pumps and two ESW pumps in System 1
b. Two Containment Spray pumps and two ESW pumps in System 2
c. One Containment Spray pump and one ESW pump in either System 1 OR 2
d. Two Containment Spray pumps and two ESW pumps in both System 1 AND 2

Answer: a

Handouts: Attachments 341-5 and 341-6

Justification: A is correct – based on the given conditions, the Primary Containment Control EOP directs placing both Containment Spray Systems in the Torus Cooling mode. EDG-1 has sufficient capacity to carry two containment spray pumps and two ESW pumps, while EDG-2 does not. In addition, since 2 Core Spray pumps are running (due to Lo-Lo level) and getting ready to inject when RPV pressure drops below ~310 psig, securing Core Spray to run 4 Containment Spray pumps is not an option. Therefore, since only one Containment Spray System can be placed in service, System 1 is the correct choice.

B is incorrect – a CAUTION in Support Procedure 25 states: “Diesel Generator overload will result if a Containment Spray Pump and ESW pump are started with a Diesel Generator load of greater than 2150 KW.” Since EDG-2 is already loaded to 1800 KW, and a containment spray/ESW pump combination will add ~ 580 KW (as shown in Attachment 341-6), EDG-2 does not have sufficient capacity to carry System 2 (C and D) containment spray pumps and System 2 (C and D) ESW pumps…it can only carry 1 containment spray pump and 1 ESW pump.

C is incorrect – according to the Primary Containment Control EOP, if torus water temperature cannot be maintained below 95 °F, two containment spray systems (4 pumps) should be operated in torus cooling, if available. Current plant conditions prevent operating all four containment spray pumps, however two pumps can and should be placed in torus cooling.

D is incorrect – in addition to the EDG load restrictions mentioned above, there is a CAUTION in Support Procedure 25 that states: "NPSH problems will develop on all operating pumps if more than 4 Containment Spray/Core Spray Main pumps are operated at the same time." With RPV level at 80 inches, Core Spray Systems 1 and 2 (2 main pumps and 2 booster pumps) would be operating, but not injecting (core spray will begin to inject when RPV pressure is ~310 psig). Since 2 core spray pumps are running, and are needed to restore RPV level when RPV pressure drops below 310 psig, only 2 containment spray pumps can be placed in service.

295026 EA1.01
Ability to operate and/or monitor the following as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE: Suppression pool cooling (CFR 41.10)

OC Learning Objective:
<u>2621.828.0.0009, Objective L:</u>
Given normal operating procedures and documents for the system, describe or interpret the procedural steps.

Cognitive Level: Comprehension or Analysis

Question Type: New

References: EMG-3200.02, Support Procedure 25, 341

46. A large un-isolable leak developed in the torus while the reactor was operating at rated power. The timeline for torus water level is as follows (<u>all (t) times are in minutes</u>):

- At t = 0, Primary Containment Control entry was required due to low torus water level
- At t = 10, makeup to the torus was commenced using Core Spray System 1
- At t = 30, torus water level was 130 inches
- At t = 50, torus water level was 120 inches

Which statement below is true for these conditions (assume no further operator actions)?

a. At t = 70 minutes, PSP will be exceeded
b. At t = 70 minutes, the EMRV tailpipes begin to uncover
c. At t = 80 minutes, Emergency Depressurization will be required
d. At t = 110 minutes, the torus vent header downcomers begin to uncover

Answer: a

Handouts: EMG-3200.02

Justification: A is correct – the present rate of drop in torus level is 0.5 inches per minute (130 – 120 = 10 inches; 10 inches/20 minutes = 0.5 inches/minute). Therefore, in 20 more minutes (t = 70 minutes), level will have dropped to 110 inches, which is the point at which the torus vent header downcomers are uncovered, and the point at which the pressure suppression function of the primary containment can no longer be assured. It is for this reason that segment A-B of the PSP curve is vertical at 110 inches.

B is incorrect – the EMRV tailpipes are not uncovered until torus level reaches 90 inches, which will occur at t = 110 minutes given the current rate of drop in torus level.

C is incorrect – Emergency Depressurization is required <u>BEFORE</u> reaching 110 inches…at t = 80 minutes, level will have dropped to 105 inches.

D is incorrect – the torus vent header downcomers are uncovered at 110 <u>inches</u>…at t = 110 <u>minutes</u>, level will have dropped to 90 inches, which is where the EMRV downcomers start to uncover.
<u>NOTE</u>: the above calculations ignore the fact that torus water level will actually drop at a quicker rate due to the round shape of the torus.

295030 EA2.01

Ability to determine and/or interpret the following as they apply to LOW SUPPRESSION POOL WATER LEVEL: Suppression pool level (CFR 41.9, CFR 41.10)

OC Learning Objective:
2621.828.0.0032, Objective J:
Identify and interpret normal, abnormal, and Emergency Operating Procedures for Primary Containment.
2621.828.0.0032, Objective T:
Interpret Primary Containment indications in terms of limits and trends.

Cognitive Level: Comprehension or Analysis

Question Type: New

References: EMG-3200.02, EOP Users Guide

8/15/06: NRC Comments
Added no operator actions.

47. Plant conditions are as follows:

- An ATWS is in progress, with reactor power currently at 8%
- SLC System 1 is injecting
- RPV water level is being maintained between 0 and –20 inches
- Fuel Zone level indicators C and D have been turned on at Panel 4F

Which Fuel Zone level instrument channels will be used to control RPV water level?

a. A and B
b. B and D
c. C ONLY
d. D ONLY

Answer: b

Handouts: None

Justification: A is incorrect – FZLI channel A and C are not accurate when SLC is injecting.

B is correct – B and D are both available and are providing accurate level indication. FZLI channels A and C utilize the SLC injection line as the variable leg and are therefore not accurate when SLC is injecting. Note that when all recirc pumps are tripped, FZLI channels A and B automatically turn on.

C is incorrect – FZLI channel A and C are not accurate when SLC is injecting.

D is incorrect – FZLI channel B is also available.

295037 EA2.02
Ability to determine and/or interpret the following as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN: Reactor water level (CFR 41.5, 41.7)

OC Learning Objective:
<u>2621.828.0.0055, Objective D:</u>
Given a set of system indications or data, evaluate and interpret them to determine limits, trends and system status.
<u>2621.828.0.0055, Objective I:</u>
Explain or describe how this system is interrelated with other plant systems.

Cognitive Level: Comprehension or Analysis

Question Type: New

References: EOP Users Guide

8/15/06: NRC Comments
Added reactor power at 8% in the question stem.

48. The control room has been evacuated due to a fire. The fire has been extinguished. ABN-29, Plant Fires, requires the following ventilation systems shutdown prior to purging the control room.

- A and B 480V Switchgear Room Ventilation System
- A/B Battery Room, MG Set Room Ventilation System
- Chemistry Laboratory Ventilation System
- Reactor Building Ventilation System

According to ABN-29, the reason this action is taken is to prevent smoke and fumes purged from the control room from being brought into these areas, which could _______________

a. prevent personnel access
b. cause damage to equipment
c. set off automatic fire suppression systems
d. cause a reaction with other hazardous materials

Answer: c

Handouts: None

Justification: A is incorrect – this is not the reason stated in ABN-29.

B is incorrect – this is not the reason stated in ABN-29.

C is correct – this is the reason stated in ABN-29.

D is incorrect – this is not the reason stated in ABN-29.

600000 AK3.04
Knowledge of the reasons for the following responses as they apply to PLANT FIRE ON SITE: Actions contained in the abnormal procedure for plant fire on site. (CFR 41.10)

OC Learning Objective:
2621.828.0.0019, Objective E:
Describe and interpret procedure sections and steps for plant emergency or off-normal conditions that involve this system including personnel allocation and equipment operation in accordance with applicable ABN, SDRP, EOP and EOP support procedures, and EPIPs.

Cognitive Level: Memory or Fundamental Knowledge

Question Type: New

References: ABN-29

49. Given the following:

- A plant startup is in progress with reactor power at 6%
- The steam chest and high pressure turbine are being warmed
- Reactor feed pump (RFP) A is in service feeding through LFRV A
- A FWLC failure causes RPV level to rise to 183 inches

Which of the following occurs as a result of this failure?

a. RFP A trips ONLY
b. The main turbine trips ONLY
c. RFP A <u>and</u> the main turbine trip
d. Neither RFP A <u>nor</u> the main turbine trip

Answer: b

Handouts: None

Justification: A is incorrect – RFP A will not trip since ROPS is bypassed.

B is correct – the main turbine must be reset if the steam chest and HP turbine are being warmed and it will trip when RPV level reaches 175 inches. For RFP A to trip, the ROPS logic must see RPV level at ≥181 inches and feedwater flow greater than 2.23 E6 lbm/hr. Feedwater flow will not get this high when feeding through the LFRV, which is rated for a maximum of 1500 gpm (1500 gpm is equal to 0.72 E6 lbm/hr).

C is incorrect – the main turbine will trip; RFP A will not trip.

D is incorrect – the main turbine will trip.

295008 G2.1.27
High Reactor Water Level / Conduct of Operations: Knowledge of system purpose and or function (CFR 41.5, 41.7)

OC Learning Objective:
<u>2621.828.0.0018, Objective A:</u>
Given plant operating conditions, describe or explain the purpose(s)/function(s) of the system and its components.
<u>2621.828.0.0018, Objective D:</u>
Describe the interlock signals and setpoints for the affected system components and expected system response including power loss or failed components.

Cognitive Level: Comprehension or Analysis

Question Type: New

References: 201, 317, RAP-H5d, RAP-H7d, ABN-10

50. The reactor was initially operating at rated power. A feedwater line break inside primary containment resulted in a high drywell pressure scram. Current plant conditions are as follows:

- RPV level is 88 inches and rising slowly with both CRD pumps injecting
- RPV pressure is being maintained at 800-900 psig with Isolation Condensers
- Drywell pressure is 13 psig and lowering slowly with drywell sprays initiated
- Torus water temperature is 145 °F and lowering slowly
- Torus water level is 168 inches and rising slowly

Which of the following is the most immediate reason for lowering torus water level?

To prevent exceeding the ____________________.

a. Torus Load Limit
b. Heat Capacity Temperature Limit
c. Primary Containment Pressure Limit
d. Maximum Pressure Suppression Primary Containment Water Level

Answer: a

Handouts: EMG-3200.02, or, providing the LARGE figures of the graphs would be preferred

Justification: A is correct – based on a torus water level of 168 inches and rising, and a reactor pressure of 800-900 psig, the Torus Load Limit is the most immediate concern since it will be exceeded before any of the other limits. From Figure E (TLL) of EMG-3200.02, the Torus Load Limit that corresponds to an RPV pressure of 800 to 900 psig is ~174 to 178 inches.

B is incorrect – since torus water temperature is lowering, the margin to the HCTL is improving.

C is incorrect – for the given torus water level and torus pressure (drywell pressure), the PCPL is of no concern…torus pressure would have to rise above 50 psig at the given torus level for this to be a concern.

D is incorrect – the MPSPCWL is 188 inches, which makes it a secondary concern relative to the Torus Load Limit.

295029 EK3.02
Knowledge of the reasons for the following responses as they apply to HIGH SUPPRESSION POOL WATER LEVEL: Lowering suppression pool water level (CFR 41.9, 41.10)

OC Learning Objective:
2621.828.0.0032, Objective J:
Identify and interpret normal, abnormal and Emergency Operating Procedures for the Primary Containment System

Cognitive Level: Comprehension or Analysis

Question Type: New

References: EMG-3200.02, EOP Users Guide

51. Consider an event in which an accident causes a high-energy radioactive system to discharge into the Reactor Building.

Assuming the radioactivity release into the Reactor Building is the same in each case, which of the following results in the <u>highest</u> off-site release rate?

	Reactor Building ΔP	Standby Gas Treatment flow
a.	-0.10 inches WG	2600 scfm
b.	0.0 inches WG	0 scfm
c.	0.10 inches WG	0 scfm
d.	0.10 inches WG	2600 scfm

Answer: d

Handouts: None

Justification: A is incorrect – in this case there is a minimal release through SGTS, which is ~99% efficient. Since SGTS is able to maintain a negative RB ΔP, there is no ground level release.

B is incorrect – with RB ΔP at zero and no SGTS flow there is no release.

C is incorrect – in this case only a ground level release is occurring since there is a positive pressure in the Reactor Building and no SGTS flow.

D is correct – since there is a positive pressure in the Reactor Building a ground level release is occurring, which is equivalent to the ground level release in choice C (based on the same RB ΔP). In addition, since SGTS is not 100% efficient (see UFSAR Table 6.5-1), there is some relatively small release through this path. Therefore, this case results in the <u>highest</u> off-site release rate.

295035 EK2.03
Knowledge of the interrelations between SECONDARY CONTAINMENT HIGH DIFFERENTIAL PRESSURE and the following: Off-site release rate (CFR 41.8, 41.9)

OC Learning Objective:
<u>2621.828.0.0042, Objective F:</u>
Given a set of system indications or data, evaluate and interpret them to determine limits, trends and system status.
<u>2621.828.0.0042, Objective L:</u>
Explain or describe how this system is interrelated with other plant systems.

Cognitive Level: Comprehension or Analysis

Question Type: New

References: UFSAR Table 6.5-1

8/15/06: NRC Comments
Corrected answer choice labels from a, b, d, c to a, b, c, d. Answer d is correct.

52. Given the following:

- The reactor is shutdown due to a forced outage
- Reactor water level is 165 inches on NR GEMAC
- Three (3) Shutdown Cooling pumps are in service
- Total Shutdown Cooling System flow is 7500 gpm

According to Procedure 305, Shutdown Cooling System Operation, raising shutdown cooling system flow rate may result in…

a. flow-induced vibration of the shutdown cooling heat exchangers
b. damage to the nuclear instrumentation due to flow-induced vibration
c. Spurious trips of the shutdown cooling pumps due to low suction pressure
d. Exceeding the maximum design tube-side flow rate of the SDC heat exchangers

Answer: c

Handouts: None

Justification: A is incorrect – Procedure 309.2 has a P&L to limit RBCCW flow to less than 3700 gpm to prevent damage to the RBCCW heat exchanger from flow induced vibration. This is not related to SDC (tube side) flow but could be a misconception.

B is incorrect – this is related to a precaution associated with Reactor Recirculation System operation (P&L 5.2.8 of Procedure 301.2), not Shutdown Cooling.

C is correct – as stated in 305, "Simultaneous operation of all three (3) SDC Pumps at high flow rates (System Flow >7500 gpm) may result in pump suction pressures near the trip setpoint (4 psig). To avoid spurious pump trips, operation at system flow > 7500 gpm should be minimized.

D is incorrect – according to 305, the maximum design tube side flow rate is 3400 gpm.

205000 A1.02
Ability to predict and/or monitor changes in parameters associated with operating the SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE) controls including: SDC/RHR pump flow (CFR 41.10)

OC Learning Objective:
<u>2621.828.0.0045, Objective P:</u>
Identify and explain the normal operating procedures for the Shutdown Cooling System.

Cognitive Level: Memory of Fundamental

Question Type: New

References: 301.2, 305, 309.2

53. Which of the following methods of makeup to the Isolation Condensers require operation of the A or B Isolation Condenser Makeup Valves, V-11-36 or V-11-34, on Panel 5F/6F?

1. Adding makeup with Demineralized Water IAW 307, Isolation Condenser System
2. Adding makeup with Fire Protection IAW 307, Isolation Condenser System (NOT from local hose stations)
3. Adding makeup with Fire Protection via local hose stations IAW 307, Isolation Condenser System
4. Core Spray makeup to the Isolation Condensers IAW 308, Emergency Core Cooling System Operation

a. 1 and 2
b. 3 and 4
c. 1 and 3
d. 2 and 4

Answer: d

Handouts: None

Justification: A, B and C are incorrect – makeup with Demineralized Water (choice 1) is the normal method of shell makeup when the IC's are in standby. This method fills the IC shells via grab sample lines…this flow path does not utilize V-11-36 or V-11-34. Makeup from Fire Protection via local hose stations (choice 3) utilizes the IC shell drain lines as the makeup flow path…does not utilize V-11-36 or V-11-34.

D is correct – adding makeup with Fire Protection IAW 307 (choice 2), and Core Spray makeup IAW 308 (choice 4), are the only methods (of those given) that utilize makeup valves V-11-36 or V-11-34, which are the normal makeup supply valves from the Condensate Transfer System.

207000 A4.06
Ability to manually operate and/or monitor in the control room: Shell side makeup valves (CFR 41.8, 41.10)

OC Learning Objective:
2621.828.0.0023, Objective C:
Describe or trace (given a simplified drawing or P&ID) the basic flow path for the following modes of Isolation Condenser operation:
1. Standby
2. Emergency Operation
3. Sources of Shell Side Makeup

Cognitive Level: Memory of Fundamental

Question Type: New

References: 307, 308 GE 148F262

8/15/06: NRC Comments
They wanted me to verify that answers b and d did require the use of the isolation condenser makeup valves V-11-34 or V-11-36. See pages 26, 29 of 307, and page E12-2 of 308. These valves do need to be operated to fill the IC shells from fire water/core spray. Later, it was realized that the question is asking for the operation of both IC makeup valves. As written, both valves may not be simultaneously opened. The question was changed to OR instead of and.

54. Given the following:

- An automatic scram occurred while operating at rated power
- All control rods did **NOT** fully insert; reactor power is 13%
- Reactor pressure band is being maintained at 800 to 1000 psig
- Reactor water level band is being maintained at –20 to +30 inches
- Standby Liquid Control (SLC) System #1 is injecting into the RPV

Which one of the following conditions ensures adequate SHUTDOWN MARGIN?

	Length of Time SLC has been Injecting	SLC Tank Concentration
a.	30 minutes	12 weight percent
b.	60 minutes	12 weight percent
c.	25 minutes	15 weight percent
d.	35 minutes	15 weight percent

Answer: b

Handouts: Tech Spec 3.2

Justification: A is incorrect – 30 gpm for 30 minutes yields 900 gallons of boron solution at 12 weight percent. This is outside the shaded area of Tech Spec Figure 3.2-1, which means insufficient boron solution would have been injected to ensure adequate SHUTDOWN MARGIN.

B is correct – SLC pump capacity is 30 gpm. 1800 gallons of boron at 12 weight percent would have been injected after 60 minutes. This is within the shaded area of Tech Spec Figure 3.2-1, which represents the acceptable values of liquid control tank volume and solution concentration which assure that, with one 30 gpm liquid control pump, the reactor can be brought to the cold shutdown condition from a full power steady state operating condition at any time in core life independent of the control rod system capabilities. (The cross-hatched area of Figure 3.2-1 represents the acceptable values of liquid control tank volume and solution concentration which assure that the equivalency requirements of 10 CFR 50.62—ATWS Rule—are satisfied. Note the Tech Spec definition of SHUTDOWN MARGIN is: "…the amount of reactivity by which the reactor would be subcritical when the control rod with the highest reactivity worth is fully withdrawn, all other operable control rods are fully inserted, all inoperable control

rods are at their current position, reactor water temperature is 68ºF, and the reactor fuel is xenon free. Determination of the control rod with the highest reactivity worth includes consideration of any inoperable control rods which are not fully inserted."
C is incorrect – this is outside the shaded area of Tech Spec Figure 3.2-1 (750 gallons at 15%).

D is incorrect – 1050 gallons of 15 weight percent of boron is outside the shaded area of Tech Spec Figure 3.2-1.

211000 K5.03
Knowledge of the operational implications of the following concepts as they apply to STANDBY LIQUID CONTROL SYSTEM: Shutdown margin (CFR 41.6)

OC Learning Objective:
<u>2621.828.0.0046, Objective A:</u>
Given plant operating conditions, describe or explain the purpose(s)/function(s) of the system and its components.
<u>2621.828.0.0046, Objective N:</u>
Given Technical Specifications, identify and explain associated actions for each section of the Technical Specifications relating to this system including personnel allocation and equipment operation.

Cognitive Level: Comprehensive or Analysis

Question Type: New

References: UFSAR 9.3.5, EOP Users Guide, Tech Spec 1.45/3.2.C, 612.4.001

8/15/06: NRC Comments
They thought that answers c and were direct lookup. They suggested using injection times for these answers, and the question was modified to that shown.

9/8/06
Rossi reviewed. Placed answers in 3-column format.

55. Given the following:

- A reactor startup is in progress
- Reactor pressure is 500 psig
- Reactor water level is 160 inches
- Reactor power is 20% on IRM Range 8
- Two steam jet air ejectors (SJAE) are in service
- The steam chest and high-pressure turbine are being warmed
- A spurious reactor isolation occurs

Which of the following describes the plant response and/or the correct action for this event?

a. Commence a normal plant shutdown IAW 203, Plant Shutdown
b. An automatic scram occurs, enter ABN-1, Reactor Scram ONLY
c. An automatic scram occurs, enter ABN-1, Reactor Scram and RPV Control – No ATWS
d. Place the startup on hold until the failure is corrected, then re-open the MSIVs IAW 301.1, Main Steam Supply System

Answer: b

Handouts: None

Justification: A is incorrect – an automatic scram will occur.

B is correct – the reactor isolation results in closure of all MSIVs and main steam line drains (in addition to some other valves). At 500 psig (<600 psig), the MSIV closure scram is bypassed. Although there is relatively little steam flow, the reactor isolation will cause reactor pressure to rise. As pressure rises, power will also rise due to collapsing steam voids. The pressure rise will cause reactor power to increase to the IRM Hi-Hi scram setpoint, which is 38% on IRM Range 8. This will occur before pressure rises to the high-pressure scram setpoint of 1045 psig. The automatic scram will terminate the pressure rise. NOTE: as stated in Tech Spec 2.3 (LSSS) Bases, "Below 600 psig, when the MSIV closure scram is bypassed, scram protection is provided by the IRMs." For the given conditions (at this point in the startup), there would be one feedwater pump in service and with relatively low steam flow and feed flow, there would not be a significant change in RPV level due to the MSIV closure. Since RPV level remains above 138 inches, and RPV pressure remains below 1060 psig, there are no EOP entry conditions.

C is incorrect – the transient will not result in any EOP entry conditions.

D is incorrect – an automatic scram will occur.

212000 A2.11
Ability to (a) predict the impacts of the following on the REACTOR PROTECTION SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Main steamline isolation valve closure (CFR 41.5, 41.6)

OC Learning Objective:
<u>2621.828.0.0037, Objective D:</u>
Describe all RPS scram logic trip signals, including the following:
1. Purpose / Design Basis
2. Setpoints
3. Conditions that allow bypassing scram signals
4. How bypassing scram signals is accomplished

<u>2621.828.0.0037, Objective F:</u>
Given a set of system indications or data, evaluate and interpret them to determine limits, trends and system status.
<u>2621.828.0.0037, Objective N:</u>
Describe and interpret procedure sections and steps for plant emergency or off-normal conditions that involve this system including personnel allocation and equipment operation IAW applicable ABN, SDRP, EOP & EOP support procedures, and EPIPs.

Cognitive Level: Comprehensive or Analysis

Question Type: New

References: ABN-1, EMG-3200.01A, Tech Spec 2.3 Bases, 237E566

56. Given the following:

- A reactor startup is in progress
- IRM Range 6/7 correlation is required by Procedure 201, Plant Startup

Which of the following support personnel, if any, are required to be notified prior to performing this task IAW Procedure 402.2, IRM Operation During Startup?

a. An I&C Technician ONLY
b. A Reactor Engineer ONLY
c. An I&C Technician and a Reactor Engineer
d. No support personnel are needed, this task is performed by Operations ONLY

Answer: a

Handouts: None

Justification: A is correct – Prerequisite 3.2 of Procedure 402.2 requires I&C to be notified to perform the IRM Range 6/7 correlation.

B is incorrect – a Reactor Engineer is not needed for IRM Range 6/7 correlation, but is needed to perform IRM calibration IAW Procedure 1001.9.

C is incorrect – only an I&C Technician is needed for IRM Range 6/7 correlation.

D is incorrect – an I&C Technician is required to support performance of this task. Specifically, they perform any required IRM adjustments.

215004 G2.1.14
Intermediate Range Monitor (IRM) System / Conduct of Operations: Knowledge of system status criteria which require the notification of plant personnel. (CFR 41.10)

OC Learning Objective:
2621.828.0.0029, Objective K:
Given Technical Specifications, identify and explain associated actions for each section of the Technical Specifications relating to this system including personnel allocation and equipment operation.

Cognitive Level: Memory or Fundamental

Question Type: New

References: 201, 402.2

57. Given the following:

- Reactor power is 70%
- Five recirc loops are in service and total recirc flow on panel 4F is 15.0 E4 gpm
- The “C” recirc loop flow transmitter that feeds the Total Recirc Flow indicator on panel 4F fails to 0 (zero)

Recirc flow as displayed on panel 4F will read __(1)__ E4 gpm, and will result in a ___(2)___.

a. (1) 12.0
(2) rod block

b. (1) 12.0
(2) scram

c. (1) 13.5
(2) rod block

d. (1) 13.5
(2) scram

Answer: a

Handouts: Attachment 202.1-2

Justification: A is correct – the total recirc flow indicator on Panel 4F receives a signal from the flow monitor on Panel 5R, which inputs to APRM Channels 5 through 8. Prior to the failure, each flow transmitter was sensing approximately 3.0 E4 gpm, which is summed to produce 15.0 E4 total recirc flow. One transmitter failing to zero results in a total indicated recirc flow of 12.0 E4 gpm. This produces a 10% mismatch between the RPS Division 1 and RPS Division 2 recirc flow monitors, causing a flow comparator rod block.

B is incorrect – reactor power at 70% is well below the scram setpoint for recirc flow at 12.0 E4 gpm…the scram setpoint from Attachment 202.1-2 is approximately 104% power.

C and D are incorrect – one could arrive at 13.5 gpm if they thought there were 10 recirc flow inputs to the total recirc flow indicator on 4F, vice only 5.

215005 A1.04
Ability to predict and/or monitor changes in parameters associated with operating the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM controls including: SCRAM and rod block trip setpoints (CFR 41.7)

OC Learning Objective:
2621.828.0.0029, Objective F:
Describe the interlock signals and setpoints for the affected system components and expected system response including power loss or failed components.
2621.828.0.0029, Objective G:
Given a set of system indications or data, evaluate and interpret them to determine limits, trends and system status.

Cognitive Level: Comprehension or Analysis

Question Type: Bank

References: 202.1, 420, UFSAR 7.5.1.8.7, RAP-H7a

58. The plant was at rated power when an event occurred which required a manual scram. All scram-related systems function as designed.

Moments later, the Shift Manager declared the Control Room un-inhabitable and must be evacuated.

Prior to leaving the Control Room, the following switch positions are noted:

- The front-panel control switch for EMRV NR108A is in OFF
- The back-panel NORMAL-DISABLED switch for EMRV NR108B is in DISABLED

Which of the following is correct regarding the ability of the EMRVs to function in the Pressure Relief mode and to perform the ADS function in the event of a small-break LOCA?

	ADS Function	Pressure Relief Mode
a.	ALL EMRVs	ONLY EMRVs B, C, D, and E
b.	ALL EMRVs	ONLY EMRVs A, C, D, and E
c.	ONLY EMRVs A, C, D, and E	ONLY EMRVs C, D, and E
d.	ONLY EMRVs B, C, D, and E	ONLY EMRVs B, C, D, and E

Answer: c

Justification: (see drawing 729E182) With the EMRV control panel switch if OFF, the associated pressure switch will not function to open the EMRV on high reactor pressure (relief mode), but the ADS function remains unaffected. With the switch in DISABLED, all functions of the associated EMRV are defeated. Therefore, EMRV A (in OFF) can function in the ADS mode but not in the relief mode. EMRV B (in DISABLED) will not work in any mode. Answer c is correct, and the other selections are incorrect since they list the incorrect valves with their available modes.

218000 K3.01
Knowledge of the effect that a loss or malfunction of the AUTOMATIC DEPRESSURIZATION SYSTEM will have on following: Restoration of reactor water level after a break that does not depressurize the reactor when required (CFR 41.7)

OC Learning Objective:
2621.828.0.0005, Objective I:
Describe the operation of the ADS controls including: Removal of ADS control logic fuses to close EMRVs.

Cognitive Level: Comprehensive or Analysis

Question Type: New

References: GE 729E182

8/15/06: NRC Comments
As originally written, they thought it was too easy: fuse removal gives it away. This question has been rewritten.

9/8/06
Rossi reviewed. Broke initial paragraph into 2 paragraphs. Placed answers in 3-column format.

59. Given the following:

- The reactor is operating at rated power
- Pressure switch PS-1A83A fails low

Using the attached drawing, GE 148F712 (see coordinates G-8), what is the effect of this failure?

a. One of the five ADS/EMRVs will NOT actuate in the ADS mode
b. One of the five ADS/EMRVs will NOT actuate in the Pressure Relief mode
c. Two of the five ADS/EMRVs will NOT actuate in the ADS mode
d. Two of the five ADS/EMRVs will NOT actuate in the Pressure Relief mode

Answer: b

Handouts: GE 148F712 (ensure large enough to read)

Justification: A is incorrect – since the failed pressure switch senses reactor pressure, and ADS functions on RPV level and Drywell pressure only, the ADS mode is not affected by this failure.

B is correct – PS-1A83A provides a high reactor pressure signal to EMRV NR108A. If this pressure switch fails low, EMRV A will not open on high reactor pressure (1065 psig).

C is incorrect – since the failed pressure switch senses reactor pressure, and ADS functions on RPV level and Drywell pressure only, the ADS mode is not affected by this failure.

D is incorrect – although 2 of 5 EMRV's open at ≤ 1085 psig, and 3 of 5 EMRV's open at ≤ 1105 psig, each EMRV has a dedicated pressure switch that provides a high reactor pressure signal to the respective EMRV actuation logic. PS-1A83A provides a reactor pressure signal to EMRV NR108A only.

239002 K6.01
Knowledge of the effect that a loss or malfunction of the following will have on the RELIEF/SAFETY VALVES: Nuclear boiler instrument system (pressure indication) (CFR 41.7)

OC Learning Objective:
<u>2621.828.0.0005, Objective E:</u>
Describe the EMRV initiation logic for both over-pressure operation and operation in the ADS mode. Include the following:

1. Initiation signals and setpoints
2. Timers and setpoints
3. Control switches
4. Panel indications

<u>2621.828.0.0005, Objective J:</u>
State how the following systems interrelate with ADS:

1. Vessel and Primary Containment Instrumentation
2. Core Spray
3. NSSS
4. Vital AC Power
5. 125 VDC Power

Cognitive Level: Comprehension or Analysis

Question Type: New

References: GE 148F712, GE 729E182, sh. 1, UFSAR 5.2.2.4

8/15/06 NRC Comment
They questioned wether a single failure could effect 2 EMRVs. Since 2 EMRVs open at the same set pressure, one pressure switch failure could possibly be thought to effect 2 EMRVs. The question remained as-is.

60. Given the following:

- The reactor is operating at rated power when a small leak develops inside the drywell
- Drywell temperature is 175 °F and drywell pressure is 2.6 psig; both are rising slowly
- The Unit Supervisor directs venting the primary containment using the Standby Gas Treatment System (SGTS) IAW Support Procedure 31

Which statement below describes how venting the primary containment IAW Support Procedure 31 affects the suppression chamber-to-drywell vacuum breakers and the reactor building-to-suppression chamber vacuum breakers? Assume the venting evolution causes containment pressure to lower.

Venting from the ___(1)___ could cause the _____(2)_____ vacuum breakers to open.

a. (1) torus
(2) suppression chamber-to-drywell

b. (1) torus
(2) reactor building-to-suppression chamber

c. (1) drywell
(2) suppression chamber-to-drywell

d. (1) drywell
(2) reactor building-to-suppression chamber

Answer: c

Handouts: None

Justification: A is incorrect – for the suppression chamber-to-drywell vacuum breakers to open, suppression chamber pressure must <u>exceed</u> drywell pressure by at least 0.5 psid. For the given conditions, venting from the torus will cause suppression chamber pressure to remain below drywell pressure.

B is incorrect – for the reactor building-to-suppression chamber vacuum breakers to open, suppression chamber pressure must be at least 0.5 psid <u>less than</u> Reactor Building pressure, which is approximately at atmospheric pressure (-0.25" WG). For the given conditions, venting the torus (to atmosphere) will not cause suppression chamber pressure to go below Reactor Building pressure by 0.5 psid.

C is correct – venting from the drywell will cause drywell pressure to lower relative to suppression chamber pressure and if drywell pressure is 0.5 psid less than suppression chamber pressure, the suppression chamber-to-drywell vacuum breakers will open.

D is incorrect – for the given conditions, venting from the drywell will not cause suppression chamber pressure to go below Reactor Building pressure.

261000 A1.06
Ability to predict and/or monitor changes in parameters associated with operating the STANDBY GAS TREATMENT SYSTEM controls including: Drywell and suppression chamber differential pressure: Mark-I (CFR 41.7, 41.9)

OC Learning Objective:
<u>2621.828.0.0042, Objective M:</u>
Describe and interpret procedure sections and steps for plant emergency or off normal conditions that involve this system including personnel allocation and equipment operations IAW applicable ABN, SDRP, EOP and EOP support procedures and EPIPs.

Cognitive Level: Comprehension or Analysis

Question Type: New

References: EMG-3200.02, EOP Users Guide, Fundamentals

61. The reactor is operating at rated power when Bus 1C undervoltage relay 27-13C fails low. Annunciator BUS 1C VOLTS LO goes into alarm. Bus 1C indications on Panel 8F/9F are normal.

How does EDG # 1 respond to this event?

EDG #1 ________

a. remains in standby
b. starts and idles at 400 RPM
c. fast starts, output breaker closes
d. fast starts, output breaker does NOT close

Answer: a

Handouts: None

Justification: A is correct – the Bus 1C (and 1D) undervoltage relays are arranged in a two-out-of-three logic scheme, which requires any two relays to trip to disconnect the bus from its normal source (Bus 1A), actuate bus load shedding, and start the EDG. If a single relay drops out on undervoltage, the annunciator will go into alarm, but the automatic actions described above will not occur until a second relay drops out on undervoltage (or fails low).

B is incorrect – EDG #1 will idle or fast start as needed.

C and D are incorrect – EDG #1 will not fast start until/unless at least one of the other two bus undervoltage relays (27-11C, 27-12C) drop out on low voltage.

262001 K3.02
Knowledge of the effect that a loss or malfunction of the A.C. ELECTRICAL DISTRIBUTION will have on following: Emergency generators (CFR 41.7)

OC Learning Objective:
2612.828.0.0013, Objective C:
Given a set of system indications or data, evaluate and interpret them to determine limits, trends and system status.
2612.828.0.0013, Objective I:
Describe the interlock signals and setpoints for the affected system components and expected system response including power loss or failed components.
2612.828.0.0013, Objective N:
State the function and interpretation of system alarms, alone and in combination, as applicable in accordance with the system RAPS.

Cognitive Level: Comprehension or Analysis

Question Type: New

References: RAP-T3a, UFSAR 8.3.1.1.1

8/15/06 NRC Comments
They thought that selection b (is prevented from starting) was not plausible. This selection has been changed to that shown.

62. Which one of the following annunciators would be accompanied by a loss of power to Main Steam Line Radiation Monitors RN06A & RN06B on Panel 2R?

a. IP-4 PWR LOST
b. CIP 3 PWR LOST
c. 24VDC PP-A PWR LOST
d. PROT SYS PNL 1 PWR LOST

Answer: d

Handouts: None

Justification: A is incorrect – none of the MSL radiation monitoring equipment is powered by IP-4 (see ABN-58).

B is incorrect – CIP-3 provides power to the MSL radiation monitor <u>recorders</u> on Panel 10F (see ABN-58).

C is incorrect – none of the MSL radiation monitoring equipment is powered by 24 VDC (see 340.2).

D is correct – MSL Radiation Monitors RN06A & B on Panel 2R are powered from Protection System Panel #1, breaker #10 (see 406.1).

262002 K1.14
Knowledge of the physical connections and/or cause-effect relationships between UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.) and the following: Main steam line radiation monitors (CFR 41.7)

OC Learning Objective:
<u>2621.828.0.033A, Objective G:</u>
Explain or describe how this system is interrelated with other plant systems.
<u>2621.828.0.033A, Objective L:</u>
State the function and interpretation of system alarms, alone and in combination, as applicable in accordance with the system RAPS.

Cognitive Level: Memory or Fundamental

Question Type: New

References: ABN-58, 406.1, 340.2, ABN-50

63. Which one of the following shows the correct correlation between the Emergency Diesel Generator governor mode of operation (droop, isochronous) and the positions of the EDG Mode Selector (PTD) Switch?

P = Peaking
T = Transfer
D = Deadline

	Droop	Isochronous
a.	P	T and D
b.	P and T	D
c.	D	P and T
d.	T and D	P

Answer: b

Handouts: None

Justification: B is correct – the three-position PTD switch has the following functions: (1) PEAKING – sets up EDG to assume 2750 KW on a normal start. Since this operation is in parallel with the grid, the governor would be in the DROOP mode. (2) TRANSFER – sets up governor control circuitry for load transfer. Load transfer occurs between the EDG and the grid, which again means parallel operation and the governor is in the DROOP mode. (3) DEADLINE – sets up EDG control circuits for isochronous operation.

A, C and D are incorrect – these are all incorrect combinations of the Mode Selector (PTD) Switch positions and the EDG governor modes of operation.

264000 K4.03
Knowledge of EMERGENCY GENERATORS (DIESEL/JET) design feature(s) and/or interlocks which provide for the following: Speed droop control (CFR 41.7)

OC Learning Objective:
<u>2621.828.0.0013, Objective H:</u>
Identify and explain system operating controls/indications under all plant operating conditions.

Cognitive Level: Memory or Fundamental

Question Type: New

References: 341

64. The plant was at rated power, when the following annunciator <u>came into alarm:</u>

- ROD CNTRL – CONTROL AIR PRESS LO

The following conditions are noted:

- INSTR AIR SUPPLY PRESS indicates 77 psig and lowering very slowly
- An NLO in the field reports the air receivers indicate 105 psig and steady
- Another NLO reports that the pre-filter Dp indicator is off-scale high
- There are no indications of any air leaks and there are no valve mis-positions.

Which of the following lists the effect on the air systems under the given conditions, and the <u>next</u> expected operator action, as outlined in ABN-35, Loss of Instrument Air? (assume no operator actions other than that listed)

	Effect	Action
a.	Service Air pressure will begin to decrease	When Service Air drops to 90 psig, then confirm the lag compressor has started
b.	Service Air pressure will begin to decrease	When Service Air pressure drops to 90 psig, then confirm Service Air valve V-6S-2 is closed and is NOT bypassed
c.	Instrument air pressure will continue to degrade	When Instrument Air pressure drops to 75 psig, then confirm that Service Air valve V-6S-2 is open and is NOT bypassed
d.	Instrument air pressure will continue to degrade	When Instrument air pressure drops to 55 psig, then scram the reactor

Answer: d

Handouts: None

Justification: The indications provided show a normal pressure at the air receivers, but a degraded air pressure downstream of the post-filters going to instrument air. Control Room instrument air pressure indication is measured just downstream of the pre-filters, air dryers and post-filters. The annunciator provided in the stem is sensed downstream of where the Control Room indication is sensed. There is no air leakage and no mis-positioned valves. The only possible cause for the degraded instrument air, is a plug of the pre-filters, or post-filters, or an air dryer failure. These components effect instrument air only.

Outside air if filtered, compressed and sent to the air receivers. Air from the air receiver outlet can either go to instrument air (after passing through the pre-filters, the air dryers, and the post-filters) or to service air (which is not filtered and dried). From the given indications, there is no abnormality with the service air side of the system. Therefore, there is no low air pressure sensed in service air, and none is expected. This would make answers a and b incorrect.

Answer b is also not a correct action from ABN-35. Answer b is incorrect.

Answer c is incorrect in that it says to confirm the service air valve open, when it should be confirmed closed.

Answer d is correct in that it refers to actions to take when instrument air is degrading to 55 psig.

300000 A2.01
Ability to (a) predict the impacts of the following on the INSTRUMENT AIR SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Air dryer and filter malfunctions (CFR 41.7, 41.10)

OC Learning Objective:
<u>2621.828.0.0043, Objective G:</u>
Describe the interlock signals and setpoints for the affected system components and expected system response including power loss or failed components.
<u>2621.828.0.0043, Objective H:</u>
Given a set of system indications or data, evaluate and interpret them to determine limits, trends and system status.
<u>2621.828.0.0043, Objective I:</u>
Identify and explain the system operating controls/indications under all plant operating conditions.
<u>2621.828.0.0043, Objective K:</u>
State the function and interpretation of system alarms, alone and in combination, as applicable in accordance with the system RAPS.

Cognitive Level: Comprehension or Analysis

Question Type: New

References: RAP-H1a, ABN-35, BR 2013, sh. 1

8/15/06 NRC Comments
They said that the correct answer did not match the ABN-35 steps, in that the intended correct response would only be taken when service air was low – not when instrument air was low. The question has been re-written.

9/8/06
Rossi reviewed. Deleted: with all systems normally aligned in the stem. Placed second stem paragraph and placed into bullets, and added indicator label for air pressure. Placed answers in 3-column format.

65. Given the following:

- The reactor is operating at rated power with the ‘A’ CRD pump in service
- The Unit Supervisor directs swapping to the ‘B’ CRD pump so maintenance can be performed on the ‘A’ CRD pump

Which of the following actions, if any, must be performed prior to starting CRD pump ‘B’?

a. Close ‘B’ CRD pump discharge valve
b. Open ‘B’ CRD pump suction valve ONLY
c. Open ‘B’ CRD pump suction <u>and</u> discharge valves
d. Take manual control of the in-service flow control valve

Answer: d

Handouts: None

Justification: A is incorrect – this is only required during initial pump startup when placing the CRD system in service.

B and C are incorrect – as stated in Procedure 302.1, NOTE 4.3.1, “during normal operation, both CRD pumps are usually valved to the system so that pump changeover for purposes other than maintenance may be made from the control room.”

D is correct – for routine pump changeover, Procedure 302.1 requires taking manual control of the in-service flow control valve (NC03A or NC03B) prior to starting the standby (alternate) CRD pump in service.

201001 A4.01
Ability to manually operate and/or monitor in the control room: CRD pumps (CFR 41.10)

OC Learning Objective:
<u>2621.828.0.0011, Objective 13:</u>
Given normal operating procedures and documents for the system, describe or interpret the procedural steps.

Cognitive Level: Memory or Fundamental

Question Type: New

References: 302.1

66. Given the following:

- The reactor is operating at rated power
- Annunciator CCW FLOW LO <u>goes into alarm</u> for the 'A' reactor recirc pump
- The crew has entered ABN-19, RBCCW Failure Response

Which one of the following describes (1) when a reactor scram is required by ABN-19, and (2) the limiting reactor recirculation system component that this action is based on?

a. (1) when one CCW FLOW LO annunciator has been in alarm for > 1 minute
(2) recirc pump seals

b. (1) when more than one CCW FLOW LO annunciator has been in alarm for > 1 minute
(2) recirc pump seals

c. (1) when one CCW FLOW LO annunciator has been in alarm for > 1 minute
(2) recirc pump motor

d. (1) when more than one CCW FLOW LO annunciator has been in alarm for > 1 minute
(2) recirc pump motor

Answer: b

Handouts: None

Justification: A is incorrect – ABN-19 directs a reactor scram when there are two or more CCW FLOW LO alarms for greater than one minute.

B is correct – when RPV temperature is > 212 °F, the mode switch is in STARTUP or RUN, and there are <u>two or more</u> CCW FLOW LO alarms for <u>greater than one minute</u>, ABN-19 directs a reactor scram and trip of all operating recirc pumps. Recirc pumps seals are the limiting component since (1) the seal temperature limits specified in ABN-19 are lower for the seals than for the pump motors and (2) high seal temperatures can cause seal failure, which is of higher consequence than high motor bearing and/or winding temperatures.

C is incorrect – ABN-19 directs a reactor scram when there are two or more CCW FLOW LO alarms for greater than one minute. Recirc pump seals are the limiting component.

D is incorrect – recirc pump seals are the limiting component.

202001 A2.17
Ability to (a) predict the impacts of the following on the RECIRCULATION SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Loss of seal cooling water (CFR 41.3, 41.10)

OC Learning Objective:
<u>2621.828.0.0035, Objective M:</u>
Using the procedure, identify and explain normal and emergency operations of the RBCCW system.

Cognitive Level: Memory or Fundamental

Question Type: New

References: RAP-E7d, RAP-E7b, ABN-19

8/15/06 NRC Comments
They asked if the CCW Low Flow alarm was to the ‘A’ recirc. pump only. Yes. Each pump has its own individual alarm. The question remains as-is.

67. The reactor was operating at rated power with the ‘B’ RWCU pump in service when a turbine trip occurred. The following conditions currently exist:

- An ATWS is in progress
- Reactor pressure is 950 psig
- RPV water level was lowered to 110”TAF and is stable
- The Reactor Operator initiates Standby Liquid Control System 1
- The following indications are observed:
  - System 1 PUMP ON light on Panel 4F is lit
  - System 1 SQUIBS light on Panel 4F is NOT lit
  - Pump discharge pressure on Panel 4F is 1080 psig
  - FLOW ON annunciator on Panel 3F is NOT in alarm
  - SQUIB VALVE OPEN annunciator on Panel 3F is NOT in alarm

What is the status of the Reactor Water Cleanup (RWCU) System?

RWCU is ____(1)____ and the ‘B’ RWCU pump is ____(2)____.

a. (1) isolated
(2) tripped

b. (1) isolated
(2) NOT tripped

c. (1) NOT isolated
(2) tripped

d. (1) NOT isolated
(2) NOT tripped

Answer: d

Handouts: None

Justification: A is incorrect – RWCU is not isolated and the ‘B’ RWCU pump is not tripped.

B is incorrect – RWCU is not isolated.

C is incorrect – the ‘B’ RWCU pump is not tripped.

D is correct – the given conditions indicate SLC System 1 pump started (PUMP ON light is lit; 1080 psig discharge pressure) but the squib valve did not fire (SQUIBS light not lit; SQUIB VALVE OPEN annunciator not in alarm) and there is no liquid poison flow into the reactor (FLOW ON annunciator not in alarm).

RWCU isolation on SLC initiation occurs based on system flow > 15 gpm as sensed by FS-IL06. This flow switch also inputs to the FLOW ON annunciator. Since the squib valve did not fire, system flow to the reactor did not reach the 15 gpm setpoint required to isolate RWCU. Since RWCU did not isolate, the 'B' RWCU pump is still running.

204000 K6.07
Knowledge of the effect that a loss or malfunction of the following will have on the REACTOR WATER CLEANUP SYSTEM: SBLC logic (CFR 41.6, 41.7)

OC Learning Objective:
<u>2621.828.0.0039, Objective D:</u>
Describe the interlock signals and setpoints for the affected system components and expected system response including power loss of failed components.
<u>2621.828.0.0039, Objective F:</u>
Explain or describe how this system is interrelated with other plant systems.

Cognitive Level: Comprehensive or Analysis

Question Type: New

References: 304, RAP-G1b, RAP-G2b

9/22/06: Validator comments: Added a bullet in the stem saying that RPV water level that was above the level setpoint for cleanup system isolation (>86").

68. Given the following:

- The reactor is operating at rated power
- LPRM detector calibrations are in progress
- One TIP detector is IN-CORE
- A leak occurs inside the drywell, causing drywell pressure to exceed 3.5 psig

As the TIP detector retracts from the reactor, when will its ball valve automatically close?

As soon as the detector ________

a. is moved outside of the indexer
b. is outside the primary containment
c. has moved past the ball valve
d. has moved into the shield chamber

Answer: d

Handouts: None

Justification: A, B and C are incorrect – the ball valve will not close until the detector is in the shield chamber, as indicated by the “in-shield” limit switch.

D is correct – when the TIP system receives a containment isolation signal due to high drywell pressure, all detectors not “in-shield” will retract to the in-shield position, the associated ball valves will close, and the purge valve will close. The ball valve receives a close signal from the detector “in-shield” limit switch (see 405.2, P&L 4.17, among other places in 405.2).

215001 A3.03
Ability to monitor automatic operations of the TRAVERSING IN-CORE PROBE including: Valve operation (CFR 41.7)

OC Learning Objective:
<u>2621.828.0.0029, Objective F:</u>
Describe the interlock signals and setpoints for the affected system components and expected system response including power loss or failed components.

Cognitive Level: Memory or Fundamental

Question Type: New

References: 405.2

8/15/06 NRC Comments
They did not think the original answer a (is outside the reactor vessel) was credible. It was changed to that shown.

69. Which of the following is a Fuel Pool Cooling System design feature that prevents inadvertent draining of the spent fuel storage pool?

a. Spent fuel storage pool liner telltale drain system
b. No penetrations in the spent fuel storage pool wall
c. Fuel pool cooling pumps trip on Lo-Lo skimmer surge tank level
d. Anti-siphon check valves in the spent fuel storage pool diffuser lines

Answer: d

Handouts: None

Justification: A is incorrect – the telltale drain system provides indication of a spent fuel storage pool liner leak…it does not prevent inadvertent draining of the pool.

B is incorrect – there are penetrations in the spent fuel storage pool wall, however none are below the level of 1 foot above the top of the stored fuel.

C is incorrect – the FPC pumps do trip on Lo-Lo skimmer surge tank level, but the purpose of this trip is to prevent loss of NPSH to the FPC pumps. Based on the design of the system (weir overflow of SFSP to SST, among others), the FPC pumps cannot drain the spent fuel storage pool.

D is correct – the diffuser lines (SFSP return) are the only lines that go below the top of the stored fuel. The anti-siphon check valves prevent inadvertent draining of the SFSP during reactor cavity letdown (siphoning from the spent fuel storage pool to the reactor cavity).

233000 K4.06
Knowledge of FUEL POOL COOLING AND CLEAN-UP design feature(s) and/or interlocks which provide for the following: Maintenance of adequate pool level (CFR 41.7, 41.9)

OC Learning Objective:
2621.828.8.0020, Objective A:
Given plant operating conditions, describe or explain the purpose(s)/function(s) of the system and its components.
2621.828.8.0020, Objective II:
Describe the operation of the anti-siphon check valves and analyze how the system may respond if they fail to perform their design function.

Cognitive Level: Memory or Fundamental

Question Type: New

References: UFSAR 9.1.2.2, 9.1.3, 205, 311, 237E756

70. Given the following:

- The plant is operating at rated power on four (4) recirc loops
- Recirc pump ‘B’ is out of service due to an oil leak in the motor
- The 4160V BUS 1B UV annunciator goes into alarm

Which of the following is the correct action to take for this event?

a. Scram the reactor and enter ABN-1, Reactor Scram
b. Confirm operating recirc pump speeds reduced to 20 to 30 Hz
c. Perform a rapid power reduction as directed by the Unit Supervisor
d. Enter 301.2, Reactor Recirculation System, for a scoop tube lockup on recirc pump ‘D’

Answer: a

Handouts: None

Justification: A is correct – an undervoltage condition on bus 1B causes a trip of recirc pump D, a trip of condensate pumps B & C, and a trip of feedwater pumps B & C. For multiple condensate and/or feedwater pump trips, ABN-17 (Feedwater System Abnormal Conditions) directs scramming the reactor and entering ABN-1. The correct action from ABN-2 (Recirculation System Failures) is to close the D pump discharge valve.

B is incorrect – this action is directed by ABN-2 for a single recirc pump trip with only 3 loops initially in service (2 pumps remaining in service). This action would not be taken when there are 3 or 4 pumps remaining in service. ABN-2 does require recirc pump speed to be less than 33 Hz when only 3 recirc pumps are in service, but it does not direct action to lower speed to the 20 – 30 Hz range.

C is incorrect – as stated in ABN-17, this is the correct action to take for a single condensate pump or single feedwater pump trip.

D is incorrect – an undervoltage condition on bus 1B causes a scoop tube lockup on the D recirc pump and this action would be necessary to control the speed of the D recirc pump if it were still running. Since it tripped on the UV condition, this action would not be taken.

259001 K2.01
Knowledge of electrical power supplies to the following: Reactor feedwater pump(s): Motor-Driven-Only (CFR 41.7)

OC Learning Objective:
2621.828.0.0017, Objective C:
Given the system logic/electrical drawings describe the system trip signals, setpoints and expected system response including power loss or failed components.

Cognitive Level: Comprehensive or Analysis

Question Type: Modified Bank

References: RAP-T4c, ABN-2, ABN-17

71. The following communication takes place over the radio between the Reactor Operator and the Reactor Building Operator while adding makeup to the “A” Isolation Condenser from the Fire Protection System:

RO: “Open V eleven forty nine, makeup valve from Fire Protection to the Isolation Condensers”

NLO: “I understand, open V eleven forty nine, Fire Protection to Isolation Condenser Makeup”

RO: “That is correct.”

This communication is __________.

a. INCORRECT because the NLO paraphrased the direction given by the RO
b. INCORRECT because the RO and NLO did not use the phonetic alphabet for “V”
c. INCORRECT because the RO and NLO did not address each other by name or title
d. CORRECT because it meets the requirements of HU-AA-101, Human Performance Tools and Verification Practices

Answer: c

Handouts: None

Justification: A is incorrect – paraphrasing is allowed by HU-AA-101.

B is incorrect – HU-AA-101 only requires use of the phonetic alphabet “as required, to ensure proper component identification.”
In this case, C is correct – HU-AA-101 requires the sender to address the receiver by name or title. HU-AA-101 also states: “For non-face-to-face verbal communication, the sender and receiver shall IDENTIFY themselves by stating their name or title.”

D is incorrect – this communication does not meet the requirements of HU-AA-101.

G2.18 Conduct of Operations: Ability to coordinate personnel activities outside the control room. (CFR 41.10)

OC Learning Objective:
<u>2621-PBIG.0002, Objective 3:</u>
Demonstrate proper communications, in accordance with Conduct of Operations Manual and Human Performance procedural requirements, including:

a. Clear and concise communications
b. Three-way communications
c. Use of phonetic alphabet
d. Specifying correct component/unit/train
e. Creating understanding
f. verifying understanding (clarification/confirmation)

Cognitive Level: Memory or Fundamental

Question Type: New

References: HU-AA-101

72. Given the following:

- The reactor is in COLD SHUTDOWN and pre-startup evolutions are in progress
- The 'B' recirc pump is being placed in service and is aligned as follows:
  - The MG set drive motor breaker is shut
  - The scoop tube is positioned at 100%
  - The WARM light has just illuminated

Which one of the following describes what happens when the STRT/NORM pushbutton is depressed?

a. The field breaker will close immediately and the scoop tube will remain at 100%.
b. The field breaker will close immediately and the scoop tube will start running back
c. The scoop tube will start running back and the field breaker will close when the scoop tube reaches the low speed position
d. The scoop tube will start running back and the field breaker will close when the scoop tube passes through the 40% to 30% range

Answer: d

Handouts: None

Justification: A, B and C are incorrect – the scoop tube will run back and the field breaker will not close until the scoop tube reaches the 40-30% range.

D is correct – as soon as the STRT/NORM pushbutton is depressed the scoop tube begins to run back. When it reaches the 40-30% position, the field breaker will close and the recirc pump will start. The scoop tube will continue to run back to the low speed position.

G2.2.1 Equipment Control: Ability to perform pre-startup procedures for the facility / including operating those controls associated with plant equipment that could affect reactivity. (CFR 41.7)

OC Learning Objective:
<u>2621.828.0.0040, Objective G:</u>
Explain the starting logic for the reactor recirc MG sets

Cognitive Level: Memory or Fundamental

Question Type: Bank

References: 301.2

73. Per RP-AA-203, Exposure Control and Authorization, occupational workers at Oyster Creek have an Administrative Dose Control Level (ADCL) of ___(1)___ mrem TEDE per year.

This limit can be raised to ___(2)___ mrem TEDE with written approval by the Radiation Protection Manager and the work group supervisor.

a. (1) 1000
(2) 2000

b. (1) 2000
(2) 3000

c. (1) 1000
(2) 4000

d. (1) 2000
(2) 4000

Answer: b

Handouts: None

Justification: A is incorrect – the initial ADCL limit is 2000 mrem TEDE.

B is correct – RP-AA-203 states: "Administrative dose control levels have been established for Total Effective Dose Equivalent Limits of 2000 mrem routine cumulative TEDE/yr." And, "To raise the ADCL to 3000 mrem TEDE in a calendar year, written approval is required by the Radiation Protection Manager and the work group supervisor."

C and D are incorrect – the initial ADCL limit is 2000 mrem TEDE; this can be raised to 4000 mrem TEDE with "written approval from the Radiation Protection Manger, a work group supervisor, and the Station/Plant Manager." The RPM and work group supervisor can only authorize an extension to 3000 mrem TEDE.

G2.3.4 Radiation Control: Knowledge of radiation exposure limits and contamination control / including permissible levels in excess of those authorized. (CFR 41.12)

OC Learning Objective:
<u>Exelon GET RWT Module (Rev. 31), Objective 17:</u>
State the Exelon Administrative Dose Control Level for TEDE and the administrative does guidelines for LDE, SDE, and TODE.
<u>Exelon GET RWT Module (Rev. 31), Objective 18:</u>

State the actions to be taken if an Exelon Administrative Dose Control Level or administrative guidelines are being approached.

Cognitive Level: Memory or Fundamental

Question Type: Modified Bank

References: RP-AA-203

74. The primary containment is being purged with air in preparation for a plant shutdown.

Why does procedure 312.9, Primary Containment Control, prohibit the simultaneous opening of Drywell and Torus vent and purge valves?

To prevent…

a. over-pressurizing the Reactor Building ventilation ducts
b. exceeding the Reactor Building ventilation exhaust fan capacity
c. loss of the positive differential pressure between the Drywell and Torus
d. creating a pathway for steam to bypass the suppression pool during a LOCA

Answer: d

Handouts: None

Justification: A is incorrect – there is no reason to believe this would occur and this is not the reason for the limitation given in 312.9.

B is incorrect – there is no reason to believe this would occur and this is not the reason for the limitation given in 312.9.

C is incorrect – this is not the reason for the limitation given in 312.9, but could be a misconception based on the requirement in 312.9 to monitor Drywell and Torus pressure during purging to maintain a positive d/p.

D is correct – as stated in P&L 7.2.6 of 312.9, "When primary containment is required, simultaneous opening of drywell and torus vent and purge valves is prohibited. Operating with both the drywell and torus valves open creates a pathway to bypass the torus-to-drywell vacuum breakers." Procedure 312.9 also references LER 97-014, which states: *"On October 21, 1997, while reviewing industry events, it was discovered that a suppression pool bypass leak path existed during purging and venting of the primary containment. Two operating procedures allowed venting and purging of the torus and drywell simultaneously. If a LOCA were to occur during this time, a pathway for steam to bypass the suppression pool was created. Calculations indicated that the allowable bypass area was exceeded but, in the unlikely event of a LOCA, the peak torus and dry-well pressure would not increase."*

G2.3.9 Radiation Control: Knowledge of the process for performing a containment purge. (CFR 41.10)

OC Learning Objective:
2621.828.0.0065, Objective F:
Given normal operating procedures and documents for the system, describe or interpret the procedural steps.

Cognitive Level: Memory or Fundamental

Question Type: Modified Bank

References: 312.9, LER 97-014

75. The reactor is operating at rated power.

Which of the following annunciators correspond to an EOP entry condition? (Assume the alarms are received individually and are valid.)

a. RX PRESS HI
b. RX LVL HI/LO
c. DW PRESS HI/LO
d. ISOL COND AREA TEMP HI

Answer: d

Handouts: None

Justification: A is incorrect – this annunciator alarms when reactor pressure reaches 1030 psig; the EOP entry condition for high reactor pressure is 1045 psig.

B is incorrect – this annunciator alarms at 146” TAF and the EOP entry on RPV water level is below 138”. answer b is incorrect.

C is incorrect – this annunciator alarms at a low DW pressure of 1.0 psig or a high DW pressure of 1.4 psig. The EOP entry condition for DW pressure is 3 psig.

D is correct – this annunciator alarms at 160 °F and requires entry into EMG-3200.11, Secondary Containment Control.

G2.4.2 Emergency Procedures/Plan: Knowledge of system set points / interlocks and automatic actions associated with EOP entry conditions. (CFR 41.10)

OC Learning Objective:
<u>2621.828.0.0023, Objective K:</u>
State the function and interpretation of system alarms, alone and in combination, as applicable in accordance with system RAPS.

Cognitive Level: Memory or Fundamental

Question Type: New

References: RAP-H3f, RAP-G7c, RAP-C5e, RAP-C8b

8/15/06 NRC Comments
They thought that the original answer b (DW TEMP HI) could be an EOP entry condition. It has been changed with that shown.