Draft Submittal

(Pink Paper)

Senior Reactor Operator Written Exam

U.S. Nuclear Regulatory Commission

Site-Specific SRO Written Examination

Applicant Information

Name:	
Date:	Facility/Unit:
Region: I ☐ II ☐ III ☐ IV ☐	Reactor Type: W ☐ CE ☐ BW ☐ GE ☐
Start Time:	Finish Time:

Instructions

Use the answer sheets provided to document your answers. Staple this cover sheet on top of the answer sheets. To pass the examination you must achieve a final grade of at least 80.00 percent overall, with 70.00 percent or better on the SRO-only items if given in conjunction with the RO exam; SRO-only exams given alone require a final grade of 80.00 percent to pass. You have 8 hours to complete the combined examination, and 3 hours if you are only taking the SRO portion.

Applicant Certification

All work done on this examination is my own. I have neither given nor received aid.

_____
Applicant's Signature

Results

RO/SRO-Only/Total Examination Values	_____ / _____ / _____	Points
Applicant's Scores	_____ / _____ / _____	Points
Applicant's Grade	_____ / _____ / _____	Percent

rec'd 9/22/08

1. Unit Two is at rated power with the following plant conditions:

   All control rods are OPERABLE.
   Rod select power is OFF.
   Control rod 10-27 scrams.
   Rod Drift alarm is received.
   20 seconds later control rod 38-11 also scrams.

   Which one of the following describes the impact on RMCS and the appropriate actions per 0AOP-02, Control Rod Malfunction / Misposition?

   A. Rod Out Block Annunciator;
      Insert Manual Scram

   B. Rod Out Block Annunciator;
      Reduce Core Flow to 65 Mlbs/hr

   C. NO Rod Out Block Annunciator;
      Insert Manual Scram

   D. NO Rod Out Block Annunciator;
      Reduce Core Flow to 65 Mlbs/hr

REFERENCE:
APP A-5 (2-2) Rod Out Block, (3-2) Rod Drift and (5-2) Rod Block RWM/RMCS Trouble
AOP-2.0 Control Rod Malfunction/Misposition

EXPLANATION:
A Rod Drift alarm is generated if an odd numbered reed switch is picked up with no "rod selected and driving" signal present. An inadvertant rod scram will cause a rod drift alarm. Below the LPAP, a rod drift/scram can cause a rod insert/withdraw from the RWM. This error will cause a Rod Block RWM alarm on A-5 (5-2) The given plant conditions are above the LPAP. No Rod Out Block alarm or Rod Block RWM alarm will be received. Per the direction of AOP-2.0, supplementary action 3.2.2, "IF greater than 25% RTP and the sum of scrammed and inoperable control rods is no more than eight, then REDUCE core flow to 65 mlbs/hr.

CHOICE "A" - Incorrect. No Rod Out Block alarm will be received. Manual Scram is an incorrect action for these conditions. If reactor power were below the LPAP, a Rod Block RWM alarm be received. If two rods had been drifting, a Scram would be appropriate per AOP-2.0.

CHOICE "B" - Incorrect. No Rod Out Block alarm will be received.

CHOICE "C" - Incorrect. Manual Scram is an incorrect action for these conditions.

CHOICE "D" - Correct Answer

201002 RMCS

A2. Ability to (a) predict the impacts of the following on the REACTOR MANUAL CONTROL SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal
conditions or operations: (CFR: 41.5 / 45.6)

A2.02 Rod drift alarm . . . . 3.2 / 3.3

SOURCE: New

LESSON PLAN/OBJECTIVE:
CLS-LP-07, Obj. 11b - describe the possible causes and required operator actions for the following alarms: A-5 3-2, Control Rod Drift.

COG LEVEL: High

2. A large break LOCA occurs on Unit Two with 2C RHR pump under clearance.

Plant conditions are as follows:

Reactor Pressure	55 psig
Reactor water level	0 inches and rising
Drywell Pressure	17.6 psig
Drywell Temperature	246° F
Torus Pressure	15.9 psig
Torus Temperature	135° F
Torus Level	-3.5 feet
Core Spray Loop B	Injecting at rated flow
Core Spray Loop A	Injecting at rated flow
RHR Loop B	Injecting at rated flow
RHR Loop A	Flow is oscillating
RHR Pump 2A Overload	In alarm
2A RHR Pump Amps	Fluctuating

Considering current plant conditions, which one of the following is a possible cause for these RHR pump indications and what actions are correct per plant procedures?

Low NPSH due to_____

A. clogging suction strainers;
Continue running 2A RHR Pump irrespective of NPSH limitations

B. clogging suction strainers;
Secure 2A RHR Pump and verify reactor water level still rising

C. high Torus Temperature / low torus level combination:
Continue running 2A RHR Pump irrespective of NPSH limitations

D. high Torus Temperature / low torus level combination:
Secure 2A RHR Pump and verify reactor water level still rising

REFERENCE:
SD-17 Residual Heat Removal System
Reactor Vessel Control Procedure
APP A-01 (4-8) RHR Pump 2A Overload

EXPLANATION:
Low NPSH is caused by insufficient pump suction head. Elevated torus temperatures as well as suction strainer clogging are potential causes. The lowest torus temperature at which NPSH limits become a concern for RHR pumps at BNP is 160°F. Suction strainer clogging has occured at several nuclear plants. Although it is less likely since the suction strainer modifications, it is still a possibility. As the strainers clog, pump amps and flows will fluctuate. Pump overload alarms may be received.

For the given conditions, (overload alarm, level above TAF and rising, multiple injection sources) the appropriate action per RVCP and the APP would be to secure the RHR pump and verify level still rising)

CHOICE "A" - Incorrect. With reactor water level above TAF and rising injection flow is not required irrespective of NPSH limitations

CHOICE "B" - Correct Answer

CHOICE "C" - Incorrect. With current torus temperature, NPSH limits are not a concern

CHOICE "D" - Incorrect. With current torus temperature, NPSH limits are not a concern

203000 RHR/LPCI: Injection Mode

A2. Ability to (a) predict the impacts of the following on the RHR/LPCI: INJECTION MODE (PLANT SPECIFIC) ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5 / 45.6)

A2.01 Inadequate net positive suction head . . . . . 3.2 / 3.4

SOURCE: New

LESSON PLAN/OBJECTIVE:
CLS-LP-18, Obj. 20. Given plant conditions, determine if indications of a clogged suction strainer exist.

COG LEVEL: High

3. Unit Two is operating at rated power.

While performing 0PT-07.2.4A, Core Spray Loop A Operability, Core Spray Room Cooler A fails to start when Core Spray Pump A is started.

The Reactor Building AO reports that the Room Cooler breaker has tripped on thermal overload.

Which one of the following identifies the action that is required by the SCO in response to the tripped Core Spray Room Cooler A breaker?

The SCO should:

A. ✓ immediately declare Core Spray Subsystem A inoperable.

B. contact Engineering to perform an operability determination.

C. direct the AO to attempt one reset of the tripped breaker and continue the test.

D. ensure that Core Spray Room Cooler B is functioning properly and continue the test.

REFERENCE:
0OI-01.08 Control of Equipment and System Status, section 5.1.2.4 ECCS Rm Clrs
AP-13 Plant Equipment Control

EXPLANATION:
Per the direction of OI-01.08, when any room cooler is determined to be inoperable, then the ECCS equipment associated with that room cooler must be declared INOP per the applicable TS.

CHOICE "A" - Correct Answer

CHOICE "B" - Incorrect. 0OI-01.08 already clarifies OPERABILITY determination.
If student is unaware of OI-01.08 guidance, they may choose this answer.

CHOICE "C" - Incorrect. Per AP-13 a tripped breaker should not be reset until an investigation has been performed, except in case of an emergency.
If this were an emergency condition, this an could be correct.

CHOICE "D" - Incorrect. Unlike RHR Room Coolers, CS room coolers are not redundant. If the question pertained to the RHR system, this answer may be correct.

209001 Low Pressure Core Spray

2.2.22 Knowledge of limiting conditions for operations and safety limits.
(CFR: 41.5 / 43.2 / 45.2)

IMPORTANCE RO 4.0 SRO 4.7

SOURCE: Bank - LOI-CLS-LP-018-A*017

LESSON PLAN/OBJECTIVE:
CLS-LP-18, Obj. 18. Given plant conditions and TS, including bases, TRM, ODCM, and COLR, determine the required actions to be taken in accordance the TS associated with the Core Spray System. (SRO/STA only)

COG LEVEL: High

4. Unit Two is operating at 23% rated power.

Grid instabilities result in the following plant conditions:

Load Reject Signal
Only one transmission line (Whiteville Line) feeding the 230 kV system

Which one of the following describes the impact these conditions will have on plant operation and the required procedural direction to mitigate these impacts?

A. Turbine Control Valve Fast Closure scram will occur;
Trip the Whiteville Line PCB's.

B. Turbine Trip/Turbine Stop Valve Closure scram will occur;
Trip the Whiteville Line PCB's.

C. Turbine Control Valve Fast Closure scram will occur;
Place the auto reclosure switches for the Whiteville Line PCB's in OFF.

D. Turbine Trip/Turbine Stop Valve Closure scram will occur;
Place the auto reclosure switches for the Whiteville Line PCB's in OFF.

REFERENCE:
SD-03 Reactor Protection System, section 3.1 RPS Trips
AOP-22 Grid Instability, step 3.2.4

EXPLANATION:
A load reject signal at any reactor power level will cause a turbine control valve fast closure scram. The load reject signal does not input into the turbine stop valve closure scram logic. During a grid instability event, with only one 230 KV line feeding the system, a supplementary action of AOP-22 is to ensure that lines PCB auto recloser is OFF.

CHOICE "A" - Incorrect
Tripping the Whiteville PCB is not an action required for these conditions.

CHOICE "B" - Incorrect
Load reject initiates a TCV fast closure scram only. A misconception of the difference between TCV and TSV scrams may cause a student to select this answer.

CHOICE "C" - Correct Answer

CHOICE "D" - Incorrect. Load reject initiates a TCV fast closure scram only.
A misconception of the difference between TCV and TSV scrams may cause a student to select this answer.

212000 RPS

A2. 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: (CFR: 41.5 / 45.6)

A2.15 Load rejection . . . . . 3.7 / 3.8

SOURCE: New

LESSON PLAN/OBJECTIVE:
CLS-LP-03, Obj. 8. List the RPS trip signals, including setpoints and how/when each signal is bypassed.

COG LEVEL: High

5. A LOCA in primary containment has caused a reactor scram.

   12 control rods are stuck at position 02
   Torus level is -28 inches

   As the reactor depressurizes, reference leg flashing occurs.

   Which one of the following identifies the response of level instrumentation and the required EOP actions if reactor water level indication cannot be determined?

   Level indication will _____;
   Enter Reactor Flood Procedure and _____.

   A. fail downscale only;
      terminate and prevent injection to the reactor and open 7 ADS valves

   B. fail downscale only;
      open 7 ADS valves (Do not terminate and prevent injection to the reactor)

   C. be erratic, cycling between upscale and downscale indication;
      terminate and prevent injection to the reactor and then open 7 ADS valves

   D. be erratic, cycling between upscale and downscale indication;
      open 7 ADS valves (Do not terminate and prevent injection to the reactor)

REFERENCE:
SD-01.2 Reactor Vessel Instrumentation, section 4.2.1

EXPLANATION:
Instrument leg flashing causes pressure transients within the lines which can cause indications to fluctuate widely from high to low. If reactor water level indication can not be determined, the Reactor Flood Procedure is entered. The actions within the RFP are determined in part by the position of the control rods. With the conditions given in the stem, the RFP requires a termination and prevention of injection prior to opening ADS valves.

CHOICE "A" - Incorrect. There are malfunctions that can occur to an instrument reference leg that will cause the instrument indication to fail downscale. (plausible)

CHOICE "B" - Incorrect. There are malfunctions that can occur to an instrument reference leg that will cause the instrument indication to fail downscale. (plausible)

CHOICE "C" - Correct Answer

CHOICE "D" - Incorrect. Would be correct if all control rods were fully inserted.

216000 Nuclear Boiler Instrumentation

A2. Ability to (a) predict the impacts of the following on the NUCLEAR BOILER INSTRUMENTATION ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal
conditions or operations: (CFR: 41.5 / 45.6)

A2.07 Reference leg flashing . . . . . 3.4 / 3.5

SOURCE: New

LESSON PLAN/OBJECTIVE:
CLS-LP-1.2, Obj. 5d. Explain the effect that the following will have on reactor vessel level and/or pressure indications: reference/variable leg flashing.

COG LEVEL: High

6. An inadvertant Group I Isolation and reactor scram have occured on Unit Two. The Group I Isolation signal is sealed in and cannot be reset.

   RCIC is injecting to maintain reactor water level
   HPCI is in the pressure control mode

   Plant conditions are as follows:

Reactor Water Level	180 inches
Reactor pressure	900 psig

   The feeder breaker from DC Bus 2B to MCC 2-XDB trips on overcurrent.

   Which one of the following identifies the effect this loss of power will have on plant operation and the operator action(s) to mitigate these effects?

   A. HPCI is not available;
      transition pressure control to SRV's per RVCP

   B. ✓ RCIC is not available;
      transition level control to CRD per RVCP

   C. HPCI is not available;
      transition pressure control to Main Steam Line drains per RVCP

   D. RCIC is not available;
      transition level control to condensate per RVCP

REFERENCE:
SD-16 Reactor Core Isolation Cooling, section 3.0
RVCP

EXPLANATION:
The primary power source for the RCIC system is Div. II 125/250 VDC.
A loss of 2-XDB causes a loss of power to the majority of the RCIC system valves.
With a loss of RCIC as a level control source, RVCP will direct the use of other available systems. With reactor pressure at 900 psig, CRD is an available makeup source. Reactor pressure is too high to utilize condensate.

CHOICE "A" - Incorrect. A loss of Div. I DC would cause a loss of HPCI

CHOICE "B" - Correct Answer

CHOICE "C" - Incorrect. A loss of Div. I DC would cause a loss of HPCI

CHOICE "D" - Incorrect. Condensate not available for injection at this reactor pressure

217000 RCIC

A2. Ability to (a) predict the impacts of the following on the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5 / 45.6)

A2.05 D.C. power loss . . . . . 3.3 / 3.3

SOURCE: New

LESSON PLAN/OBJECTIVE:
CLS-LP-16, Obj. 8. Identify the power supply (bus and voltage) for the following RCIC components: Valves, Logic, flow controller, Vacuum pump, and condensate pump.

COG LEVEL: High

7. Severe weather has caused a complete loss of off-site power.

All Emergency Diesel Generators have failed to start, as required.

Trouble shooting of the failure to start has been underway for 20 minutes.

(reference provided)

Which one of the following describes the appropriate EAL to declare?

A. Unusual Event.

B. Alert.

C. Site Area Emergency.

D. No declaration is required.

REFERENCE:
PEP-2.1 Initial Emergency Actions, 6.0 Electrical and Power Failures

EXPLANATION:
The inability to power wither 4KV bus from off-site power AND loss of all on-site AC power capability indicated by failure of diesel generators to start or synchronize AND
lasting more than 15 minutes = Site Area Emergency

CHOICE "A" - Incorrect

CHOICE "B" - Incorrect

CHOICE "C" - Correct Answer

CHOICE "D" - Incorrect

264000 EDGs

2.4.41 Knowledge of the emergency action level thresholds and classifications.
(CFR: 41.10 / 43.5 / 45.11) |

IMPORTANCE RO 2.9 SRO 4.6

SOURCE: New

LESSON PLAN/OBJECTIVE:
CLS-LP-301, Obj. 4c. Given PEP-02.1, discuss/perform the following actions: Classify emergency events (SRO Only)

COG LEVEL: Low

8. I&C has requested removing annunciator card AOG System Disch Rad High from service for trouble shooting of the annunciator.

   The trouble shooting activity will take place early in the shift and last 2 hours.

   Which one of the following identifies the ODCM entry requirement and Annunciator Removal From Service Form completion requirement?

   A. ODCM Specification must be entered;
      Annunciator Removal From Service Form must be completed

   B. ODCM Specification must be entered;
      Annunciator Removal From Service Form is not required if approved by SCO

   C. ODCM Specification entry is not required provided the conditions identified in the Specification are met;
      Annunciator Removal From Service Form must be completed

   D. ODCM Specification entry is not required provided the conditions identified in the Specification are met;
      Annunciator Removal From Service Form is not required if approved by SCO

REFERENCE:
OI-01.08 Section 5.2.5 "Disabling Annunciators"

EXPLANATION:
Per OI-01.08, ODCM annunciators may be removed from service for up to 30 minutes without entering the associated spec. Also, if an annunciator is to be disabled for a period of time not to exceed shift turnover then the Removal from Service form can be waived.

CHOICE "A" - Incorrect, see explanation.

CHOICE "B" - Correct Answer

CHOICE "C" - Incorrect, see explanation.

CHOICE "D" - Incorrect, see explanation.

272000 Radiation Monitoring

2.2.14 Knowledge of the process for controlling equipment configuration or status. (CFR: 41.10 / 43.3 / 45.13)

IMPORTANCE RO 3.9 SRO 4.3

SOURCE: New

LESSON PLAN/OBJECTIVE:
CLS-LP-201-D, Obj. 10d. Explain the following regarding annunciator status per 0OI-1.08: disabling an annunciator.

COG LEVEL: High

9. Unit One is operating at rated power when the 1A Recirc pump trips.

Core support plate Delta-P is 4.5 psid
APRM's indicate 70% power.

(reference provided)

Based on these indications which one of the following would be the calculated total core flow and what action should be taken, if any, in accordance with 1AOP-4.0, Low Core Flow.?

A. 40.5 Mlb/hr
No action required.

B. 40.5 Mlb/hr
Core flow should be reduced to below 30.8 Mlb/hr

C✓ 37.5 Mlb/hr
Power should be reduced to less than 50% power.

D. 37.5 Mlb/hr
Core flow should be reduced to below 30.8 Mlb/hr.

REFERENCE:
1OP-02 Attachment 1 Rev. 74 / 1AOP-4.0 Rev. 21

EXPLANATION:
If WTCF is unavailable then the operator will have to use the graph to determine the total core flow. Due to recent events while in single loop power must be reduced to less than 50% to prevent instability scrams.

SRO Only - Assessing plant conditions and determining what action written into a plant procedure is required.

CHOICE "A"
could be chosen if the examinee started from the bottom with the percent power lines instead of from the top. Core flow should be maintained between 30.8 and 45 Mlbs/hr to prevent cooldown of the idle loop. Reactor power should be reduced to less than 50% power to prevent a scram from instabilities as seen in a recent scram.

CHOICE "B"
could be chosen if the examinee started from the bottom with the percent power lines instead of from the top. Core flow should be maintained between 30.8 and 45 Mlbs/hr to prevent cooldown of the idle loop. Operation not allowed below 30.8.

CHOICE "C"
Correct answer: using the 70% power line at 4.5 psid falls on the line between 35 and 40 which would be interpolated as 37.5. Core flow should be maintained between 30.8 and 45 Mlbs/hr to prevent cooldown of the idle loop. Reactor power should be reduced to less than 50% power to prevent a scram from instabilities as seen in a recent scram.

CHOICE "D"
using the 70% power line at 4.5 psid falls on the line between 35 and 40 which would be interpolated as 37.5. Core flow should be maintained between 30.8 and 45 Mlbs/hr to prevent cooldown of the idle loop. Operation not allowed below 30.8.

295001 Partial or Complete Loss of Forced Core Flow Circulation

AA2. Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION : (CFR: 41.10 / 43.5 / 45.13)

AA2.03 Actual core flow..... 3.3 / 3.3

SOURCE: New

LESSON PLAN/OBJECTIVE:
CLS-LP-302-C, Recirculation System Related AOPs.
Obj. 12. Describe the methods to determine core flow using core plate d/p.

COG LEVEL: Higher order.

10. Unit Two is operating at rated power when the following alarms are received:

DG-4 CTL Power Supply Lost
DG-4 Lo Start Air Press
DG4/E4 ESS Loss of Norm Power
DG-2 CTL Power Supply Lost

Which one of the following is the cause of these alarms and what action should be directed per 0AOP-39, Loss of DC Power?

There is a loss of 125V DC Distribution Panel:

A. 1B and direct I&C to confirm ESS Panels have transferred to its alternate control power.

B. 1B and direct I&C to confirm DG2 has auto transferred to its alternate control power.

C. 2B and direct I&C to confirm ESS Panels have transferred to its alternate control power.

D. 2B and direct I&C to confirm DG4 has auto transferred to its alternate control power.

REFERENCE:
0AOP-39

EXPLANATION:
Alarms on DG4 indicate that loss is from 2B. Alarm on DG2 is from alternate supply being lost. I/C assistance is needed to measure voltage on the ESS panels, light indication is lost due to normal power being lost.

CHOICE "A" The DC panel that is lost is not 1B.

CHOICE "B" The DC panel that is lost is not 1B and control power is transferred manually only.

CHOICE "C" Correct answer.

CHOICE "D" DG2 has lost its alternate control power and control power is transferred manually only.

295004 Partial or Complete Loss of D.C. Power

AA2. Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER : (CFR: 41.10 / 43.5 / 45.13)

AA2.01 Cause of partial or complete loss of D.C. power..... 3.2 / 3.6

SOURCE: new

LESSON PLAN/OBJECTIVE:
CLS-LP-302-G Obj. 4, Given plant conditions an any of the following AOP's, determine the required supplemental actions: 0AOP-39, Loss of DC Power.

COG LEVEL: Higher Order

11. Unit One is operating at 100% power with one control rod scram accumulator inoperable. The associated control rod scram time was within the limits of TS Table 3.1.4-1, Control Rod Scram Times, during the last scram time test.

Which one of the following describes the Tech Spec required action(s)?

The affected control rod must be declared:

A. slow only.

B. inoperable only.

C. slow or inoperable.

D. slow and inoperable.

REFERENCE:
TS 3.1.5

EXPLANATION:
Control rod scram accumulators shall be operable in Modes 1and 2.
One control rod scram accumulator inoperable with reactor steam dome pressure >950 psig the required action is to declare the associated control rod scram time slow (only applicable if it was within the limits of Table 3.1.4-1 during the last scram time surv.) or declare the associated control rod inoperable within 8 hours.

CHOICE "A" Incorrect, the control rod may be declared inoperable.

CHOICE "B" Incorrect the control rod may be declared slow.

CHOICE "C" Correct answer.

CHOICE "D" Incorrect, it may be one or the other but not both in accordance with the TS.

295006 SCRAM

2.2.37 Ability to determine operability and/or availability of safety related equipment.
(CFR: 41.7 / 43.5 / 45.12)

IMPORTANCE RO 3.6 SRO 4.6

SOURCE: New

LESSON PLAN/OBJECTIVE:
CLS-LP-08, Obj. 18. given plant conditions and TS, including the bases, TRM, ODCM, and COLR, determine the required actions to be taken in accordance with TS associated with CRD system. (SRO/STA Only)

COG LEVEL: Low/fund.

12. An ATWS with a Group I isolation occurred on Unit Two with the following plant conditions:

Reactor Power	APRM downscales
Control Rods	15 rods not full in
Current Reactor Pressure	1000 psig and lowering
Peak Reactor Pressure	1145 psig
Recirc Pumps	Running
Scoop tubes	Locked at 50% speed

Based on the above observations, which one of the following would be considered the status of the SRV's and what action(s) should be taken with respect to the recirc pumps?

A. Only 7 SRV's should have opened.
Recirc pumps should be tripped.

B. 8 SRV's should have opened.
Recirc pumps should be tripped.

C. Only 7 SRV's should have opened.
Scoop tubes unlocked and speed controllers set to 10%.

D. 8 SRV's should have opened.
Scoop tubes unlocked and speed controllers set to 10%.

REFERENCE:

EXPLANATION: SRVs are designed to lift at 1130, 1140 and 1150 psig. At 1130 4 SRVs open, at 1140 another 4 SRVs open and at 1150 the remaining 3 SRVs open. Based on the highest pressure reading of 1145 then 8 SRVs should have opened. ARI should have auto initiated because of reactor pressure being greater than 1137.8 psig, which would have tripped the pumps. Since the auto action has not occurred then it should be made to happen.

CHOICE "A" Incorrect, 7 SRVs is the number of ADS valves.

CHOICE "B" correct answer

CHOICE "C" Incorrect, 7 SRVs is the number of ADS valves. Speed controllers to 10% is an action from the scram hard card.

CHOICE "D" Incorrect, Speed controllers to 10% is an action from the scram hard card.

295007 High Reactor Pressure

2.1.7 Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.
(CFR: 41.5 / 43.5 / 45.12 / 45.13)

IMPORTANCE RO 4.4 SRO 4.7

SOURCE: New

LESSON PLAN/OBJECTIVE:
CLS-LP-20, Obj. 9. List the SRV pressure relief setpoints.

COG LEVEL: Higher Order

13. Given the following alarms on Unit Two:

Isophase Bus Cooling Wtr Flow-Low
Isophase Bus Fan Trip
Isophase Bus Return Air Temp - High
M-G Bearing & Oil Temp-Hi

Which one of the following is the cause of these alarms and what procedures should be entered?

A. Partial loss of TBCCW.
Enter 0AOP-17, TBCCW System Failure and 0AOP-19, CSW System Failure.

B. Complete loss of TBCCW.
Enter 0AOP-17, TBCCW System Failure and the Reactor Scram Procedure.

C. Partial loss of CSW.
Enter 0AOP-19, CSW System Failure and 0AOP-17, TBCCW System Failure.

D. Complete loss of CSW.
Enter 0AOP-19, CSW System Failure and the Reactor Scram Procedure.

REFERENCE:
0AOP-17

EXPLANATION:
The Isophase air temp and MG oil temp could be indicative of either CSW or TCC failure.
The cooling water flow low is from a loss of TCC to the Isophase Bus Duct Cooler which causes a fan trip.
If it was a partial loss then the water flow low alarm would not be in.
The AOP for TCC should be entered which for a complete loss tells you to insert a scram and perform RSP concurrently.

CHOICE "A" With the water flow low alarm it indicates that it is a complete loss. would not have to enter the AOP for a loss of CSW.

CHOICE "B" correct answer.

CHOICE "C" With the fan trip indicates a loss of TCC not CSW.

CHOICE "D" With the fan trip indicates a loss of TCC not CSW.

295018 Partial or Complete Loss of Component Cooling Water

AA2. Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER : (CFR: 41.10 / 43.5 / 45.13)

AA2.03 Cause for partial or complete loss..... 3.2 / 3.5

SOURCE: new

LESSON PLAN/OBJECTIVE:
CLS-LP-302H, Obj. 1a. Given plant conditions, determine if the following AOPs should be entered: 0AOP-17, TBCCW System Failures.

COG LEVEL: Higher Order

14. Unit One is at full power with the B CRD pump operating when all offsite power was lost. The following is the status of the Emergency Diesel Generators:

DG1	Locked out on fault
DG2	Running and loaded
DG3	Running and loaded
DG4	Running and loaded

Which one of the following is the status of the CRD system and what action should be taken in accordance with 0AOP-36.1, Loss of Any 4160V Buses or 480V E-Buses?

A. B CRD Pump is running with no power to the flow controller.
Swap flow controllers locally.

B. No CRD pumps are running with no power to the flow controller.
Start B CRD Pump and transfer 1AB to its alternate power supply.

C. B CRD Pump is running with a loss of power to its cooling water solenoid.
Shift 1AB-TB to alternate power supply.

D. No CRD pumps are running with a loss of power to the cooling water solenoids.
Cross-tie E5 and E6 to start B CRD pump, and shift 1AB to alternate power supply.

REFERENCE:
0AOP-36.1

EXPLANATION:
with a loss of all offsite power the E-Buses will strip the loads (CRD Pumps), there are no auto starts for these pumps, so both CRD pumps will be off. DG1 is lost which means E1 is lost and A CRD pump will not be able to be started. E5 to E6 would not be crosstied unless emergency conditions exist. The CRD flow controller is powered from 1AB which will need to be transferred to its alternate power supply. The cooling solenoid for the A CRD pump is powered from 1A. The actions in the AOP state to restart CRD per the OP and transfer 1AB-RX, 31AB, and 1AB to their alternate power supply.

CHOICE "A" B CRD is not running it would have been load stripped. There is no power to the controller and swapping them locally will not change that fact.

CHOICE "B" correct answer.

CHOICE "C" B CRD is not running it would have been load stripped. If 1AB-TB is transferred then it will not have any power, its normal power is from E6 and alternate power is from E5.

CHOICE "D" Crosstieing E5 and E6 will not get any power to the pump but will get power to the cooling water solenoids. Since an emergency condition does not exist then this action should not be taken. If E5 to E6 crosstie is performed then there is no reason to swap 1AB to its alternate power supply.

295022 Loss of Control Rod Drive Pumps

AA2. Ability to determine and/or interpret the following as they apply to LOSS OF CRD PUMPS : (CFR: 41.10 / 43.5 / 45.13)

AA2.02 CRD system status..... 3.3 3.4

SOURCE: new

LESSON PLAN/OBJECTIVE:
CLS-LP-302G, Obj. 4c. given plant conditions and any of the following AOP's, determine the required supplementary actions: AOP-36.1.

COG LEVEL: Higher Order

15. Unit Two is in a refueling outage when a fuel bundle is dropped in the spent fuel pool and the following alarms are received:

    Area Rad Refuel Floor High
    Process Rx Bldg Vent Rad Hi
    Rx Bldg Roof Vent Rad High

    0AOP-5.0, Radioactive Spills, High Radiation, and Airborne Activity, is entered.

    Which one of the following is the appropriate course of action?

    A. Continue in 0AOP-5.0.
       Secure and Isolate Reactor Building Ventilation.

    B. Enter Radioactivity Release Control Procedure.
       Verify Secondary Containment Isolation has occurred.

    C. Exit 0AOP-5.0 and enter Radioactivity Release Control Procedure.
       Verify CREV automatically initiated.

    D. Enter Radioactivity Release Control Procedure and perform 0AOP-5.0 concurrently.
       Calculate Site Boundary Dose per 0PEP-3.4.7.

REFERENCE:
0AOP-5.0 / EOP-RRCP

EXPLANATION:
All three of these alarms are symptoms for the AOP and the last one is an entry condition for the EOP. Unlike 0AOP-14 when an entry condition exists for the EOP you do not exit the AOP, instead it is completed concurrently with the EOP. If turbine building hi rad conditions exist or if an alert or higher on rad conditions exist then Once thru is placed in recirc (recent mod). conditions do not exist for SCI (SBGT start, Group VI, and RBV isolation). CREV should be manually started, no auto start signal exists. An action from the EOP is to do a 3.4.7 calculation.

CHOICE "A" AOP-5.0 should be executed, but also EOP-RRCP should be entered. Once thru is not placed in recirc unless TB rad is a problem or an ALERT condition on rad exists.

CHOICE "B" SCI signal does not exist. If RB Vent Hi Hi was in this would be a correct answer.

CHOICE "C" There is not an auto start signal for CREV under these condition. AOP would not be exited.

CHOICE "D" correct answer.

295023 Refueling Accidents

2.4.8 Knowledge of how abnormal operating procedures are used in conjunction with EOPs. (CFR: 41.10 / 43.5 / 45.13)

IMPORTANCE RO 3.8 SRO 4.5

SOURCE: New

LESSON PLAN/OBJECTIVE:
CLS-LP-302J, Obj. 1. Given plant conditions, determine if the AOP-5.0 should be entered.

COG LEVEL: Higher Order

16. An event on Unit One has resulted in the following plant conditions:

Reactor pressure:	1000 psig
Reactor Water Level	120 inches
Drywell pressure:	3 psig
Supp. Pool pressure:	2 psig
Supp. Pool water temp:	150° F
Supp. Pool water level:	-4 feet

(Reference Provided)

Based on the above conditions which one of the following is the required action?

A. Reduce reactor pressure as necessary to remain in the safe region of the heat capacity temperature limit.

B. Anticipate Emergency Depressurization and control injection from HPCI/RHR/CS/Condensate.

C. Perform Emergency Depressurization and control injection from HPCI/RHR/CS/Condensate.

D. Reduce Suppression Pool temperature as necessary to remain in the safe region of the heat capacity temperature limit.

REFERENCE:
Heat Capacity Temperature Graph, PCCP.

EXPLANATION:
Once the HCTL has been exceeded then ED is required. As the HCTL is approached then it is appropriate to lower pressure/torus temp to remain in the safe region.

CHOICE "A" This would be the appropriate action as the HCTL is being approached, not after the limit has been exceeded.

CHOICE "B" This would be the appropriate action as the HCTL is being approached, not after the limit has been exceeded.

CHOICE "C" correct answer.

CHOICE "D" This would be the appropriate action as the HCTL is being approached, not after the limit has been exceeded.

295026 Suppression Pool High Water Temperature

EA2. Ability to determine and/or interpret the following as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE: (CFR: 41.10 / 43.5 / 45.13)

EA2.03 Reactor pressure..... 3.9 / 4.0

SOURCE: new

LESSON PLAN/OBJECTIVE:
CLS-LP-300L, Obj. 5a, Given the PCCP, determine the appropriate actions if any of the following limits are approached or exceeded: Heat Capacity Temperature Limit.

COG LEVEL: higher order

17. 0MST-PCIS21Q, PCIS Rx Water LL2 and LL3 Div I Trip Unit Chan Cal and Func Test, was performed and the following is the as left data:

Instrument	Calibration Current
B21-LT-N024A-1-1	11.59 mAdc
B21-LT-N024A-1-2	7.40 mAdc
B21-LT-N025A-1-1	11.64 mAdc
B21-LT-N025A-1-2	7.43 mAdc

(Reference provided)

Based on the above information which one of the following is the status of the Div I trip system and the required action?

A. LL2 function is inoperable.
LL3 function is operable.
Place LL2 in a trip condition within 12 hours

B. LL2 function is operable.
LL3 function is inoperable.
Place LL3 in a trip condition within 24 hours.

C. LL2 function is inoperable.
LL3 function is operable.
Restore isolation capability within one hour.

D. LL2 function is operable.
LL3 function is inoperable.
Isolate the affected penetration flowpath within one hour.

REFERENCE:
Given the acceptance criteria of 0MST-PCIS21Q pages 5/6
Given 0OI-18 page 13
TS 3.3.6.1

EXPLANATION:
From the acceptance criteria, LL2 must have a current reading of greater than 11.7 mAdc and LL3 must have a current reading of greater than 4.99 mAdc.
Tech spec - LL2 function is outside of its allowable isolation setpoint so it is not operable. From the bases isolation functions are considered to be maintaining isolation capability when sufficient channels are operable or in trip, such that one trip system will generate a trip signal from the given function on a valid signal. For functions 1a. (LL3) this would require both trip systems to have a total of three channels. For function 5g. (LL2) this would require one trip system to have two channels, each operable or in trip.
From OI-18 A1 and A2 are the affected instruments.
The LL3 trip logic is A1 or A2 and B1 or B2. (which still would work)
The LL2 trip logic is A1 and B1 for half and A2 and B2 for the other half of the isolation.
Based on this the LL2 function is inoperable and unable to provide isolation capability on a valid signal.

CHOICE "A"

CHOICE "B"

CHOICE "C"

CHOICE "D"
295031 Reactor Low Water Level

2.2.39 Knowledge of less than or equal to one hour Technical Specification action statements for systems. (CFR: 41.7 / 41.10 / 43.2 / 45.13)

IMPORTANCE RO 3.9 SRO 4.5

SOURCE: new

LESSON PLAN/OBJECTIVE:
CLS-LP-1.2, Obj. 13. Given plant conditions and TS, including the bases, TRM, ODCM, and COLR, determine the required actions to be taken in accordance with TS associated with Reactor Vessel Instrumentation system. (SRO/STA Only)

COG LEVEL: High

18. Unit Two has an unisolable high energy line break with all rods in and the following annunciators in alarm:

South CS Rm Flood Level Hi
South CS Rm Flood Level Hi-Hi
South RHR Rm Flood Level Hi

Which one of the following is the probable cause of the alarms and what action should be taken in accordance with the Secondary Containment Control Procedure?

A. Pipe break in the HPCI Turbine Steam Supply Line.
Consider Anticipation of Emergency Depressurization.

B. Pipe break in the HPCI Turbine Steam Supply Line.
Perform Emergency Depressurization of the Reactor.

C. Pipe break in the RWCU System.
Consider Anticipation of Emergency Depressurization.

D. Pipe break in the RWCU System.
Perform Emergency Depressurization of the Reactor.

REFERENCE:
System knowledge/location
0EOP-01-SCCP

EXPLANATION:
First have to determine that the leak has to be from the RWCU based on knowledge of system flowpath and location of components. Then based on having two areas at max norm and one area at max safe the operator should consider anticipation of ED. If more than one area is exceeding max safe then ED is required.

CHOICE "A" Incorrect. Pipe break would be in the HPCI room which has submarine doors to maintain the leak within that room.

CHOICE "B" Incorrect. Pipe break would be in the HPCI room which has submarine doors to maintain the leak within that room and ED would not be required until two areas above max safe.

CHOICE "C" Correct answer.

CHOICE "D" Incorrect. ED would not be required until two areas above max safe.

295036 Secondary Containment High Sump / Area Water Level

EA2. Ability to determine and/or interpret the following as they apply to SECONDARY CONTAINMENT HIGH SUMP/AREA WATER LEVEL : (CFR: 41.10 / 43.5 / 45.13)

EA2.03 Cause of the high water level..... 3.4 / 3.8

SOURCE: new

LESSON PLAN/OBJECTIVE:
CLS-LP-300M, Obj. 8b. Given plant conditions and the SCCP, determine if any of the following are required: Consider anticipation of emergency depressurization.

COG LEVEL: Higher Order

19. Which one of the following identifies the earliest point during a reactor startup that the requirement can be relaxed for two CO's to be in the Main Control Room for the unit involved in the startup per 0OI-01.02, Shift Routines and Operating Practices.

A. After rated reactor power is achieved

B. After rated reactor pressure is achieved

C. After the second Reactor Feed Pump is in service

D. After the Main Generator is synchronized to the grid

REFERENCE:
0OI-01.02 Shift Routines and Operating Practices, section 5.1.5

EXPLANATION:
OI-01.02 states that Two Control operators are required until the Main Generator is synchronized to the grid. All the other answer options are plant milestones for a reactor startup and plausible responses.

CHOICE "A" - Incorrect, see explanation.

CHOICE "B" - Incorrect, see explanation.

CHOICE "C" - Incorrect, see explanation.

CHOICE "D" - Correct Answer

2.1.4 Knowledge of individual licensed operator responsibilities related to shift staffing, such as medical requirements, "no-solo" operation, maintenance of active license status, 10CFR55, etc.
(CFR: 41.10 / 43.2)

IMPORTANCE RO 3.3 SRO 3.8

SOURCE: Bank LOI-CLS-LP-201-D*01C (1)

LESSON PLAN/OBJECTIVE:

COG LEVEL: Low

20. During an accident, the Reactor Flooding Procedure is being executed.

Plant conditions are as follows:

RPV Water Level	Unknown
Control Rods	One rod full out, all others full in
Supp Chamber Pressure	10 psig
SRVs	7 open
ECCS pumps	All available pumps injecting

Which one of the following identifies when the reactor can be detemined to have been flooded to the Top of Active Fuel?

When RPV pressure has been no less than:

A. 50 psig for the Minimum Core Flooding Interval

B. 60 psig for the Minimum Core Flooding Interval

C. 50 psig for the Maximum Core Uncovery Time Limit

D. 60 psig for the Maximum Core Uncovery Time Limit

REFERENCE:
Reactor Flooding Procedure (Step 60)

EXPLANATION:
Minimum reactor flooding pressure requires an RPV pressure of at least 50 psig above suppression chamber pressure for the minimum core flooding interval to assure the core is flooded to TAF.

CHOICE "A" - Incorrect. RPV pressure must be maintained 50 psig "above" suppression pool pressure.

CHOICE "B" - Correct Answer

CHOICE "C" - Incorrect. RPV pressure must be maintained 50 psig "above" suppression pool pressure. Maximum Core Uncovery Time Limit is utilized in the Reactor Flood Procedure but does not apply for these conditions.

CHOICE "D" - Incorrect. Maximum Core Uncovery Time Limit is utilized in the Reactor Flood Procedure but does not apply for these conditions.

2.1.7 Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation. (CFR: 41.5 / 43.5 / 45.12 / 45.13)

IMPORTANCE RO 4.4 SRO 4.7

SOURCE: Bank LOI-CLS-LP-300-F*12C (5)

LESSON PLAN/OBJECTIVE:
CLS-LP-300-F Objective 8

COG LEVEL: High

21. A fire in the control building fire area requires entry into 0PFP-013, General Fire Plan, and 0ASSD-01, Alternative Safe Shutdown Procedure.

Which one of the following operator actions is directed from 0ASSD-01 following the manual scram of each reactor?

A. Trip reactor recirculation pumps

B. Place MSIV control switches in close

C. Reduce reactor pressure to 700 psig

D. Place condensate booster pump mode selector switches to manual

REFERENCE:
0ASSD-01 Alternate Safe Shutdown Procedure, section 3.5.2

EXPLANATION:
All of the available responses are actions required for AOP-32 Plant Shutdown from Outside the Control Room, therefore plausible options.
Of these actions, the only one directed from the applicable section of ASSD-01 is to place the MSIV control switches to close.

CHOICE "A" - Incorrect, see explanation.

CHOICE "B" - Correct Answer

CHOICE "C" - Incorrect, see explanation.

CHOICE "D" - Incorrect, see explanation.

2.4.27 Knowledge of "fire in the plant" procedures.
(CFR: 41.10 / 43.5 / 45.13)

IMPORTANCE RO 3.4 SRO 3.9

SOURCE: New

LESSON PLAN/OBJECTIVE:
CLS-LP-304, Obj. 12. Given plant conditions with an ASSD fire and the ASSD procedures, determine the appropriate operator actions to be performed for the fire.

COG LEVEL: High

22. In accordance with 0AI-147, Systematic Approach to Troubleshooting, which one of the following identifies the trouble shooting activities that must be approved by the Plant General Manager?

A. all high risk activities ONLY

B. high risk activities performed during max/safe/gen periods of operation ONLY

C. medium and high risk activities performed during max/safe/gen periods of operation.

D. medium risk activities performed during max/safe/gen periods of operation and all high risk activities

REFERENCE:
0AI-147 "Systematic Response to Troubleshooting"

EXPLANATION:
Per AI-147, the Plant General Manager is required to approve troubleshooting activities classified as medium or high risk which are performed during max/safe/gen periods.
Each of the available choices present options that a student may conclude reasonable, therefore plausible.

CHOICE "A" - Incorrect, see explanation.

CHOICE "B" - Incorrect, see explanation.

CHOICE "C" - Correct Answer

CHOICE "D" - Incorrect, see explanation.

2.2.20 Knowledge of the process for managing troubleshooting activities.
(CFR: 41.10 / 43.5 / 45.13)

IMPORTANCE RO 2.6 SRO 3.8

SOURCE: New

LESSON PLAN/OBJECTIVE:

COG LEVEL: Low

23. During the performance of 1OP-30, Condenser Air Removal and Off-Gas Recombiner System, it is determined that a temporary procedure change is required due to an error in the procedure.

Per PRO NGGC-0204 Procedure Review and Approval, which one of the following describes how this temporary change is categorized and the required expiration date?

A. One-Time-Use-Only,
not to exceed 21 days from interim approval date

B. Permanent Revision to Follow,
not to exceed 21 days from interim approval date

C. One-Time-Use-Only,
not to exceed 4 months from interim approval date

D. Permanent Revision to Follow,
not to exceed 4 months from interim approval date

REFERENCE:
PRO-NGGC-0204 Procedure Review and Approval, section 9.3 TC Process

EXPLANATION:
Temporary changes can be classified as either "One Time Use" or "Permanent Revision to Follow". A revision to correct a mistake is a procedure is classified as "Permanent Revision to Follow". The required expiration date for a Brunswick TC is "not to exceed 4 months from interim approval". For a TC at Robinson, the time frame would be 21 days. Both time frames are specified in the common procedure.

CHOICE "A" - Incorrect, see explanation.

CHOICE "B" - Incorrect, see explanation.

CHOICE "C" - Incorrect, see explanation.

CHOICE "D" - Correct Answer

2.2.6 Knowledge of the process for making changes to procedures.
(CFR: 41.10 / 43.3 / 45.13)

IMPORTANCE RO 3.0 SRO 3.6

SOURCE: New

LESSON PLAN/OBJECTIVE:
CLS-LP-201C , Obj. 5b. State the definition of the following in accordance with PRO-NGGC-0204, as they apply to temporary changes: Permanent revision to follow.

COG LEVEL: High

24. An unisolable RWCU leak in secondary containment has resulted in the following reactor building radiation levels as reported by E&RC:

Time	50' Sample Station	20' Drywell Entrance
0800	2200 mrem/hr	1500 mrem/hr
0810	1800 mrem/hr	1800 mrem/hr
0820	1800 mrem/hr	2100 mrem/hr

(reference provided)

What action is required by the Secondary Containment Control Procedure?

A. Shutdown the reactor per GP-05.

B. Scram and cooldown <100° F/hr

C. Scram and cooldown >100° F/hr

D. Scram and emergency depressurize the reactor

REFERENCE:
Secondary Containment Control Procedure

EXPLANATION:
At 0800 MaxSafe operating value is exceeded for the 50' area.
At 0810 MaxSafe is no longer exceed for the 50' area.
at 0820 MaxSafe operating value is exceeded for the 20' area.
Two areas have now exceeded their max safe values, although not concurrently.
Since the parameter exceeded was radiation levels, SCCP bases allows resetting of the parameter if it goes back below the MaxSafe value. If this were not the case, SCCP would require an emergency depressurization. If the exceeded parameter had been temperature, a reset would not be allowed and an ED would be required.

CHOICE "A" - Incorrect. If the leak had been isolated, this answer would be correct.

CHOICE "B" - Correct Answer

CHOICE "C" - Incorrect. Cooldown rate is not to exceed 100F/hr.

CHOICE "D" - Incorrect, ED not required for radiation.

2.3.15 Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc. (CFR: 41.12 / 43.4 / 45.9)

IMPORTANCE RO 2.9 SRO 3.1

SOURCE: New

LESSON PLAN/OBJECTIVE:
CLS-LP-300M, Obj. 6e. Given plant conditions and the SCCP, determine if any of the following have been exceeded: Max safe/normal operating radiation levels.

COG LEVEL: High

25. Which one of the following statements is correct with respect to approval of a Radioactive Liquid Release Permit?

Release tanks shall be verified recirculated a minimum of _____ tank volume(s) and release approval is required by _____.

A. One;
Unit SCO only

B. One;
Unit SCO and Shift Superintendent

C. Two;
Unit SCO only

D. Two;
Unit SCO and Shift Superintendent

REFERENCE:
1OP-6.4 Discharging Radioactive Liquid Effluents the Discharge Canal, section 3.2
Attachment 4 Liquid Release Permit

EXPLANATION:
Per the precautions section of OP-6.4, all releases must be recirculated a minimum of 2 tank volumes prior to release. Also, all Release Permits require the approval of both the Unit SCO and the Shift Superintendent.

CHOICE "A" - Incorrect. SS approval also required; If student is unaware of requirement to recirc 2 tank volumes, 1 tank volume is plausible option.

CHOICE "B" - Incorrect. If student is unaware of requirement to recirc 2 tank volumes, 1 tank volume is plausible option.

CHOICE "C" Incorrect. SS approval also required;

CHOICE "D" - Correct Answer

2.3.6 Ability to approve release permits. (CFR: 41.13 / 43.4 / 45.10)

IMPORTANCE RO 2.0 SRO 3.8

SOURCE: New

LESSON PLAN/OBJECTIVE:
CLS-LP-6.3, Obj. 5. Given a level in one of the Radwaste Release Tanks, calculate the minimum time required for recirculation.

COG LEVEL: High