EXAMINATION ANSWER KEY

Dresden Station 2019-301 NRC Exam - SRO

76 **ID: 13826** **Points: 1.00**

Per DFP 0800-01, MASTER REFUELING PROCEDURE, which of the following is required to verify the Shutdown Margin is adequate prior to start of fuel moves?

A. Shift Manager.

B. Unit Supervisor.

C. Refuel Floor Supervisor.

D. Qualified Nuclear Engineer.

Answer: D

Question 76 Info	
Topic:	76 - Generic.2.1.40
Comments:	Objective: 23400LK005 Reference: DFP 0800-01 K/A: Generic 2.1.40 -- / 3.9 K/A: Knowledge of refueling administrative requirements. CFR: 43.5 Safety Function: N/A Level: Memory Pedigree: Bank History: 2008 NRC, 2013 Cert Comments: A - Incorrect. Plausible because the Shift Manager must be notified when core alterations have commenced or are in progress, this is not related to Shutdown Margin B - Incorrect. Plausible because the Unit Supervisor controls access to roped off areas underneath the Vessel during core alterations. C - Incorrect. Plausible because the Refuel Floor Supervisor maintains control of Refuel bridge operations and maintains logs of all Move Sheets. D - Correct. DFP 800-01 requires that EITHER a Reactor Engineer OR Qualified Nuclear Engineer perform the Shutdown Margin calculation is adequate for fuel moves. SRO Criteria: 5 **REQUIRED REFERENCES: None.**

77 **ID: 13808** **Points: 1.00**

In order to return a SRO Licensee to ACTIVE status from INACTIVE status, the Licensee must . . . . .

A. obtain special permission from the NRC Regional office for reactivation.

B. at a minimum, have received a passing grade on a special reactivation exam.

C. complete a minimum of 60 hours of shift functions under the direction of an operator or senior operator and in the position to which the individual will be assigned.

D. participate in a complete plant tour as part of a minimum of 40 hours of shift functions under the direction of an operator or senior operator and in the position to which the individual will be assigned.

Answer: D

Question 77 Info	
Topic:	77 - Generic.2.1.04
Comments:	Objective: DRE 299LK187 State the actions necessary to reactivate a license Reference: OP-AA-105-102, 10 CFR 55.53 K/A: Generic.2.1.04 --- / 3.8 K/A: Knowledge of individual licensed operator responsibilities related to shift staffing, such as medical requirements, “no-solo” operation, maintenance of active license status, 10CFR55, etc. CFR: 43.2 Safety Function: N/A Level: Memory Pedigree: Bank History: 2008 NRC, 2013 Cert Comments: A - Incorrect. Incorrect As long as the individual has no NRC restrictions special permission is not required. This is plausible as this would be correct if the reason for deactivating the license was NRC restrictions, i.e. INPO assignment, college, military, or foreign interexchange program. B - Incorrect. Incorrect There is no special reactivation exam as long as the individual is current in the requalification program. This is plausible due to the requirement of current in a training program (not more than 1 cycle behind). C - Incorrect. The number of hours under direction or observation is 40 hours not 60. This plausible because 60 hours are required per quarter if the individual is on 12 hour shifts. D - Correct. Before resumption of functions authorized by a license issued under this part, an authorized representative of the facility licensee shall certify the following: That the licensee has completed a minimum of 40 hours of shift functions under the direction of an operator or senior operator as appropriate and in the position to which the individual will be assigned. The 40 hours must have included a complete tour of the plant. SRO Criteria: 2 This an SRO function as the Shift Operations Superintendent will document and approve each individual apply for reactivation status. **REQUIRED REFERENCES: None.**

78	ID: 24186	Points: 1.00

Use of DOP 2000-111, Waste Surge Tk Radwaste Discharge to River with the Off-Stream Liquid Effluent Monitor NOT in Operation requires the authorization of the Shift Manager as well as the _______.

A. Unit Supervisor

B. Chemistry Manager

C. Shift Operations Supervisor

D. Radiation Protection Manager

Answer: B

Question 78 Info	
Topic:	78 - Generic 2.3.06
Comments:	Objective: DRE268LN001.14 Reference: DOP 2000-111 K/A: Generic.2.3.06 -- / 3.8 K/A: Ability to approve release permits. CFR: 43.4 Safety Function: N/A Level: Memory Pedigree: Bank History 14-1 NRC exam Explanation: With the discharge monitor not in use, then use DOP 2000-111 to perform a River Discharge. Prerequisite D.1 states that permission MUST be obtain from Shift Manager AND Chemistry Manager prior to implementation of this procedure. A. Incorrect This is plausible due to the Unit Supervisor must grant permission to perform a River discharge valve lineup. This is after the Reviews of the River Card. B. Correct With the monitor not in use DOP 2000-111 Attachment A requires signatures from both the Shift Manager and the Chemistry Manager prior to discharge. C. Incorrect With the monitor not in use the Shift Operations supervisor can verify calculations, but Shift Manager and Chemistry Manager are required signatures. D. Incorrect Radiation Protection Manager signature is not required in-spite of the radiation monitor not being used. This plausible because RP Manager signature is required during off normal events leading to plant releases. RP Manager must also be notified in DOP 2000-110 with monitor operable SRO per criteria 4. **REQUIRED REFERENCES: None.**

79	ID: 13377	Points: 1.00

Per the Technical Specification Bases, concerning a Unit 3 complete loss of AC power:

The Unit 3 AND Unit ____(1)____ EDGs must be capable of starting and connecting to their buses.

This is designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the design limits are NOT exceeded for ____(2)__ .

A. (1) 2 **AND** 2/3
(2) Fuel **AND** Reactor Coolant System **ONLY**

B. (1) 2 **ONLY**
(2) Fuel, Reactor Coolant System, **AND** Containment.

C. (1) Unit 2 **AND** 2/3
(2) Fuel, Reactor Coolant System, **AND** Containment

D. (1) 2/3 ONLY
(2) Fuel **AND** Reactor Coolant System **ONLY.**

Answer: C

Question 79 Info	
Topic:	79 - 295003.G.2.25
Comments:	Objective: 299LN001-2 Reference: Tech Spec Bases 3.8.1 K/A: 295003.G.2.25 3.2 / 4.2 K/A: Partial or Complete Loss of A.C. Power: Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits. Safety Function: 6 CFR: 43.2 Level: Memory Pedigree: Bank History: N/A Comments: A. Incorrect For a loss of voltage on any given unit, the unit EDG and common EDG (not opposite unit) must be capable of starting and connecting to its own bus(on an undervoltage), to count as a qualified onsite circuit. Therefore the U2 EDG is not required. The first part is plausible due to the ability to cross-tie Unit Diesels and the second part is plausible because it is partially correct. B. Incorrect For a loss of voltage on any given unit, the unit EDG and common EDG (not opposite unit) must be capable of starting and connecting to its own bus(on an undervoltage), to count as a qualified onsite circuit. The first part is plausible due to the ability to cross-tie Unit Diesels. The second part of the answer is correct for the 2/3 EDG. C. Correct For a loss of voltage on any given unit, the unit EDG and common EDG (not opposite unit) must be capable of starting and connecting to its own bus(on an undervoltage), to count as a qualified onsite circuit. This design ensures sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the designed safety limits of the fuel, Reactor Coolant System, and containment are not exceeded. Two qualified circuits between the offsite transmission network and the onsite Class 1E AC Electrical Power Distribution System, two separate and independent DGs, one qualified circuit between the offsite transmission network and the opposite unit's Division 2 onsite Class 1E AC Electrical Power Distribution subsystem capable of supporting equipment required to be OPERABLE by LCO 3.6.4.3, "Standby Gas treatment (SGT) System," LCO 3.7.4, "Control Room Emergency Ventilation (CREV) System" (Unit 3 only), and LCO 3.7.5, "Control Room Emergency Ventilation Air Conditioning (AC) System" (Unit 3 only), and the opposite unit's DG capable of supporting the equipment required to be OPERABLE by LCO 3.6.4.3, LCO 3.7.4 (Unit 3 only), and LCO 3.7.5 (Unit 3 only), ensure availability of the required power to shut down the reactor and maintain it in a safe shutdown condition after an anticipated operational occurrence (AOO) or a postulated DBA. D. Incorrect The 2/3 EDG is required per part 1 of the question. In addition to Fuel and Reactor Coolant system, Containment limits are also protected. SRO Criteria: 2 **REQUIRED REFERENCES: None.**

80	ID: 22439	Points: 1.00

Unit 2 is operating at near rated power with TIRS 2-1640-200A, TORUS TEMP MON DIV I temperature recorder O.O.S., when a transient occurred, resulting in the following set of conditions:

- Drywell pressure is 1.55 psig.
- RPV water level is +12 inches.

An NSO reported that the TIRS 2-1640-200B TORUS TEMP MON DIV II temperature recorder currently indicates the following:

- Point 1 103°F
- Point 2 107°F
- Point 3 113°F
- Point 4 115°F
- Point 5 93°F
- Point 6 115°F
- Point 7 121°F
- Point 8 120°F

____(1)____ requires entering DEOP 200-1, PRIMARY CONTAINMENT CONTROL, and the Unit Supervisor must direct starting all available ____(2)____.

A. (1) Torus temperature **ONLY**
(2) Torus cooling **ONLY**.

B. (1) Torus temperature **ONLY**;
(2) Torus cooling **AND** direct a scram.

C. (1) Torus temperature **AND** Drywell pressure
(2) Torus cooling and Torus Sprays **ONLY**.

D. (1) Torus temperature **AND** Drywell pressure
(2) Torus cooling and Torus Sprays **AND** direct a scram.

Answer: B

Question 80 Info	
Topic:	80 - 295013.G.4.47
Comments:	Objective: 29502LK011 Reference: DEOP 200-1 K/A: 295013.G.4.47 --/4.2 K/A: High Suppression Pool Temperature: Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material. Safety Function: 5 CFR: 43.5 Level: High Pedigree: Bank History: NRC 2009 Comments: A - Incorrect The torus bulk temperature exceeds the DEOP 200-1 entry condition. Starting torus cooling would be a correct action but because temperature is also above 110 and requires a reactor scram. B - Correct The average (bulk) Torus temperature is 110.9°F (which is above the DEOP 200-1 entry condition of 95°F). The required actions for this is to start all available Torus Cooling and also when average Torus temperature is above 110°F, the required actions are to scram. While a Drywell pressure of 1.5 psig is an Operations department action level, it is NOT an entry condition for DEOP 200-1. Starting Torus Sprays would be correct if DEOP 200-1 was entered on Drywell pressure. C - Incorrect Torus temperature has exceeded the DEOP 200-1 entry condition of 95 degrees. Drywell pressure has not exceeded the DEOP entry of 2 psig. Torus cooling would be a correct action but because 2 psig has not been exceeded Torus Sprays are not needed. Starting Torus Sprays would be correct if DEOP 200-1 was entered on Drywell pressure and entry is plausible because a Drywell pressure of 1.5 psig is an Operation department action level. D - Incorrect Torus temperature has exceeded the DEOP 200-1 entry condition of 95 degrees. Drywell pressure has not exceeded the DEOP entry of 2 psig. Torus cooling would be a correct action but because 2 psig has not been exceeded Torus Sprays are not needed. Additionally the Scram threshold has been met. Starting Torus Sprays would be correct if DEOP 200-1 was entered on Drywell pressure and entry is plausible because a Drywell pressure of 1.5 psig is an Operation department action level. SRO per Criteria: 5 **REQUIRED REFERENCES: DEOP 200-1 with the entry conditions blanked out.**

81	ID: 24233	Points: 1.00

Given the following:

- Unit 2 is at rated conditions
- IMD is installing a jumper in the 902-55 panel to troubleshoot the SO-2-2499-1A, DW H2/O2 MON INLET VLV that failed to open during DOS 0040-07, VERIFICATION OF REMOTE POSITION INDICATION FOR VALVES INCLUDED IN INSERVICE TESTING (IST) PROGRAM.
- The work order package does NOT have a 10CFR50.59 screening.

What is the MAXIMUM time this jumper may remain installed without having a 10CFR50.59 screening?

A. 30 days

B. 60 days

C. 90 days

D. 120 days

Answer: C

Question 81 Info	
Topic:	81 - Generic 2.2.05
Comments:	Objective: 29900LK148 Reference: LS-AA-104-1000 K/A: Generic 2.2.05 2.2/3.2 K/A: Knowledge of the process for making design or operating changes to the facility. CFR: 43.3 PRA: No Level: Memory Pedigree: Bank History: 2015 NRC Comments: A - Incorrect. This is time allowed for a 50.59 screening. B - Incorrect. This is plausible if the candidate believes 90 days will require a 50.59 screening. C - Correct. IAW LS-AA-104-1000 4.2.2, a temporary alternation is exempt from performing a 50.59 review as long as it is in direct support of maintenance. Temporary alternations that support maintenance including jumpering terminals, lifting leads, etc. An approved procedure must exist (CC-AA-112) for tracking and controlling the 90 day period. Otherwise at a minimum a 50.59 screening must be performed. CC-AA-112 step 2.5 declares: although the "90 days from the current date" is conservative, the addition of this date will result in a "flag" in the schedule that reflects the earliest possible date that the temporary configuration change has to be removed, UNLESS the maintenance activity that it supports is completed sooner. Note, that MR90s are required to be removed when the maintenance activity is completed or within 90 days of the temporary change installation, whichever comes first. 50.59 uses terms like "90 days at power" and not just 90 days. While "90 days at power" are the absolute limits, the ability to control the limits of temporary change installation is based on just 90 days. The reason for using 90 days and not taking credit for the "at power" is that the NEI guidance has defined "at power" as beginning when the reactor goes critical. The difficulty in controlling duration limits based on a "reactor criticality date" that may come earlier or later than scheduled introduces a variable that is difficult to use in establishing an easily identified removal date. Risk considerations associated with the temporary change are addressed as part of the maintenance activity being performed. Procedure WC-AA-101 governs the risk evaluation for the for the on-line maintenance activity. MR90 work orders should have the term 'MR90' in their title to further identify that it is a Maintenance Rule (a)(4) temporary change. D - Incorrect. This is plausible if the candidate incorrectly applies SR 3.0.2 which allows which allows for a 25% extension of completion times for Technical Specification surveillance Requirements. SRO CRITERIA: 3. Facility licensee procedures required to obtain authority for design and operating changes in the facility. **Required References: None.**

82 **ID: 27489** **Points: 1.00**

Unit 2 was operating at 100% power when the 2A RR Pump tripped.

- Total core flow is 55%
- Reactor Power is 48%
- OPRM Indicating Lights on the 902-5 panel are OFF

(1) What is the status of the OPRMs?
(2) What is the Unit Supervisor required to direct?

A. (1) OPRMs are operable.
(2) Raise running Recirc pump speed.

B. (1) OPRMs are inoperable.
(2) Raise running Recirc pump speed.

C. (1) OPRMs are operable.
(2) Manually enable the OPRMs on the 902-37 panel.

D. (1) OPRMs are inoperable
(2) Manually enable the OPRMs on the 902-37 panel.

Answer: D

Question 82 Info	
Topic:	82 - 295001.A2.02
Comments:	Objective: 29900LK157 Reference: DGP 03-03, TS 3.3.1.1 K/A: 295001.A2.02 3.1/3.2 K/A: Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION: Neutron monitoring CFR: 43.5 Safety Function: 1 Level: High Pedigree: New History: N/A Comments: A - Incorrect. OPRMs are inoperable, however the 902-5 panel indications should be green vice white. The second part is plausible because raising RR pump speed will move the unit out of the OPRM enabled region, however this is not the appropriate response. B - Incorrect. The first part is correct OPRMs are inoperable, however the 902-5 panel indications should be green vice white. The second part is plausible because raising RR pump speed will move the unit out of the OPRM enabled region, however this is not the appropriate response. C - Incorrect. OPRMs are inoperable, and not enabled. Manual enabling is done on the 902-37 panel. This is plausible because the OPRMs can be operated from the 902-5 panel, however the control is limited to bypass only. D - Correct. OPRMs are inoperable, but once enabled will become operable again. Manual enabling is done on the 902-37 panel. SRO Criteria: 5 **REQUIRED REFERENCES: None.**

83	ID: 27491	Points: 1.00

Unit 2 is operating at 100% power with 2B TBCCW pump OOS.

- Annunciator 923-1 C-1, U2 OR U3 TBCCW PP TRIP, is in alarm.

1 Minute later 923-1 B-5, U2 OR U3 INST AIR COMP TRIP, alarms.

- DOA 4700-01, INSTRUMENT AIR SYSTEM FAILURE, is entered.

If annunciator 902-5 A-1, SCRAMVLV AIR SUPPLY PRESS LO, is received, what procedures will the Unit Supervisor be required to enter?

1. DGP 2-3, REACTOR SCRAM
2. DEOP 100, RPV CONTROL
3. DGP 3-1, POWER CHANGES

A. (1) **ONLY**

B. (1) **THEN** enter (2)

C. (2) **THEN** enter (1)

D. (3) **THEN** enter (1)

Answer: B

Question 83 Info	
Topic:	83 - 295018 G.2.4.05
Comments:	Objective: 29900LP061 Reference: DAN 923-1 C-2, DAN 923-1 B-5, DOA 4700-01, DEOP 100 K/A: 295018 G.2.4.05 4.3 K/A: Knowledge of the organization of the operating procedures network for normal, abnormal, and emergency evolutions: Partial or Total Loss of CCW. CFR: 43.5 Safety Function: 8 Level: High Pedigree: New History: N/A Comments: A. Incorrect In addition to DGP 2-3 for the Reactor Scram, DEOP would also have to be entered for low reactor water level. This is plausible because not all reactor scrams require entry into DEOP 100 the candidate must determine that at full power setpoint setdown will take level below the entry level. B. Correct A loss of all TBCCW will cause a trip of all instrument air compressors in less than a minute. When in the DOA for loss of instrument air and annunciator 902-5 A-1 comes in then a Reactor Scram is required and entry into DGP 2-3. With the unit at 100% power, setpoint setdown will take Reactor water level below 8 inches and require an entry into DEOP 100. C. Incorrect With a loss of instrument air the alarm for Scram Air supply pressure requires a Manual Reactor Scram. This would take place prior to DEOP 100 entry. This is plausible due to other alarms that come in prior to Scram Air supply are related to FW Reg vlvs on backup air or FW Reg vlvs lockup. This would indicate that DEOP entry condition may occur prior to automatic scram. D. Incorrect There is not procedural guidance in DOA 4700-01 to take an emergency load drop, therefore DGP 3-1 is not entered. As stated above from full power DEOP 100 must also be taken. This is plausible because other DOA's require emergency load drop in the case of impending scram conditions to minimize the transient on the plant. SRO Criteria: 5 With a loss of TBCCW the Instrument Air system will trip in less than a minute. With a loss of Instrument air the SRO must determine when to take a reactor scram and with the Unit at full power what is the impact of setpoint setdown and the actions needed for recovery. **REQUIRED REFERENCES: None.**

84	**ID: 27492**	**Points: 1.00**

Unit 2 is in refuel, with fuel moves in progress.

- A failure of the grapple caused a fuel bundle to drop into the core.
- Refuel Floor Area Radiation Monitor (ARM) High Radiation is reading 1.2 R/Hr.
- The Standby Gas Treatment System auto started.

The SRO is required to direct evacuation of the ____(1)____ and declare an ____(2)____.

A. (1) Refuel Floor **ONLY**;
(2) Unusual Event

B. (1) Refuel Floor **ONLY**;
(2) Alert

C. (1) Drywell **AND** Refuel Floor;
(2) Unusual Event

D. (1) Drywell **AND** Refuel Floor;
(2) Alert

Answer: D

Question 84 Info	
Topic:	84 - 295023.A2.01
Comments:	Objective: 23400LK001 Reference: DFP 0850-03, DOA 0010-08, DAN 923-5 A-1, EP-AA-1004 Addendum 3 K/A: 295023.A2.01 3.6/4.0 K/A: Ability to determine and/or interpret the following as they apply to REFUELING ACCIDENTS: Area rad levels CFR: 43.5 Safety Function: 8 Level: High Pedigree: New Comments: A - Incorrect. A dropped fuel bundle is a symptom of drywell evacuation DOA 0010-08. The first part is plausible because the accident and indications are located on the refuel floor. The second part is plausible because unplanned ARM readings rising is one condition that factors into an Unusual Event classification. B - Incorrect. A dropped fuel bundle is a symptom of drywell evacuation DOA 0010-08. The first part is plausible because the accident and indications are located on the refuel floor. The second part is correct. C - Incorrect. This is plausible due to the first part being correct. The second part is plausible because unplanned ARM readings rising is one condition that factors into an Unusual Event classification but an Unusual event requires loss of water level AND unplanned ARM readings rising. D - Correct. Local ARM reading above the alert value, and confirmation of high value (SBGT autostarts at 100 mr/hr on the refuel floor) Alert (RA2) criteria has been reached. With a dropped fuel bundle and local rad levels confirmed to be above alarm setpoint, the SRO is required to direct evacuation of BOTH the Drywell and refuel floors. SRO per Criteria: 5 EP classification is a SRO only task at Dresden Station **REQUIRED REFERENCES: EP-AA-1004 Addendum 3.**

85 **ID: 27493** **Points: 1.00**

Unit 2 is at 100% power and Unit 3 is in Refuel.

Work on the 923-2, 345 Kv switchyard panel MOD is in progress.

Cutting and grinding causes a small fire with a large amount of smoke.

The fire is extinguished within 3 minutes.

The Unit Supervisor directs...

A. Position CRM ISOL switch to ISOLATE
CREVS remains operable while in Isolate Mode

B. Position CRM ISOL switch to ISOLATE
Declare CREVS inoperable while in Isolate Mode

C. Position CRM AIR FLOW CONTROL switch to OUTSIDE
CREVS remains operable while in Purge Mode.

D. Position CRM AIR FLOW CONTROL switch to OUTSIDE
Declare CREVS inoperable while in Purge Mode.

Answer: D

Question 85 Info	
Topic:	85 - 600000 A.2.05
Comments:	Objective: 28800LK004 Reference: DOA 5750-04, T.S. 3.7.4 K/A: 600000 A.2.05 --/3.0 K/A: Ability to determine and/or interpret the following as they apply to Plant Fire On Site: Ventilation alignment necessary to secure affected area: CFR: 45.8 Safety Function: 8 Level: High Pedigree: New History: N/A A. Incorrect With the source of the smoke from in the Control Room then the correct mode would be would be PURGE. This is plausible because step D.2 of DOA 5750-04, has a decision point on whether to go to D.5 or D.8. This would be correct if D.8 was chosen. B. Incorrect With the source of the smoke from in the Control Room then the correct mode would be would be PURGE. In addition CREWs would be operable in Isolate Mode. This is plausible because step D.2 of DOA 5750-04, has a decision point on whether to go to D.5 or D.8. This would be correct if D.8 was chosen. The second part is correct C. Incorrect The first part is correct. With a fire in the control room causing smoke or noxious fumes entry is required into DOA 5750-04, SMOKE, NOXIOUS FUMES OR AIRBORNE CONTAMINANT IN THE CONTROL ROOM. If the origin of the smoke is from inside the control room then Step D.5 requires placing Main Control Room HVAC to the PURGE MODE. In the Purge Mode with the dampers selected for OUTSIDE, the T.S for Control Room Envelope is not met and the system must be declared INOP. D. Correct With a fire in the control room causing smoke or noxious fumes entry is required into DOA 5750-04, SMOKE, NOXIOUS FUMES OR AIRBORNE CONTAMINANT IN THE CONTROL ROOM. If the origin of the smoke is from inside the control room then Step D.5 requires placing Main Control Room HVAC to the PURGE MODE. While in the Purge Mode of operation CREVs is inoperable per Tech Spec 3.7.4. SRO Criteria : This question meets Criteria 5 for SRO only due to the fact that the On Site Fire would require entry into multiple DOA.s and include Operability calls based on Tech Spec compliance **Required References: None.**

86	ID: 27494	Points: 1.00

Unit 2 was operating at 100% when a scram occurred.

Aux Power fast transfer failed to occur.
- EDG's are operating as designed.
- HPCI has failed to start.
- Power has been restored to Bus 23.
- Bus 24 is overcurrent.
- RPV pressure is 800 psig and lowering slowly.
- RPV level is -148" and lowering slowly.

What is the SRO required to direct NEXT?

A. Control RPV water level between +8" and +48" using condensate and feed system.

B. Start 1 SBLC pump and 2A CRD pump to control RPV level between +8" and + 48"

C. Start both SBLC pumps and 2A CRD pump to control RPV level between +48" and -170"

D. Exit DEOP 100, RPV CONTROL and enter DEOP 400-2 EMERGENCY DEPRESSURIZATION.

Answer: C

Question 86 Info	
Topic:	86 - 295031.G.2.1.23
Comments:	Objective: 29501LK024 Reference: DEOP 100, DOP 1200-02 K/A: 295031.G.2.1.23 4.3/4.4 K/A: Ability to perform specific system and integrated plant procedures during all modes of plant operation: Reactor Low Water Level CFR: 43.5 Safety Function: 2 Pedigree: New Level: High History: N/A Comments: A - Incorrect. This is prohibited due to the initial loss of all condensate pumps. Restart of condensate system is not allowed. This is plausible because it would be the correct answer if aux power had transferred as designed. B - Incorrect. This would be the correct number of SBLC pumps to start for an ATWS, however when using SBLC for level, both pumps are required. This is plausible because the level band would be correct if SBLC was not used. (i.e. only CRD) C - Correct. Due to RPV level above TAF, attempts to restore RPV level shall be made. The only high pressure injection sources available are SBLC and 1 CRD pump. If the decision to use SBLC is made, the band for RPV level is above TAF D - Incorrect. This is plausible because this would be correct if RPV pressure was less than 500 psig. SRO Criteria: 5 **REQUIRED REFERENCES: None.**

87	**ID: 27497**	**Points: 1.00**

Unit 2 is operating at 100% power.

DOP 0700-06, TRAVERSING INCORE PROBE (TIP) SYSTEM OPERATION, is being performed in AUTO MODE.

A FWLC transient occurs causing Rx water level to drop to +3 inches.

TIPS DRIVE CONTROL CH 'C' indicates detector position at BOTTOM CORE LIMIT and will not move.

Ball Valve Ch 3 red position indication is illuminated.

What is the expected impact on the TIPs system **AND** what action MUST be taken?

A. Group 2 signal to isolate TIPs
Place the MAN VALVE CONTROL to CLOSED for channel 3 Ball Valve.

B. Group 3 signal to isolate TIPS
Place the MAN VALVE CONTROL to CLOSED for channel 3 Ball Valve.

C. Group 3 signal to isolate TIPs
Isolate TIP tube by turning keylock switch to FIRE

D. Group 2 signal to isolate TIPs
Isolate TIP tube by turning keylock switch to FIRE.

Answer: D

Question 87 Info	
Topic:	87 - 215001 A2.01
Comments:	Objective: 215LN001.08 Reference: DAN 902-5 E-5. DOP 0700-06 K/A: 215001 A2.01 --/2.9 K/A: Ability to predict the impacts of the following on the Traversing In-core Probe: and based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operation: Low reactor water level. CFR: 43.5 Safety Function: 7 Level: High Pedigree: New History: N/A Comments: A. Incorrect With Tip operation in Auto Mode and a group 2 signal present, TIP detectors should automatically runback to In-shield position and Ball vlv will close. Rx water level below +8 inches will cause a Gr 2 isolation. With indication of detector at bottom position and a red light indication on the channel 3 ball valve, attempting to close the ball valve will not be effective. B. Incorrect Water level at +3 inches would cause a group 3 initiation but would not signal TIPs to isolate. With indication of detector at bottom position and a red light indication on the channel 3 ball valve, attempting to close the ball valve will not be effective. C. Incorrect With low reactor water level present a group 3 would occur but this would not affect TIPs valve operations. The actions to isolate the TIP tube would be a correct for a Group 2. This is plausible due to the student must understand which valve are affected. And that a red light indication on the ball valve indicates that the valve is still open. D. Correct With Tip operation in Auto Mode and a group 2 signal present, TIP detectors should automatically runback to In-shield position and Ball vlv will close. Rx water level below +8 inches will cause a Gr 2 isolation. If the auto action does not occur, then Attachment C of DOP 0700-06 provides contingency actions that must be taken. The SRO Must verify conditions and then Fire the Shear valve to isolate TIPs. SRO Criteria: 5 The chosen K/A does not have a 43.b CFR tie. This question meets the SRO only criteria for assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. Per DOP 0700-06 the action to fire the sheer vlvs requires SRO permission. (see step 1.d of Attachment C) **REQUIRED REFERENCES: None.**

88	ID: 27495	Points: 1.00

Unit 2 is at 100% power.

- Annunciator 902-3 A-1, AREA RAD HI, is in alarm
- Annunciator 902-3 G-2, AREA TEMP HI, is in alarm

RWCU pump area rad level is 26 mr/hr.
RWCU pump area temperature is 215 degrees F.
The leak can **NOT** be isolated

3 minutes later...

- Annunciator 902-3 A-3, RX BLDG VENT CH B RAD HI HI, alarms
- Annunciator 902-3 F-14, RX BLDG VENT CH A RAD HI HI, alarms

What is the highest emergency action level thresholds and classification that must be declared?

A. Declare an ALERT, FA1

B. Declare an ALERT, RA3

C. Declare UNUSUAL EVENT, RU3

D. Declare SITE AREA EMERGENCY, FS1

Answer: D

Question 88 Info	
Topic:	88 - 295034 G.2.04.41
Comments:	Objective: 29501LP032A Reference: DAN 902-3 A-3, DAN 902-3, DEOP 300-1, EP-AA-1004 K/A: 295034 G.2.04.41 -- / 4.6 K/A: Knowledge of the emergency action level thresholds and classifications. Secondary Containment Ventilation High Radiation. CFR: 43.5 Safety Function: 9 Level: High Pedigree: New History: N/A Comments: A. Incorrect The setpoint for 902-3 A-3 and F-14 is 4 mr/hr. This is the same threshold for entry into DEOP 300-1. Classification of FA1 threshold is met but the information meets the higher threshold of FS1. This is plausible because while the potential loss of Reactor Coolant based on Max Normal temperature level is exceeded, a loss of containment is also met. B. Incorrect DEOP 300-1 threshold for Area Rad is 30 mr/hr. This has not been exceeded. ALERT RA3 threshold for UNPLANNED elevated radiation levels may be exceeded, but per Table R3 they are applicable only in Modes 3, 4 and 5. C. Incorrect RU1 would be correct if the candidate does not differentiate between RB Vent Rad Hi-Hi and Offgas Rad Hi-Hi D. Correct With RWCU pump area temperature of 215 degrees the Max Safe threshold has been exceeded. This would require DEOP entry for Area Temperature above max normal. Being above Max Safe temperature would be a loss of Containment and a Potential Loss of Reactor Coolant System. The threshold for FS1 is a Loss or Potential Loss of ANY two barriers. SRO Criteria: 5 EAL classifications are a SRO only task. **Required References: EP-AA-1004 Addendum 3 and DEOP 300-1**

89	ID: 27498	Points: 1.00

Unit 2 was operating at 100% power when the 902-4 panel indicating lights for 2B Reactor Recirc Pump Precharge indicated amber and green.

What is the appropriate ARPM Flow Biased Neutron Flux - High setpoint to maintain RPS System operable?

A. $\leq$ 0.56W + 67.4% RTP and $\leq$ 118.5% RTP

B. $\leq$ 0.56W + 67.4% RTP and $\leq$ 122% RTP

C. 0.56W + 63.2% RTP and $\leq$ 118.5% RTP

D. 0.56W + 63.2% RTP and $\leq$ 122% RTP

Answer: C

Question 89 Info	
Topic:	89 - 215005.G.2.1.07
Comments:	Objective: 215LN005.07 Reference: TS 3.4.1, TS 3.3.1.1, DGP 03-03 K/A: 215005.G.2.1.07 4.4/4.7 K/A: Average Power Range Monitors: Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation CFR: 43.2 Safety Function: 7 Pedigree: New Level: High History: N/A Comments: A - Incorrect. This is plausible because this is the TS required value with both RR pumps running. B - Incorrect. This is plausible because the MAX CTP is clamped, however this is not the only required TS adjustment. C - Correct. With only 1 RR pump running, TS 3.3.1.1 requires adjustment to the APRM Flow Biased Neutron Flux-High RPS actuation setpoint. This is the correct formula. D - Incorrect. While the formula based on flow is correct, the MAX CTP clamp is incorrect but plausible because it would be correct for both pumps running. SRO Criteria: This knowledge objective is SRO ONLY at Dresden Station as it requires knowledge of information contained in a table (beyond the requirements of RO Only Knowledge). SRO Criteria 2. **REQUIRED REFERENCES: T.S. 3.3.1.1**

90 **ID: 27040** **Points: 1.00**

Unit 3 was operating at 100% power with the 3A Fuel Pool Cooling (FPC) pump O.O.S. for motor replacement.
Bus 39 de-energized due to an overcurrent condition.

The Unit Supervisor is required to direct **LOCAL** monitoring of fuel storage pool water temperature ____(1) once per hour locally and ____(2)____ to ensure limit(s) are not exceeded.

A. (1) **ONLY**
(2) start additional RBCCW pumps, per DOP 3700-02 REACTOR BUILDING CLOSED COOLING WATER SYSTEM OPERATION

B. (1) **ONLY**
(2) align a SDC pump to the fuel pool system, per DOP 1000-04, FUEL POOL COOLING MODE OF OPERATION OF SHUTDOWN COOLING SYSTEM

C. (1) **AND** level
(2) start additional RBCCW pumps, per DOP 3700-02, REACTOR BUILDING CLOSED COOLING WATER SYSTEM OPERATION

D. (1) **AND** level
(2) align a SDC pump to the fuel pool system, per DOP 1000-04, FUEL POOL COOLING MODE OF OPERATION OF SHUTDOWN COOLING SYSTEM

Answer: D

Question 90 Info	
Topic:	90 - 233000.G2.01.32
Comments:	Objective: DRE233LN001.08 Reference: DOA 1900-01 K/A: 233000.G2.01.32 --/4.0 K/A: FUEL POOL COOLING AND CLEANUP. Ability to explain and apply system limits and precautions. CFR: 43.2 Safety Function: 9 Level: High Pedigree: Bank History: 2015 NRC Explanation: A. Incorrect - With loss of power to both FPC pumps, fuel pool levels and temperatures must be monitored locally. MCR level indication is lost due to power supply failure and temperature indications may be erroneous due to water stagnation. The first part is plausible because it is partially correct. Starting additional RBCCW pumps would be correct if FPC system was not maintaining temperature with forced flow present. B. Incorrect - The first part is plausible because it is partially correct. Alignment of SDC pump to FPC is correct action for given conditions. Due to water stagnation, temperature indications (although available) in the MCR may be erroneous and should be monitored locally. C. Incorrect - The first part is correct. Temperature and level of Fuel Storage Pool shall be monitored locally. Starting additional RBCCW pumps is incorrect. Starting additional RBCCW pumps would be correct if FPC system was not maintaining temperature with forced flow present. D. Correct - Due to lack of power to both FPC pumps, MCR alarm capability is removed for spent fuel pool level. (Actually based on water level in the spent fuel pool surge tank). Local temperature monitoring is required due to possible erroneous MCR indications caused by water stagnation. SRO Criteria: 5, The candidate is required to assess facility conditions and select the appropriate procedure during abnormal situations **REQUIRED REFERENCES: None.**

91	ID: 27499	Points: 1.00

Unit 2 is in MODE 2 with SRM 21 bypassed.

- IRMs are on ranges 3 and 4
- SRM 23 becomes Inoperable.

What are the TS/TRM implications?

A. SRM Instrumentation is INOP per TS 3.3.1.2.

B. Control Rod Block Instrumentation is INOP per TS 3.3.2.1.

C. Control Rod Block Instrumentation is INOP per TRM 3.3.a.

D. Post Accident Monitoring (PAM) Instrumentation is INOP per TRM 3.3.b

Answer: C

Question 91 Info	
Topic:	91 - 215004.A2.04
Comments:	Objective: 205LN004.07 Reference: TRM 3.3.a, TRM 3.3.b, 3.3.2.1, 3.3.1.2 K/A: 215004.A2.04 3.5/3.7 K/A: Ability to (a) predict the impacts of the following on the SOURCE RANGE MONITOR (SRM) SYSTEM; and based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Up scale and downscale trips CFR: 43.2 Safety Function: 7 Pedigree: New Level: High History: N/A Comments: A - Incorrect. This is plausible because the required number of SRMs are not met, however this is only applicable when IRMs are less than Range 2. B - Incorrect. This is plausible because while the required number of SRMs are not met for Control Rod Block Instrumentation, the applicability is not per TS. C - Correct. Per TRM 3.3.a 3 SRM channels are required given the conditions in the stem. D - Incorrect. This is plausible because SRMs are part of PAM instrumentation and 2 channels are required for PAM to be operable. SRO Criteria: 2 **REQUIRED REFERENCES: None.**

92 **ID: 27500** **Points: 1.00**

DFPS 4195-03, AUXILIARY ELECTRICAL EQUIPMENT ROOM HALON DAMPER TEST, is being performed.

One of the dampers fails to close.

What actions are required to be taken?

A. Declare CO2 inoperable **ONLY**
Establish Hourly Fire Watch

B. Declare Halon **AND** CO2 inoperable
Establish Hourly Fire Watch

C. Declare Halon **AND** CO2 inoperable
Establish Continuous Fire Watch

D. Declare Halon inoperable **ONLY**
Establish Continuous Fire Watch

Answer: C

Question 92 Info	
Topic:	92 - Generic 2.4.25
Comments:	Objective: 286002.16 Reference: DFPS 4195-03, OP-MW-201-007, TRM 3.7.n K/A: Generic 2.4.25 -- / 3.7 K/A: Knowledge of fire protection procedures. CFR: 43.5 Safety Function: 8 Level: High Pedigree: New History: N/A Comments: A. Incorrect CO2 does have to be declared inoperable but Halon must also be declared inop. The fire watch established must be continuous. The first part is plausible because it is partially correct. The second part is plausible due to the fact that CO2 is an AEER system and TRM 3.7.n has an option for an hourly fire watch in Condition A. B. Incorrect Both Halon and CO2 must be declared inoperable. With Halon inoperable the procedure also requires a continuous Fire watch as opposed to Hourly. Part one is correct. Part two is plausible due to the fact that CO2 is an AEER system and TRM 3.7.n has an option for an hourly fire watch in Condition A. C. Correct If when performing DFPS 4195-03 a damper fails to close, the DFPS states that Halon and CO2 must be declare inoperable. With Halon and CO2 inop a continuous fire watch must be established. D. Incorrect Halon does have to be declared inoperable but CO2 must also be declared inop. With Halon inoperable a continuous Fire Watch must be established. Part one is plausible because it is partially correct. The second part is correct. SRO Criteria: 5 **REQUIRED REFERENCES: None.**

93	ID: 27501	Points: 1.00

An annunciator in the Radwaste Control Room has been determined to be a nuisance alarm.

Who must be informed and how is the disabled alarm tracked?

A. The Shift Manager must be informed. The alarm is documented in the Turnover, and a yellow sticker is placed on annunciator window

B. The Shift Manager must be informed. The alarm is documented in IR, and a green dot placed on annunciator window

C. The Field Supervisor must be informed. The alarm is documented in the Turnover, and a green dot placed on annunciator window

D. The Field Supervisor must be informed. The alarm is documented in IR, and a yellow sticker is placed on annunciator window

Answer: C

Question 93 Info	
Topic:	93 - Generic.2.43
Comments:	Objective: 298L042 Review and Maintain the Equipment Status log (SRO ONLY) Reference: OP-DR-108-101-1002, OP-AA-103-102 K/A: G.2.2.43 3.0/3.3 K/A: Knowledge of the process used to track inoperable alarms CFR: 43.5 Level: Memory Pedigree: New History: N/A Comments: A - Incorrect. OP-DR-108-101-1002 directs documentation in the Turnover. This is plausible because it would be correct in the main control room with the exception of the yellow dot. B - Incorrect. OP-DR-108-101-1002 directs documentation in the Turnover. Yellow stickers indicate time delays so alarms do not become a nuisance. The correct dot would be green. This is plausible because the green dot is correct. An IR is plausible because it would be required if major equipment was not available. C - Correct. The Field Supervisor must supervise annunciator response in the RWCR. OP-DR-108-101-1002 directs documentation in the Turnover and application of a green dot to annunciator window. D - Incorrect. OP-DR-108-101-1002 directs application of green dot to annunciator window. Yellow stickers indicate time delays so alarms do not become a nuisance. This is plausible because the first part is correct. An IR is plausible because it would be required if major equipment was not available. SRO Criteria: This is SRO ONLY required knowledge at Dresden Station. The position of Field Supervisor is an SRO position. As such the responsibility of these positions is SRO ONLY knowledge. **REQUIRED REFERENCES: None.**

94 **ID: 27502** **Points: 1.00**

U2 is operating at 100% power when a transient occurs.

- Torus Bottom Pressure is 20 psig and going up at .5 psig/min.
- Torus water level is 15 feet and steady.
- Reactor water level is -140 inches and lowering at 2 in/min.
- Drywell temperature is 200 degrees F and going up 10 degrees/min.
- Reactor Pressure is 800 psig and lowering 30 lbs/min

Which safety function is the first priority and why?

A. Containment Integrity
Blowdown due Drywell Temperature

B. Containment Integrity
Blowdown due to Pressure Suppression Limit (PSP)

C. Reactor Water Inventory Control
Entering DEOP 400-3 for Steam Cooling

D. Reactor Water Inventory Control
Blowdown due to Rx water level below TAF

Answer: D

Question 94 Info	
Topic:	94 - Generic 2.4.22
Comments:	Objective: 29800LP041 Reference: DEOP 100, DEOP 200-1, DEOP 400-3 K/A: Generic 2.4.22 --/4.4 K/A: Knowledge of the bases for prioritizing safety functions during abnormal/emergency operations. CFR: 43.5 Safety Function: 2 and 5 Level: High Pedigree: New History: N/A Comments: A. Incorrect The limit for Drywell temperature is 338 degrees. This would be met in 14 minutes at the current rate. This is plausible because if it was not recognized that the correction factor was no longer in effect, then reactor water level would not have reached TAF for 15 minutes. In addition in the previous revision of the DEOPs the limit was 281 degrees which would have been met first. B. Incorrect The limit for PSP is 26 psig for Torus level at 15 feet. This would be met in 12 minutes at the current rate. This is plausible because if it was not recognized that the correction factor was no longer in effect, then reactor water level would not have reached TAF for 15 minutes. C. Incorrect The limit for reactor water level entry into STEAM COOLING with no injection source is 162 inches. This would be at the 11 minute mark. In addition there is no mention of which injection sources are available. This is plausible because if it was not recognized that the correction factor was no longer in effect, then reactor water level would not have reached TAF for 15 minutes. D. Correct With water level dropping at 2 in/min, in addition to reactor pressure dropping at 30 psig/min, will cause the level correction factor to not be in affect after 10 minutes due to reactor pressure below 500 psig. This will require a Reactor Blowdown due to Rx water level below TAF. SRO Criteria : 5 **REQUIRED REFERENCES: None.**

95	ID: 27504	Points: 1.00

Unit 2 is in MODE 3.

DOS 1100-03, STANDBY LIQUID CONTROL INJECTION TEST, is in progress.

The SBLC INJECTION CONTROL keylock switch is taken to SYS 1 position.

The 2 1201-1, RX OUTLET ISOL valve fails to close.

What is the status of Group 3 PCIS **AND** what actions MUST be taken:

A. INOP
Isolate both trains of SBLC per DOP 1100-01

B. INOP
Isolate RWCU system per DOP 1200-01

C. OPERABLE
Isolate 2A SBLC per DOP 1100-01

D. OPERABLE
Isolate RWCU system per DOP 1200-01

Answer: B

Question 95 Info	
Topic:	95 - 223002 A2.11
Comments:	Objective: 223LN001.07b Reference: DOS 1100-03, T.S. 3.3.6.1 K/A: 223002 A2.11 --/3.9 K/A: Ability to predict the impacts of the following on the PCIS/Nuclear Steam Supply Shutoff; and based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Standby liquid initiation. CFR: 43.5 Safety Function: 5 Level: High Pedigree: New History: N/A Comments: A. Incorrect PCIS logic inop if one channel from SBLC is not operable. The corrective action is to either declare 2A SBLC inop or isolate RWCU. This is plausible because it would take a decision on whether 1 or both trains of SBLC are operable. B. Correct With one channel of SBLC logic to PCIS inoperable, PCIS Group 3 is INOP and either 2A channel of SBLC must be declared INOP or RWCU system must be isolated. C. Incorrect With DOS 1100-03 failure of the 2A SBLC isolation logic to RWCU, The PCIS system is declared INOP. SBLC or RWCU must be isolated. Part one is plausible because 1 train of SBLC is still available. D. Incorrect With DOS 1100-03 failure of the 2A SBLC isolation logic to RWCU, The PCIS system is declared INOP. SBLC or RWCU must be isolated. Part one is plausible because both trains of SBLC are still available. SRO Criteria: 5 **REQUIRED REFERENCES: None.**

96	ID: 27507	Points: 1.00

Unit 2 was operating at 100% power when a manual reactor scram was required.

During DGP 02-03 hardcard actions
- ARI has been actuated.
- Reactor power remains 100%
- RPV level is determined to be unknown.

Over the next 3 minutes the following occur:
- Annunciator for PCIS GRP II - Reset
- 902-5 panel RPV level Lo - Reset
- HPCI Turbine Trip - Illuminated
- Main Turbine Trip - Illuminated

The SRO can determine RPV level to be in which of the following bands?

A. 0 to +15"

B. +16" to +30"

C. +31" to +45"

D. > +46"

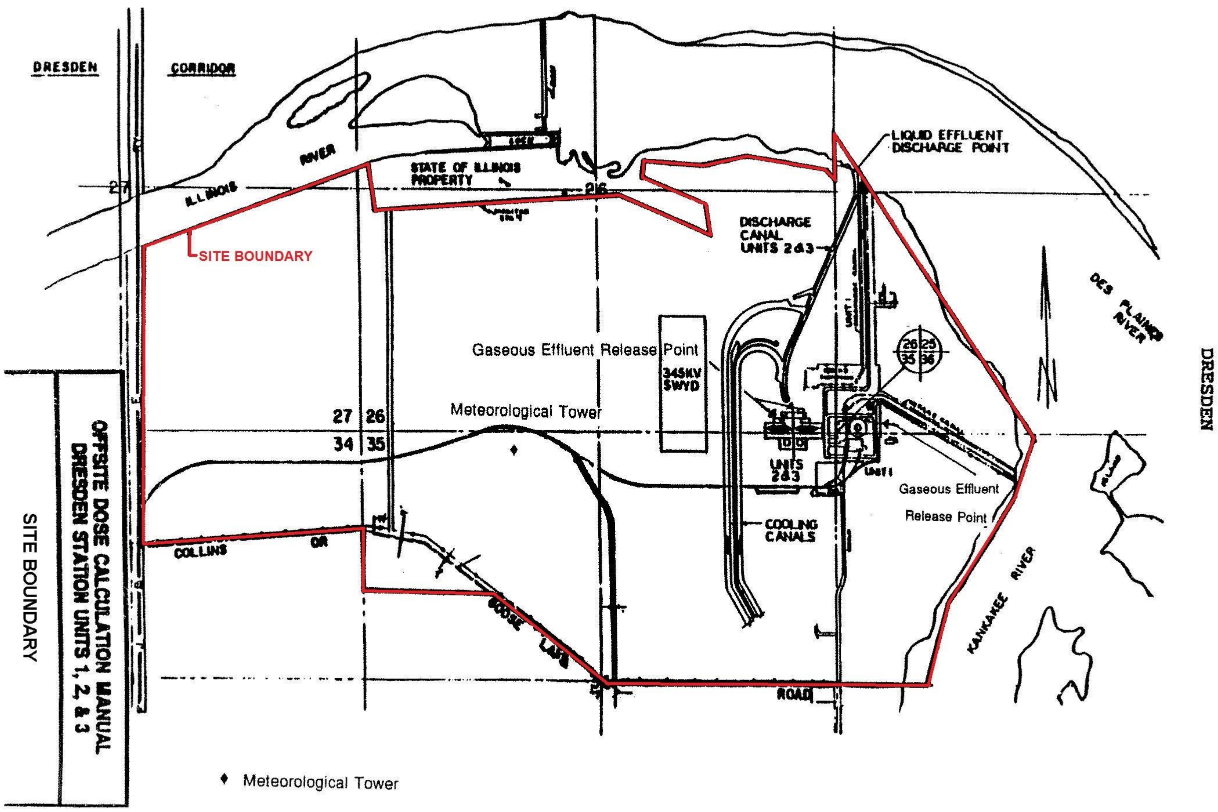
Answer: D

Question 96 Info	
Topic:	96 - 295005.A2.07
Comments:	Objective: 29502LK070 Reference: DEOP 0010, DAN 902(3)-5 K/A: 295005.A2.07 3.5/3.6 K/A: Ability to determine and/or interpret the following as they apply to MAIN TURBINE GENERATOR TRIP: Reactor Water Level CFR: 43.5 Safety Function: 3 Pedigree: New Level: High History: N/A Comments: A - Incorrect. This would be correct based on the PCIS Group II resetting (+8") B - Incorrect. This would be correct based on the RPV level lo resetting (+25") C - Incorrect. This would be correct based on the HPCI turbine trip (+43") D - Correct. This is correct based on the turbine being tripped. This will occur at +46” Therefore the SRO can adequately determine that RPV level is somewhere above +46". RPV level remains unknown, but indication remains. SRO CRITERIA: 5 SRO must determine if RPV level is known or unknown with respect to DEOP actions, SRO must validate inputs into each annunciator to verify if they are independent or redundant information. **REQUIRED REFERENCES: None.**

97 **ID: 27496** **Points: 1.00**

Which of the following gamma dose rates as measured by a field survey will meet or exceed the SITE AREA EMERGENCY threshold for off-site release?

1) 150 mR/hr at the Meteorological Tower
2) 180 mR/hr in the 345 KV switchyard
3) 110 mR/hr at Lift Station
4) 120 mR/hr in the Training Building parking lot



A. 3 **ONLY**

B. 1 and 2 **ONLY**

C. 3 and 4 **ONLY**

D. 1, 2, **AND** 4

Answer: A

Question 97 Info	
Topic:	97 - 295017.A2.01 (Print in Color)
Comments:	Objective: 29502LK056 Reference: EP-AA-1000, ODCM, EP-AA-1004 Addendum 3 K/A: 295017.A2.01 2.9*/4.2* K/A: Ability to determine and/or interpret the following as they apply to HIGH OFF-SITE RELEASE RATE: Off-site release rate: Plant Specific CFR: 43.5 Safety Function: 9 Pedigree: New Level: High Explanation: A - Correct. Although this is not the highest rad level, this is the only location outside the site-boundary (off-site). B - Incorrect. Although these rad levels are in excess of Site Area EAL thresholds, the 345 kv switchyard is on site. This location is commonly thought of as off-site C - Incorrect. Although these rad levels are in excess of Site Area EAL thresholds, this location is on site. This location is commonly thought of as off-site D - Incorrect. Although these rad levels are in excess of excess of Site Area EAL thresholds, some of these locations are on site. It is plausible a candidate may interpret off-site as outside the protected area. SRO Criteria: 5 **REQUIRED REFERENCES: EP-AA-1004 Addendum 3**

98 **ID: 27506** **Points: 1.00**

U2 is operating at 100% power.

Annunciator 902-8 D-9, U2 250 VDC BATT GROUND, alarms.

The Operator dispatched reports a 200Vdc ground (85 k ohms) with no buttons pushed.

What level of ground is present AND whose approval is required to perform DOP 6900-04, UNIT 2 250V DC GROUND DETECTION?

A. Level II
Unit Supervisor

B. Level II
Shift Manager

C. Level III
Unit Supervisor

D. Level III
Shift Manager

Answer: A

Question 98 Info	
Topic:	98 - 263000 A2.01
Comments:	Objective: 263LN001.05 Reference: DAN 902-8 D-9, DOP 6900-04 K/A: 263000 A2.01 --/3.2 K/A: Ability to predict the impacts of the following on the DC Electrical Distribution: and based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Grounds. CFR; 43.5 Safety Function: 6 Level: High Pedigree: New History: N/A Comments: A. Correct The alarm threshold for DAN 902-8 D-9, U2 250 VDC BATT GROUND, is 183.6 VDC, this is the same as the threshold for a Level II ground with no buttons pushed. The threshold for a Level III ground is 229.4 VDC. The Unit Supervisor must give permission to perform ground checks for a level II ground. Shift Manager notification to contact General Manager BWR Operations is required for a Level III ground. B. Incorrect The correct Level of ground is II based on 200 Vdc with no buttons pushed. Unit Supervisor approval is required to perform ground checking procedure. In addition the SM would have to notified if a Lvl III ground was occurring. C. Incorrect The Lvl III threshold of 229 VDC has not been met. This is plausible because the 200 Vdc while less than the 229 of a Lvl III ground it is well above the Lvl threshold, D. Incorrect The Lvl III threshold of 229 VDC has not been met. Unit Supervisor approval is required to perform ground checking procedure. This is plausible due to 200 Vdc while less than the 229 of a Lvl III ground it is well above the Lvl threshold, In addition the SM would have to notified if a Lvl III ground was occurring. SRO Criteria: 5 **REQUIRED REFERENCES: None.**

99	**ID: 27508**	**Points: 1.00**

U2 is operating at 100% power.

A Hydraulic ATWS occurs. The RO reports the following:

- Rx power is 20%.
- Rx water level is 0 inches in auto.
- Torus temperature is 95 degrees and slowly rising.
- 902-5 Panel Hard Card actions have been completed.

What is the SRO required to direct with regard to the FWLC system and the DEOP bases for the action?

A. lower level to -35 inches in manual
to minimize boron dilution

B. lower level to -35 inches in manual
to mitigate consequences of large irregular neutron flux oscillations.

C. lower level to -35 inches in auto
to minimize boron dilution

D. lower level to -35 inches in auto
to mitigate consequences of large irregular neutron flux oscillations.

Answer: B

Question 99 Info	
Topic:	99 - 259002 G.2.04.18
Comments:	Objective: 29502LK037 Reference: DEOP 400-05, EPG Contingency 5 chapter 14 K/A: 259002 G.2.04.18 --/4.0 K/A: Knowledge of the specific bases for EOPs, Reactor Water Level Control System. CFR: 43.1 Safety Function: 2 Level: High Pedigree: Bank History: N/A Comments: A. Incorrect With power reported at greater than 6% and reactor water level at 0 inches, then terminate to at least -35 inches is required. This must be done in manual to close the feed reg valves. With boron injecting into the vessel systems used for control of water level must be operated so as to minimize boron dilution. This is plausible because actions taken by the hardcard are to lower level to -40 inches in auto, but once power and reactor level have been announced, the action to terminate and prevent to -35 must be taken. B. Correct With a Scram required and power above 6% water level must be less than -35 inches. With level currently 0 inches, power at 20%, and Torus temp less than 110 degrees, level must be lowered to less than -35 inches. This is to minimize subcooling and minimize consequences of large irregular neutron flux oscillations. C. Incorrect With power reported at greater than 6% and reactor water level at 0 inches, then terminate to at least -35 inches is required. Level must be lowered to less than -35 inches in manual regardless of system actuations. This is plausible because if a terminate and prevent are not required, then this is how RPV level would be controlled. This is also plausible because with boron injecting into the vessel systems used for control of water level must be operated so as to minimize boron dilution D. Incorrect This is the how the FWLC is normally operated to control RPV level 2 ft below the feedwater spargers. FLWC setpoint is driven to -40" and FWLC remains in automatic. However since terminate and prevent conditions exist, the actions may no longer be performed in auto. SRO Criteria: 1 **REQUIRED REFERENCES: None.**

100	**ID: 27505**	**Points: 1.00**

Unit 2 is operating at 100% power when RPV level reached 0"

- DGP 02-03 Hard Card actions are being taken.
- Recirc Pumps are running at minimum speed.
- 3 APRM downscale lights are illuminated.
- Torus Temperature is 109 F.

The US ____(1)____ direct tripping Recirc Pumps ____(2)____.

A. WILL
This is required to provide a prompt reduction in power.

B. WILL
This is required because they did not trip on ARI initiation.

C. Will **NOT**
would reduce boron mixing efficiency.

D. Will **NOT**
because reactor power is less than 6%.

Answer: A

Question 100 Info	
Topic:	100 - 202001.G.2.4.09
Comments:	Objective: 29502LK046.C Reference: EPG B-6-52-54, DGP 02-03, DEOP 0400-05 K/A: 202001.G.2.4.09 3.8/4.2 K/A: Recirculation System: Knowledge of low power/shutdown implications in accident (e.g. loss of coolant accident or loss of residual heat removal) mitigation strategies. CFR: 43.5 Safety Function: 1 Pedigree: New Level: High History: N/A Comments: A - Correct. The purpose of tripping the RR pumps is to provide a prompt reduction in power. This is only required if reactor power is above 6% (APRM downscales) B - Incorrect. Although ARI will trip the RR pumps, this is only if ARI is automatically initiated via logic. If manually initiated from the control room, the RR pumps do not trip. C - Incorrect. While tripping the RR pumps will reduce the boron mixing efficacy if boron is injected, the reduction in power is of precedence. D - Incorrect. Reactor power is not below 6%, therefore the application of the decision to trip the RR pumps needs still be addressed. SRO Criteria: 5 **REQUIRED REFERENCES: None.**