Attached is an early version of the NRC-authored written examination.

The questions in this document have been peer-reviewed by Region III examiners with comments documented on Form 401-9.

This document consisted of questions provided to the licensee for their first onsite-validation of the written exam performed the week of August 19, 2013. This exam is incomplete as questions were still being generated at the time of the first onsite validation.

The following questions were not included in this document:

43, 66, 67, 68, 70, 72, 73, 78, 80, 81, 85, 87, 88, 90 – 97, 99

Question # 001R1

Unit 2 was operating at near 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 29, 2013

Question # 002R1

Unit 2 Reactor was operating at rated power with a normal electrical lineup. The control room operators observed the following:

- a trip of 2B Reactor Feedwater pump;
- a trip of both C & D Condensate/Condensate Booster pumps;
- a Unit 2 reactor scram, and;
- a Unit 2 main generator trip.

The operators determine the cause of this event is…

a. A lockout of Bus 22 ONLY

b. A lockout of Bus 24 ONLY

c. A failure of Transformer 86

d. A trip of main generator output breakers

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: DAN902-6,F-5. Dresden High Voltage Distribution Dwg 262LN003-012.
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: 10/29/2013

Question # 003

Unit 3 was operating at rated power with a normal lineup and the following plant conditions:

- 125 VDC Battery Charger 3A is OOS for maintenance
- 125 VDC Battery Charger 3 then failed.

With a 3.8 volt/hour loss of voltage from normal loads on the U3 125 VDC system, what is the MAXIMUM time it would take BEFORE the U3 125 VDC battery drops below its minimal discharge voltage? (Assume no operator action was taken.)

a. 2.5 hours.

b. 3.5 hours.

c. 4.5 hours.

d. 5.5 hours.

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: C, Per UFSAR 8.3.2.2.1, 125 VDC minimum discharge voltage is 105 VDC.
125 V – (3.8 V/hr X 4.5 hours) = 107.9 V. 4.5 hrs is the maximum time.
A is not correct, 115.5 is still above the minimum discharge voltage.
B is incorrect, in 3.5 hours, voltage will drop to 111.3 VDC, Not below the minimum of 105 VDC.
D is below the minimum discharge voltage (100.3 volts) but it is not the maximum time!

References: UFSAR 8.3.2.2.1

Direct/New/Modified: New
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: Walton
Exam Date: 10/29/2013

Question # 004

Unit 3 was at 100% power. With a fault in the EHC System that results 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: 10/29/2013

Question # 005R.1

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: 10/29/2013

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 # 006L1

When scramming the reactor per DSSP 100-C, “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
   (2) bypassess the MSIV closure function on low RPV pressure

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

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

d. (1) depresses MANUAL SCRAM CH A AND MANUAL SCRAM CH B pushbuttons 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: **~~C/A~~** **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: 10/29/2013

Question # 007R1

Unit 3 was at rated power when control room operators observed the following:

- A slight increase in drywell pressure
- A slight increase in drywell temperature
- An increase in both recirculation pump seal temperatures

An equipment operator reported back that the RBCCW expansion tank level is stable but the RBCCW/SW TCV has failed closed. An attempt to start the standby RBCCW pump was unsuccessful. (The 2/3 RBCCW pump is lined up to Unit 2).

The operators must FIRST _____(1)_______ before _______(2)_______.

a. (1) lower reactor power and recirculation pump speed
   (2) there are any high temperature alarms on the recirculation pumps.

b. (1) lineup 2/3 RBCCW pump to Unit 3
   (2) RBCCW high temperature alarms.

c. (1) trip the recirculation pumps
   (2) the reactor is manually scrammed.

d. (1) manually scram the reactor
   (2) tripping the recirculation pumps.

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: D is correct sequence 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.
C is incorrect since reactor must be tripped before tripping the recirculation pumps.

References: DOA 3700-01, Loss of RBCCW procedure, Step C.3 and Section H.
Direct/New/Modified: **New**
Memory/Comprehension-Analysis: **C/A**
Level (SRO/RO); **RO**
K/A: 295018, Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF CCW: System loads.
Author: Walton
Exam Date: 10/29/2013

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 Inlet Drain Pot 2A Trap Bypass valve (2-2301-31) 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: B. Per DOA 4700-01, Table 1, 2-2301-31 fails closed, the 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: **C/A**
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: 10/29/2013

Question # 009

Unit 2 is in Mode 5 with ‘C’ shutdown cooling system in operation IAW DOP 1000-03 at 130°F. A system fault resulted in the SW TCV to the in-service RBCCW heat exchanger failing closed. Which plant process computer screen will reveal this deficient condition first?

a. Screen 30 SPDS Primary

b. BOP Unit Operating Factors Screen

c. System Health Display Screen

d. Alarm Summary Screen

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: D Per reference
A is incorrect since this condition will not be seen on this higher level screen
B is incorrect since this screen produces daily and monthly operating performance of the unit.
C is incorrect since System Health will not display primary plant parameters(?)

References: Plant Computer System Lesson Plan. DAN 902-4, A23. DOP 1000-03, Precaution E7. DFP 0800-01, “Master Refueling Procedure,” Sect. H. DOP 1000-04, “Fuel Pool Cooling…” Section H.2. DOP 1000-03, “SDC Mode of Operation,” F.1.d, “Maintain RPV temperature less than 140°F.”

Direct/New/Modified: **New**
Memory/Comprehension-Analysis: **Memory**
Level (SRO/RO) **RO**
K/A: 295021 G2.1.19. Loss of SDC, Ability to use plant computers to evaluate system or component status.
Author: walton
Exam Date: 10/29/2013

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 hoist loaded and over the core.

b. rod worth minimizer in OPERATE.

c. either rod block monitor downscale.

d. prevents movement of a second control rod when the first control rod is NOT fully inserted.

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. this is not a rod block, but a rod select interlock.
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: 10/29/2013

Question # 011L1

Unit 2 was operating at full rated power when operators note an increase in drywell pressure. Drywell pressure is 1.22 psig and rising .3 psig each minute. An ECCS initiation signal, is required by Tech Specs to occur 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) = 1.2 = 1.70 psig (actual pressure for initiation). Upon initiation, drywell spray valves are interlocked closed and must be manually overridden open (using key lock) AND with DW pressure > 1.0 psig.
(0.3 psig X 3 minutes = 2.0 psig (Technical Specification Limit)
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: 10/29/2013

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 field breakers trip (no time delay)

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

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

ANSWER: B.
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: 10/29/2013

Question #13R1

Annunciator 902-4, B-18, “DIV 1 TORUS Local Water Temperature High” alarms. What indicator would the operators use 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

UNRESOLVED COMMENT: Is this a proper question to ask of the operators? Should the operators know the location of this indicator??

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: C/A
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: 10/29/2013

Question # 014

Unit 3 was operating in Mode 1. After a transient, 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. A safety relief valve has failed open.

b. Containment is breached following a water break LOCA

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

d. Containment is functioning normally following a bypass path discharge into the suppression chamber airspace has occurred.

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER:
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: 10/29/2013

Question # 015R1

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

- Torus water temperature is 92°F and rising slowly
- Torus 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. insufficient volume for ECCS makeup.

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

c. Incomplete steam condensation resulting in 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 ECCS makeup or 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: C/A
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: 10/29/2013

Question # 016R2

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.

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 SLC (with Boron) are injecting,
RPV level is -200 inches and increasing,
MSIVs are open.

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

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

References: DEOP 400-5, "Failure to Scram" and DEOP 400-3, "Steam Cooling"

Direct/New/Modified: ~~New~~ **BANK**
Memory/Comprehension-Analysis: ~~Memory~~ **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: 10/29/2013

Question # 017R1

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 at -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) reduces 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: Memory
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: 10/29/2013

From General Electric EPG/SAG, Appendix B, Contingency 5.

The process by which reactor power is reduced by lowering RPV water level occurs as follows:

1. The reactor is in a natural circulation mode following recirculation pump trip (accomplished in EPG Step RC/Q-2 or Step RC/Q-5). The natural circulation driving head is a function of the fluid density difference between the regions inside and outside of the shroud (void fraction directly affects the fluid density inside the shroud) and the height of the fluid columns (RPV water level).
2. As RPV water level is lowered, the height of the fluid columns is reduced, thereby reducing the natural circulation driving head.
3. As the natural circulation driving head is reduced, the natural circulation flow through the core is reduced.
4. The reduced core flow results in a reduced rate of steam removal from the core.
5. The reduced rate of steam removal results in an increased void fraction inside the shroud.
6. The increased void fraction adds negative reactivity to the reactor.
7. The negative reactivity drives the reactor slightly subcritical and power begins to decrease.
8. The reduced reactor power results in a reduced steam generation rate.
9. The reduced steam generation rate results in a reduced void fraction.
10. When the void fraction drops to its original value (with some slight adjustment to account for reduced Doppler reactivity), the reactor returns to criticality at a lower power.

Question # 018R1L1

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 down wind were to receive an acute dose of 300 rads, what biological effects would occur about 30 days later in the correct sequence?

1. Death (to 50% of the population)
2. Hematopoietic Syndrome (decrease in blood cell count)
3. Gastrointestinal Syndrome (nausea, vomiting and diarrhea)

a. 2 ONLY

b. 2 then 1 ONLY

c. 3 then 2 ONLY.

d. 3 then 2 then 1.

(NEED COMMENTS ON THIS ONE!)

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: A: Hematopoietic (or blood) Syndrome –there will be observable decreases in blood cell count at doses of about 100 rad.
B & D are incorrect due to insufficient dose to cause death. 400 rads is lethal dose to 50% of the population in 60 days.
C. is incorrect since gastrointestinal (or GI) Syndrome occurs with a dose of about 500 rads: will result in nausea, vomiting, and diarrhea.
REFERENCES: NRC Study Materials (see attached)

Direct/New/Modified: **New**
Memory/Comprehension-Analysis: **C/A**
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: 10/29/2013

Question # 019L1

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). This condition was produced by _____(1)______ in alarm for the AEER. If this condition detects an actual fire, the automatic suppression for the AEER is ______(2)_______.

	(1)	(2)
a.	one detector	water sprinkler system
b.	two detectors	water sprinkler system
c.	one detector	halon dispersal system
d.	two detectors	halon dispersal system.

NEED LICENSEE COMMENTS ON THIS ONE!

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: D. The magnitude of the sensitivity change is what the XL3 uses to determine whether a detector goes into trouble or an alarm condition.
AEER Halon System. (Requires 2 detectors sensing conditions that indicate a fire.)
A & B are incorrect since there is no water sprinkler system in the AEER.
C is incorrect since per the reference, two detectors are required to signal a fire in the AEER.

References: XL3 Fire Computer Lesson Plan, pg

Direct/New/Modified: **New**
Memory/Comprehension-Analysis: **C/A**
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: 10/29/2013

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: 325 KV

What are the required actions from the Operating team AND the reason for these actions?

a. Adjust TR 32 Tap Changer;
   to restore system operability

b. Adjust TR 32 Tap Changer;
   to reduce circulating currents

c. Adjust TR 86 Tap Changer;
   to restore system operability

d. Adjust TR 86 Tap Changer;
   to reduce circulating currents

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: A. Unit 2 TR 86 (Unit 3 TR 32) LTC in AUTO, adjust LTC. IF TR 86(32) LTC is operating in AUTO, AND switchyard voltage is greater than the maximum percent voltage drops OR minimum voltages provided in Table 1 of Attachment A(B) OR D(E), THEN
the LTC will maintain site voltages above the minimum required voltage for operability.:

References: DOA 6500-12, Low Switchyard Voltage

Direct/New/Modified: Direct from **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: 10/29/2013

Question #21

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 peaked at +60 inches AND is currently +35 inches and lowering.

Assuming the above trends continue, the HPCI system:

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

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

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. The system will reinitiate, when reactor water level drops to low level initiation set point.

References:
DRE206LN001, High Pressure Coolant Injection System Lesson Plan; Revision 4

Direct/New/Modified: MODIFIED from bank question QQ23577
Memory/Comprehension-Analysis: **Comprehension-Analysis**
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: 10/29/2013

Question #22

Given the following:

- Unit 2 is in Mode 4 with Reactor Recirculation system secured.
- 2A + 2B Shutdown Cooling loops are running at full 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: **Comprhension-Analysis**
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: 10/29/2013

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 from 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.
d. INCORRECT – Isolation Condenser/DW Atmos Temps Recorder 2(3)-1340-1 individual data points but not the average..

References:
Unit 2(3) DOS 1600-29 Unit 2 and 3 Drywell Temperature Surveillance; Rev 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: 10/29/2013

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: 10/29/2013

Question # 025L1

Unit 3 was operating at rated power when a transient occurred, requiring entry into DEOP 500, FAILURE TO SCRAM.

- RPV water level is being maintained within a band of -100 to -50 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: DEOP 500, Failure to Scram.
DEOP 500-5, Alternate Insertion of Control Rods; Rev 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: 10/29/2013

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. removal of radioactive noble gases.

b. removal of radioactive halogen gases.

c. maintaining a slightly negative pressure in the Reactor Building.

d. increasing the dilution air flow through the Reactor Building Ventilation Stack.

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

a. INCORRECT – Noble gases will pass through the SBGT train and will be exhausted to the 310 foot Chimney.
b. CORRECT – Radioactive halogen gases are removed by adsorbtion in the SBGT charcoal adsorber beds.
c. INCORRECT – The building is normally maintained at a slightly negative pressure. Simply maintaining a negative pressure does not reduce the concentration of radioactive material released. Maintaining a negative pressure ensures that radioactive material doesn’t exit the reactor building without first being processed by the SBGT train.
d. 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.

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: 10/29/2013

Question # 027

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

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

b. the Reactor Building Exhaust fan on low Reactor Building pressure.

c. all Rx Building Vent and Exhaust Fans ONLY.

d. all 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: **COMPREHENSION-ANALYSIS**
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: 10/29/2013

Question # 028

(DELETED Q, reworking)

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: Reeser
Exam Date: 10/29/2013

Question # 029

The unit was in Mode 4 with Shutdown Cooling (SDC) in operation, with one 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 48 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 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: **COMPREHENSION-ANALYSIS**
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: 10/29/2013

Question # 030L1

Following a Group 1 isolation and Reactor trip, 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?

a. HPCI flow will be automatically maintained at the flow controller setpoint.

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

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

d. HPCI flow will 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: **Comprehension-Analysis**
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: 10/29/2013

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 automatically starts on low IC shell side water level and transfers water from the Clean Demineralized Water Tank through an automatic level control valve.

d. One of the two Diesel Driven IC Makeup Pumps is manually started on low IC shell side water level and transfers water from the Clean Demineralized Water Tank through a remote manually operated makeup valve.

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: D 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.
C 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: 10/29/2013

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: 10/29/2013

Question # 033L1

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: 10/29/2013

Question # 034

Given the following plant conditions:

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

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: 10/29/2013

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: **Comprehension-Analysis**
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: 10/29/2013

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 $1x10^5$ 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 300 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: **Comprehension-Analysis**
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: 10/29/2013

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: 10/29/2013

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: 10/29/2013

Question # 039

Primary Containment Isolation System (PCIS) sensor and trip relays 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: b.

PCIS sensor and trip relays, and control logic relays are normally energized. The tripped state for sensor and trip relays do not seal-in. Control logic relays do seal-in and require actuation of a reset switch to re-energize.

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: 10/29/2013

Question # 040

Following a Group 1 isolation and Reactor trip, the NSO was directed to stabilize RPV pressure within a band of 900 to 1060 psig. To accomplish this task the NSO would:

a. Verify automatic actuation of the Isolation Condenser and manually cycle ADSVs, in the preferred sequence, as needed to maintain RPV pressure within the band.

b. Verify automatic actuation, and monitor operation, of the Isolation Condenser ONLY.

c. Manually cycle ADSVs, in the preferred sequence, as necessary to maintain RPV pressure until the Isolation Condenser can be manually placed in service.

d. Manually place the Isolation Condenser in service and monitor operation ONLY.

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

a & b. INCORRECT – With an uncomplicated Group 1 Isolation and reactor trip, the Isolation Condenser (IC) is not expected to automatically initiate. Following the MSIV closure pressure will increase until one or more ADSVs open. Pressure will then decrease until two or more ERVs are closed (approx. 1030-1050 psig). Pressure will then cycle on the ERV setpoints (1030-1120 psig) until manual control of the ERVs is taken and IC is initiated. RPV pressure must remain above 1068 psig for greater than 12 seconds. This is not expected as long as the ERVs function properly.

c. CORRECT

d. INCORRECT – DEOP 0100-00 directs that if ADSVs are cycling to take manual control of ADSVs until pressure is below 945 psig and then to cycle valves (in the preferred sequence if possible) as necessary to maintain PPV pressure below 1060 psig.

References:
DRE207LN001, Isolation Condenser Lesson Plan; Revision 7
DEOP 0100-00, RPV Control; Revision 10
DAN 902(3)-3 C-13, D-9, D-13, E-12 & E-13, Electromatic (Target Rock) Relief Valve Open; Revisions 17-19
Direct/New/Modified: **NEW**
Memory/Comprehension-Analysis: **COMPREHENSION-ANALYSIS**
Level (SRO/RO): **RO**
K/A: 239002.A2.06 Ability to (a) predict the impacts of the following on the RELIEF/SAFETY VALVES ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Reactor high pressure
Author: Reeser

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
- 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 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.

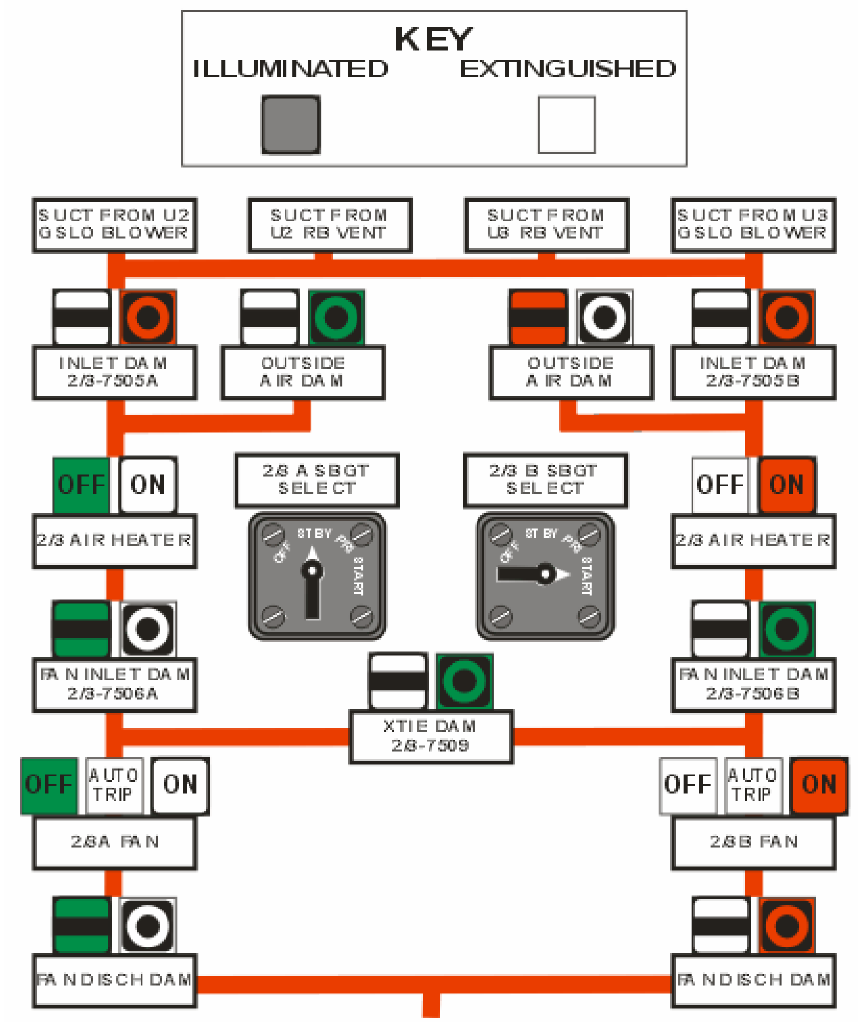
ANSWER: d. maintain the 3B FRV's position for one (1) second. After one (1) second, it will pulse down to 30% of its current demand.
Upon a scram signal, the system takes the setpoint to –10" and locks the FRVs in place for 1 second. At 1 second (T1) the programming looks at reactor water level and makes 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: **Direct** (2012 NRC)
Memory/Comprehension-Analysis: **Comprehension-Analysis**
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: 10/29/2013

Question # 042

Both units were operating at rated power with SBGT being operated due to maintenance on the Rx Bldg Vent systems, when **Unit 3** experienced a sustained loss of Instrument Air.

Given the initial system lineup below, what actions (if any) are required per DOA 4700-01, INSTRUMENT AIR SYSTEM FAILURE, for the SBGT system?



a. No actions are required.

b. Valve in the backup air supply to the SBGT system.

c. Reference Tech Specs, for exceeding SBGT flow limit.

d. Verify automatic start of the 'A' SBGT train and trip of the 'B' SBGT train.

ANSWER: c. Reference Tech Specs, for exceeding SBGT flow limit.

FROM LP DRE261LN001 - Unit 3 IA system supplies 2/3-7510-B, B SBGT TRN FAN SUCT AO VLV. Therefore, on a loss of Unit 3 IA system, ONLY 2/3-7510-B damper would fully OPEN, then the 2/3A SBGT Train would be the preferred train to operate if needed. If a loss of Instrument Air occurred, the affected SBGT train(s) will function, but the flow rate through the SBGT system may exceed the upper Tech Spec flow limit of 4400 scfm. This will reduce the residence time of iodine in the charcoal filters and therefore reduce the effectiveness of iodine removal. Thus it has been determined there is no concern of Control Room dose or off-site dose limits exceeding acceptable values.

References: None provided

Direct/New/Modified: **Modified** from bank Q#22234

Memory/Comprehension-Analysis: **Comprehension-Analysis**

Level (**RO**)

K/A: 261000 (SGTS): A4.09 Ability to manually operate and/or monitor in the control room: Ventilation valves/dampers.
Author: BANK - C. Zoia
Exam Date: 10/29/2013

Question # 044

During the performance of a test on an inoperable 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 for these non-emergency conditions?

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

a. approved by any SRO.

b. the Shift Manager is notified.

c. approved by any two SROs.

d. logged on the abnormal component position sheet.

ANSWER: c. The steps to isolate the leak can be taken when approved by any two SROs.. Configuration Control requires approval of two SROs per OP-AA-108-101, Operational Configuration Control. The other choices apply in different situations.

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: 10/29/2013

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: Direct (2009 Cert Exam)

Memory/Comprehension-Analysis: Memory

Level (**RO**)

K/A: G2 262002 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: 10/29/2013

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 175 volts and lowering.

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: **Direct** 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: 10/29/2013

Question # 047

Both units are operating at rated power with DC electrical systems in a normal lineup when the control room receives Annunciators 902-8, C-10, and 903-8 C-10; 250 VDC POWER FAILURE. The SRO enters DOA 6900-4, "Failure of Unit 2(3) 250 VDC Power Supply." A dispatched EO reports 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: 10/29/2013

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 DC Bus 2A-1.

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: 10/29/2013

Question # 049

A loss of offsite power (LOOP) has occurred on site coincident with a loss of coolant accident (LOCA) on Unit 3. Diesel Generator 3 has loaded onto Bus 34-1. However, a ground on bus 34-1 has resulted in actuation of the 5 minute timer. If the under-voltage condition persists, which of the following loads on Bus 34-1 will trip?

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. 2, 3 AND 4 ONLY.

d. 1, 2, 3, AND 4.

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: B. per “U 2/3 Auxiliary Power Lesson Plan,” pg. 22, the LPCI pumps and SDC pumps will trip. 480 VAC Emergency Bus and 3B RBCCW pumps will NOT trip.

References: U 2/3 Auxiliary Power Lesson Plan, pgs 22, 23, DOP 6500-27, “Removing 4KV Bus 34 from Operation for Maintenance/Testing,” Section D.3.
Direct/New/Modified: **Modified**
Memory/Comprehension-Analysis: **Memory**
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: 10/29/2013

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: 10/29/2013

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: 10/29/2013

Question # 052

Unit 2 was operating at full 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.

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.	an after cooler auto drain failure,	manually open the drain valve.
d.	a low instrument air system pressure,	open the dryer bypass valve.

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: A, per the DAN, if system pressure is >60 psig, then problem is either a plugged pre-filter or a malfunctioning dryer tower.
B & C are 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: **C/A**
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: 10/29/2013

Question # 053

Unit 2 was operating at rated power with CCSW status as follows:

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

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

a. (1) go full closed
   (2) keep the CCSW pump discharge piping full of water.

b. (1) will throttle to maintain 20 psid
   (2) maintain a positive d/p across the heat exchanger tubes.

c. (1) go full open
   (2) keep the CCSW piping downstream of the heat exchanger full of water

d. (1) will remain in present position.
   (2) to prevent damage from constant starting current.

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 started on a LPCI initiation.
C is incorrect, valve goes full closed with CCSW pumps tripping on a LPCI initiation.
D is incorrect. Reason is correct and why 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: 10/29/2013

Question # 054

Unit 3 was in Mode 1 when Annunciator 923-1, G6; U3 INST AIR PRESS LOW energized. 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?

a. Control rods have NOT scrammed, accumulators are charging.

b. Control rods have scrammed, accumulators are NOT charging.

c. Control rods have scrammed, no flow through the CRD cooling water header.

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

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: C: Per DOA 4700-01, Instrument Air Abnormal Procedure, Table A, Scram inlet and outlet valves fail open (rods scram) and CRD FCVs fail closed (no flow through the cooling water header). Accumulator charging is unaffected by loss of instrument air.

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: 10/29/2013

Question # 055

The unit is in Startup mode 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.

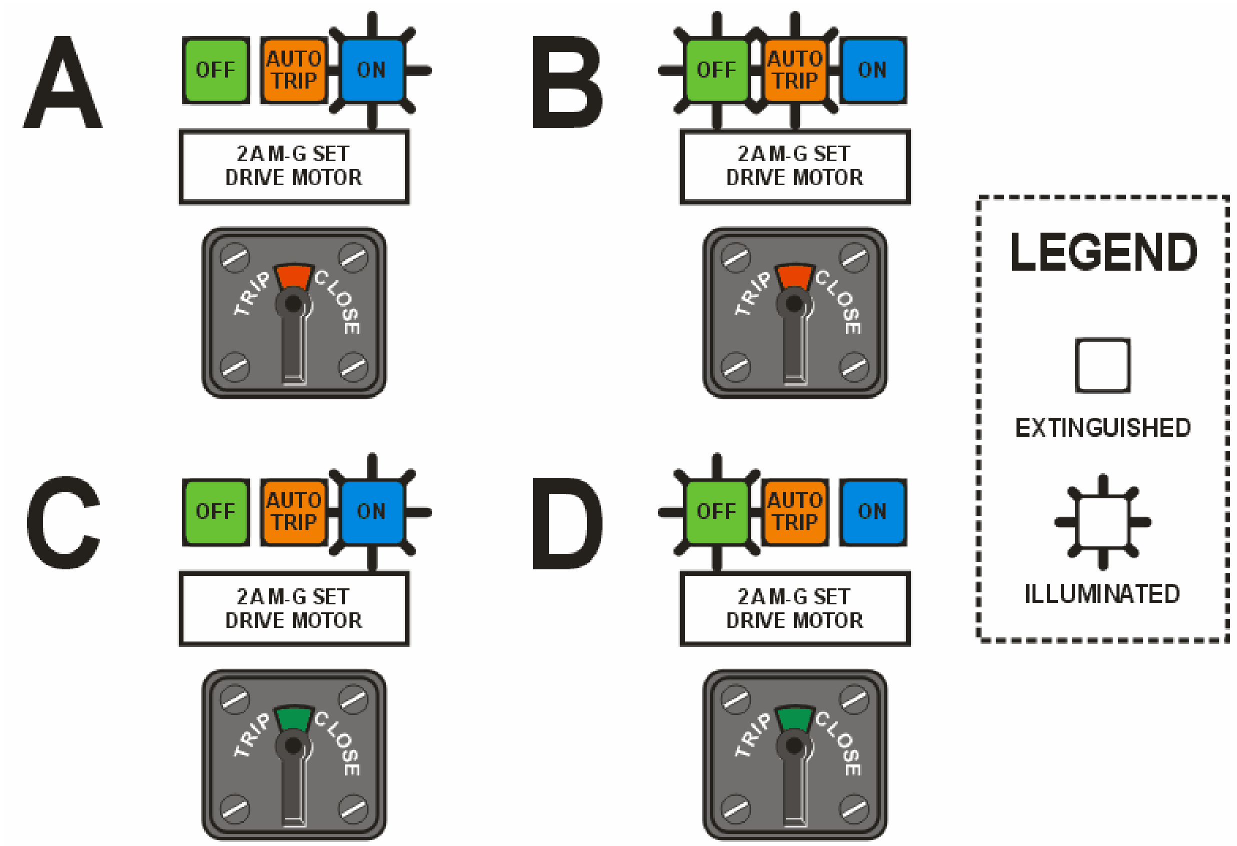
References: Reactor Theory, Chapter 5, Control Rods.

Direct/New/Modified: **New**
Memory/Comprehension-Analysis: **M**
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: 10/29/2013

Question # 056

Unit 2 was operating at near 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
Exam Date: 10/29/2013

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: 10/29/2013

Question # 058R1

Unit 2 is in refueling mode with the reactor vessel head removed:

- RWCU pump B is maintaining reactor water level at the flange level.
- The auxiliary RWCU pump is OOS.

While draining RPV inventory through the RWCU system to the radwaste system, the NSO identified RMC 2-1290-14, Drain Flow Controller indicated that FCV 2-1220, Drain Flow control valve, had closed. This condition was caused by an 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: **Memory**
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: 10/29/2013

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.	Powerplex.	withdrawn from
d.	Powerplex	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: 10/29/2013

Question # 060

Unit 2 is at 35% power in 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: 10/29/2013

Question # 061R1

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 deenergizes due to a fault and can NOT be reenergized. What effect 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 A is true, CCSW pumps powered by Bus 23. MOV's fail as is.
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.
C is false, power to LPCI pumps is from 23-1 bus
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: 10/29/2013

Question # 062

Unit 3 is starting up with the main generator connected to the grid at 22% 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: 10/29/2013

Question # 063

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: 10/29/2013

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 in 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) must 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: 10/29/2013

Question # 065

Conditions exist on Unit 3 such that there is an unisolable leak through both primary and secondary containments and the off-site release is approaching a General Emergency level. The SRO has entered into DEOP 300-02, “Radioactive Release Control.” Regarding management expectations for usage of DEOP 300-02, during these conditions, which one of the following is NOT true:

a. Operators are to maximize cooldown of the unit (may exceed 100°F/hr rate).

b. Operators are to make a PA announcement informing all station personnel of a release in progress AND its location.

c. The crew is to consider using the isolation condenser in anticipation of blowdown.

d. Senior Operators are to announce to the crew transition into and out of DEOP 300-02 AND the reason why.

Since reference was from OP-DR procedure, is this considered RO knowledge?

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: A is correct since operators are NOT to exceed the TS limit of 100°F/hr.
See references for reasons.
References: OP-DR-103-102-1002, Sections 4.3.2 and 4.3.7.

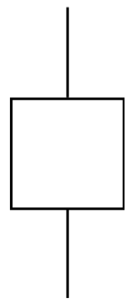
Direct/New/Modified: **New**
Memory/Comprehension-Analysis: **Memory**
Level (SRO/RO) **RO** (NOT SRO since references are from admin procedure, nothing from any DEOP!)
K/A: 27200 Radiation Monitoring, G2.4.13 Knowledge of crew roles and responsibilities during EOP usage.
Author: Walton
Exam Date: 10/29/2013

Question # 066

(Still working)

Question # 069

When reviewing an electrical print for a tag-out, the assigned operator asks you what the following symbol represents AND its function:



You reply that this symbol represents:

a. Circuit Recloser; provides momentary circuit protection for transient events.

b. Circuit Recloser; provides circuit protection for faults.

c. Oil Circuit Breaker; provides circuit protection for faults.

d. Oil Circuit Breaker; provides momentary circuit protection for transient events.

ANSWER: c: Oil Circuit Breaker provides circuit protection for faults.
All other answers have either a different looking symbol or function.
References: None provided. SEE Electrical Print Reading Lesson DRE101LN001. Figure 1.

Direct/New/Modified: **New**
Memory/Comprehension-Analysis: **Memory**
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: C. Zoia
Exam Date: 10/29/2013

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 parameters 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 1.11 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: 10/29/2013

Question # 073L1

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 MU16 (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 principals, if you operate MU16 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: 10/29/2013

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 +13 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, you expect the SRO will have you ……….

e. remain in DGP 02-03.

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

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

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

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: A. Do not meet entry conditions for entry into DEOPs.
B is incorrect, RPV parameters do not meet DEOP entry conditions.
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: 10/29/2013

Q#75
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) deenergizing various non-vital 250 VDC 2A and 2B loads.
   (2) prolong 250 VDC battery life.

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: Knowledge of RO Tasks performed outside the main control room during an emergency and the resultant operational effects.
Author: Walton
Exam Date: 10/29/2013

Q#76
Unit 3 is operating at rated power. I&C technicians are mistakenly working on 3B Recirculation pump differential pressure detector 3-260-4B. They open the detector's equalizing valve. A transient occurs. The SRO immediately enters ...

a. DGP 02-03, Reactor Scram.

b. DGP 03-03, Single Recirculation Loop Operation.

c. DOA 0202-01, Recirculation Pump Trip – One or Both Pumps.

d. DOA 0500-01, Inadvertent Entry into the Unstable Power/Flow Region.

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: C. Opening the detectors equalizing valve causes the detector to go to 0 psid. At 4 psid, DAN 903-4, A-7, "3B Recirc PP DP LO" alarms. The DAN requires entry into DOA 0202-01.
A is incorrect. Must enter DOA 0202-01 first, DGP 02-03 is referenced in DOA0202-01 flow chart, but conditions do not require an immediate reactor scram!
B & D distractors are not IAW DAN 903-4 A-7. These procedures may eventually be used, but not entered into immediately!

References: DAN 903-4, A-7. DOA 0202-01.

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 #77

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 due to a loss of RPS Bus A.

The following conditions exist after the scram signal was received:

- RPV Pressure is 1060 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.
- SBLC was initiated and is injecting.

Which of the following lists the order of procedures that the Unit Supervisor must enter OR direct the Operating team to perform?

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

Answers:

a. Enter #2; then enter #3; then enter #4.

b. Enter #1; then enter #3; then enter #4.

c. Enter #1; then enter AND exit #2; then enter #3;

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

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: D. 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: 10/29/2013

Question #79

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
- "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: 12 psig and rising slowly.
- Drywell temperature is: 240°F and rising slowly.
- Torus Bulk Temperature is: 225°F and steady.
- Torus Level is: 14 feet and steady.
- Torus Pressure is: 9.2 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. start one LPCI pump with flow <2750 gpm.

b. reduce CS flow to <2750 gpm, then start 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 starting one LPCI pump, keep LPCI flow <4000 gpm.

REFERENCE PROVIDED: DEOP 200-01, "Primary Containment Control" with entry conditions removed.

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: A; Per DEOP 200-01, Primary Containment Pressure Leg, Torus pressure 9.2 psig plus head of torus water level of 14 feet (adds ~6 psig to torus bottom pressure) equals ~15 psig pressure on Fig W. With torus bulk temp at 225°F must keep ECCS flow <10750 gpm. With 8000 gpm CS flow, have 2750 gpm flow remaining.
B is incorrect since do not want to divert any CS flow from core. Would be correct if RPV level was rising.
C is incorrect. Plausible if applicant does not add head of torus water to torus pressure and uses 10 psig line on Fig W.
D is incorrect since do not want to secure any CS flow injection to core since level is just maintaining.
References: 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

Q#82
The radwaste panel operator reports 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 reports 1000 liters of .07 milli-Ci was discharged to the river. The isotopic analysis of the discharge was 20% Tritium 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.
iii. To be included in the stations Annual Effluents Report.

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

REFERENCES PROVIDED: 10CFR Part 20, Appendix B.
Reportability Manual, Section RAD 1.4.

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: C: .07 mCi x 1000uCi/mCi = 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 and iii. 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.
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: 10/29/2013

Q#83
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.
- “A” RWCU pump is in operation.

The Refuel Coordinator from the refuel floor reports they would like to improve the clarity in the refuel cavity. The SRO directs:

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

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

c. 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.”

d. Place additional temporary cavity filtration equipment in service per DFP 0800-01, “Master Refueling Procedure.”

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: B: 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 & C 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: M
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: 10/29/2013

Q#84L1
Unit 3 is at rated power. Before I&C Technicians calibrate reactor water level instrument 3-263-57A, the SRO would review Technical Specification(s) _____(1)______ for equipment operability determination. During this maintenance activity, should a containment isolation signal occur, the reason for this would be a spurious actuation of reactor water level instrument(s) _____(2)_______.

a. (1) 3.3.6.1, Primary Containment Isolation Instrumentation AND 3.3.6.2, Secondary Containment Isolation Instrumentation.
   (2) 3-263-57B.

b. (1) 3.3.6.1, Primary Containment Isolation Instrumentation AND 3.3.6.2, Secondary Containment Isolation Instrumentation.
   (2) 3-263-58A OR 3-263-58B.

c. (1) 3.3.6.1, Primary Containment Isolation Instrumentation ONLY
   (2) 3-263-58A OR 3-263-58B.

d. (1) 3.3.6.2, Secondary Containment Isolation Instrumentation ONLY
   (2) 3-263-57B

REFERENCES REQUIRED??

Hidden Text below: FILE; OPTIONS; DISPLAY
ANSWER: B. Logic of ½ twice means 3-263-58A OR 3-263-58B 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. Technical Specs 3.3.6.1 and 3.3.6.2.

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: 10/29/2013

Q#86
Following a LOCA on Unit 3, and given the following conditions:

- Procedure DEOP 0500-03, Alternate Water Injection Systems, was selected, and G.12, INJECTION USING ONE CORE SPRAY PUMP OR ONE LPCI PUMP WITH SUCTION ALIGNED TO CST, is being evaluated.
- Condensate Storage Tank (CST) level is 15 feet.
- The 3B Core Spray (CS) is inoperable.
- The 3A CS suction pressure is 16 psig.
- The 3B LPCI suction pressure is 17 psig.
- The 3A LPCI suction pressure is 14 psig.

Which of the following actions should be directed by the Unit Supervisor (US)?

a. Align the 3A CS pump to the CST per G.12.b (UNIT 3 ONLY) ALIGN 3A CORE SPRAY PUMP SUCTION TO CST.
b. Align the 3A LPCI pump to the CST per G.12.c ALIGNING 2(3)A LPCI PUMP SUCTION TO CST.
c. Align the 3B LPCI pump to the CST per G.12.d ALIGNING 2(3)A LPCI PUMP SUCTION TO CST.
d. None of the actions above are appropriate due to CST level

ANSWER: b. Align the 3A LPCI pump to the CST per G.12.c ALIGNING 2(3)A LPCI PUMP SUCTION TO CST, because its suction pressure is <15 psig.
3A CS would normally be preferred if BOTH CS pumps were available. Also, The 3B LPCI pump would be the preferred LPCI pump due to the accessibility of its suction valve. However, neither can be used because their suction pressure is >15 psig. Finally, CST level is adequate for action b.
References: DEOP 0500-03, Alternate Water Injection Systems at section G.12, INJECTION USING ONE CORE SPRAY PUMP OR ONE LPCI PUMP WITH SUCTION ALIGNED TO CST (NOT provided during the exam)
New:
Comprehension-Analysis
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: C. Zoia
Exam Date: 10/29/2013

Q#89

During the performance of Technical Specifications (TS) surveillance SR 3.3.6.1.2, Perform CHANNEL FUNCTIONAL TEST, on Unit 2, a blown fuse within the PCIS instrumentation caused annunciator 902-5 E-5, GROUP 2 ISOLATION INITIATED, to come in. 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)

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: 10/29/2013

Q#98
As the SRO for a shift, you are in the third quarter and have to decide which NSO can substitute for a 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 has not stood a watch as an NSO in the 2nd quarter
James – has reported to the medical staff last week a condition requiring use of a prescription drug. 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.

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ANSWER: C: Both James and Julia meet requirements for standing watch.
Jim 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: 10/29/2013

Q#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: 10/29/2013