Question # 001

Unit 2 was operating at rated power when BOTH Recirc Pumps tripped.

Prior to the transient, RPV water level was ___(1)___ than INDICATED Fuel Zone RPV water level AND the difference will get ___(2)___ following the transient.

a. (1) lower (2) smaller

b. (1) lower (2) larger

c. (1) higher (2) smaller

d. (1) higher (2) larger

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: A; Explanation: Fuel Zone indications are affected by anything that causes flow through the monitored jet pump. Therefore recirc pump flow causes the Fuel Zone instruments to be inaccurate in the non- conservative direction (read higher than actual).

References:

Direct/New/Modified: **Bank**
Memory/Comprehension-Analysis: **C/A**
LEVEL (SRO/RO): **RO**
K/A: 295001 AA1.07, Partial or complete loss of forced core flow circulation: Nuclear Boiler Instrumentation System
Author: Licensee/Walton
Exam Date: October 30, 2013

Question # 002

Unit 2 Reactor was operating at rated power with a normal electrical lineup when a scram and main generator trip occurred. The control room operators observed the following:

- all Reactor Feedwater pumps trip;
- all Condensate/Condensate Booster pumps trip;

The operators determine the cause of this event is…

a. A lockout of Bus 24 ONLY

b. A failure of Transformer 2

c. A failure of Transformer 86

d. A failure of Transformer 21

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: C; Transformer 86 feeds Transformer 22 (RAT) which feeds Busses 22, 24 and 24-1. Bus 22 feeds RFP B, Bus 24 feeds C & D Condensate/Condensate Booster pumps.
A is incorrect since a lockout on Bus 22 would not explain a trip of the C & D C/CB pumps.
B is incorrect since a lockout on Bus 24 would not explain a trip of the B RFP.
D is incorrect since main generator output breakers opened due to the scram; not the cause of the scram.
References: DAN 902-6, F-5, Dresden High Voltage distribution DWG262LN003-012.

Direct/New/Modified: **New**
Memory/Comprehension-Analysis: **C/A**
Level (SRO/RO) **RO**
K/A: 295003 AA2.01; Ability to determine and/or interpret the following as they apply to PARTIAL or COMPLETE LOSS OF AC POWER; cause of partial or complete loss of AC power.
Author: Walton
Exam Date: October 30, 2013

Question # 003

The Unit 2 125 VDC and 250 VDC systems are aligned as follows:

- The 2A battery charger is supplying the Unit 2 125VDC busses.
- The Unit 2 250VDC battery charger is out of service.
- The 2/3 250VDC battery charger is being supplied from its Unit 2 power source and is supplying the Unit 2 250VDC busses.
- A fire causes Bus 29 to become de-energized.

What effect, if any, does this have on the Unit 2 125VDC and 250VDC system?

a. NEITHER the Unit 2 125VDC OR 250VDC systems would be affected.

b. BOTH the Unit 2 125VDC AND 250VDC systems would be powered from the battery ONLY.

c. The Unit 2 125VDC would be powered from the battery ONLY;
The Unit 2 250VDC system would be powered from the 2/3 battery charger.

d. The Unit 2 125VDC system would be powered from the 2A battery charger;
The Unit 2 250VDC system would be powered from the battery ONLY.

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: D; 2A 125 Vdc charger is supplied from MCC 28-2 (from Bus 28) and is NOT affected. The student must also know that the 2/3 250 Vdc charger is being powered from MCC 29-2 (Bus 29) and with a fire on Bus 29, MCC 29-2 becomes de-energized, causing the 250 Vdc system to lose all chargers (U2 charger is OOS), thus the system would be on the batteries ONLY.

Direct/New/Modified: **DIRECT** Bank Question 13241
Memory/Comprehension-Analysis: **C/A**
Level (SRO/RO) **RO**
K/A: 295004; Partial or complete loss of DC control power G2.1.32 Ability to explain and apply system limits and precautions.
Author: Licensee/Walton
Exam Date: October 30, 2013

Question # 004

Unit 3 was at 100% power. With a fault in the EHC System that resulted in the bypass valves failing to open, the #1 and #4 main turbine generator stop valves (SV) slowly go closed. What parameter FIRST causes the reactor scram and why?

As _____(1)_____, a reactor scram will occur as a result of _____(2)______.

	(1)	(2)
a.	reactor temperature rises	doppler coefficient of reactivity
b.	reactor pressure rises	void coefficient of reactivity
c.	#1 and #4 SV close	#1 & #4 SV open limit switches opening
d.	reactor vessel level lowers	2 of 4 low reactor vessel level pressure switches actuating

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: b: Closing a SV with TBV failing to open from full power causes pressure increase in reactor vessel. Reactor trip is based on void coefficient of reactivity.
A is incorrect due to reactor temperature increase being slower than void coefficient
C is incorrect since turbine stop valves 1 & 4 will not cause an RPS trip. Need 3 SV closure signals to cause RPS actuation signal
D is incorrect since reactor vessel lowering is slower than the reactor pressure rise.

References: DRE241LN001A (EHC Lesson Plan); Operational Physics May 2010, pg 16 of 83.
RPS Lesson plan, Pg 15 of 58.
Direct/New/Modified: **New**
Memory/Comprehension-Analysis: **C/A**
Level (SRO/RO): **RO**
K/A: 295005 AK01.01; Knowledge of the operational implications of the following concepts as they apply to MAIN TURBINE GENERATOR TRIP, Pressure effects on reactor power.
Author: walton
Exam Date: October 30, 2013

Question # 005

One minute after a reactor scram occurs, the predominate neutron population in the core originates from…

a. neutrons emitted from fission in fuel.

b. subcritical neutron multiplication in the fuel.

c. delayed neutron fraction from fission products.

d. prompt neutron fraction from fission products.

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: C is correct from the first seconds after a scram to several minutes after a scram.
A. is correct answer for at power operation.
B. correct for neutron levels in the source range several hours after the reactor scram.
D: Correct for neutrons created at the time of the scram.

References: Lesson Plan: Operational Physics, (Instructors Guide) May 2011, pgs 60 & 61 of 83. (See text below)
Direct/New/Modified: **Modified**
Memory/Comprehension-Analysis: **Memory**
Level (SRO/RO) **RO**
K/A: 295006 K2.06: Knowledge of Interrelations between SCRAM and the following: Reactor Power.
Author: Walton
Exam Date: October 30, 2013

A-B: The rapid insertion of a large amount of negative reactivity causes the prompt neutron population to decrease rapidly. Segment A-B of the curve depicts this prompt drop.
B-C: During this period, the neutron population is dominated by the appearance of delayed neutrons from shorter- and intermediate-lived delayed neutron precursors. These precursors, which were formed when the reactor was at 100% power, decay within a few minutes.
C-D: Once the shorter-lived precursors have effectively all decayed, neutron population is controlled by the appearance of delayed neutrons from the longest-lived precursors. From this point, power falls at a constant exponential rate of –1/3 DPM until neutron population is low enough for the effect of source neutrons to be seen and a subcritical equilibrium is reached.

Question # 006

When scramming the reactor per DSSP 100-CR, “Hot Shutdown Procedure – Control Room Evacuation,” the NSO ______(1)______ because this/these action(s) ______(2)______.

a. (1) rotates the RX MODE SWITCH to the SHUTDOWN position ONLY
   (2) bypasses the MSIV closure function on low RPV pressure

b. (1) depresses MANUAL SCRAM CH A AND CH B pushbuttons ONLY
   (2) retains the MSIV closure function on low RPV pressure

c. (1) rotates the RX MODE SWITCH to the SHUTDOWN position ONLY
   (2) allows an RPV high water level condition to exist to ensure adequate core cooling

d. (1) depresses MANUAL SCRAM CH A AND CH B pushbutton followed by a reset
   (2) prevent a high level condition in the scram discharge volume

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: B. Per DSSP 100-CR, this is the correct action and correct reason.
A & C are incorrect since the Mode Switch is required to remain in RUN, not SHUTDOWN!
D is incorrect since must not reset the scram after depressing CH A and CH B scram PB.
References: DSSP 0100-CR, “Control Room Evacuation.” DOA 0010-10, “Fire/Explosion.”
Direct/New/Modified: **Modified** from Bank: Q14008.
Memory/Comprehension-Analysis: **MEMORY**
Level (SRO/RO) **RO**
K/A: 295016, Knowledge of the reasons for the following responses as they apply to CONTROL ROOM ABANDONMENT: Reactor Trip.
Author: walton
Exam Date: October 30, 2013

Question # 007

Unit 3 was at full power operation when an equipment operator, misunderstanding an out of service, calls control room operators to inform them he has closed valve 3-3702 (Loop 1 RBCCW Isolation Valve) and is unable to reopen the valve.

RBCCW system loads affected by this error are ____(1)____ and the operators response is to ____(2)_____.

a. (1) shutdown cooling heat exchangers
   (2) ensure that the system is not in operation

b. (1) RWCU non-regenerative heat exchangers
   (2) isolate the RWCU system

c. (1) recirculation pumps seal and motor cooling
   (2) manually scram the reactor

d. (1) both units pumpback air compressors
   (2) shutdown all pumpback air compressors

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: C is correct for a degraded RBCCW system per DOA 3700-01.
A. would be correct if there was a slight delay in starting the standby RBCCW pump.
B is incorrect since lining up 2/3 RBCCW pump to Unit 3 would take longer than 1 minute.
D is incorrect since these loads are supplied by 3701 valve.

References: RBCCW Lesson Plan, Sections III.D & III.F & VI.A.
Direct/New/Modified: **New**
Memory/Comprehension-Analysis: **C/A**
Level (SRO/RO); **RO**
K/A: 295018, AA1.02: Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF CCW: System loads.
Author: Walton
Exam Date: October 30, 2013

Question # 008

With Unit 2 in Mode 1 operations, control room operators have entered into DOA 4700-01, "Instrument Air System Failure." Should the instrument air system continue to depressurize, what effect would this have on the Unit 2 HPCI inlet drain pot operations?

The U2 HPCI Steamline Inlet Drain Pot Drain valve (2-2301-28) would fail ____(1)_____ and the U2 HPCI inlet drain pot flow path would be directed to the ____(2)_______.

	(1)	(2)
a.	closed	main condenser
b.	closed	torus
c.	open	main condenser
d.	open	torus

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: D. Per DOA 4700-01, Table 1, 2-2301-28 fails open, both HPCI drain pots would be directed from the condenser to the torus.

References: DOA 4700-01, Instrument Air Abnormal Procedure, Table 1. PNID M-51

Direct/New/Modified: **New**
Memory/Comprehension-Analysis: **Memory**
Level (SRO/RO) **RO**
K/A: 295019 AA2.02: Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR: Status of Safety related instrument air system loads.
Author: Walton
Exam Date: October 30, 2013

Question # 009

Unit 2 is in MODE 3 with the following set of conditions:

- One turbine bypass valve is full open
- RPV water temperature at 316°F and slowly lowering
- RPV pressure is 245 psig, slowly lowering.
- 2B Recirc pump is running at minimum speed.
- 2A EHC Pump is out of service.

Then an overcurrent condition occurs on Bus 27.

For these conditions, what is the preferred method of heat removal from the RPV?

a. Open an additional turbine bypass valve fully.

b. Place Isolation Condenser System in service per DOP 1300-03.

c. Initiate HPCI System in pressure control mode per DOP 2300-03.

d. Alternate opening of Electromatic Relief Valve(s) at five minute intervals.

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: B is correct. Per DOP 1300-03, F.8, the preferred order of systems to be used for RPV pressure control are 1. Isolation Condenser, 2. HPCI in pressure control, 3 ADS System relief valves. TBV's are unavailable since the last EHC pump has lost power from Bus 27.
Reference: DOP 1300-03, Manual Operation of the Isolation Condenser, Step F.8.
Direct/New/Modified: **MODIFIED** (Q22591)
Memory/Comprehension-Analysis: **C/A**
Level (SRO/RO) **RO**
K/A: 295021 A1.04: Ability to Operate and/or monitor the following as they apply to LOSS OF SHUTDOWN COOLING: Alternate Heat Removal Methods.
Author: Walton
Exam Date: October 30, 2013

Question # 010

With the Unit in MODE 5 and the MODE Switch in the REFUEL position, what condition below produces an interlock to prevent inadvertent criticality during fuel movements via a control rod withdraw BLOCK?

a. fuel grapple loaded and over the core.

b. rod worth minimizer in OPERATE.

c. either rod block monitor downscale.

d. refuel bridge track load trip plate enabled.

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: A
B. is not true since the RWM is used to prevent rod select error (rod drop accident) not an inadvertent criticality.
C true for MODE Sw in RUN
D is not true. It provides indication to circuit of refuel bridge position, but is only an input to the correct answer A.
References: UFSAR Sect. 7.7.1.2.2 – Rod Blocks

Direct/New/Modified: **New**
Memory/Comprehension-Analysis: **Memory**
Level (SRO/RO) **RO**
K/A: 295023. AK01.03, Knowledge of the operational implications of the following concepts as they apply to REFUELING ACCIDENTS: Inadvertent criticality.
Author: walton
Exam Date: October 30, 2013

Question # 011

Unit 2 was operating at rated power when operators note an increase in drywell pressure. Drywell pressure is 1.21 psig and rising .3 psig each minute. An ECCS initiation signal, is required by Tech Specs to occur in no later than _____(1)_____. Drywell spray valves (2-1501-27 and 2-1501-28) _____(2)______.

	(1)	(2)
a.	2 minutes	will automatically open
b.	2 minutes	must be overridden to open
c.	3 minutes	will automatically open
d.	3 minutes	must be overridden to open

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: B 0.3 psig/min X 2 minutes = 0.6 psig. 0.6 psig + 1.21 psig = 1.81 psig (Technical Specification Limit). Upon initiation, drywell spray valves are interlocked closed and must be manually overridden open (using key lock) AND with DW pressure > 1.0 psig.
References: DAN 902-3, A-13, Drywell Pressure High. LPCI Instrumentation drawing. LPCI Lesson Plan.
Direct/New/Modified: **New**
Memory/Comprehension-Analysis: **C/A**
Level (SRO/RO) **RO**
K/A: Knowledge of the interrelationships between HIGH DRYWELL PRESSURE and RHR/LPCI.
Author: walton
Exam Date: October 30, 2013

Question # 012

Unit 2 was operating at 90% power on the 100% rod line when a transient occurred. The NSO observed the following:

- reactor steam dome pressure increased to 1215 psig
- reactor water level lowered to -20"
- RPS and EHC systems failed to respond.
- Annunciator 902-5, A-8, "ATWS Ch A/B TRIP" alarmed
- Annunciator 902-5, F7, "ATWS LVL/PRESS ABNORMAL" alarmed.

Assuming no operator actions are taken, how do the recirculation pumps respond?

a. Recirculation MG Set supply breakers trip after a 9-second time delay

b. Recirculation MG Set supply breakers trip (no time delay)

c. Recirculation MG Set field breakers trip after a 9-second time delay

d. Recirculation MG Set field breakers trip (no time delay)

ANSWER: D
ATWS-RPT trips the reactor recirculation motor-generator (RRMG) set field breakers; immediately on high pressure, or after a 9 second time-delay on low-low reactor water level."
"The 9-second time delay relays are for the Lo-Lo Level RPT. ATWS LOCA trip, trips the MG set Field Breaker (instead of the supply breaker)…"
Per referenced DAN: High Pressure actuates at 1211 psig, Rx low level actuates at -51 inches.

References: ATWS Lesson Plan, DAN 902-5, A-8 and F-7.

Direct/New/Modified: **New**.
Memory/Comprehension-Analysis: **C/A**
Level (SRO/RO) **RO**
K/A: 295025 EA1.07, Knowledge of the reasons for the following responses as they apply to HIGH REACTOR PRESSURE: ARI/RPT/ATWS.
Author: walton
Exam Date: October 30, 2013

Question # 013

Annunciator 902-4, B-18, “DIV 1 TORUS Local Water Temperature High” alarms. What indicator would the operators be REQUIRED to use per the DAN to verify this alarm?

1) TR 2-1641-9, Suppression Pool Bulk Temperature on Panel 902(3)-4.
2) TIRS 2-1640-200A, Torus Water Temperature Recorder, on Panel 902(3)-36.
3) TIRS 2-1640-200B, Torus Water Temperature Recorder, on Panel 902(3)-36.

Answers:

a. 2 & 3 ONLY

b. 1 & 2 ONLY

c. 1 & 3 ONLY

d. 1, 2, & 3

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: B; Since B-18 alarms at 197°F, annunciator has operator check front panel indicator and Div 1 indicator on back panel.
C. True for annunciator D-18, Div 2 Torus local temperature alarm. But no DIV 2 alarm was included in stem of question.
A. & D are false since can not read Div 1 alarm on Div 2 back panel indicator.
References: DAN 902-4, A-18, B-18, and D-18. DOS 1600-20, “Suppression Pool Temperature Monitoring.”

Direct/New/Modified: **New**
Memory/Comprehension-Analysis: **Memory**
Level (SRO/RO) **RO**
K/A: 295026 EA1.03 Ability to operate and/or monitor the following as they apply to SUPRESSION POOL HIGH WATER TEMPERATURE: Temperature Monitoring.
Author: Walton
Exam Date: October 30, 2013

Question # 014

Unit 3 was operating in Mode 1. A transient occurred and the following containment parameters were observed:

- Drywell pressure 12 psig rising.
- Drywell air temperature 240°F rising.
- Torus pressure 7 psig rising.
- Torus water temperature 105°F rising.

No operator action has been taken. Which of the following is indicated?

a. An Electromatic relief valve has failed open.

b. Containment is breached following a water break LOCA

c. Containment is functioning as designed following a high pressure discharge into the drywell.

d. Containment is functioning as designed following a bypass path discharge into the suppression chamber airspace.

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: C
A & B incorrect due to first two bullets
D incorrect due to Torus Pressure and Drywell pressure should be similar and there would be no rising torus water temperature.

References: Containment Lesson Plan, Sect VII.C.

Direct/New/Modified: **Direct**, From LaSalle NRC Exam 2003.
Memory/Comprehension-Analysis: **C/A**
Level (SRO/RO) **RO**
K/A: 295028, EA2.05, Ability to determine and/or interpret the following as they apply to HIGH DRYWELL TEMPERATURE: Torus/Suppression chamber pressure.
Author: pt young/walton
Exam Date: October 30, 2013

Question # 015

With Unit 3 in Mode 1 operations, the torus conditions are as follows:

- Torus water temperature is 92°F and rising slowly
- Torus narrow range water level is -5.0 inches and lowering slowly

Should a LOCA occur during these conditions, what is the FIRST concern operators would have for primary containment?

a. Incomplete steam condensation.

b. Insufficient scrubbing of iodine from steam discharged during a LOCA.

c. Condensate oscillation and chugging loads.

d. Excessive clearing loads from steam discharges and pool swell could result in damage to the torus and its supports.

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: A. Low torus water level could result in insufficient inadequate quenching of steam discharged from T-quenchers or LOCA, or lower volume would absorb less steam energy before heating up excessively.
B is incorrect, not discussed in lesson plan as a function of primary containment
C is true for torus high temperature conditions.
D. true for high torus water level.
References: DEOP 200-1, Primary Containment. Technical Specifications 3.6.2.1 & 3.6.2.2.

Direct/New/Modified: **New**
Memory/Comprehension-Analysis: **Memory**
Level (SRO/RO) RO
K/A: 295030 Low Suppression Pool water level G.2.1.28 Knowledge of the purpose and function for major system components and controls.
Author: walton
Exam Date: October 30, 2013

Question # 016

Which of the following constitutes "Adequate Core Cooling?”

NOTE: Only the injection sources stated are injecting.

a. ATWS in progress,
the feed water system is maintaining level between -185 inches and -143 inches,
MSIVs are open
RPV pressure 700 psig.

b. All rods in,
IC operating,
MSIV/ADS valves are closed,
RPV level is -200 inches and decreasing,
RPV pressure is 200 psig.

c. ATWS in progress,
CRD, HPCI and SBLC (with Boron) are injecting,
RPV level is -200 inches and increasing,
RPV pressure 400 psig,
MSIVs are open.

d. All rods in,
HPCI is injecting,
1 ADS valve is open,
RPV level at -200 inches and increasing,
RPV pressure 400 psig,
MSIVs and IC-1301-3 are closed.

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: a.
References: DEOP 400-5, “Failure to Scram” and DEOP 400-3, “Steam Cooling”
Direct/New/Modified: **BANK**
Memory/Comprehension-Analysis: **HIGHER**
Level (SRO/RO): **RO**
K/A: 295031, EK1.01, Knowledge of the Operational implications of the following concepts as they apply to REACTOR LOW WATER LEVEL: Adequate core cooling.
Author: Walton
Exam Date: October 30, 2013

Question # 017

A transient on Unit 2 reactor resulted in the following conditions:

- NOT all control rods are inserted,
- APRM Downscale lights are NOT lit,
- Recirculation pumps are tripped,
- SBLC is injecting.

DEOP 400-5, "Failure to Scram" was entered. The SRO ordered reactor water level be maintained less than -35 inches. Lowering reactor water level under these conditions lowers reactor power by _____(1)______ which ____(2)_____.

a. (1) maximizing core inlet subcooling
   (2) mitigates potential thermo-hydraulic instabilities

b. (1) increasing natural circulation flow
   (2) adds negative reactivity via the void coefficient

c. (1) reducing thermal driving head in the core
   (2) adds negative reactivity via the void coefficient

d. (1) raising fuel temperature
   (2) adds negative reactivity via the doppler coefficient

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: C
A. incorrect, lowering reactor water level <u>minimizes</u> core inlet subcooling.
B. incorrect, lowering reactor water level <u>decreases</u> natural circulation flow
D. as fuel temperature rises, reactivity coefficient becomes less negative.
References: DEOP 400-5, Failure to Scram. Station Nuclear Engineering NEDE-24810. General Electric EPG/SAG, Appendix B, Contingency 5, (see next page). BWR Power Response Lesson Plan.
Direct/New/Modified: **New**
Memory/Comprehension-Analysis: **Higher**
Level (SRO/RO) RO
K/A: 295037 EK2.09 Knowledge of the interrelations between SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN and following: Reactor water level.
Author: Walton
Exam Date: October 30, 2013

Question # 018

Unit 3 was operating at rated power when a loss of coolant accident occurred that caused a fuel element failure. Coincident to this, containment has failed.

If members of the public downwind were to receive an acute dose of 150 rem, what biological effects are expected to occur?

1. Death (to 50% of the population).
2. Slight decrease in blood cell count.
3. Nausea/vomiting to <50% of population within 3 hours.
4. Loss of hair after 2 weeks.

a. 2 ONLY

b. 2 AND 3 ONLY

c. 2, 3 AND 4 ONLY.

d. 1, 2, 3 AND 4.

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: B & C
B is correct for radiation workers per Reg Guide 8.29. For a 150 rem exposure, a slight decrease in blood cell count with 50% of the population experiencing nausea/vomiting is expected.
C is considered correct also for public exposure per EPA guidance provided on web site.
D are incorrect, for a 150 rem exposure, death and loss of hair are not expected
REFERENCES: Exelon Generic Rad Worker Training, July 2013, pg 96.

Direct/New/Modified: **New**
Memory/Comprehension-Analysis: **Memory**
Level (SRO/RO) **RO**
K/A: 295038 EK1.01 Knowledge of the operational implications of the following concepts as they apply to HIGH OFF-SITE RELEASE RATE: Biological effects of radioisotope ingestion.
Author: Walton
Exam Date: October 30, 2013

Question # 019

Control room operators hear the fire alarm and observe the red XL-3 alarm light illuminated between the 902-8 and 903-8 panels. The operators refer to the XL-3 alarm printer to determine there is indication of a fire in the Auxiliary Electric Equipment Room (AEER). The alarm condition was sensed by _____(1)______ and the extinguishing gas will be admitted into the room through _____(2)_______.

	(1)	(2)
a.	heat detectors	a Cardox valve.
b.	heat detectors	pilot manifold valves.
c.	smoke detectors	a Cardox valve.
d.	smoke detectors	pilot manifold valves.

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: D. AEER uses smoke detectors. Halon is used as extinguishing agent in AEER and is admitted through a series of solenoid valves after a 60 second evacuation time delay.
A & B are incorrect since there are no heat detectors in the AEER. Heat detectors are found in the EDG rooms.
C is incorrect. Pilot manifold admits halon into AEER. Cardox valve admits CO2 into EDG rooms.

References: Gaseous Fire System Lesson Plan, Section IV.

Direct/New/Modified: **New**
Memory/Comprehension-Analysis: **Memory**
Level (SRO/RO) **RO**
K/A: 600000, AK2.01; Knowledge of the interrelations between PLANT FIRE ON SITE and the following: Sensors/detectors and valves.
Author: Walton
Exam Date: October 30, 2013

Question # 020

Unit 2 and Unit 3 were operating at rated power when Transmission Systems Operations (TSO) notified the Control Room that the predicted post Unit trip with LOCA switchyard voltages were:

- Unit 2: 350 KV
- Unit 3: 335 KV

What are the required actions (1) from the Operating team AND the reason (2) for these actions assuming the TR32 and TR86 load tap changers are in manual?

a. (1) Adjust TR 32 Tap Changer;
   (2) to restore system operability

b. (1) Adjust TR 32 Tap Changer;
   (2) to reduce circulating currents

c. (1) Adjust TR 86 Tap Changer;
   (2) to restore system operability

d. (1) Adjust TR 86 Tap Changer;
   (2) to reduce circulating currents

Hidden Text below: FILE; OPTIONS; DISPLAY
Answer: A: The LTC will maintain site voltages above the minimum required voltage for operability. Given the voltages, only Unit 3 is below the procedural setpoint. The actions are to adjust the TR-32 Tap Changer (to position 31) to raise VOLTS (not VARs) to restore system operability
References: DOA 6500-12, Low Switchyard Voltage

Direct/New/Modified: **Bank**, Q14699, used on 2012 NRC exam.
Memory/Comprehension-Analysis: **C/A**
Level (SRO/RO) **RO**
K/A: 700000 AK3.02 Knowledge of the reasons for the following responses as they apply to GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES: Reactor and turbine trip criteria.
Author: Walton
Exam Date: October 30, 2013

Question # 021

Given the following conditions:

- A small break LOCA has occurred.
- Drywell pressure is currently 2.5 psig and slowly rising.
- Reactor water level peaked at +60 inches AND is currently +35 inches and lowering.

Assuming the above trends continue, and NO operator actions taken, the HPCI system has:

a. remained in standby but will automatically initiate when reactor water level drops to the low level initiation setpoint.

b. automatically initiated, is currently in operation, but unable to keep up with the leak rate.

c. automatically initiated, is currently NOT in operation, and will NOT automatically restart.

d. automatically initiated, is currently NOT in operation, but will automatically initiate when reactor water level drops to low level initiation setpoint.

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: d

a. INCORRECT – The HPCI system automatically initiates on EITHER low RPV water level OR High Drywell pressure.
b. INCORRECT – The HPCI system initiated automatically and injected into the RPV, but the HPCI turbine trips on High (+50 inches) RPV water level.
c. INCORRECT – The HPCI system automatically started on HIGH Drywell pressure and the HPCI Turbine tripped on HIGH RPV water level. Automatic reset of the RPV High Water Level HPCI turbine trip is inhibited by the concurrent High Drywell pressure, but the system will automatically reinitiate when reactor water level drops to low level initiation setpoint.
d. CORRECT – The HPCI system automatically started on HIGH Drywell pressure and the HPCI Turbine tripped on HIGH RPV water level. The HPCI turbine trip will automatically reset, and the system will reinitiate, when reactor water level drops to low level initiation setpoint.

References:
DRE206LN001, High Pressure Coolant Injection System Lesson Plan; Revision 4
Direct/New/Modified: **MODIFIED** from bank question QQ23577
Memory/Comprehension-Analysis: **C/A**
Level (SRO/RO): **RO**
K/A: 295008.AK2.05 Knowledge of the interrelations between HIGH REACTOR WATER LEVEL and the following: HPCI
Author: Reeser
Exam Date: October 30, 2013

Question # 022

Given the following:

- Unit 2 is in Mode 4 with Reactor Recirculation system secured.
- 2A and 2B Shutdown Cooling loops are running at 50% flow, lined up to both Reactor Recirculation loops.

Maintaining reactor water above __________ will ensure that the unit remains in Mode 4.

a. +50 inches

b. +30 inches

c. +10 inches

d. -50 inches

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: a.

a. CORRECT – Reactor water level must be maintained above +48 inches (second stage of the moisture separators) to ensure that a flow path through the reactor core is maintained.
b. INCORRECT – Plausible since +30 inches is the normal control band.
c. INCORRECT – Plausible since +10 inches is just above the Group 3 (SDC) isolation setpoint.
d. INCORRECT – Plausible if applicant believes the Group 3 isolation setpoint is associated with low-low level.

References:
DOP 1000-3, Shutdown Cooling Mode of Operation; Revision 74
DRE205LN001, Shutdown Cooling (SDC) System Lesson Plan; Revision 7
DRE223LN004, Reactor Pressure Vessel and Internals Lesson Plan; Revision 6
Direct/New/Modified: **MODIFIED** from Dresden Exam Bank Question 13803
Memory/Comprehension-Analysis: **Memory**
Level (SRO/RO): **RO**
K/A: 295009.AK1.05 Knowledge of the operational implications of the following concepts as they apply to LOW REACTOR WATER LEVEL: Natural circulation
Author: Reeser
Exam Date: October 30, 2013

Question # 023

The Unit Drywell Bulk Average Temperature:

a. is displayed on the SPDS Summary Display.

b. must be calculated using data obtained from both inside and outside the Main Control Room.

c. is displayed on Drywell Temperature Recorder TR 2(3)-5741-19 located in the Reactor Building.

d. is displayed on Isolation Condenser/DW Atmos Temps Recorder 2(3)-1340-1 located on Main Control Room Panel 902(3)-3.

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: b.

a. INCORRECT – The Drywell Temperature displayed on SPDS is an average of only two data points (A Recirc Pp Mtr Area and B Recirc Pp Mtr Area).
b. CORRECT – The Unit Drywell Bulk Average Temperature is a volume weighted average, calculated using data collected in the Main Control Room (including DW Atmosphere Temperatures indicated on Isolation Condenser/DW Atmos Temps Recorder 2(3)-1340-1 located on Main Control Room Panel 902(3)-3) and locally Drywell Temperature Recorder TR 2(3)-5741-19 in the Reactor Building.
c. INCORRECT – Drywell Temperature Recorder TR 2(3)-5741-19 displays individual data points but not the average.INCORRECT – Isolation Condenser/DW Atmos Temps Recorder 2(3)-1340-1 individual data points but not the average.on on

References:
Unit 2(3) DOS 1600-29, UNIT 2 and 3 Drywell Temperature Surveillance; Revision 5:
Direct/New/Modified: **NEW**
Memory/Comprehension-Analysis: **Memory**
Level (SRO/RO): **RO**
K/A: 295012.AA2.01 Ability to determine and/or interpret the following as they apply to HIGH DRYWELL TEMPERATURE : Drywell temperature
Author: Reeser
Exam Date: October 30, 2013

Question # 024

When using ADSVs to control pressure in accordance with DEOP 100, “RPV Control,” the ADSVs are operated in a preferred sequence to:

a. minimize cyclic stresses on ADSV tailpipes.

b. minimize uneven heating of the Torus water.

c. balance hydraulic stresses on the Torus wall.

d. prevent overheating of the pilot valve solenoid.

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: b.

a. INCORRECT – Cyclic stresses of SRV tailpipes is minimize through manual control of pressure below ADSV automatic setpoints.
b. CORRECT – Per 295L-S1, DEOP 100, RPV Control Lesson Plan the preferred sequence allows for even heat distribution in the Torus water. These valves discharge to different bays in the Torus and cycling the valves assures that one section will not become excessively hotter than another.
c. INCORRECT – Plausible if examinee believes that SRV discharge produces a high localized dynamic stress on the Torus wall.
d. INCORRECT -- Plausible if examinee believes that repeated cycling of power within a short time period could cause overheating of the solenoid.

References:
295L-S1, DEOP 100, RPV Control Lesson Plan; Revision 06

Direct/New/Modified: **NEW**
Memory/Comprehension-Analysis: **MEMORY**
Level (SRO/RO): **RO**
K/A: 295013.AK1.01 Knowledge of the operational implications of the following concepts as they apply to HIGH SUPPRESSION POOL TEMPERATURE : Pool stratification
Author: Reeser
Exam Date: October 30, 2013

Question # 025

Unit 3 was operating at rated power when a transient occurred, requiring entry into DEOP 400-05, FAILURE TO SCRAM.

- RPV water level is being maintained within a band of -100 to -60 inches using the Condensate/Feedwater system
- RPV pressure is being maintained within a band of 800 to 1060 psig using the Turbine Bypass Valves

The Unit Supervisor has directed insertion of control rods by alternate methods in accordance with DEOP 500-05, ALTERNATE INSERTION OF CONTROL RODS.

Which of the following “alternate” control rod insertion methods REQUIRE operations to be performed outside of the Main Control Room?

a. Use of Scram Test Switches

b. Manually driving Control Rods

c. Repeated scram/resets defeating RPS logic

d. Pull power supply fuses for scram solenoids

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: c.

a. INCORRECT – is performed on a Main Control Room panel 902(3) – 16
b. INCORRECT - requires control panel operations to maximize CRD Drive Water Pressure, is performed on Main Control Room panels 902(3) – 5.
c. CORRECT – requires fuses to be pulled in the Auxiliary Electrical Equipment Room to prevent automatic ARI actuation on low RPV water level.
d. INCORRECT – Done from CR on 902(3)-15 & 17 panels.

References:
DEOP400 -05, FAILURE TO SCRAM:
DEOP 500-5 ALTERNATE INSERTION OF CONTROL RODS; Revision 17,
Direct/New/Modified: **NEW**
Memory/Comprehension-Analysis: **MEMORY**
Level (SRO/RO): **RO**
K/A: 295013 G2.4.35 Knowledge of local auxiliary operator tasks during an emergency and the resultant operational effects.
Author: Reeser
Exam Date: October 30, 2013

Question # 026

Automatic initiation of a Standby Gas Treatment (SBGT) train upon detection of Reactor Building Ventilation exhaust high radiation reduces the concentration of radioactive material being exhausted from the reactor building by:

a. using filters to remove radioactive particulates and inert gases.

b. using charcoal adsorbers to remove radioactive gasses.

c. diluting SBGT air flow through the Reactor Building Ventilation Stack.

d. maintaining building pressure slightly negative that allows for radioactive decay of shorter lived radioactive isotopes.

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: b.

a. INCORRECT – filters will remove particulates but not inert gases.
b. CORRECT – Radioactive halogen gases are removed by adsorbtion in the SBGT charcoal adsorber beds.
c. INCORRECT – Effluent from the SBGT train is exhausted to the 310 foot Chimney and not to the Reactor Building Vent Stack. Flow to the Reactor Building Vent stack is actually terminated by the same signals that initiate SBGT.
d. INCORRECT – The building is normally maintained at a slightly negative pressure. Simply maintaining a negative pressure does not reduce the concentration of shorter lived radioactive material released. Maintaining a negative pressure ensures that radioactive material doesn't exit the reactor building without first being processed by the SBGT system.

References:
DRE261LN001, Standby Gas Treatment System Lesson Plan; Revision 6
DRE288LN001, Reactor Building Ventilation System Lesson Plan; Revision 4

Direct/New/Modified: **NEW**
Memory/Comprehension-Analysis: **MEMORY**
Level (SRO/RO): **RO**
K/A: 295034.EK3.02 Knowledge of the reasons for the following responses as they apply to SECONDARY CONTAINMENT VENTILATION HIGH RADIATION : Starting SBGT/FRVS
Author: Reeser
Exam Date: October 30, 2013

Question # 027

A break in the air supply line to Unit 2 Reactor Building Ventilation Outlet Isolation Dampers will result in the trip of:

a. Unit 2 Reactor Building Vent (Supply) fans on high Reactor Building pressure.

b. Unit 2 Reactor Building Exhaust fan on low Reactor Building pressure.

c. all Unit 2 Rx Building Vent and Exhaust Fans ONLY.

d. all Unit 2 Rx Building Vent and Exhaust Fans AND automatic actuation of the Standby Gas Treatment System.

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: c.

a. INCORRECT – Plausible because closure of an outlet isolation damper isolates the suction path for the exhaust fans and building pressure would increase until the supply fans tripped on high pressure if the fans were not trips directly by repositioning of the damper.
b. INCORRECT – Plausible if examinee believes building pressure is sensed between the outlet isolation damper and the exhaust fans.
c. CORRECT – A failure of the air supply to a Reactor Building Ventilation isolation dampers will result in the closure of the damper. All Reactor Building Vent and Exhaust Fans will trip if any one of the four Reactor Building Ventilation isolation dampers is not full open.
d. INCORRECT – Plausible if the examinee believes that the Standby Gas Treatment System actuates directly due to the trip of the Reactor Building Vent and Exhaust fans or because of high Reactor Building pressure (low negative pressure).

References:
DRE261LN001, Standby Gas Treatment System Lesson Plan; Revision 6
DRE288LN001, Reactor Building Ventilation System Lesson Plan; Revision 4
Direct/New/Modified: **NEW**
Memory/Comprehension-Analysis: **C/A**
Level (SRO/RO): **RO**
K/A: 295035.EA1.01 Ability to operate and/or monitor the following as they apply to SECONDARY CONTAINMENT HIGH DIFFERENTIAL PRESSURE: Secondary containment ventilation system
Author: Reeser
Exam Date: October 30, 2013

Question # 028

Given the following conditions:

- Dresden 2 has just experienced a loss of offsite power.
- Drywell pressure has rapidly risen to +2.5 psig.
- The diesels fast started.

Which of the following states the timed starting sequence for the emergency bus equipment?

a.
  - When reactor pressure reaches 350 psig AND 8.5 minutes have elapsed from initiation, then the diesel generator breaker closes,
  - then the first low pressure coolant injection (LPCI) pump starts
  - followed by the core spray pump start 5 seconds later
  - followed by the second LPCI pump start 5 seconds later.

b.
  - When reactor pressure reaches 350 psig AND 8.5 minutes have elapsed from initiation, then the diesel generator breaker closes,
  - then the first LPCI pump starts immediately
  - followed by the second LPCI pump start 5 seconds later
  - followed by the core spray pump start 5 seconds later.

c.
  - The diesel generator breaker closes within 10 seconds of the diesel start,
  - then the core spray pump starts
  - followed by 5 seconds later by the first LPCI pump start,
  - followed by the second LPCI pump start 5 seconds later.

d.
  - The diesel generator breaker closes within 10 seconds of the diesel start,
  - then the first LPCI pump starts
  - followed by the second LPCI pump start 5 seconds later
  - followed by the core spray pump start 5 seconds later.

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: D: CORRECT –
A, B, and C: INCORRECT – incorrect sequences

References: Dresden Lesson Plan, LPCI, DRE203LN001, pg 46.

Direct/New/Modified: ~~NEW~~ **BANK**
Memory/Comprehension-Analysis: **C/A**
Level **RO**

K/A: 203000.A1.08; Ability to predict and/or monitor changes in parameters associated with operating the RHR/LPCI: INJECTION MODE (PLANT SPECIFIC) controls including: Emergency generator loading
Author: BANK
Exam Date: October 30, 2013

Question # 029

The unit was in Mode 4 with one Shutdown Cooling (SDC) loop in operation, aligned to both Reactor Recirculation loops, but with only one Reactor Recirculation pump in operation.

Which of the following sets of actions, performed independently, will mitigate the effects of a trip of the running Reactor Recirculation pump?

a. Place a second SDC loop in operation and maximize flow in both SDC loops.

b. Shut the pump suction valve for the tripped Reactor Recirculation pump.

c. Place a second SDC loop in operation, maximize flow in both SDC loops, AND raise RPV water level to at least 30 inches.

d. Place a second SDC loop in operation, maximize flow in both SDC loops, AND shut the tripped Reactor Recirculation pump's suction valve.

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: D is Correct – increases core flow and prevents bypass flow through the idle loop.
A is INCORRECT – Some SDC return flow will be short cycled through the Reactor Recirculation loop, bypassing the reactor core.
B is INCORRECT – This will prevent short cycle flow through the Recirculation loop, but with no Reactor Recirculation pump in operation, there will not be enough flow through the core.
C is INCORRECT – While raising RPV water level will promote natural circulation through the core, cooling will be degraded due to short cycling through the Reactor Recirculation loop.

References:
UNIT 2(3) DOP 1000-03, SHUTDOWN COOLING MODE OF OPERATION; Revision 74
Direct/New/Modified: **NEW**
Memory/Comprehension-Analysis: **C/A**
Level: **RO**
K/A: 205000.A2.11 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: Recirculation pump trips
Author: Reeser
Exam Date: October 30, 2013

Question # 030

Following a Group 1 isolation and Reactor scram, you have been directed to maintain RPV pressure within a control band of 800-1060 psig using the ERVs. HPCI is in operation. You become distracted performing another assigned activity and RPV pressure increases to the Electromatic relief valves opening automatic setpoint.

Which of the following describes how HPCI flow will be affected by the increased RPV pressure? HPCI flow will…

a. be automatically maintained at the flow controller setpoint.

b. be terminated due to reaching the shut-off head of the HPCI pump.

c. decrease below the flow controller setpoint due to HPCI speed being limited by the MSC High Speed Stop.

d. decrease below the flow controller setpoint due to HPCI speed being limited by the MGU High Speed Stop.

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: A is CORRECT – The HPCI system is design to provide rated flow up to a RPV pressure of 1150 psig. The ERV setpoints are approximately 1110/1134 psig
B is INCORRECT – The shut-off head of the pump is approximately 1250 psig. The ERV setpoints are approximately 1110/1134 psig
C is INCORRECT – The MSC HSS is approximately 4000 rpm and is higher than the speed necessary produce rated flow at the ERV setpoint. Additionally the speed control circuit selects the lower of the MSC or MGU outputs to control turbine speed. The MSC ramps from 0 to 4000 rpm during startup with the MGU taking control as flow increases to setpoint.
D is INCORRECT – The MGU HSS is approximately 4000 rpm and is higher than the speed necessary produce rated flow at the ERV setpoint.
References:
DRE206LN001, High Pressure Coolant Injection Lesson Plan; Revision 4
DRE239LN001, Main Steam System Lesson Plan; Revision 10
Direct/New/Modified: **NEW**
Memory/Comprehension-Analysis: **Memory**
Level (SRO/RO): **RO**
K/A: 206000.A3.04 Ability to monitor automatic operations of the HIGH PRESSURE COOLANT INJECTION SYSTEM including: Reactor pressure
Author: Reeser
Exam Date: October 30, 2013

Question # 031

Which of the following describe how the Isolation Condenser (IC) shell side water level is normally maintained?

a. An automatic level control valve supplies water from the Clean Demineralized Water System header to maintain level.

b. A remote manually operated makeup valve is cycled to supply water from the Condensate Transfer System to maintain level.

c. One of the two Diesel Driven IC Makeup Pumps is manually started and transfers water from the Clean Demineralized Water Tank through a remote operated makeup valve.

d. One of the two Diesel Driven IC Makeup Pumps automatically starts on low IC shell side water level and transfers water from the Clean Demineralized Water Tank through an automatic level control valve.

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: C is correct. Diesel Driven IC Makeup pumps must be manually started and the makeup valve is manually operated as needed to maintain level.
A is INCORRECT – There are no automatically control components.
B is INCORRECT – The Condensate Transfer System is potentially contaminated and is used only as a last resort.
D is INCORRECT – Diesel Driven IC Makeup pumps must be manually started and the makeup valve is manually operated.

References:
DRE207LN001, Isolation Condenser Lesson Plan; Revision 7
Direct/New/Modified: **NEW**
Memory/Comprehension-Analysis: **Memory**
Level (SRO/RO): **RO**
K/A: 207000.A4.06 Ability to manually operate and/or monitor in the control room: Shell side makeup valves
Author: Reeser
Exam Date: October 30, 2013

Question # 032

Which of the following combinations of ECCS subsystems will ensure adequate core cooling during a DBA LOCA?

a. One LPCI subsystem

b. One Core Spray subsystem

c. One Core Spray subsystem AND the 5 ADS valves

d. One Core Spray Subsystem AND one LPCI subsystem

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: D is correct. At least two Core Spray subsystems, or One Core Spray subsystem AND one LPCI subsystem are necessary to ensure that at least 4500 gpm spray is provided to cool the top third of the core with RPV level maintain at 2/3 core coverage.
A is INCORRECT – While two LPCI pumps will restore RPV level to 2/3 core coverage it will not ensure cooling to the upper 1/3 of the core. Either a Core Spray system is needed to cool the upper 1/3 or the core must be completely covered which would require additional LPCI flow.
B is INCORRECT – One Core Spray pump will not ensure to 2/3 core coverage
C is INCORRECT – Core depressurization will occur without ADS but One Core Spray pump will not ensure to 2/3 core coverage

References:
DRE209LN001, Core Spray System Lesson Plan; Revision 8
DRE203LN001, Low Pressure Coolant Injection Lesson Plan; Revision 8
USAR Section 6.3.3 [Emergency Core Cooling System] Performance Evaluation; Revision 10 (June 2013)
Direct/New/Modified: **NEW**
Memory/Comprehension-Analysis: **Memory**
Level (SRO/RO): **RO**
K/A: 209001.2.4.21 Knowledge of the parameters and logic used to assess the status of safety functions, such as reactivity control, core cooling and heat removal, reactor coolant system integrity, containment conditions, radioactivity release control, etc.
Author: Reeser
Exam Date: October 30, 2013

Question # 033

DOP 1100-05, SBLC TANK WATER ADDITION AND AIR SPARGING, requires a dedicated operator be on the floor by SBLC with communications to the control room to realign the system if required.

Why must an operator be available to realign the SBLC from its air sparge condition to its accident position?

a. The addition of air will dilute the boron concentration.

b. SBLC Tank Level will indicate higher than actual.

c. SBLC pump suctions must be closed to prevent potential air binding.

d. The air sparging valve does not automatically reposition upon actuation of SBLC.

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER:

a. INCORRECT – Addtion of air does not dilute the boron concentration.
b. INCORRECT – Tank level indicator measures Instrument Air back pressure and is not affected by air sparging which uses Service Air.
c. INCORRECT – Suction valves are not closed during the evolution.
d. CORRECT – SR3.1.7.6 cannot be satisfied while the air sparger valve is open.

References:
Dresden Nuclear Power Station Technical Specification Bases B3.1.7.
DOP 1100-05, SBLC TANK WATER ADDITION AND AIR SPARGING
Direct/New/Modified: **NEW**
Memory/Comprehension-Analysis: **Memory**
Level (SRO/RO): **RO**
K/A: 211000.K1.03 Knowledge of the physical connections and/or cause-effect relationships between STANDBY LIQUID CONTROL SYSTEM and the following: Plant Air Systems
Author: Reeser
Exam Date: October 30, 2013

Question # 034

Unit 3 is operating at 85% reactor power with the following conditions:

- RPS Bus A is powered from its NORMAL power source and
- RPS Bus B is powered from its ALTERNATE power source due to maintenance on its normal source.

Under these conditions, if the NORMAL power source to RPS Bus A is lost, RPS Channel A will _____________ its ALTERNATE power source.

a. remain energized due to a bumpless transfer to

b. lose power but can be manually re-energized from

c. lose power and cannot be manually re-energized from

d. momentarily lose power, but automatically re-energize from

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: c

a. INCORRECT – There are no automatic source transfers.
b. INCORRECT – There is only one ALTERNATE power source for the RPS buses and a mechanical interlock prevents the ALTERNATE source from supplying both RPS buses at the same time.
c. CORRECT – There is only one ALTERNATE power source for the RPS buses and a mechanical interlock prevents the ALTERNATE source from supplying both RPS buses at the same time.
d. INCORRECT – There are no automatic source transfers and a mechanical interlock prevents the ALTERNATE source from supplying both RPS buses at the same time.

References:
DRE212LN001, Reactor Protection System Lesson Plan; Revision 3
DRE262LN005, Low Voltage Distribution System Lesson Plan; Revision 6
Direct/New/Modified: **NEW**
Memory/Comprehension-Analysis: **Memory**
Level (SRO/RO): **RO**
K/A: 212000.K2.02 Knowledge of electrical power supplies to the following: Analog trip system logic cabinets
Author: Reeser
Exam Date: October 30, 2013

Question # 035

A reactor startup is in progress with the plant at the point of adding heat. The output amplifier for IRM 15 fails such that IRM 15 reads 3/125, resulting in:

a. no RPS trips.

b. the trip of RPS Channel A.

c. the trip of RPS Channel B.

d. the trip of both RPS Channels A and B.

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: a

a. CORRECT – a downscale failure without an INOP trip condition will result in only a rod block with no RPS channels tripped.
b. INCORRECT
c. INCORRECT
d. INCORRECT

References:
DRE215LN003, Intermediate Range Monitoring System Lesson Plan; Revision 3
Direct/New/Modified: **NEW**
Memory/Comprehension-Analysis: **Memory**
Level (SRO/RO): **RO**
K/A: 215003.K3.01 Knowledge of the effect that a loss or malfunction of the INTERMEDIATE RANGE MONITOR (IRM) SYSTEM will have on following: RPS
Author: Reeser
Exam Date: October 30, 2013

Question # 036

A plant start-up is in progress. Which of the following conditions will result in a rod block?

a. SRM detectors fully Inserted, the highest indication is 1×10^5 CPS, all IRMs indicate on scale range 8

b. SRMs detectors partially withdrawn, the lowest indication is 150 CPS, IRMs indicate on scale ranges 2 and 3

c. SRM detectors fully withdrawn, the lowest indication is 320 CPS, IRMs indicate on scale on ranges 2 and 3

d. SRM detectors are fully withdrawn, the highest indication is 150 CPS, all IRMs indicate on scale range 4

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER:

a. INCORRECT – SRM Detector High rod block is bypassed when all IRMs indicate on Range 8 or higher.
b. CORRECT – Detector Wrong Position rod block is actuated at < 290 cps with any IRM below Range 3.
c. INCORRECT – Detector Wrong Position rod block is actuated at < 290 cps with any IRM below Range 3.
d. INCORRECT – Detector Wrong Position rod block is bypassed when all IRMs indicate on scale on Range 3 or higher.

References:

DRE215LN004, Source Range Monitoring System Lesson Plan; Revision 7
UNIT 2(3) DOP 0700-01, Source Range Monitor Operation (SRM); Revision 14

Direct/New/Modified: **MODIFIED**
Memory/Comprehension-Analysis: **C/A**
Level (SRO/RO): **RO**
K/A: 215004.K4.06 Knowledge of SOURCE RANGE MONITOR (SRM) SYSTEM design feature(s) and/or interlocks which provide for the following: IRM/SRM interlock
Author: Reeser
Exam Date: October 30, 2013

Question # 037

Each APRM channel receives inputs from LPRMs located in ____(1)_______ axial plane(s) distributed across ____(2)_______ quadrant(s), and provides an indication of the bulk thermal power for _____(3)______.

a. (1) four
(2) a single
(3) that quadrant

b. (1) one upper and one lower
(2) all four
(3) the entire core

c. (1) the upper or lower two
(2) a single
(3) that quadrant

d. (1) four
(2) all four
(3) the entire core

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: d
The LPRMs are assigned in an orderly pattern across the core. The locations are selected so that every APRM will give a good representative core average power signal. The assignments are symmetrical, on diagonals through the LPRM assemblies at core location 32-33. APRM Channels 1, 2, and 3 and LPRM Group 1 are on one set of diagonals; APRM Channels 4, 5, and 6, and LPRM Group 2 are on the alternate set of diagonals. Each LPRM detector is assigned to only one APRM channel or LPRM group
References:
DRE215LN005, Average Power Range Monitoring Lesson Plan; Revision 6
Direct/New/Modified: **NEW**
Memory/Comprehension-Analysis: **Memory**
Level (SRO/RO): **RO**
K/A: 215005.K5.04 Knowledge of the operational implications of the following concepts as they apply to AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM: LPRM detector location and core symmetry
Author: Reeser
Exam Date: October 30, 2013

Question # 038

What MINIMUM combination of low pressure ECCS pumps would have to FAIL to prevent automatic actuation of the ADS system?

a. BOTH Core Spray (CS) pumps ONLY

b. ALL four Low Pressure Coolant Injection (LPCI) pumps ONLY

c. ALL Division 1 low pressure ECCS pumps (CS and LPCI); OR
ALL Division 2 low pressure ECCS pumps (CS and LPCI)

d. ALL low pressure ECCS pumps

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: D

Any single low pressure ECCS pump operating at greater than approximately 100 psig will permit ADS to automatically actuate.

References:

DRE218LN001, Automatic Depressurization System Lesson Plan; Revision 3

Direct/New/Modified: **NEW**
Memory/Comprehension-Analysis: **Memory**
Level (SRO/RO): **RO**
K/A: 218000.K6.01 Knowledge of the effect that a loss or malfunction of the following will have on the AUTOMATIC DEPRESSURIZATION SYSTEM : RHR/LPCI system pressure
Author: Reeser
Exam Date: October 30, 2013

Question # 039

Primary Containment Isolation System (PCIS) logic relays for Groups 1 and 3 are normally _____(1)______ and the tripped state ____(2)_______.

a. (1) energized
   (2) seals-in

b. (1) energized
   (2) does not seal-in

c. (1) de-energized
   (2) seals-in

d. (1) de-energized
   (2) does not seal-in

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: a.
PCIS logic relays, are normally energized. The tripped state for logic relays seal-in.
References:
DRE223LN005, Primary Containment Isolation System Lesson Plan; Revision 4
Direct/New/Modified: **NEW**
Memory/Comprehension-Analysis: **Memory**
Level (SRO/RO): **RO**
K/A: 223002.A1.04 Ability to predict and/or monitor changes in parameters associated with operating the PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF controls including: Individual system relay status
Author: Reeser
Exam Date: October 30, 2013

Question # 040

Unit 2 was operating at rated power when the 125 VDC 2A-1 Distribution panel was de-energized for maintenance. After the maintenance was complete, 125 VDC 2A-1 Distribution panel was re-energized.

What is the current power supply to the Unit 2 Safety Relief Valve 203-3A solenoid?

a. U2 ESS Bus

b. U2 Instrument Bus

c. 125VDC Distribution Panel 2A-1

d. 125 VDC Distribution Panel 2B-1

Hidden Text below: FILE; OPTIONS; DISPLAY
Answer: C: The 203-3A SRV solenoid has two power supplies. The normal supply is the 2A-1 Dist Panel and the alternate supply is the 2B-1 Dist Panel. When the normal supply (2A-1) is lost a “normal seeking” automatic transfer device will transfer to the alternate supply (2B-1). Upon the normal supply being re-energized, the “normal seeking” automatic transfer device will transfer power from the alternate supply (2B-1) to the normal supply (2A-1).
A is incorrect: The ESS Bus is the power supply to the Acoustic monitor for the ERVs.
B is incorrect. The Instrument Bus is the power supply to the Tailpipe temperature monitor for the ERVs.
D is incorrect. 2B-1 would be the power supply if the automatic transfer device didn’t transfer back to the normal supply.

Direct/New/Modified: **Bank** (Q23867)
Memory/Comprehension-Analysis: **Memory**
Level (SRO/RO): **RO**
K/A: 239002.Relief/Safety Valves: K2.01 Knowledge of the electrical power supplies to the following: SRV solenoids
Author: Licensee/Walton
Exam Date: October 30, 2013

Question # 041

Unit 3 is operating with the following conditions:

- Reactor power is 35% and steady,
- Reactor water level is being maintained at +30 inches,
- Feedwater level control is in Master Auto,
- 3B FRV and LFRV are in Automatic; the 3A FRV is closed in manual,
- 3B RFP is running; and 3A and 3C Condensate Pumps are running,
- A spurious full scram signal is received.

How will the FWLC system initially react?

The FWLC system will…

a. maintain reactor water level at +5 inches without regard to feedwater flow.

b. maintain reactor water level at □30 inches without regard to feedwater flow.

c. close the 3B FRV for 25 seconds. After 25 seconds, it will control level at the current setpoint.

d. maintain the 3B FRV's position for one (1) second. After one (1) second, it will pulse down to 30% of its current demand.

ANSWER: D. maintain the 3B FRV's position for one (1) second. After one (1) second, it will pulse down to 30% of its current demand.
Upon a scram signal, the system takes the setpoint to –10" and locks the FRVs in place for 1 second. At 1 second (T1) the programming looks at reactor water level and makes one of the following decisions: If reactor water level is greater than –30", then: The system "pulses" the FRVs to 30% of current demand, any valve in manual, receives a closed signal. 25 seconds later (T26) the system releases the valves in auto to control at the current setpoint. 19 seconds later (T45) the system ramps back to a setpoint of +30", at a rate of 10"/minute.
If reactor water level is less than –30", then: The system releases the FRVs to control at the current setpoint (assuming water level is low enough to NOT enter the HPCI and IC steam lines)

References: None provided.
Direct/New/Modified: **BANK** (2012 NRC)
Memory/Comprehension-Analysis: **C/A**
Level (**RO**)
K/A: 259002 (RX WLC): A3.06 Ability to monitor automatic operations of the REACTOR WATER LEVEL CONTROL SYSTEM including: Reactor water level setpoint setdown following a reactor scram.
Author: BANK - C. Zoia
Exam Date: October 30, 2013

Question # 042

Both units were operating at near rated power with SBGT being operated due to maintenance on the Rx Bldg Vent systems, with the following set of conditions:

- The 2/3 A SBGT SELECT control switch is in STBY.
- The 2/3 B SBGT SELECT control switch is in START.

Then **Unit 3** experiences a sustained loss of Instrument Air.

Concerning the SBGT system, what occurs to the SBGT TRN FAN 2/3 SUCT AO VLV(s) (2/3-7510)?

a. 2/3-7510-B fails full open.

b. 2/3-7510-B fails full closed.

c. 2/3-7510-A AND 2/3-7510-B fail full open.

d. 2/3-7510-A AND 2/3-7510-B fail full closed.

ANSWER: A: Each unit supplies its own SBGT train's SUCT VLV. With 'B' train operating and a loss of U2 Inst Air, this would have no effect on the 'B' train.
References: None provided

Direct/New/Modified: **NEW**
Memory/Comprehension-Analysis: **Memory**
Level (**RO**)

K/A: 261000 (SGTS): A4.09 Ability to manually operate and/or monitor in the control room: Ventilation valves/dampers.
Author: Walton
Exam Date: October 30, 2013

Question #43 was removed from the exam due to resolution of post exam comments.

Question # 043

~~Unit 3 was operating at rated power when a transient occurred, resulting in the following conditions:~~

- ~~RPV water level is -72 inches and trending up.~~
- ~~The TR-32 Sudden Pressure Relay (SPR) activated.~~
- ~~The Unit 3 EDG started but its output breaker subsequently tripped on over-current.~~

~~The Unit Supervisor has directed the crew to enter and execute DGA-12, PARTIAL OR COMPLETE LOSS OF AC POWER.~~

~~The required electrical lineup is to power Bus 33-1 from __(1)__ and Bus 34-1 from __(2)__ .~~

	~~(1)~~	~~(2)~~
~~a.~~	~~Bus 23-1;~~	~~U3 SBO~~
~~b.~~	~~U3 SBO;~~	~~Bus 24-1~~
~~c.~~	~~2/3 EDG;~~	~~U3 SBO~~
~~d.~~	~~2/3 EDG;~~	~~Bus 24-1~~

~~ANSWER: c. (1) 2/3 EDG; (2) U3 SBO.~~
~~Upon a scram, loss of off-site power, and failure of the Unit 3 EDG, DGA-12 directs powering Bus 33-1 from the 2/3 EDG and Bus 34-1 from the U3 SBO (via backfeed from Bus 34). The distractors are possible lineups based on failures of the 2/3 EDG and/or both unit EDGs.~~

~~References: None provided.~~
~~Direct/New/Modified: **BANK** (2011 CERT)~~
~~Memory/Comprehension-Analysis: **C/A**~~
~~Level (**RO**)~~
~~K/A: 262001 (AC Electrical Distribution): K1.01 Knowledge of the physical connections and/or cause-effect relationships between A.C. ELECTRICAL DISTRIBUTION and the following: Emergency generators.~~
~~Author: BANK - C. Zoia~~
~~Exam Date: October 30, 2013~~

Question # 044

During the performance of a test on a NON-SAFETY-related system, a leak occurs in the system. If isolating the leak requires repositioning a component to a position different from that listed in its normal valve line-up, and no applicable procedure currently exists, which of the following is required by OP-AA-108-101 “Control of Equipment and System Status” for these non-emergency conditions?

The steps needed to isolate the leak can be taken once…

1. approved by any SRO.
2. approved by any two SROs.
3. documented on the abnormal component position sheet.

ANSWERS:

a. 1 & 2 ONLY

b. 2 & 3 ONLY

c. 1 & 3 ONLY

d. 1, 2 & 3.

ANSWER: C. The steps to isolate the leak can be taken when approved by one SRO.
For Safety related systems, configuration Control requires approval of two SROs per OP-AA-108-101, Operational Configuration Control for safety related systems and a 50.59 evaluation.

References: None provided. See OP-AA-108-101, Operational Configuration Control and LP DRE-N-ADM-2, Configuration Control.

Direct/New/Modified: **New**
Memory/Comprehension-Analysis: **Memory**
Level (**RO**)

K/A: G2.1.20 Ability to interpret and execute procedure steps.

Author: C. Zoia
Exam Date: October 30, 2013

Question # 045

Unit 2 was operating at rated power when a fire de-energized the U2 ESS Bus.

Which of the following components would lose power?

a. TIP valve controls

b. SBLC Pump Indication

c. Rod Position Indication

d. Main Generator Voltage Regulator

ANSWER: c. Rod Position Indication.
The RPIS is powered from the ESS Bus. The other choices are powered from the Instrument Bus (This is a common misconception at Dresden).

References: None provided.
Direct/New/Modified: **BANK** (2009 Cert Exam)
Memory/Comprehension-Analysis: **Memory**
Level (**RO**)
K/A: G2262002 UPS (AC/DC); K3.06 Knowledge of the physical connections and/or cause-effect relationships between UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.) and the following: Rod position indication..

Author: BANK / C. Zoia
Exam Date: October 30, 2013

Question # 046

Unit 2 was operating at rated power, when the following annunciators were received:

- 902-8 E-8 ESS UPS ON DC OR ALTERNATE AC
- 902-8 F-8 ESS UPS TROUBLE

The equipment operator dispatched to the AEER reported the following on the 902-63B panel:

- Normal A/C power has FAILED to ESS Bus.
- The LOW DC VOLTAGE light is illuminated.
- The DC VOLT meter indicates a degraded voltage condition.

What is supplying power to the Unit 2 Essential Service (ESS) Bus?

a. MCC 28-2

b. Bus 25

c. Bus 29

d. Unit 2 250 VDC system

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: B Knowledge of the ESS circuit paths is needed to answer this question. The normal supplies to the ESS Bus, in the descending order they feed is: Bus 29, U2 T.B. 250Vdc battery system, Bus 25, then MCC 28-2 (emergency). Given a degraded condition or loss of the 250Vdc supply, then both NORMAL supplies (Bus 29 and 250 VDC, via the inverter) are lost. The next power supply to feed is the ALTERNATE AC source of Bus 25.

References:DOP 6800-01 & DAN 902-8 F-8.

Direct/New/Modified: **BANK** Q12903
Memory/Comprehension-Analysis: **C/A**
Level (SRO/RO) **RO**
K/A: 262002 UPS K6.02; Knowledge of the effect that a loss or malfunction of the following will have on the UNINTERRUPTABLE POWER SUPPLY (AC/DC): DC electrical power.
Author: Facility/Walton
Exam Date: October 30, 2013

Question # 047

Both units were operating at rated power with DC electrical systems in a normal lineup when the control room received Annunciators 902-8, C-10, and 903-8 C-10; 250 VDC POWER FAILURE. The SRO entered DOA 6900-4, “Failure of Unit 2(3) 250 VDC Power Supply.” A dispatched EO reported back that Turbine Building MCC 3 is damaged and is de-energized. This condition will result in _____(1)______ because ______(2)_______.

a. (1) Unit 2 HPCI being able to start but unable to run
(2) the gland steam condenser exhauster would trip.

b. (1) Unit 2 HPCI being unable to start
(2) there is no oil pressure to open the stop and control valves.

c. (1) Unit 3 HPCI being able to start but unable to run
(2) the gland steam condenser exhauster would trip.

d. (1) Unit 3 HPCI being unable to start
(2) there is no oil pressure to open the stop and control valves.

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: B: TB MCC 3 feeds Unit 2 MCC 2A and 2B panels. Both Unit 2 HPCI gland steam condenser exhauster and HPCI Aux Oil pump are powered from MCC 2A. Per the lesson plan, HPCI will not start since the Aux Oil pump won’t start and can’t develop oil pressure sufficient to open the HPCI stop and control valves.
A is incorrect. HPCI can NOT start, but can still run without the gland steam condenser exhauster.
C & D are both incorrect. Unit 3 HPCI is unaffected by a loss of MCC 2A. Unit 3 HPCI auxiliaries are powered from MCC 3A.

References: 250 VDC Distribution Lesson Plan, pg 17. DOA 6900-04. Dan 902-8, C-10, “250 VDC Power Failure.” DOP 6900-04, 250 VDC Load List, pg 25.
Direct/New/Modified: **New**
Memory/Comprehension-Analysis: **C/A**
Level (SRO/RO) **RO**
K/A: 263000 DC Electrical Distribution K2.01, Knowledge of electrical power supplies to the following: Major DC loads.
Author: Walton
Exam Date: October 30, 2013

Question # 048

Given the following plant conditions:

- Both Units were in MODE 1 operations with DC busses normally aligned.
- Subsequently, carpenters working near 2A-1 distribution panel damaged, and caused a loss of power on the 125 VDC 2A-1 distribution panel.

Which electrical breakers will lose their control power?

a. RBCCW Pump 2A

b. Reactor Feedwater Pumps 2A and 2C.

c. Bus 21 Main Feed Breaker from UAT 21

d. Bus 23 Main Feed Breaker from UAT 21

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: D; 2A-1 DC distribution bus provides power to Bus 23 MF Bkr from UAT 21
A is incorrect, RBCCW 2A receives power from the U2 Rx Bldng Dist. Pnl 2.
B is incorrect, RFW pumps 2A & 2C receive control power from Bus 2A-2.
C is incorrect, Bus 21 MF Bkr from UAT 12 receives control power from Bus 2A-2.
References: DOA 6900-T1, Table 3

Direct/New/Modified: Modified from 2008 NRC exam Q14497
2013 Audit Exam question Q23473. **BANK**
Memory/Comprehension-Analysis: **Memory**
Level (SRO/RO) **RO**
K/A: 263000DC Electrical Distribution; K3.02, Knowledge of the effect that a loss or malfunction of the DC ELECTRICAL DISTRIBUTION will have on the following: Components using DC Control Power (ie; breakers).
Author: Walton
Exam Date: October 30, 2013

Question # 049

A loss of coolant accident has occurred on Unit 3. Emergency Diesel Generator (EDG) 3 has loaded onto Bus 34-1. However, a partial ground on Unit 3 EDG has resulted in Bus voltage dropping to 3700 VAC. If the degraded voltage condition persists, which of the following loads on Bus 34-1 will trip (if any) after 5 minutes?

1. 480 VAC Emergency Bus (Bus 39)
2. 3C & 3D LPCI Pumps
3. 3B & 3C Shutdown Cooling Pumps
4. 3B RBCCW Pump

Answers:

a. 1 AND 4 ONLY.

b. 2 AND 3 ONLY.

c. 1, 2, 3 AND 4.

d. No loads will trip.

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: D is correct since the 5-minute timer is out of the circuit due to an ECCS actuation (EDG breaker being closed).
B is incorrect.. In “U 2/3 Auxiliary Power Lesson Plan,” pg. 22, the LPCI pumps and SDC pumps will trip if there was no ECCS actuation present, the 5-minute timer trip these pumps.
A & C are incorrect since 480 VAC Emergency Bus and 3B RBCCW pumps will NOT trip.

References: U 2/3 Auxiliary Power Lesson Plan, pgs 22, 23,& 24, DOP 6500-27, “Removing 4KV Bus 34 from Operation for Maintenance/Testing,” Section D.3.
Direct/New/Modified: **Modified**
Memory/Comprehension-Analysis: **C/A**
Level (SRO/RO) **RO**
K/A: 264000 K3.01; Knowledge of the effect that a loss or malfunction of the EMERGENCY GENERATORS will have on the following: ECCS.
Author: Walton
Exam Date: October 30, 2013

Question # 050

Which ONE is NOT a design feature of the emergency diesel generator governor.

a. Trips the EDG on overspeed.

b. Controls the position of the EDG fuel racks.

c. Limits the diesel generator maximum output.

d. Controls the change in EDG speed with respect to changes in real load.

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: A is correct. The EDG over speed trip device is connected to the cam shaft and does NOT interact with the EDG governor.
B & C & D. are all incorrect. The EDG governor does all these functions.

References: EDG Auxiliary Systems Lesson Plan, pgs 33 & 45.

Direct/New/Modified: **New**
Memory/Comprehension-Analysis: **Memory**
Level (SRO/RO) **RO**
K/A: 264000 K3.01, Knowledge of EMERGENCY GENERATORS design feature(s) and/or interlocks which provide for the following: Governor control.
Author: Walton
Exam Date: October 30, 2013

Question # 051

With Unit 3 Instrument Air system in its normal lineup, a leak develops resulting in the instrument air system pressure lowering. Control room operators receive Alarm 923-1, G-6, “U3 INST AIR PRESS LO.” Based on this alarm, AO 3-4701-500, U3 Service Air to IA X-tie valve _____(1)______. To reposition AO 3-4701-500 to close, _______(2)________.

a. (1) opens automatically due to low air receiver pressure
   (2) an operator must reset the valve locally

b. (1) opens automatically due to low air header pressure
   (2) system pressure rising will cause the valve to automatically close

c. (1) must be manually opened
   (2) an operator must reset the valve locally

d. (1) must be manually opened
   (2) system pressure rising will cause the valve to automatically close

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: A; per DAN 923-1, “IF < 85 psig in U3 Main IA Receiver, THEN AO 3-4701-500, U3 SERV AIR TO INST AIR AUTO X-TIE VLV, OPENS. (Valve will remain open until reset at control box West of U3 Main IA Receiver.)
B & D are incorrect since system pressure rising will not cause the valve to reset.
C is incorrect since valve will auto open.
References: DAN 923-1, G-6, “U3 INST AIR PRESS LO.”

Direct/New/Modified: **New**
Memory/Comprehension-Analysis: **C/A**
Level (SRO/RO) **RO**
K/A: 300000 K4.02; Knowledge of (INSTRUMENT AIR SYSTEM) design feature and or interlocks which provide for the following: Cross over to other air systems.
Author: Walton
Exam Date: October 30, 2013

Question # 052

Unit 2 was operating at rated power, when Annunciator 923-1, E-4, “2A INST AIR DRYER TROUBLE” alarmed. The NSO dispatched an EO who reported back that the instrument air system pressure downstream of the dryer is 95 psig and the dryer towers are taking longer than 1 minute to cycle and the bypass valve is NOT open.

The reason for this is _____(1)______ and action to be taken is to ______(2)_______.

	(1)	(2)
a.	a plugged pre-filter	replace the pre-filter
b.	a plugged after-filter	replace the after-filter
c.	dryer switching failure	contact EMD to troubleshoot
d.	a low instrument air system pressure	open the dryer bypass valve

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: C, per the DAN, if system pressure is >60 psig, then problem is either a plugged pre-filter or a malfunctioning dryer tower.
A is incorrect since the bypass valve did not open.
B is incorrect, these conditions would not produce this alarm.
D is incorrect, system pressure is >60 psig. Would be correct if system pressure was <60 psig.

References: DAN 923-1, E-4, “2A INST AIR DRYER TROUBLE”
Direct/New/Modified: **New**
Memory/Comprehension-Analysis: **Memory**
Level (SRO/RO) **RO**
K/A: 300000 Instrument Air K5.13; Knowledge of the operational implications of the following concepts as they apply to the INSTRUMENT AIR SYSTEM: Filters.
Author: Walton
Exam Date: October 30, 2013

Question # 053

Unit 2 was operating at rated power with the CCSW system in the following lineup.

- 2A and 2B CCSW pumps were operating for a surveillance test.
- 2-1501-3A CCSW/LPCI Heat Exchanger outlet isolation valve is in MANUAL.

Unit 2 experienced a LOCA with a LPCI initiation. Assuming no operator intervention, the 2-1501-3A CCSW/LPCI Heat Exchanger outlet isolation valve will ______(1)________.

a. go full closed

b. will throttle to maintain 20 psid

c. go full open

d. will remain in present position

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: A. With a LPCI initiation, CCSW pumps will trip and hx'er discharge valve goes closed. Valve closes when both CCSW pumps are off to keep the discharge piping full.
B would be true if the CCSW pumps remained running on a LPCI initiation.
C is incorrect, valve goes full closed with CCSW pumps tripping on a LPCI initiation.
D is incorrect. valve is normally controlled in MANUAL from CR.
References: CCSW Lesson Plan, Section III.D.

Direct/New/Modified: **New**
Memory/Comprehension-Analysis: **C/A**
Level (SRO/RO) **RO**
K/A: 400000 Component Cooling water: K6.01; Knowledge of the effect that a loss or malfunction of the following will have on the CCWS: Valves.
Author: Walton
Exam Date: October 30, 2013

Question # 054

Unit 3 was in Mode 1 when Annunciator 923-1, G6; U3 INST AIR PRESS LOW alarmed. Equipment operators were dispatched to investigate but they were unable to:

- restart U3 instrument air compressors,
- open U3 service air to instrument air cross tie valve.
- open instrument air cross tie valve from Unit 2.

What is the status of the CRD system as Instrument Air pressure continues to drop to <45 psig? Assume operators have completed DGP 02-03, “Reactor Scram” actions for the CRD system to prevent RPV overfill.

a. Accumulators are charging, flow control valves have failed closed.

b. Accumulators are NOT charging, the flow control valves have failed closed.

c. Accumulators are NOT charging, the flow control valves have failed open.

d. Accumulators are charging, there is flow through the CRD cooling water header.

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: B: With air pressure at 45 psig, operators must scram the plant. Per DGP 02-03, Reactor Scram, operators must close the accumulator charging valve to prevent run out on the CRD pump. Per DOA 4700-01, Instrument Air Abnormal Procedure, Table A, Scram inlet and outlet valves fail open (rods scram) and CRD FCVs fail closed (minimal flow through the cooling water header due to designed leakage

References: DOA 4700-01, Instrument Air Abnormal Procedure, Section F; DAN 923, G6; Instrument Air System Lesson Plan.

Direct/New/Modified: **New**
Memory/Comprehension-Analysis: **C/A**
Level (SRO/RO) **RO**
K/A: 201001 A1.10, Ability to predict and/or monitor changes in parameters associated with operating the CONTROL ROD DRIVE HYDRAULIC SYSTEM controls including: CRD Cooling water flow.
Author: Walton
Exam Date: October 30, 2013

Question # 055

The unit is in Startup mode at the point of adding heat, increasing in pressure per DGP 01-01, "Unit Startup." Regarding control rod worth, as the reactor pressure is raised, control rod worth…

a. Increases with increasing pressure.

b. Decreases with increasing pressure.

c. Does not change as pressure increases.

d. Does not change during startup but increases when voids are formed.

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: A. From Reactor Theory, Chapter 5, Control Rods:
MODERATOR TEMPERATURE EFFECTS ON CONTROL ROD WORTH
For a neutron to be absorbed in a control rod, it must travel some distance to reach the control rod. As moderator temperature increases during a reactor heatup, the density of the moderator decreases. This decreased density yields longer slowing down lengths and longer thermal diffusion lengths. Hence, neutrons from deep inside a fuel bundle have a greater probability of interacting with the control rods. Rod worth increases as the moderator temperature increases.

In general, an increase in voids causes rod worth to decrease. In the voided regions of the core, increased slowing down length results in a considerable increase in the average energy of the neutron flux at the control rods. This causes a decrease in the average thermal flux at the rods, therefore, decreasing rod worth.

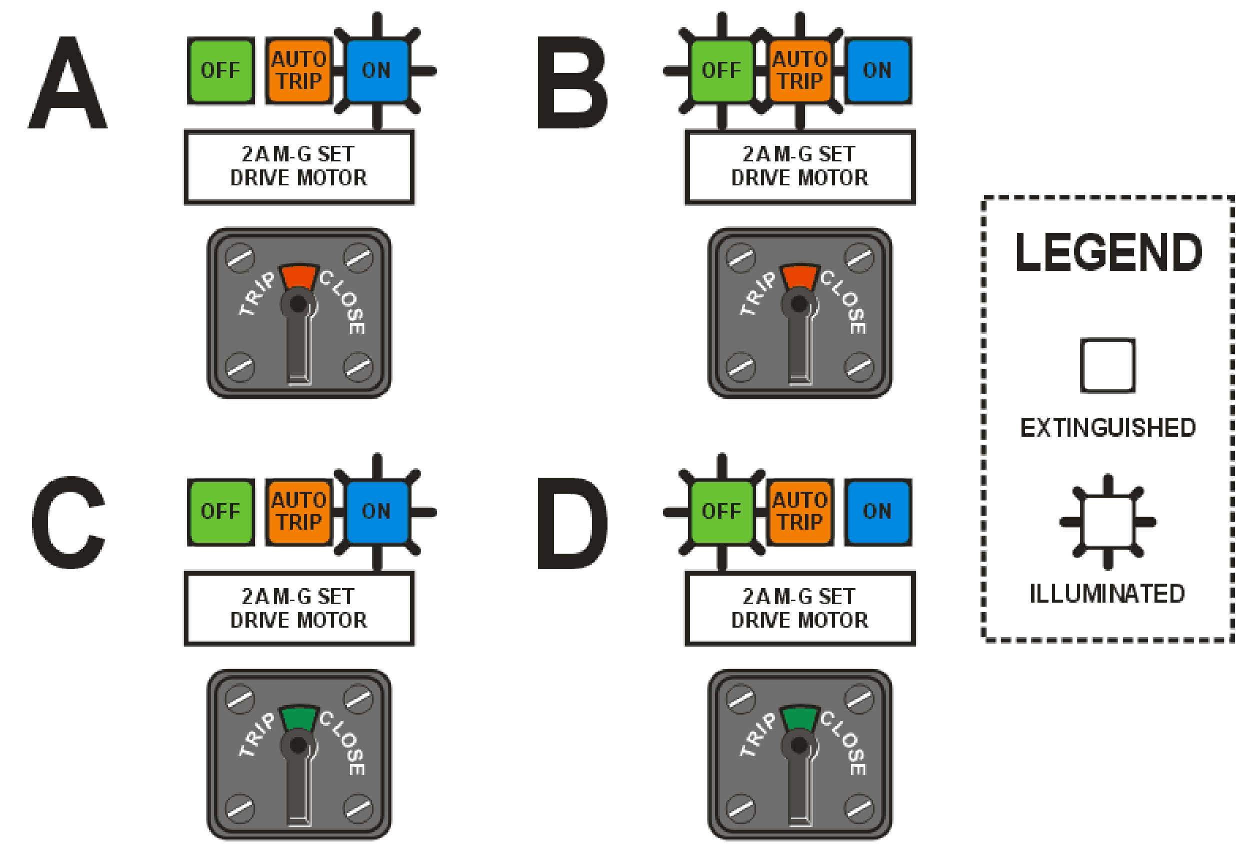
References: Reactor Theory, Chapter 5, Control Rods.

Direct/New/Modified: **New**
Memory/Comprehension-Analysis: **Memory**
Level (SRO/RO) **RO**
K/A: 201003 K5.06, Knowledge of the operational implications of the following concepts as they apply to CONTROL ROD AND DRIVE MECHANISM: How control rod with varies with moderator temperature and voids..
Author: Walton
Exam Date: October 30, 2013

Question # 056

Unit 2 was operating at rated power, when the Recirc system MO 2-202-5A, 2A PP DISCH VLV began drifting closed.

Which of the following would reflect the expected indications for the 2A M-G SET DRIVE MOTOR, when the discharge valve reaches 89% open?



Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: B.

References:
Direct/New/Modified: Direct from **Bank** Q14654
Memory/Comprehension-Analysis: **C/A**
Level (SRO/RO) **RO**
K/A: 202001, A3.04 Ability to monitor automatic operations of the RECIRCULATION SYSTEM including: Lights and alarms.
Author: Facility/Walton

Question # 057

Unit 2 is at full power operations with both recirculation pumps in Master Manual operation. The reactor operator sees both recirculation pump speed indicators lowering. Both reactor recirculation pump speeds settle to 68%. This runback was caused by a/an _____(1)______ runback. It occurred due to a ______(2)_______.

a. (1) anti-cavitation
   (2) recirculation pump discharge valve not being fully open

b. (1) anti-cavitation
   (2) loss of a feedwater/condensate pump

c. (1) feedwater level control
   (2) loss of a feedwater/condensate pump

d. (1) feedwater level control
   (2) recirculation pump discharge valve not being fully open

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: C. FWLC runback decelerates recirc speed at 7.5%/sec until speed equals 68%. It is caused by a low feedwater flow.
A & B are incorrect. Recirc speed decelerates at 2.5%/sec until pump speed equals 30%. This condition is caused by a recirc pump valve disch valve going closed or a loss of cond/feed pump.
D is incorrect. A feedwater level control runback is not generated from a recirc pump discharge valve going closed.
References: DOP 02-03, Reactor Recirculation Flow Control System Operation, Sect H.6.
DOA 0202-04, Reactor Recirculation Adjustable Speed Drive (ASD) Alarm Response, Sect F.

Direct/New/Modified: **New**
Memory/Comprehension-Analysis: **C/A**
Level (SRO/RO) **RO**
K/A: 202002; Knowledge of the RECIRCULATION FLOW CONTROL SYSTEM design features and/or interlocks which provide for the following: Minimum and maximum pump speed setpoints.
Author: Walton
Exam Date: October 30, 2013

Question # 058

Unit 2 is preparing for a startup with RPV pressure 150 psi. Additional plant conditions are:

- RWCU pump B is maintaining reactor water level,
- The auxiliary RWCU pump is OOS.

While draining RPV inventory through the RWCU system to the radwaste system, the NSO identified a loss of drain flow. An equipment operator reported FCV 2-1220, Drain Flow control valve, had closed. This condition was caused by a RWCU ___________.

a. drain line low flow condition

b. drain line high flow condition

c. drain line low pressure condition

d. demineralizer inlet low flow condition

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: C. "Upon sensing a low upstream pressure of 5 psig, the drain flow regulator closes. This is to prevent draining the system high points (demins) which are above the normal reactor water level, when the reactor is not at pressure and the Aux pump is not operating."
A & D are incorrect since an RWCU low flow condition (demin inlet or drain line) has no automatic actions, DAN 902-4, B-11.
B. Is incorrect. High flow condition will not cause valve to close.
References: RWCU Lesson Plan, Section III.G. DAN 902-4, B-11.

Direct/New/Modified: **New**
Memory/Comprehension-Analysis: **C/A**
Level (SRO/RO) **RO**
K/A: 204000 A4.03: Ability to manually operate and/or monitor in the control room: RWCU drain flow regulator.
Author: Walton
Exam Date: October 30, 2013

Question # 059

During a TIP scan of the core, the _____(1)______ computer gathers the neutron flux profile data from the TIP detector when the detector is being ____(2)_____ the core.

	(1)	(2)
a.	Plant process	withdrawn from
b.	Plant process	inserted into
c.	WCMS	withdrawn from
d.	WCMS	inserted into

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: A. (See below)

References: From TIPS Lesson Plan: At equilibrium xenon and steady state power, TIP scans are performed. The process computer gathers TIP data, LPRM readings and APRM readings for each TIP trace. This information is transferred from the process computer to the POWERPLEX computer. The computer compares the POWERPLEX data, after being machine normalized and full power adjusted, to the LPRM readings and calculates GAFs that will make the LPRMs correspond to the given TIP data. During LPRM calibration, the common channel (channel 10) is to be traversed by each TIP machine. The computer normalizes (averages) the five scans to cross-calibrate the TIPs before using the TIP outputs to calibrate the LPRMs.Scan - The withdrawal of a detector from the core top to the core bottom with data gathering of the neutron flux profile.

Direct/New/Modified: **New**
Memory/Comprehension-Analysis: **Memory**
Level (SRO/RO) **RO**
K/A: 215001 K1.02, Knowledge of the physical connections and/or cause-effect relationships between TRAVERSING IN-CORE PROBE and the following: Process computer.
Author: Walton
Exam Date: October 30, 2013

Question # 060

Unit 2 is at 35% power during a startup. The operator selects control rod H-8. At that time, the following annunciators alarm:

- 902-5, A-7, RBM High/Inop
- 902-5, C-7, RBM Downscale
- 902-5, C-3, Rod Out Block

On the 902-5 panel, the DWNSCL and INOP lights for RBM 8 are also energized, and half of the RPS white lights on the 902-5 panel are de-energized.

This condition was caused by ___________.

a. RPS Bus A de-energizing

b. RPS Bus B de-energizing

c. A peripheral rod was selected for withdraw

d. An LPRM has failed low resulting in <50% of assigned inputs to an APRM

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: B RBM Channel 8 is powered from RPS Bus B. A loss of RPS B will also result in loss of power to APRM's 4, 5 & 6 causing a ½ scram.
A is incorrect, RBM Channel 7 is powered from RPS Bus A
C is incorrect, the rod selected is a center rod. Selection of an edge rod would result in an automatic bypass of the RBM. This condition would result in BYPASS being illuminated on 902-5 panel. This condition would NOT cause a ½ scram.
D is incorrect since this condition would not cause a ½ scram.
References: Rod Block Monitor Lesson Plan. APRM Lesson Plan, Sect II.G, Power Supplies.

Direct/New/Modified: **New**
Memory/Comprehension-Analysis: **C/A**
Level (SRO/RO) **RO**
K/A: 215002 K2.01: Knowledge of electrical power supplies to the following: RBM channels.
Author: Walton
Exam Date: October 30, 2013

Question # 061

Unit 2 was operating at rated power with Division I LPCI in suppression pool cooling after completion of a HPCI surveillance test. At this time, Bus 23-1 de-energizes due to a fault and can NOT be reenergized. What effect (if any) would this condition have on LPCI in suppression pool cooling mode?

a. A loss of CCSW cooling occurs.

b. A loss of suppression pool inventory occurs.

c. A loss of LPCI system pressure and flow occurs.

d. NO change in LPCI suppression pool cooling system parameters.

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER C is true, LPCI pumps powered from 23-1 bus
A is false, would be true if Bus 23 de-energized.
B is false, LPCI pumps powered by Bus 23-1, valves are MOV's – fails as is. No high pressure conditions expected, no relief valve lifting expected, no system drain valves would open.
CCSW pumps powered by Bus 23. MOV's fail as is.
D: is false. LPCI pumps and MOVs (fail as is) continue to operate, but no CCSW cooling.
References: Dan 902-5 E-5 (Groups 2 PCIS list) & D-5 (Group 3 PCIS list). DOP 6700-18, Attachment A, MCC 28-1 loads. LPCI Lesson Plan
Direct/New/Modified: **New**
Memory/Comprehension-Analysis: **C/A**
Level (SRO/RO) **RO**
K/A: 219000 K6.01: Knowledge of the effect that a loss or malfunction of the following will have on the RHR/LPCI: TORUS/SUPPRESSON POOL COOLING MODE: AC Power.
Author: Walton
Exam Date: October 30, 2013

Question # 062

Unit 3 is in Mode 1 with the main generator connected to the grid at 30% power. Recirculation flow is 62%. The following plant conditions exist:

- 2 condensate/condensate booster pumps (C/CBP) are in operation,
- Feedwater pump 'A' is in operation,
- Feedwater heaters have been placed in operation.

Under these conditions, the extraction steam MOVs from the LP turbines all go closed and the extraction steam bypass valves open. This results in ____(1)_____. To address this issue, the operators will _____(2)_____.

a. (1) LP Feedwater heater flash tank levels rising
   (2) ensure proper operation of heater drain bypass AOVs

b. (1) MSDT level rising
   (2) ensure proper operation of the MSDT spill valves

c. (1) an unplanned entry into the Unstable Power/Flow region
   (2) reduce reactor power by inserting CRAM rods

d. (1) feedwater temperature lowering
   (2) maintain previous power level by reducing reactor recirculation flow

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: D. Correct action per DGA-07 for loss of FW heating.
A is incorrect since levels in the flash tanks would be expected to lower – loss of input!
B is incorrect, since MSDT level should be unaffected by this event.
C is incorrect since power/flow point is to the right of the unstable region. Correct action per DOA 500-01.
References: Feedwater Heating Lesson Plan. DGA -07, "Unexpected Reactivity Change," Step D.3. DOA 500-01, "Inadvertent Entry into the Unstable Power/Flow Region," Step D.1. Feedwater Heater Extraction Steam Flow Drawing, M-13.
Direct/New/Modified: **New**
Memory/Comprehension-Analysis: **C/A**
Level (SRO/RO) **RO**
K/A: 256000 K1.06, Knowledge of the physical connections and/or cause-effect relationships between REACTOR CONDENSATE SYSTEM and the following: Extraction steam.
Author: Walton
Exam Date: October 30, 2013

Question # 063

With the Unit 2 radwaste system in a normal lineup, which of the following sumps would be directly affected by a piping obstruction that prevents all inputs to the Radwaste Waste Collector Tank (WCT)?

a. U2 Drywell Floor Drain (DWFDS)

b. U2 Turbine Building Floor Drain (TBFDS)

c. U2 East Reactor Building Floor Drain (RBFDS)

d. U2 West Reactor Building Floor Drain (RBFDS)

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: A All Drywell equipment and floor drain sumps pump to the WCT. The DWFDS are pumped to the WCT because it is high quality water. The Turbine Building Equipment Drain Sumps pump to the Turbine Building Floor Drain Sumps which are then pumped to the Floor Drain Collector Tank. All Reactor Building Floor Drain sumps pump to the Floor Drain Collector Tank.
References: DOP 2000-24.
Direct/New/Modified: Direct Q13361 **Bank**
Memory/Comprehension-Analysis: **Memory**
Level (SRO/RO) **RO**
K/A: 268000 K3.04, Knowledge of the effect that a loss or malfunction of the RADWASTE will have on the following: Drain sumps.
Author: Dreden/Walton
Exam Date: October 30, 2013

Question # 064

Unit 3 is in Startup Mode at 300 psig with the following plant conditions:

- The condenser vacuum pump is running
- The ‘B’ SJAE/Off Gas recombiner train has been placed in service.
- Condenser vacuum is 15 inches HG and rising.
- RP personnel report rising radiation levels from the main steam lines and off gas system.

Chemistry personnel later confirm the presence of a fuel element defect. To minimize the operational effects, operators will ____(1)_____ because ____(2)_______.

a. (1) de-energize the condenser vacuum pump
(2) it is a source of an unmonitored release path

b. (1) immediately isolate main control room ventilation and start the air filtration unit
(2) it could result in control room personnel unnecessarily being exposed to radiation/contamination

c. (1) de-energize the condenser vacuum pump
(2) it will trip on main steam line high radiation signal

d. (1) reset the off gas radiation monitors
(2) the offgas system will automatically isolate in <5 minutes

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: C
A is incorrect, the condenser vacuum pump discharges to the stack, a monitored release path.
B is incorrect. Must perform these actions within 40 minutes of receiving an RBX system hi hi radiation signal
D is incorrect since either MSL or OG system radiation alarms will set an OG timer circuit that will isolate OG discharge in 15 minutes.
References: DGA-16, Coolant High Activity/Fuel Element Failure, D.3 - D.5.
Direct/New/Modified: **New**
Memory/Comprehension-Analysis: **C/A**
Level (SRO/RO) **RO**
K/A: 271000A2.03; Ability to (a) Predict the impacts of the following on the OFFGAS SYSTEM and (b) based on those predictions, use procedures to correct, control or mitigate the consequences of those abnormal conditions or operations: Main Steamline High radiation.
Author: Walton
Exam Date: October 30, 2013

Question # 065

Unit 2 was operating at near rated power with MCC 28-2 out of service, when the following annunciators are received:

- 902-8 B-8, 120/240 ESS BUS VOLT LO.
- 902-8 E-10, 120/240 ESS BUS ON EMERG SPLY.

An impact of this transient is that the _______ rad monitor(s) become(s) de-energized.

a. Channel 'A' Off Gas

b. Channel 'A' Refuel Floor

c. 2B and 2D Main Steam Line (MSL)

d. Outboard channel 'B' Reactor Building and Fuel Pool high rad aux relays.

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: C: With MCC 28-2 OOS, the RESERVE power supply to the U2 ESS bus is lost. The two alarms that are received indicate that the ESS Bus ABT has transferred (irregardless of the reserve power supply being energized or not). These events remove ALL power from the ESS bus, which is the power supply for the B and D MSL rad monitors.
A & B are incorrect: Channel A Off Gas rad monitor and Channel A Refuel Floor rad monitor are powered from RPS Bus A, which is still energized (from MCC 29-2 via the B RPS MG Set - cross divisional power).
D is incorrect. Outboard channel B RBX & fuel pool high rad aux relays are powered from the Inst Bus (which is still energized from MCC 25-2).

Direct/New/Modified: **Bank** (Q23971)
Memory/Comprehension-Analysis: **C/A**
Level (SRO/RO) **RO** K/A: 27200 Radiation Monitoring, G2.2.37: Ability to determine operability and/or availability of safety related equipment.
Author: Licensee/Walton
Exam Date: October 30, 2013

Question # 066

Unit 2 TBCCW system has been lost and cannot be restored.

Which of the following actions are required?

a. Verify Stator Cooling runback occurs, due to high temperatures.

b. Verify the CRD pump oil coolers AUTO transfer to the Service Water system.

c. Announce loss of Service Air on the PA system to warn personnel who may be using U2 Service Air.

d. Reduce Recirc pump speeds to maintain MG Set motor winding temperatures below procedural limits.

ANSWER: c
a., b, & d: are incorrect. ALL of these situations require a PA announcement.
Reference: DOA 4600-01, DOP 3800-01, Dan 923-1, C-5.

Direct/New/Modified: **BANK**
Memory/Comprehension-Analysis: **Memory**
Level **RO**
History: 2007 NRC, 2009 Cert
Explanation: A loss of TBCCW causes a loss of cooling to, and the subsequent loss of, the Service Air System. Per the referenced DOA, a PA announcement must be made to warn personnel using Service Air as breathing air. Verify runback is not required since the Stator Water coolers, even though in the Turbine Building, are not cooled by TBCCW (common mis-conception). The CRD coolers are normally cooled by TBCCW and backed up from Service Water, but this is a manual operation only (not auto). Reduce Recirc pump speeds to maintain MG set temps is not required since the MG sets, even htough in the Turbine Building are not cooled by TBCCW (common mis-conception).
K/A: G 2.1.14: Knowledge of criteria for conditions that require plant-wide announcements.
Author: BANK
Exam Date: October 30, 2013

Question # 067

Post-maintenance testing of the SBGT system is in progress. IF an automatic initiation of SBGT occurs, what MANUAL action(s) is/are expected to be taken?

a. INITIATE Reactor Building Isolation.

b. PLACE the control switch for the non-running train to A(B) OFF.

c. PLACE the control switch for the non-running train to A(B) PRI.

d. CLOSE both units' Reactor Building Ventilation Isolation Dampers.

ANSWER: c: PLACE the control switch for the non-running train to A(B) PRI.
b. incorrect position.
a & d. are VERIFICATIONS – manual action is not expected.
References: None provided. Limitations and Actions from DOS 7500-02.

Direct/New/Modified: **New**
Memory/Comprehension-Analysis: **C/A**
Level (**RO**)
K/A: G 2.1.32: Ability to explain and apply system limits and precautions.
Author: C. Zoia
Exam Date: October 30, 2013

Question # 068

Chemistry has completed sampling the U2 SBLC tank with the following results:

- Tank volume: 3500 gallons
- Tank temperature: 111°F
- Sodium Pentaborate concentration: 14.5%

You review Tech Spec Figures 3.1.7-1 and 3.1.7-2 (provided) and determine __________ in order to comply with these curves.

a. no changes are required

b. tank temperature must be lowered to $\leq$110°F.

c. tank level must be lowered by >400 gallons.

d. sodium pentaborate concentration must be raised to 15%

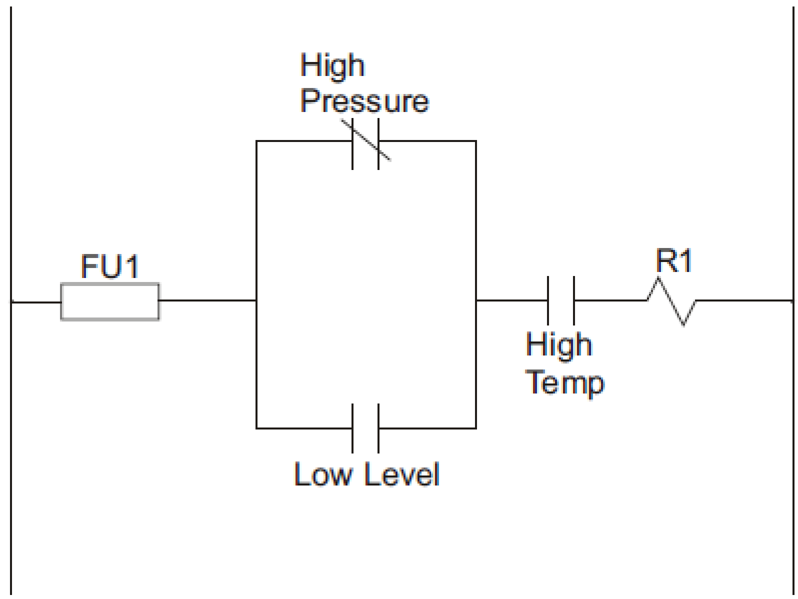
ANSWER: B. Per Fig 3.1.7-1, must lower tank temperature to <110°F.
A is incorrect. Presently the SBLC is inoperable.
C & D are incorrect. Lowering tank level by 400 gallons or raising pentaborate concentration will not meet reqts.

Provide Tech Spec Figures 3.1.7-1 and 3.1.7-2

Direct/New/Modified: **New**
Memory/Comprehension-Analysis: **C/A**
Level (**RO**)
K/A: G 2.1.34: Knowledge of primary and secondary chemistry limits.
Author: Walton
Exam Date: October 30, 2013

Question # 069

Assume in the drawing below all contacts are shown in the de-energized (shelf) state.
Which of the following conditions will result in Relay R1 energizing?
The low level condition is _____(1)______, OR high pressure condition is ____(2)_____ AND high temperature condition is _____(3)______.



	(1)	(2)	(3)
a.	clear	clear	clear.
b.	actuated	clear	actuated.
c.	clear	actuated	actuated.
d.	actuated	clear	clear.

ANSWER: B Either a high pressure clear condition OR a Low Level actuated condition AND a high temperature actuated condition will cause the relay to energize.
Direct/New/Modified: **New**
Memory/Comprehension-Analysis: **C/A**
Level (**RO**)
K/A: G 2.2.15: Ability to determine the expected plant configuration using design and configuration control documentation, such as drawings line-ups, tag-outs, etc..
Author: Walton
Exam Date: 10/30/2013

Question # 070

With Unit 2 in Mode 2, Operational Leakage (Unidentified - Technical Specification LCO 3.4.4) is currently 5.2 gallons per minute (gpm), but has doubled (2.6 gpm increase) within the past 24 hours.

In order to change Modes to Mode 1, Unidentified Operational Leakage must be:

a. DECREASED to $\leq$5 gpm prior to the Mode change ONLY.

b. DECREASED to a 2 gpm increase within 24 hours prior to the Mode change ONLY.

c. MAINTAINED at current values with TS Actions being met.

d. DECREASED to a 2 gpm increase within 24 hours AND $\leq$5 gpm prior to the Mode change.

ANSWER: D. DECREASE unidentified leakage to $\leq$5 gpm prior to the Mode change and must must decrease 2 gpm increase within 24 hours.
All other answers do not meet TS requirements OR apply extra requirements for entry into Mode 1.

References: None provided. See Introduction to TS Lesson DRE299LN001 and TS 3.4.4.
Direct/New/Modified: **New**
Memory/Comprehension-Analysis: **C/A**
Level (**RO**)
K/A: G 2.2.22: Knowledge of LCOs and Safety Limits
Author: C. Zoia
Exam Date: October 30, 2013

Question # 071

Unit 2 experienced a failure to scram from a turbine trip at 98% power. The NSO successfully initiated a manual scram approximately 4 seconds after the turbine trip. Post trip analysis revealed the following data for the event:

- Peak Reactor Power 116%
- Peak Reactor Pressure 1255 psig
- Minimum Vessel Level -124 inches
- Most Limiting MCPR 1.08

Which one of the following Safety Limits was exceeded?

a. MCPR

b. RPV Level

c. Thermal Power

d. RPV Pressure

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: A

Reference: T.S. Section 2 Safety Limits
Comments: With the successful scram, all parameters stayed within safety limits except MCPR which dropped to 1.08, below the limit of 1.12 for dual recirc loop operation.
Direct/New/Modified: Direct Q13056 **BANK**
Memory/Comprehension-Analysis: **Memory**
Level (SRO/RO) **RO**
K/A: G 2.2.22; Knowledge of limiting conditions for operations and safety limits.
Author: Facility/Walton
Exam Date: October 30, 2013

Question # 072

The station is performing a discharge to the river from the waste surge tank. During this discharge, the rad waste panel operator informs you of the following annunciator he just received:

- Annunciator 2223-6 A-12, 2/3 RADWASTE DISCHARGE HIGH RADIATION

Assuming the off-stream liquid effluent monitor is operable, what are the expected actions for this condition?

A grab sample _____(1)_____ from the discharge monitor sample loop and the discharge ______(2)_______.

	(1)	(2)
a.	must be manually drawn	stops automatically
b.	will automatically be drawn	stops automatically
c.	must be manually drawn	must be stopped manually
d.	will automatically be drawn	must be stopped manually

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: D
A, B and C are incorrect since according to the reference, a grab sample automatically is drawn and the discharge must be stopped manually..
References: DAN 2223-6: A-12, 2/3 Radwaste Discharge High Radiation annunciator

Direct/New/Modified: **New**
Memory/Comprehension-Analysis: **C/A**
Level (SRO/RO) **RO**
K/A: G 2.3.5: Ability to use radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.
Author: Walton
Exam Date: October 30, 2013

Question # 073

You are assigned to an LLRT during the outage. You have received 70 mrem exposure for the day but have one last task to perform. You are to operate MU-16 (see attached map, valve operator is on top of valve). Your RWP allows a maximum dose of 150 mrem/day and maximum dose rate of 100 mrem/hr. Using ALARA principles, if you operate MU-16 for a total of 10 minutes, what is your MAXIMUM REMAINING dose at the end of this task?

a. 65 mrem

b. 70 mrem

c. 75 mrem

d. 80 mrem

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: B Allowed 150 mrem (limit) – 70 mrem (previous exposure) = 80 mrem (allowed). 60 mrem/hr x 10min/60min/hr = 10 mrem. 80 mrem – 10 mrem = 70 mrem.
A is incorrect. If you operate valve from 90 mrem/hr side: 90 mrem allowed 10/60 = 15 mrem. 80-15 = 65 mrem.
C is incorrect. Operate wrong valve (MU26) in 30 mr/hr field. 30 mrem X 10/60 = 5 mrem. 80 mrem – 5 = 75 mrem.
D is incorrect. 80 mrem means you forgot to subtract the dose accumulated from the job.
References:

Direct/New/Modified: **New**
Memory/Comprehension-Analysis: **C/A**
Level (SRO/RO) **RO**
K/A: G 2.3.7; Ability to comply with radiation work permit requirements during normal or abnormal conditions.
Author: Walton
Exam Date: October 30, 2013

Question # 074

Unit 3 was operating at 75% of full rated power with HPCI operating for a surveillance test and torus cooling in operation when a scram occurred. The plant conditions are now:

- One rod at position 04, all other rods at position 00;
- RPV water level lowered to +6 inches and is recovering to normal post-scram levels;
- RPV pressure remained within a band of 980 psig and 1030 psig;
- Drywell pressure is at 1.4 psig and slowly rising;
- Torus temperature is 93°F and steady.

All other parameters are as expected post-scram. As the NSO you perform scram choreography per DGP 02-03, "Reactor Scram." Based on the given conditions, the operating crew is required to…

a. enter DGP 02-03 ONLY

b. enter DGP 02-03 and enter DEOP 100, "RPV Control."

c. enter DGP 02-03 and enter DEOP 400-5, "Failure to Scram."

d. enter DGP 02-03 and enter DEOP 200, "Primary Containment Control."

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: B Meet entry conditions for entry into DEOP 100 and DGP 02-03.
A is incorrect, since one rod greater than 2 requires entry into DEOP 100.
C is incorrect, One rod at position 04 will still cause the reactor to remain shutdown under all conditions.
D is incorrect since torus temperature is <95°F entry condition to DEOP 200.

References: DEOP 100, 400-5, 200, DGP 02-03.
Direct/New/Modified: **New**
Memory/Comprehension-Analysis: **C/A**
Level (SRO/RO) **RO**
K/A: G.2.4.8: Knowledge of how abnormal operating procedures are used in conjunction with EOPs.
Author: Walton
Exam Date: October 30, 2013

Question # 075

In accordance with DSSP 0100-CR, “Hot Shutdown Procedure – Control Room Evacuation,” as Unit 2 NSO, your responsibilities outside the control room include ____(1)_____. These actions are performed to ______(2)_______.

a. (1) de-energizing various non-vital 250 VDC 2A and 2B loads
   (2) to prevent a hot short from causing multiple spurious actuations

b. (1) removing ERV and Target Rock valves control power
   (2) prevent spurious shorts from opening relief valves

c. (1) verifying Isolation Condenser valves 2-1301-1 and 2-1301-4 are open
   (2) ensure decay heat is being removed from the reactor

d. (1) verifying breakers tripped on Buses 21 and 22 and pulling their breaker closed fuses
   (2) ensure that a bus ground fault does not effect supply transformers

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: A. As required by Attachment A of DSSP 100-CR
B & D are incorrect since these actions are performed by Operator 2
C is incorrect since these actions are performed by Operator 1.

References: DSSP 100-CR, Hot Shutdown – Control Room Evacuation.
Direct/New/Modified: **New**
Memory/Comprehension-Analysis: **C/A**
Level (SRO/RO) **RO**
K/A: G2.4.34. Knowledge of RO Tasks performed outside the main control room during an emergency and the resultant operational effects.
Author: Walton
Exam Date: October 30, 2013

Question # 076

Unit 3 is operating at full power (on 100% FCL) when the 3B Recirculation pump trips. The SRO enters DOA 0202-01, “Recirc Pump Trip - One or Both Pumps.” The SRO orders _______(1)_______ and enters ______(2)_______.

a. (1) Scram the reactor
   (2) DGP 02-03, Reactor Scram

b. (1) Lower Recirc pump speed to <77% AND insert CRAM Rods to lower reactor power to between 25% and 30%
   (2) DGP 03-03, Single Recirculation Loop Operation

c. (1) Reduce remaining recirc pump speed to <77%
   (2) DOA 0202-01, Recirculation Pump Trip – One or Both Pumps.

d. (1) Insert CRAM Rods to lower reactor power to between 25% and 30%
   (2) DGP 03-03, Single Recirculation Loop Operation.

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: D is correct. With the reactor at full power (100% FCL and 100% power), using DOA 0202-01 Hard Card, must insert CRAM rods to lower Rx power to <30% enter DOA 0202-01
A is incorrect. Conditions do not require an immediate reactor scram!
B & C are incorrect. FCL is NOT less than 90%, so inserting CRAM rods and lower recirc pump speed to <77% is not applicable.
References: Provide DOA 0202-01, Recirculation pump Trip.

Direct/New/Modified: **New**
Memory/Comprehension-Analysis: **C/A**
Level (SRO/RO) **SRO** only; Criteria II.E
K/A: 295001 PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION.
G2.1.20; Ability to interpret and execute procedure steps.
Author: Walton
Exam Date: 10/29/2013

Question # 077

Unit 3 was operating with the MODE switch in RUN and with all APRMs reading between 12 to 14% power, when an automatic scram signal was generated.

The following conditions exist after the scram signal was received:

- RPV Pressure is 1000 psig.
- Drywell Pressure is 1.19 psig.
- IRMs were inserted and are on range 6.
- RPV water level dropped to +15 inches and is currently at +30 inches.
- All East Bank Control Rods, that were withdrawn, inserted to position 00.
- All West Bank Control Rods, that were withdrawn, inserted to position 18.
- All eight Scram Solenoid Group Lights (A1 through A4 and B1 through B4) extinguished.

Which of the following lists the order of procedures that the Unit Supervisor entered OR directed the crew to perform?

1) DGP 2-3, REACTOR SCRAM
2) DEOP 100, RPV CONTROL
3) DEOP 200-1, PRIMARY CONTAINMENT CONTROL
4) DEOP 400-5, FAILURE TO SCRAM
5) DEOP 500-5, ALTERNATE INSERTION OF CONTROL RODS

a. Enter #2;
then enter #4;
then enter #5

b. Enter AND exit #2;
then enter #4;
then enter #1

c. Enter #1;
then enter AND exit #2;
then enter #4;
then enter #5

d. Enter #1;
then enter AND exit #2;
then enter #3;
then enter #4;
then enter #5

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: C. With the conditions provided post-scram, the Unit is in a hydraulic ATWS condition (all scram pilot lights extinguished with rods not full in). Based on the post-scram data provided, a DEOP 100 Entry condition does not exist (RPV Level/Pressure, DW Pressure, Reactor Power), therefore, the Unit Supervisor should direct entry into DGP 2-3, Reactor Scram. Per DGP 2-3, since all rods are not at 02 or 00, DEOP 100 is entered. Upon entering DEOP 100, the Unit Supervisor exits DEOP 100 and enter DEOP 400-5, Failure to Scram (reactor won't stay shutdown under all conditions). Since boron is injecting the override in 400-5 is not met and rods should be inserted per DEOP 500-5 using repeated scram/resets.
References: DGP 2-3, Reactor Scram. DEOP 100, RPV Control. DEOP 400-5, Failure to Scram. DEOP 500-5, Alternate Insertion of Control Rods.

Direct/New/Modified: Direct from **Bank**, Q22001
Memory/Comprehension-Analysis: **C/A**
Level (SRO/RO) **SRO** Only, Criteria II.E
K/A: 295006 AA2.02; Ability to determine and/or interpret the following as they apply to SCRAM: Control rod position.
Author: Dresden/Walton
Exam Date: October 30, 2013

Question # 078

During performance of DSSP 100-CR, “Hot Shutdown Procedure – Control Room Abandonment,” what MINIMUM responsibilities does the SRO have when actions may conflict with Technical Specifications?

a. Do NOT allow operators to perform actions that violate Technical Specifications.

b. Allow operators to violate Technical Specifications, but only after another SRO concurs.

c. Must implement Title 10 CFR 50.54.x prior to allowing any violations of Technical Specifications.

d. Must implement Title 10 CFR 50.54.x AND must get Plant Manager concurrence prior to allowing operations in violation of Technical Specifications.

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: C is correct per DSSP 100-CR, Section F.5.
Distractors A & B do not comply with DSSP 100-CR
Distractor D is only partially correct, do NOT need PM concurrence.
References: DSSP 100-CR, Section F.5.

Direct/New/Modified: **New**
Memory/Comprehension-Analysis: **Memory**
Level (SRO/RO) **SRO** Only, Criteria II.A
K/A: 295016 Control Room Abandonment, G2.2.40; Ability to apply Technical Specifications for a system.
Author: Walton
Exam Date: October 30, 2013

Question # 079

Unit 3 was operating at full power when an event occurred. Later during the accident, the following plant conditions exist:

- HPCI pump is out of service
- LPCI pumps running but NOT injecting to the RPV.
- “A” & “B” Trains of Core Spray are injecting into the RPV at 4000 gpm each.
- RPV Water level is -45 inches and steady.
- Drywell pressure is 7 psig and rising slowly.
- Drywell temperature is 240°F and rising slowly.
- Torus Bulk Temperature is 200°F and steady.
- Torus Level is 14 feet and steady.
- Torus Bottom Pressure is 10.2 psig and rising slowly.

The SRO is performing steps from DEOP 100, “RPV Control” and DEOP 200-01, “Primary Containment Control.” To spray the drywell, AND to prevent ECCS pump cavitation the SRO orders…

a. spray with one LPCI pump with flow <2750 gpm.

b. reduce CS flow to <2750 gpm, then inject with one LPCI pump at 4000 gpm.

c. the drywell can NOT be sprayed without cavitating existing ECCS pump flow.

d. secure one CS pump before injecting with one LPCI pump, keep LPCI flow <4000 gpm.

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: C is correct. The way this question is worded and that drywell spray cannot be throttled, means that there is no condition that can exist without cavitating the ECCS pumps.
A; Per DEOP 200-01, Primary Containment Pressure Leg, Torus bottom pressure 10.2 psig on Fig W. With torus bulk temp at 200°F must keep ECCS flow <10750 gpm. With 8000 gpm CS flow, have 2750 gpm flow remaining. However since drywell spray cannot be throttled, this answer is incorrect.
B is incorrect since do not want to divert any CS flow from core. Would be correct if RPV level was rising.
D is incorrect since do not want to secure any CS flow injection to core since level is just maintaining.
References: Provide DEOP 200-01, Primary Containment Control
Direct/New/Modified: **New**
Memory/Comprehension-Analysis: **C/A**
Level (SRO/RO) **SRO** Only Criteria II.E

K/A: 295024 High Drywell Pressure, EA2.06 Ability to determine and/or interpret the following as they apply to HIGH DRYWELL PRESSURE: Suppression Pool temperature.
Author: Walton
Exam Date: 10/29/2013

Question # 080

A transient from rated power has caused Unit 2 RPV pressure to exceed DEOP 100 entry conditions for reactor pressure. The SRO has entered into DEOP 100, "RPV Control." The plant conditions are as follows:

- Drywell pressure 1.1 psig and steady,
- Reactor power below IRM Range 6 and lowering,
- RPV Pressure is cycling between 1100 – 1200 psig.

What method would the SRO FIRST direct to control RPV pressure?

a. Start HPCI in pressure control mode.

b. Initiate IC and open ADSVs to lower RPV pressure to 945 psig.

c. Stabilize pressure below 1060 psig using turbine bypass valves.

d. Reduce RPV pressure to <920 psig using the turbine bypass valves opening jack.

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: B. is correct per pressure leg of DEOP 100, RPV Control.
A would be correct but not the FIRST direction!
C would be correct for ADSV's NOT cycling
D is not allowed to be performed by operators as shown in Note for Step E.29 in DGP 02-03, "Reactor Scram." May be used for later plant cooldown.

References: DEOP 100, "RPV Control." DGP 02-03, Reactor Scram, Step E.21 & E.29 Note.

Direct/New/Modified: **New**
Memory/Comprehension-Analysis: **C/A**
Level (SRO/RO) **SRO** Only, Criteria II.E
K/A: High Reactor Pressure, G2.4.20; Knowledge of the operational implications of EOP warnings, cautions, and notes.
Author: Walton
Exam Date: October 30, 2013

Question # 081

An ATWS occurred on Unit 3. The SRO has entered into DEOP 400-5, “Failure to Scram” and has directed initiation of SBLC. Which of the following conditions would DEOP 400-5 direct the SRO to stop injection of SBLC?

a. Standby Liquid Control tank level reaches 36%.

b. Standby Liquid Control tank level reaches 8%.

c. Neutron counts are less than IRM range 7.

d. When SBLC tank level indicates that volume and concentration limits of Tech Spec Figures 3.1.7-1 and 3.1.7-2 have been injected.

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: B per DEOP 400-5
A is incorrect. SBLC tank level of 36% is used in DEOP 400-5 pressure control leg. It is not a criteria to stop injection of SBLC.
C is incorrect. IF neutron counts less than range 7 AND boron has NOT been initiated, SRO directed to leave DEOP 400-5 and returning to RPV Control DEOP.
D is incorrect. This is the design basis per Tech Spec 3.1.7, not the DEOP 400-5 requirements.
References: DEOP 400-5. Tech Spec 3.1.7 Basis.
Direct/New/Modified: **New**
Memory/Comprehension-Analysis: **Memory**
Level (SRO/RO) **SRO** Only, Criteria II.E since information is contained in DEOP 400-5 procedure steps and not an EOP entry level condition.
K/A: 29035 EA2.03, Ability to determine and/or interpret the following as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN: SBLC tank level.
Author: Walton
Exam Date: October 30, 2013

Question # 082

The radwaste panel operator reported that the Floor Drain Sample Tank was inadvertently sent to the river in lieu of the Waste Collector tank due to an equipment lineup error. Chemistry reported 1000 liters of waste water containing 70 uCi was discharged to the river over a 1 hour period. The isotopic analysis of the discharge was 20% H-3 and 80% Co-60.

(1) Is this reportable?
(2) If so, what are the reporting requirements?

i. 60-day report due per Rad 1.4, Liquid Effluent Release. Curie content exceeded 20 times the limits of 10 CFR 20, Appendix B, Table 2, Column 2, in an unrestricted area.
ii. 30-day report due per Rad 1.4, Liquid Effluent Release. Curie content exceeded 10 times the limits of 10 CFR 20, Appendix B, Table 2, Column 2, in an unrestricted area.

	(1)	(2)
a.	No, NOT Reportable.	N/A
b.	Yes, Reportable per 50.73(a)(2)(viii)(B) ONLY.	i ONLY.
c.	Yes, Reportable per 20.2203(a)(3) ONLY.	ii ONLY.
d.	Yes, Reportable per 50.73(a)(2)(viii)(B) and 20.2203(a)(3).	i, AND ii.

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: C: 70 uCi. 1000 liters = 1E6 milliliters.
70uCi/1E6 milliliters = 7E-5 uCi/ml. 80% is Co-60 or 5.6E-5 uCi/ml Co-60. Discharge is >3E-6 limit by about 18X, so is reportable per ii. Tritium discharged is well below the limit.
References: TRM 3.7d, Liquid Holdup Tanks and Basis document. Rad Waste Tank Locations Drawing 286LN002-004. Floor Drain System Lesson Plan.
References Provided: 10 CFR 20, Appendix B; Reportability Manual, Section RAD 1.4.
Direct/New/Modified: **New**
Memory/Comprehension-Analysis: **C/A**
Level (SRO/RO): **SRO** Only, Criteria II.A, Reportability
K/A: 295038 High Offsite Release Rate G2.4.30: Knowledge of events related to system operation/status that must be reported to internal organizations or external agencies such as the State, the NRC, or the transmission system operator.
Author: Walton
Exam Date: October 30, 2013

Question # 083

Unit 2 is in Refuel Mode with water level in the refuel cavity 3 inches below the fuel pool ventilation ducts being maintained by the weir walls. Other plant conditions are as follows:

- Shutdown Cooling Pump “A” is in Shutdown Cooling Mode
- “A” Fuel Pool Cooling Pump is in service.
- RWCU auxiliary pump is in operation.

The Refuel Coordinator from the refuel floor reports they would like to improve reactor cavity clarity after a crud burst. The SRO directs:

a. Lowering SDC flow per DOP 1000-03, “Shutdown Cooling Mode of Operation.”

b. Raising SDC flow per DOP 1000-03, “Shutdown Cooling Mode of Operation.”

c. Raising RWCU flow per DOP 1200-01, “RWCU System Operation During Startup and Shutdown”

d. Placing an additional FPC pump in operation for Fuel Pool Cooling per DOP 1000-04 “Fuel Pool Cooling Mode of Operation of Shutdown Cooling System.”

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: C: Per DFP 0800-01, F.24.b, “Chemistry controls and operation of refueling water purification systems during plant operation and while shutdown should support and maintain adequate refueling water clarity.” This is accomplished by use of lights, RWCU system and temporary cavity filtration equipment.
A & B are incorrect since changing SDC flow rate will not improve pool clarity, but will change pool temperature.
D is incorrect since DFP 0800-01, is the incorrect reference for placing temporary filtration system into service.
References: DFP 0800-01, “Master Refueling Procedure.”DOP 1000-03, “Shutdown Cooling Mode of Operation.” DOP 1200-01, “RWCU System Operation During Startup and Shutdown” DOP 1000-04 “Fuel Pool Cooling Mode of Operation of Shutdown Cooling System.”
Direct/New/Modified: **New**
Memory/Comprehension-Analysis: **Memory**
Level (SRO/RO) **SRO** Only, Criteria II.E Procedure choice
K/A: 295008 AA2.03; Ability to determine and/or interpret the following as they apply to HIGH REACTOR WATER LEVEL: Reactor Water Cleanup Blowdown flow.
Author: Walton
Exam Date: October 30, 2013

Question # 084

Unit 2 is at rated power. Before I&C Technicians place the 2-263-57A instrument in a tripped condition for maintenance, the SRO would review Technical Specification(s) _____(1)______ for equipment operability determination. During this maintenance activity, a spurious trip of reactor water level instrument(s) _____(2)______ would cause a containment isolation to occur.

a. (1) 3.3.6.1, Primary Containment Isolation Instrumentation AND 3.3.6.2, Secondary Containment Isolation Instrumentation
   (2) 2-263-58A.

b. (1) 3.3.6.1, Primary Containment Isolation Instrumentation AND 3.3.6.2, Secondary Containment Isolation Instrumentation.
   (2) 2-263-58B.

c. (1) 3.3.6.1, Primary Containment Isolation Instrumentation ONLY
   (2) 2-263-58A OR 2-263-58B.

d. (1) 3.3.6.2, Secondary Containment Isolation Instrumentation ONLY
   (2) 2-263-57B.

REFERENCES PROVIDED: Dwg. 12E-6822, Dwg.12E-2501, Sheets 1 & 3 and
Dwg. 12E-2464 Sheets 1 & 2

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: B. Logic of ½ twice means 2-263-58A OR 2-263-58B would need to actuate. SRO would need to review both primary and secondary ctmt isolation TS. A & D are incorrect since 3-263-57B would not meet ½ twice logic.

References: DAN 903-5, D-5 and E-5, Group 2 (3Z) Isolation Initiated annunciators. Tech. Specs. 3.3.6.1 and 3.3.6.2. Dwgs 12E-6822 and 12E-2501 sheets 1 & 3.
References Provided: Dwg. 12E-6822, Dwg. 12E-2501, Sheets 1 & 3 and
Dwg. 12E-2464, Sheets 1 & 2.
Direct/New/Modified: **New**
Memory/Comprehension-Analysis: **C/A**
Level (SRO/RO) **SRO** Only, Criteria IIE, Tech Specs.
K/A: 295020 AA2.06; Ability to determine and/or interpret the following as they apply to INADVERTENT CONTAINMENT ISOLATION: Cause of isolation.
Author: Walton
Exam Date: October 30, 2013

Question # 085

Unit 2 was in Mode 5 with Rod F-5 at position 48 for testing. CRD pump 2A was out of service for maintenance when a transient occurred causing the following alarms:

Annunciator 902-5, D-2, “2B ROD DRIVE PP SUCT LO”
Annunciator 902-5, B-2, “ROD DRIVE PP TRIP”

The SRO enters ______(1)_______ and directs ______(2)_______.

a. (1) DGP 02-01, Unit Shutdown
   (2) reducing reactor recirculation flow

b. (1) DOA 0300-01, Control Rod Drive System Failure
   (2) scramming the reactor on the first accumulator trouble light

c. (1) DOA 0300-01, Control Rod Drive System Failure
   (2) scramming the reactor on the second accumulator trouble light

d. (1) DOA 0300-01, Control Rod Drive System Failure
   (2) Insert all withdrawn control rods using individual scram test switches

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: D is correct per DOA 0300-01 for unit in Mode 3, 4, 5.
C: DOA 0300-01, Control Rod Drive System Failure is entered. With 2nd accumulator fault on rod not fully inserted, a scram is inserted.
A is incorrect. DAN references DOA 0300-01, not DGP 02-01.
B is incorrect. With unit in Mode 2, this would be correct!

References: DAN 902-5, D-2. DOA 0300-01, Control Rod Drive System Failure.
Direct/New/Modified: **New**
Memory/Comprehension-Analysis: **C/A**
Level (SRO/RO) **RO**
K/A: 295022 Loss of CRD pumps G2.2.23: Ability to perform specific system and integrated plant procedures during all modes of plant operation.
Author: Walton
Exam Date: October 30, 2013

Question # 086

Unit 2 was operating at near rated power, when a transient occurred, resulting in the following set of conditions:

- RPV pressure is 300 psig.
- RPV water level is -71 inches.
- Drywell pressure is 3.0 psig and rising.
- NO Recirc system valves have repositioned.
- All LPCI valve GREEN indicating lights are illuminated.
- 'A' Recirc loop pressure is 4 psig higher than 'B' Recirc loop.

The LPCI system response is: _____(1)_______.
The Unit Supervisor directs: ______(2)_______.

a. (1) NO LPCI injection
   (2) Force LPCI logic to Div II per DOP 1500-09, LPCI LOOP SELECT DEFEAT

b. (1) NO LPCI injection
   (2) Manually position the required valves per DAN 902-3 A-4, LPCI LOOP SELECTION LOGIC

c. (1) LPCI injecting to BOTH loops
   (2) Force LPCI logic to Div II per DOP 1500-09, LPCI LOOP SELECT DEFEAT

d. (1) LPCI injecting to BOTH loops
   (2) Manually position the required valves per DAN 902-3 A-4, LPCI LOOP SELECTION LOGIC

ANSWER: B: For the given conditions LPCI Loop select logic should have chosen the 'A' loop for injection and closed the Recirc Discharge vlv on the 'A' loop and closed the LPCI 21 vlv for the 'B' loop of LPCI. The valves did not reposition therefore both 22 vlvs are in the normal closed position and no injection can occur. The correct actions to properly position the valves are directed in DAN 902-3 A-4.
DIRECT from **Bank** (Q22707)
C/A
Level (**SRO**) per E. Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. [10 CFR 55.43(b)(5)]
K/A: 203000 RHR/LPSI / 2.1.20 Ability to interpret and execute procedure steps.
Author: Licensee/Walton
Exam Date: October 30, 2013

Question # 087

Unit 2 was operating at near rated power, when the following set of conditions occurred, in the order listed:

- The reactor remained at power following a main turbine trip.
- A Group 1 Isolation occurred while lowering RPV level
- RPV water level is being maintained in a band of -40 to -50 inches
- Reactor power is varying between 5% and 15%.
- Torus temperature was observed to be 100°F and rising slowly
- The NSO initiated SBLC as directed by placing the SBLC keylock switch to the "SYS 1" position.
- 10 seconds later, Bus 23-1 locked-out due to a fault.

_____________ SBLC pump(s), is (are) operating.
As the Unit Supervisor you direct _________________________.

a. Neither
the Field Supervisor to initiate Alternate SBLC Injection

b. Neither
the NSO move the SBLC keylock switch to the "SYS 2" position

c. One
the NSO to start the second SBLC Pump by placing the SBLC keylock switch to the "SYS 1 & 2" position

d. Both
the NSOs to lower RPV water level by terminating and preventing all RPV injection except boron and CRD until power is below 6%, OR RPV water level drops to TAF, OR All ADSVs remain closed with Drywell Pressure below 2 psig

ANSWER: b.

a. INCORRECT – Neither pump will be operating (pump A started initially but is lost when Bus 23-1 locks out), but SBLC Pump “B” is still available and per DOP 1100-02, INJECTION OF STANDBY LIQUID CONTROL is started by placing the SBLC keylock switch to the “SYS 2” position.
b. CORRECT – Neither pump will be operating (pump A started initially but is lost when Bus 23-1 locks out), but SBLC Pump “B” is still available and per DOP 1100-02, INJECTION OF STANDBY LIQUID CONTROL is started by placing the SBLC keylock switch to the “SYS 2” position.
c. INCORRECT – Pump A started initially but is lost when Bus 23-1 locks out. Per DOP 1100-02 only one SBLC pump is started in response to an ATWS (two pumps are started for Alternate RPV injection). Placing the SBLC control switch to the “SYS 1 & 2” position would start SBLC Pump B but would not be in accordance with the approved procedure.

d. INCORRECT -- Pump A started initially but is lost when Bus 23-1 locks out. Additionally, RPV water level would not be lowered further until Torus Temperature exceeded 110°F.

References:
DRE211LN001, Standby Liquid Control (SBLC) Lesson Plan; Revision 8
DOP 1100-02, INJECTION OF STANDBY LIQUID CONTROL; Revision 18
DEOP 400-05, FAILURE TO SCRAM; Revision 16
DEOP 500-01, ALTERNATE STANBBY LIQUID CONTROL INJECTION; Revision 16

Direct/New/Modified: **MODIFIED** from Bank Question QQ 23815 used on the 2008 Cert Exam
Memory/Comprehension-Analysis: **C/A**
Level (SRO/RO): **SRO**
Level (SRO) per E. Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. [10 CFR 55.43(b)(5)]
K/A: 211000/A.2.03 – Ability to predict impacts of AC power failures on SBLC and then use appropriate procedures.
Author: D. Reeser
Exam Date: October 30, 2013

Question # 088

During the execution of DOS 0500-06, APRM GAIN ADJUSTMENT FACTOR VERIFICATION, at 100% core thermal power (CTP), the performer reported that the as-found APRM Gain Adjustment Factors (AGAFs) for APRMs 1 and 2 failed to meet the acceptance criteria. The AGAFs were reported to be 0.977 and 0.960 for APRM 1 and 2 respectively.

Which of the following actions MUST be taken?

a. Within 2 hours, restore the AGAF of BOTH APRM channels to within specification by performing APRM gain adjustments per DOP 0700-09, APRM Gain Adjustment.

b. Within 6 hours, place either APRM Channel 1 OR 2 in TRIP per Technical Specification LCO 3.3.1.1 Reactor Protection System (RPS) Instrumentation.

c. Within 12 hours, restore the AGAF of APRM Channel 1 OR 2 to within specification by performing APRM gain adjustments per DOP 0700-09, APRM Gain Adjustment.

d. Within 12 hours, place BOTH APRM Channels 1 AND 2 in TRIP per Technical Specification LCO 3.3.1.1 Reactor Protection System (RPS) Instrumentation.

ANSWER: C

a. INCORRECT –The 2 hour limit applies to the high AGAF limit (1.02). TS entry is not required unless the AGAF cannot be adjusted.
b. INCORRECT – Tech Spec entry is not required unless the AGAF cannot be adjusted. The 6 hour time limit is plausible if examinee believes the APRMs 1 and 2 are in separate trip systems.
c. CORRECT – Within 12 hours, perform an APRM Gain Adjustment per DOP 0700-09, APRM Gain Adjustment. TS entry is not required unless the AGAF cannot be adjusted.
d. INCORRECT – Tech Spec entry is not required unless the AGAF cannot be adjusted. The 12 hour limit would be applicable if the AGAF of either channel could not be restored.

References: Technical Specification LCO 3.3.1.1, DOP 0700-09, APRM Gain Adjustment, and DOS 0500-06, APRM GAIN ADJUSTMENT FACTOR VERIFICATION. (NOT provided during the exam)

Reference Provided: Tech Spec 3.3.1.1

New

C/A

Level (**SRO**) per E. Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. [10 CFR 55.43(b)(5)] and Facility operating limitations in the TS and their bases. [10 CFR 55.43(b)(2)]

K/A: 212000(RPS) / 2.2.12. Knowledge of Surveillances

Author: C. Zoia (D. Reeser)

Exam Date: October 30, 2013

Question # 089

During the performance of Technical Specifications (TS) surveillance SR 3.3.6.1.2, Perform CHANNEL FUNCTIONAL TEST, on Unit 2, Fuse F595-704B blew in PCIS instrumentation causing annunciator 902-5 E-5, GROUP 2 ISOLATION INITIATED, to alarm. Which of the following actions must be ordered to address this once the blown fuse is replaced?

a. Verify Group 2 isolation automatic actions have occurred as expected per DAN 902(3)-5 E-5.

b. Reset annunciator 902-5 E-5, GROUP 2 ISOLATION INITIATED, ONLY per DAN 902(3)-5 E-5.

c. On the 902-5 panel, reset the Group 2 isolation signal using the Group 2 & 3 ISOL RESET switch per DAN 902(3)-5 E-5.

d. Reset the TIP Group 2 isolation per DOP 0700-06, TRAVERSING INCORE PROBE (TIP) SYSTEM OPERATION.

ANSWER: b. Reset the annunciator ONLY per DAN 902(3)-5 E-5. Since a single blown fuse would only affect one sensor, per the NOTE in the DAN, only an alarm would be expected. All other distractors assume a Group 2 occurred and are part of the process of verifying and resetting it.
References: DAN 902(3)-5 E-5 (NOT provided during the exam)
References Provided: Dwg 12E-2501, Sheet 2; Dwg. 12E-2510, Sheet 1 and 12E-2511.
New
Memory:
Level (**SRO**) : per E. Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. [10 CFR 55.43(b)(5)]
K/A:223002 (PCIS)/A2.08, Predict impacts of Surveillace Testing, and based on those impacts, use procedures to correct.
Author: C. Zoia
Exam Date: October 30, 2013

Question # 090

A Reactor Coolant System (RCS) leak is suspected on Unit 2 due to increasing trends indicated on the Drywell Continuous Air Monitor rising Drywell pressure (1.2 psig and rising slowly). During the investigation, annunciator 923-1 F-1, U2 RBCCW HEAD TANK LVL HI/LO is received. Which of the following actions must be ordered FIRST to address RBCCW conditions?

a. Isolate Drywell RBCCW ONLY.

b. Manually SCRAM the reactor, trip the Reactor Recirculaton Pumps, then isolate Drywell RBCCW.

c. Dispatch an Equipment Operator to determine U2 RBCCW Head Tank level and trend; monitor RBCCW system parameters to verify proper system operation.

d. Isolate RBCCW flow to non-essential and/or out of service loads and monitor remaining RBCCW system loads for increasing temperatures.

ANSWER: c.

a. INCORRECT – Immediate isolation of Drywell RBCCW is only required if Drywell Pressure exceeds 2 psig concurrently with a LOCA (DAN 923-1 F-1 and DOA 3700-01)
b. INCORRECT – Manual SCRAM of the reactor and trip of the Recirc Pumps is required only if all RBCCW flow is lost (DOA 3700-01).
c. CORRECT – Per DAN 923-1 F-1, an operator is dispatched to determine level and trend. Per DOA 3700-01, parameters are checked to ensure proper system operation. Subsequent actions will be based on those reports.
d. INCORRECT – Neither the DAN nor DOA direct this action, but is plausible since the DOA does have a step to locate and isolate any leaks that may be the source of the problem.

References:
DAN 923-1 F-1, Uw RBCCW HEAD TANK LVL HI/LO; Rev 7.
DOA 3700-01, LOSS OF COOLING BY REACTOR BUILDING CLOSED COOLING WATER (RBCCW) SYSTEM; Revision 19
New
C/A
Level (**SRO**) : per E. Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. [10 CFR 55.43(b)(5)]
K/A: 400000 (CCW) / A2.02, Predict impacts of High/Low Surge Tank Level, and based on those impacts, use procedures to correct.
Author: C. Zoia/D. Reeser
Exam Date: October 30, 2013

Question # 091

Unit 3 was at 80% CTP when annunciators 903-5 A-3, ROD DRIFT and 903-5 G-2, ACCUMULATOR LVL HI/PRESS LO alarmed.

- The NSO reported that 3 control rods have “ACCUMULATOR TROUBLE” and “SCRAM” indicators illuminated.
- The 3 control rods, which were previously withdrawn, now indicate “green dashes” and the rod motion was NOT observed.
- No further rod motion is occurring.

Based on the conditions above, what action is required to be ordered?

a. Scram the reactor and enter DGP 02-03, REACTOR SCRAM.

b. Initiate an “Emergency Load Decrease” to < 25% CTP and contact the Qualified Nuclear Engineer for guidance on further power changes.

c. Return the unit to the pre-transient power level and contact the Qualified Nuclear Engineer for guidance on further power changes.

d. Discontinue all control rod movement and recirculation flow increases, evaluate whether or not core thermal limits are within specifications, and contact the Qualified Nuclear Engineer for guidance on further power changes.

ANSWER: d.

a. INCORRECT – SCRAM not required unless 4 rods scram to 00
b. INCORRECT – Such a large power reduction would not be required unless thermal limits have been exceeded
c. INCORRECT – Per DOA 0300-12, MISPOSITIONED CONTROL ROD
d. CORRECT – Per DOA 0300-12, MISPOSITIONED CONTROL ROD

References:
DAN 903-5 A-3, ROD DRIFT; Revision 16
DAN 903-5 G-2, ACCUMULATOR LVL HI/PRESS LO; Revision 19

DRE201LN001, Control Rod Drive (CRD) System Lesson Plan; Revision 11
DRE201LN002, Reactor Manual Control System (RMCS) and Rod Position Indication System (RPIS) Lesson Plan; Revision 3.

New
C/A
Level (**SRO**) : per E. Assessment of facility conditions and selection of appropriate procedures

during normal, abnormal, and emergency situations. [10 CFR 55.43(b)(5)]
K/A: 201002 (RMCS) / A2.02, Predict impacts of Rod Drift Alarm, and based on those impacts, use procedures to correct.
Author: C. Zoia
Exam Date: October 30, 2013

Question # 092

Unit 2 was at full power when annunciator 902-7 F-5, Turb Gen BRG Vibration Hi alarmed. A few minutes later, 902-7 E-5, Turb Gen BRG Vibration Hi-Hi alarmed. The NSO reported that bearing 11 was reading 11 mils. Based on these conditions, what actions MUST the SRO order?

a. Manually trip the turbine, enter DOA 5600-01, “Turbine Trip,” THEN transition to DGP 02-03, “Reactor Scram.”

b. Manually scram the reactor AND transition to DGP 02-03, “Reactor Scram,” THEN enter DOA 5600-01, “Turbine Trip.”

c. Verify an automatic turbine trip, enter DOA 5600-02, “Turbine High Vibration,” THEN transition to DGP 02-03, “Reactor Scram.”

d. Verify an automatic turbine trip, THEN enter DOA 5600-01, “Turbine Trip,” THEN transition to DGP 02-03, “Reactor Scram.”

ANSWER: A. Manually trip the turbine, enter DOA 5600-01, TURBINE TRIP, THEN transition to DGP 02-03, REACTOR SCRAM. All remaining actions are improperly sequenced under the stated conditions.

References: DOA 5600-01, Turbine Trip, DOA 5600-02, Turbine High Vibration.

New
C/A
Level (**SRO**) : per E. Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. [10 CFR 55.43(b)(5)]
K/A: 245000 (Main Turbine Gen.) / G2.1.31, Ability to explain and apply system limits and precautions.
Author: C. Zoia
Exam Date: October 30, 2013

Question # 093

Unit 2 was at full power when MCC 20-2, 120/240V Distribution Panel, Circuit #1 tripped on overcurrent. It was reported that the 2/3 Diesel Fire Pump (DFP) control cabinet in the 2/3 crib house lost AC control power. The SRO will declare ____________.

1. the fire protection system inoperable per TRM 3.7.i, Condition A.
2. the fire protection system inoperable per TRM 3.7.i, Condition B.
3. the water suppression system inoperable per TRM 3.7.j, Condition A.

Answers:

a. 1 ONLY

b. 2 ONLY

c. 1 & 2 ONLY

d. 1, 2 AND 3

References provided: TRM 3.7.i and TRM 3.7.j.

ANSWER: A. Consider the 2/3 DFP inoperable, and enter TRM TLCO 3.7.i, Fire Water Supply System, Condition A. All remaining actions are not required under the stated conditions. The cause of the failure is given, not spurious, and is specifically addressed by DAN XL3 51-13. The Fire Protection System Lesson Plan, DRE286LN001 says the DFP can be manually started at the local controller, however, the battery chargers and the engine block heater will be lost.
References: TRM 3.7.i, and TRM 3.7.j. The Fire Protection System Lesson Plan, DRE286LN001
References Provided: TRM 3.7.i and TRM 3.7.j
New
C/A
Level (**SRO**) : per E. Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. [10 CFR 55.43(b)(5)]
K/A: 286000 (Fire Protection) / A2.03, Ability to predict impacts of an AC failure in Fire Protection and utilize procedures to address.
Author: Zoia/Walton
Exam Date: October 30, 2013

Question # 094

Unit 3 conditions are as follows:

- A LOCA has occurred;
- The TSC estimates 10% core fuel is damaged;
- An uncontrolled release to the environment is occurring via the 310' chimney;
- A site area emergency has been declared.

As the Emergency Director, you order equipment operator Jack to manually, locally, close valve 3-1601-92, (Drywell Vent to Chimney Isolation Valve). This emergent task will result in 300 mrem exposure to both Jack and RP Tech, Jill. (Both Jack and Jill are 44 years old).
Their TEDE dose histories are as follows:

	Lifetime Exposure	Annual Exposure	Daily exposure
Jack	45.5 Rem	850 mrem	0 mrem
Jill	2.5 Rem	1800 mrem	0 mrem

In order to start this task, per RP-AA-203, "Exposure Control and Authorization," which of the following forms must be completed?

- Attachment 1: Dose Control Level Extension Forms
- Attachment 2: High Lifetime Dose Control Level Extension Form
- Attachment 3: Planned Special Exposure Form

ANSWERS:

a. Both Jack and Jill need Attachment 1 completed, ONLY.

b. Jack needs Attachment 2 completed ONLY.
Jill needs Attachment 1 completed ONLY.

c. Both Jack and Jill need Attachment 2 completed ONLY.

d. Jack needs Attachment 2 AND Attachment 3 completed.
Jill needs Attachment 1 AND Attachment 3 completed.

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: B is the correct answer. Jack's exposure is considered a "High Lifetime" exposure since his exposure exceeds his age. He needs Attachment 2 completed. Jill needs Attachment 1 completed since she will exceed the 2000 mrem admin limit.

A would be correct if Jack's dose was not expected to exceed the 2000 mrem annual admin dose limit.
C is incorrect. Jill does not need Attachment 2 since she does not qualify for "high lifetime" exposure.
D is incorrect since even though both are performing a task for "protection of large populations," their dose limits do not exceed NRC limits and PSE does not apply.
References: RP-AA-203, "Exposure Control and Authorization.
Reference Provided: RP-AA-203, "Exposure Control and Authorization"
Direct/New/Modified: **New**
Memory/Comprehension-Analysis: **C/A**
Level (SRO/RO) **SRO** Only, Criteria II.E
K/A: 2.3.13 Knowledge of radiation safety principles pertaining to licensed operator duties, such as containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc..
Author: Walton
Exam Date: October 30, 2013

Question # 095

Unit 2 is in Mode 1, Unit 3 is in REFUEL with core alterations in progress when the refuel bridge operator reported dropping a control rod on a fuel bundle in the spent fuel pool. The operator reported gas being releasing from the damaged bundle. Everyone is evacuating the refuel bridge.

The NSO began observing refueling floor radiation levels rising on RR 3-1705-21, RX BLDG VENT EXH DUCT RAD MON. DAN 903-3, E-16, RX BLDG FUEL POOL CH 'B' RAD HI alarm energized. As Unit Supervisor, you reviewed the DAN and found that NONE of the automatic actions occurred for this alarm. To minimize the hazard of spreading radiation/contamination, you ordered the NSO to trip RX BLDG ventilation and to start SBGT. The NSO completed the task and reported the manual actuation was successful without any problems.

You declare the system is inoperable and determine ___________.

a. Tech Spec 3.3.6.2, Condition A is applicable with a 12 hour completion time.

b. Tech Spec 3.3.6.2, Condition A is applicable with a 24 hour completion time.

c. Tech Spec 3.3.6.2 is NOT applicable for MODE 5.

d. no Technical Specification entries were required since the system needs 2 Refuel Radiation Monitor alarms for system actuation.

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: B is correct. Problem is with instrumentation. Rad Monitors are allowed a 24 hour completion time.
A is incorrect since 12 hour completion time are not applicable for rad monitors.
C is incorrect. In applicable Mode column of Table 3.3.6.2-1, a footnote specifies applicability for movement of recently irradiated fuel.
D is incorrect since only 1 of 2 refuel floor radiation monitors are required for system actuation.
References: DAN 902(3)-3, E-16. SBGT Lesson Plan. Technical Specifications.
Reference Provided: Tech Spec 3.3.6.2
Direct/New/Modified: **New**
Memory/Comprehension-Analysis: **C/A**
Level (SRO/RO) **SRO** Only, Criteria II.F, Technical Specifications.
K/A: 2.3.14 Knowledge of radiation or contamination hazards that may arise during normal, abnormal or emergency conditions or activities.
Author: Walton
Exam Date: October 30, 2013

Question # 096

Units 2 and 3 were operating at rated power when an earthquake occurred. Both units scrammed. The following conditions now exist on Unit 2:

- An ATWS has occurred with reactor power at 12% and steady;
- The SRO has entered DEOP 400-5, “Failure to Scram”
- RPV water level is at -12” slowly lowering;
- RPV pressure is 1100 psig and cycling;
- A steam line break is located upstream of 2-2301-3, HPCI STEAM ADMISSION VLV:
- 2-2301-4 & 5, HPCI STEAM ISOL VLVs failed to close;
- HPCI Room temperature is 212°F rising slowly
- SDC heat exchanger room is 185°F and slowly rising.

As the Unit 2, Unit Supervisor, you direct _____(1)______ because ______(2)_______.

a. (1) DEOP 400-5, Level Leg: Lower RPV level to at least -35” by terminating and preventing ALL RPV injection
   (2) lowering RPV water level will lower reactor power

b. (1) DEOP 400-5, Pressure Leg: Stabilize RPV pressure below 1060 psig using main turbine bypass valves
   (2) lowering RPV pressure below the ADSV setpoint will prevent shrink/swell of RPV water level and minimize dynamic loads on the SRV tailpiece and RPV

c. (1) a transition from DEOP 300-1, “Secondary Containment Control” to DEOP 400-2, “Emergency Depressurization”
   (2) reduces the driving head and flow of primary systems that are unisolated and discharging into the secondary containment

d. (1) DEOP 400-5, Power Leg: Open MSIVs and use main condenser as a heat sink
   (2) discharging heat to the condenser will preserve the energy suppression design feature of containment

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: C is correct. Per NOTE in DEOP 400-2, “Emergency Depressurization,” this procedure overrides DEOP 400-5 (Pressure and Level Legs)
A & B are incorrect, since these are actions from DEOP 400-5, level and pressure legs.
D is incorrect. Since conditions are met for ED, must ED in lieu of discharging heat to main condenser.
References: DEOP 400-5, Failure to Scram. DEOP 400-2, Emergency Depressurization.
REFERENCES PROVIDED: DEOP 400-5 with initial conditions removed and SBLC tank levels removed from pressure control leg (36%) AND power control leg (8%). (See Q81)

DEOP 300-01 “Secondary Containment Control” with initial conditions removed.
Direct/New/Modified: **New**
Memory/Comprehension-Analysis: **C/A**
Level (SRO/RO) **SRO** Only, Criteria IIE, EOP usage.
K/A: G2.4.22: Knowledge of the basis for prioritizing safety functions during abnormal/emergency operations.
Author: Walton
Exam Date: October 30, 2013

Question # 097

Unit 2 was operating at full power when the unit experienced a LOCA with a fuel cladding failure and a breach of primary containment. Given that:

- there are no bridges or road work ongoing in the EPZ;
- public exposure is expected to be 190 mrem;
- RPV level is -195 inches and lowering slowly;
- Wind is from 350° North.

As the Emergency Director, what initial protective action recommendations are applicable (if any)?

a. No PARS required.

b. Shelter subareas 1, 3, 4, ONLY.

c. Evacuate subareas 1, 3, 4, 7, 9, ONLY.

d. Evacuate subareas 1, 3, 4, ONLY.

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: C from the initial conditions given, the applicant is to determine that a GE exists. With dose projections < assumed on Fig 4-1, no evidence of a hostile action and the road conditions given, the applicant should select C.
A is incorrect since a GE exists.
B is incorrect since there are no transportation impediments so sheltering is not an option.
D is incorrect. Could be misread as winds from 170° rather than to 170°.
References: EP-AA-1004, Exelon Nuclear Radiological Emergency Plan – Dresden.
Reference Provided: EP-AA-1004, Figure 4-1; EP-AA-1004 EAL Hot and Cold Charts.
Direct/New/Modified: **New**
Memory/Comprehension-Analysis: **C/A**
Level (SRO/RO) **SRO** Only, Criteria D, Radiological hazard protection.
K/A: G.2.4.44: Knowledge of emergency plan protective action recommendations.
Author: Walton
Exam Date: October 30, 2013

Question # 098

As the SRO for a shift, you are in the third quarter and have to decide which NSO can substitute for an NSO who had to leave in the middle of your shift. Your present NSO has a “no-solo” license with no other restrictions. From this available list provided, who is eligible for the 2nd NSO position on your crew?

John – reactivated his license in the 1st quarter, but did NOT stand a watch as an NSO in the 2nd quarter

James – has reported to the medical staff last week a condition requiring a reduction of dosage of a prescription drug he is currently taking. A license change has been submitted to the NRC for review.

Julia – has a “no-solo” license and is a declared pregnant worker.

a. Julia ONLY

b. James ONLY

c. James and Julia.

d. John & James.

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: C: Both James and Julia meet requirements for standing watch.
John must stand quarterly watches to maintain proficiency. Presently, his license is not active.
James may stand watch while a medical condition is being reviewed by the NRC.
Julia may stand watch. Two licensed operators with no-solo conditions on their license can still stand watch together.
References: OP-AA-105-100-102“

Direct/New/Modified: **New**
Memory/Comprehension-Analysis: **C/A**
Level (SRO/RO) **SRO** only.
K/A: G 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.
Author: Walton
Exam Date: October 30, 2013

Question # 099

As the WEC supervisor, you are reviewing upcoming planned work activities to determine the scope of work to be performed. You have separated out the following packages for further review:

1. W/O 13XXXXXXX, requires a planned entry into TS 3.0.3.
2. W/O 13YYYYYYY, work package will breach primary containment.
3. W/O 13ZZZZZZZZ, Paragon risk assessment results in online risk going yellow.

IAW WC-AA-106, "Work Screening and Processing," which of these work requests will require the work to be performed during an outage?

a. 1, AND 2 ONLY.

b. 2, AND 3 ONLY.

c. 1, AND 3 ONLY

d. 1, 2, AND 3.

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: A per WC-AA-106, "Work Screening and Processing," Attachment 3, TS 3.0.3 and breaching primary containment must be done off line per Attachment 3. Per WC-AA-101 Sect. 4.5.11 and Att 3, CDF risk increase of 8X is Yellow risk and can be done online. Item #3 is the only activity that can be done on line.
References: WC-AA-106, "Work Screening and Processing." WC-AA-101, "On-Line Work Control Process," Sect. 4.5.11 & 12 and Attachment 1 and 3.
Direct/New/Modified: **New**
Memory/Comprehension-Analysis: **C/A**
Level (SRO/RO) **SRO** Only Criteria
K/A: G 2.2.18: Knowledge of the process for managing maintenance activities during shutdown operations, such as risk assessments, work prioritization, etc.
Author: Walton
Exam Date: October 30, 2013

Question # 100

Surveillance Requirement 3.8.3.1, requires that stored diesel engine fuel oil be tested every 31 days since fuel oil degradation during long term storage could result in…

a. dissociation of the fuel oil.

b. improper ignition in the diesel engine.

c. fouling of the filters and fuel oil injection equipment.

d. increased soot production that could foul the exhaust system.

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: C "Fuel oil degradation during long term storage shows up as an increase in particulate, mostly due to oxidation. The presence of particulate does not mean that the fuel oil will not burn properly in a diesel engine. The particulate can cause fouling of filters and fuel oil injection equipment, however, which can cause engine failure."
A is incorrect since diesel fuel does not dissociate.
B is incorrect as referenced in above statement.
D is incorrect since soot production in a diesel is not related to fuel intake but due to improper combustion.

References: Technical Specifications 3.8.3.1 Basis
Direct/New/Modified: **New**
Memory/Comprehension-Analysis: **Memory**
Level (SRO/RO) **SRO** Only Criteria II.B TS Basis
K/A: G2.2.25; Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits.
Author: Walton
Exam Date: October 30, 2013.

ANSWER KEY

RO EXAM	RO EXAM	RO EXAM	SRO EXAM
1 A	26 B	51 A	76 D
2 C	27 C	52 C	77 C
3 D	28 D	53 A	78 C
4 B	29 D	54 B	79 ~~A~~ C (changed)
5 C	30 A	55 A	80 B
6 B	31 C	56 B	81 B
7 C	32 D	57 C	82 C
8 D	33 D	58 C	83 C
9 B	34 C	59 A	84 B
10 A	35 A	60 B	85 D
11 B	36 B	61 C	86 B
12 D	37 D	62 D	87 B
13 B	38 D	63 A	88 C
14 C	39 A	64 C	89 B
15 A	40 C	65 C	90 C
16 A	41 D	66 C	91 D
17 C	42 A	67 C	92 A
18 B & C (changed)	~~43 C~~ deleted	68 B	93 A
19 D	44 C	69 B	94 B
20 A	45 C	70 D	95 B
21 D	46 B	71 A	96 C
22 A	47 B	72 D	97 C
23 B	48 D	73 B	98 C
24 B	49 D	74 B	99 A
25 C	50 A	75 A	100 C