EXAMINATION ANSWER KEY

19-1 (2020-301) NRC Exam - SRO

1	ID: 27600	Points: 1.00

Unit 2 is operating at 100% power.

Instrument Air header pressure is slowly dropping.
DOA 4700-01, INSTRUMENT AIR SYSTEM FAILURE, is being performed.

As pressure continues to drop, backup nitrogen will be supplied to the Feed Reg valves at___(1)___ psig.

BY DESIGN, backup nitrogen is expected to operate for **AT LEAST** ___(2)____ minutes.

A. (1) 65
(2) 30

B. (1) 65
(2) 60

C. (1) 83
(2) 30

D. (1) 83
(2) 60

Answer: C

Answer Explanation

Per DAN 902(3)-6 H-10 FW REG VLVS BACKUP AIR ACTIVE, on decreasing instrument air pressure of 83 psig, the feedwater regulating valves are supplied with nitrogen for operation from the backup supply.
Per DAN 902(3)-6 H-10 FW REG VLVS BACKUP AIR ACTIVE,Feedwater Regulation Valves Backup Supply will supply N2 for FWRV Operation for at least 30 minutes.

Question 1 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	3.00
System ID:	27600
User-Defined ID:	27600
Cross Reference Number:	
Topic:	01 - 295019 A1.01
Num Field 1:	
Num Field 2:	
Text Field:	

Comments:	Objective: 278LN001.08 References: DAN 902(3)-6 H-10, DOA 4700-01 K/A: 295019 A1.01 3.5/3.3 K/A: Ability to operate and/or monitor the following as they apply to Partial or Total Loss of Inst. Air: Backup air supply. Safety Function: 8 CFR: 41.7/45.6 PRA: No Level: Memory Pedigree: New History: N/A Justification for Memory level: FWLC Lesson Plan, Objective 12, covers knowledge of the effect that a total loss or malfunction of various plant systems has on the plant.. The pressure at which backup nitrogen will activate, and the length of time the backup nitrogen will function is covered under this objective, which is required to be known from memory. This knowledge is important since it gives the operators knowledge of how long the backup air would be expected to be available if a loss or reduction in normal air supply pressure occurs. Explanations: A. Incorrect - (1) with lowering instrument air pressure the backup nitrogen system will open at 83 psig. (2) The second part of the answer is correct. Plausible because (1) 65 psig is the pressure at which the feed reg valves will lockup if pressure continues to drop. (2) The second part of the answer is correct. B. Incorrect - (1) with lowering instrument air pressure the backup nitrogen system will open at 83 psig. (2) Feedwater Regulation Valves Backup Supply will supply N2 for FWRV Operation for at least 30 minutes. Plausible (1) because 65 psig is the pressure at which the feed reg valves will lockup if pressure continues to drop. (2) Part 2 is plausible because of the number of systems and requirements that are 1 hour. DAN 902(3)-6 H-10 FW REG VLVS BACKUP AIR ACTIVE states that it will last at least 30 minutes. May be misinterpreted as 60 minutes C. Correct - Per DAN 902(3)-6 H-10 FW REG VLVS BACKUP AIR ACTIVE, on decreasing instrument air pressure of 83 psig, the feedwater regulating valves are supplied with nitrogen for operation from the backup supply. Per DAN 902(3)-6 H-10 FW REG VLVS BACKUP AIR ACTIVE,Feedwater Regulation Valves Backup Supply will supply N2 for FWRV Operation for at least 30 minutes. D. Incorrect - (1) The first part of the answer is correct. (2) DAN 902-6(3) H-10 FW REG VLVS BACKUP AIR ACTIVE states that will supply for at least 30 minutes. Plausible (1) because part 1 is correct and (2) Part 2 is plausible because of the number of systems and requirements that are 1 hour. DAN 902(3)-6 H-10 FW REG VLVS BACKUP AIR ACTIVE states that it will last at least 30 minutes. May be misinterpreted as 60 minutes **Required References: None**

2	**ID: 23463**	**Points: 1.00**

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

- Drywell temperature is 170ºF.
- Primary Containment water level is 18 ft.
- Drywell/Torus pressure is rising.

Which one of the following is the **LOWEST** pressure that containment integrity can **NO LONGER** be assured?

A. Drywell pressure of 62 psig.

B. Torus bottom pressure of 26 psig.

C. Torus bottom pressure of 62 psig.

D. Drywell/Torus differential pressure of +2 psid.

Answer: C

Answer Explanation

Given a Primary Containment (Drywell or Torus) water level of 18 feet, a Torus bottom pressure of 62 psig is the lowest value listed that would threaten the integrity of containment.

Question 2 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	3.00
System ID:	23463
User-Defined ID:	23463
Cross Reference Number:	
Topic:	02 - 295024.K1.01
Num Field 1:	
Num Field 2:	
Text Field:	

Comments:	Objective: DRE223LN001.12 Reference: DEOP 200-1, EOP-DEOP TB K/A: 295024.K1.01 4.1 / 4.2 K/A: Knowledge of the operational implications of the following concepts as they apply to HIGH DRYWELL PRESSURE: Drywell Integrity: Plant-Specific. CFR: 41.8 Safety Function: 5 PRA: No Level: Memory Pedigree: Bank History: 2013 Cert Justification for Memory Level: Lesson Plan DRE223LN001, Objective 3, covers knowledge of the Drywell components under normal operating conditions. This knowledge, to be known from memory, includes knowledge of design limits for temperature and pressure. This is important information to an operator, since it provides the basis for knowledge of when component operating limits have been exceeded. Explanations: A. Incorrect - Drywell pressure is nominally 4 to 5 psig lower than torus bottom pressure. Therefore 62 psig torus bottom pressure would occur first. Plausible because candidate must determine that the 62 psig limit used in DEOP technical bases is torus bottom versus DW pressure, and that although this value threatens containment limits, it is not the lowest value that does. B. Incorrect - Torus bottom pressure would violate PSP if the torus was at its normal level and would therefore require a blowdown but by itself would not impact DW integrity. Plausible because at normal torus level, this pressure would require a blowdown, and if blowdown could not be performed the ability for the torus to quench all of the steam would be in jeopardy. C. Correct - Given a Primary Containment (Drywell or Torus) water level of 18 feet, a Torus bottom pressure of 62 psig is the lowest value that would threaten the integrity of containment. D. Incorrect - Drywell/Torus DP of 2 psig would exceed normal operating limits but integrity would not be challenged with a positive value. Plausible because candidate must determine that a negative DP not a positive DP would cause damage. **REQUIRED REFERENCES: None.**

Question 2 Table-Item Links

General Question Data - Site Ownership

Dresden

General Question Data - Ops Program

Reactor Operator

3	ID: 23979	Points: 1.00

Unit 3 was operating at near rated power when a transient occurred, resulting in an ATWS condition, requiring SBLC injection.

SBLC injects ___(1)___ the Bottom Core Plate

and

___(2)___ boron is needed to keep the reactor shutdown as the moderator temperature goes down.

A. (1) below;
(2) less

B. (1) above;
(2) less

C. (1) below;
(2) more

D. (1) above;
(2) more

Answer: C

Answer Explanation

As boron concentration in the core goes up, the amount of THERMAL neutrons available for fissioning of fuel goes down, thereby reducing reactor power. More boron is needed to keep the reactor shutdown under cold conditions verses hot conditions due to positive reactivity added by moderator temperature coefficient.

Question 3 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	2.00
System ID:	23979
User-Defined ID:	23979
Cross Reference Number:	
Topic:	03 - 295037.K1.03
Num Field 1:	
Num Field 2:	
Text Field:	

Comments:	Objective: DRE211LN001.01 Reference: UFSAR 9.3.5, M-12, EOP-DEOP TB K/A: 295037.K1.03 4.2 / 4.4 K/A: Knowledge of the operational implications of the following concepts as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN: Boron effects on reactor power (SBLC). K/A Justification: Location of boron injection applies to this K/A due to the possibility of RWCUs taking boron out of the core. From UFSAR 9.3.5.3 - In the event that the SBLC system interlock fails to isolate the reactor water cleanup system, the cleanup system would start to remove the boron from the core. This removal rate would be extremely small because of the flow path of the boron; the boron is inserted into the bottom of the vessel, moves up through the core, then to the outside of the core shroud. Safety Function: 1 CFR: 41.8 to 41.10 PRA: No Level: Memory Pedigree: Bank History: 2011 Cert Explanations: A. Incorrect - (1) The flowpath for boron injection comes in below the bottom core plate (2) due to temperature coefficient more boron would have to be added. Plausible because (1) part 1 is correct and (2) it must be known if net positive or negative reactivity is added due to decrease in temperature with boron injection. Moderator temperature coefficient becomes less negative with rising boron concentration. Less boron is plausible because as the reactor coolant *boron concentration* increases, the *moderator temperature coefficient* becomes less negative. This *is* because a 1°F increase in reactor coolant *temperature* at higher *boron concentrations* results in a larger increase in the thermal utilization factor. B. Incorrect - (1) The flowpath for boron injection comes in below the bottom core plate and (2) due to colder water more boron would need to be added. (1) Plausible because (1) the candidate must know the flowpath of the SBLC system and (2) it must be known if net positive or negative reactivity is added due to decrease in temperature with boron injection. Moderator temperature coefficient becomes less negative with rising boron concentration. Less boron is plausible because as the reactor coolant *boron concentration* increases, the *moderator temperature coefficient* becomes less negative. This *is* because a 1°F increase in reactor coolant *temperature* at higher *boron concentrations* results in a larger increase in the thermal utilization factor. C. Correct - (1) The flowpath for boron injection comes in below the bottom core plate. (2) As boron concentration in the core goes up, the amount of THERMAL neutrons available for fissioning of fuel goes down, thereby reducing reactor power. More boron is needed to keep the reactor shutdown under cold conditions verses hot conditions due to positive reactivity added by moderator temperature coefficient. D. Incorrect - (1) The flowpath for boron injection comes in below the bottom core plate and (2) due to colder water more boron would need to be added. Plausible because (1) the candidate must know the flowpath of the SBLC system and (2) Part 2 of the answer is correct. **REQUIRED REFERENCES: None.**

Question 3 Table-Item Links

General Question Data - Site Ownership

Dresden

General Question Data - Ops Program

Reactor Operator

4 **ID: 27601** **Points: 1.00**

Unit 2 is operating at full power, when a report of an oil fire in the 517' elevation of the U2 TB is received.

- The Incident Commander reports that the entire area from the 2B IAC to the U2 EHC pumps is engulfed in flames.
- DOA 0010-10, Fire / Explosion was entered.

One of the required actions for a fire in this area is to post the EDG rooms and generator exciter housings with DANGER signs.

This is done because of the potential for a severe, Appendix R type fire to induce multiple shorts resulting in discharge of ___(1)___ in the affected room(s) without the associated ___(2)___.

A. (1) Halon
(2) alarms **ONLY**

B. (1) Halon
(2) alarms **OR** delay timer

C. (1) Cardox
(2) alarms **ONLY**

D. (1) Cardox
(2) alarms **OR** delay timer

Answer: D

Answer Explanation

(1) Per DOA 0010-10, these actions are taken because there is the potential for a severe, Appendix R type fire to induce multiple spurious shorts resulting in spurious Cardox actuation (2) This could occur without the associated alarms or delay timer

Question 4 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	4
Difficulty:	3.00
System ID:	27601
User-Defined ID:	27601
Cross Reference Number:	
Topic:	04 - 600000 K3.04
Num Field 1:	
Num Field 2:	
Text Field:	

Comments:	Objective: DRE286LN002 Obj 12 Reference: DOA 0010-10 K/A: 600000 K3.04 2.8/3.4 K/A: Knowledge of the reasons for the following responses as they apply to PLANT FIRE ON SITE: Actions contained in the abnormal procedure for plant fire on site Safety Function: 8 CFR: 41.10 PRA: No Level: Memory Pedigree: New History: N/A Explanations: A. INCORRECT - (1) Per DOA 0010-10, the actions are taken because of the potential spurious discharge of CARDOX. (2) Per DOA 0010-10, this action is taken because spurious Cardox actuation in the diesel generator rooms without the associated alarms or delay timer could occur. Plausibility: This is plausible because (1) Halon is also a gaseous suppression system, which is used in the AEER. The candidate must know that CARDOX is used in the EDG rooms and generator exciters, not Halon. (2) The candidate must be familiar with the DOA 0010-10 requirements, which specify that this is to done to because of the potential for spurious Cardox actuation in the diesel generator rooms without the associated alarms or delay timer. B. INCORRECT - (1) Per DOA 0010-10, the actions are taken because of the potential spurious discharge of CARDOX. (2) The second part of the answer is correct. Plausibility: This is plausible because (1) Halon is also a gaseous suppression system, which is used in the AEER. The candidate must know that CARDOX is used in the EDG rooms and generator exciters, not Halon. (2) The second part of the answer is correct. C. INCORRECT - (1) The first part of the answer is correct (2) Per DOA 0010-10, this action is taken because the CARDOX actuation could occur without the associated alarms OR delay timer. Plausibility: This is plausible because (1) The first part of the answer is correct. (2) The candidate must be familiar with the DOA 0010-10 requirements, which specify that this is to done to because of the potential for spurious Cardox actuation in the diesel generator rooms without the associated alarms or delay timer. D. CORRECT - (1) Per DOA 0010-10, these actions are taken because there is the potential for a severe, Appendix R type fire to induce multiple spurious shorts resulting in spurious Cardox actuation (2) This could occur without the associated alarms or delay timer **REQUIRED REFERENCES: None**

None

5	ID: 27390	Points: 1.00

Unit 2 is in **STARTUP** following a refueling outage.

- The 2B CRD pump is OOS
- RPV Pressure <150 psig in preparation for the low pressure HPCI run.

THEN the 2A CRD pump caught fire.

At 0010 ACCUMULATOR TROUBLE light illuminated for a CRD at position 00
At 0020 ACCUMULATOR TROUBLE light illuminated for a CRD at position 00
At 0030 ACCUMULATOR TROUBLE light illuminated for a CRD at position 12

At what time is a manual reactor scram **FIRST** required to be inserted?

A. 0010

B. 0020

C. 0030

D. 0050

Answer: C

Answer Explanation

With an accumulator trouble light illuminated for a CRD not at 00, an immediate RX scram is required.

Question 5 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	4.00
System ID:	27390
User-Defined ID:	27390
Cross Reference Number:	
Topic:	05 - 295022.K2.03
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	Objective: 201LN001.08 Reference: DOA 0300-01 K/A: 295022.K2.03 3.4/3.4 K/A: Knowledge of the interrelations between LOSS OF CRD PUMPS and the following: Accumulator pressures K/A Justification: If CRD Charging Water Pressure can NOT be immediately restored to greater than or equal to 940 psig, Reactor Steam Dome Pressure < 900 psig and any ACCUMULATOR TROUBLE light illuminates on the full core display for a control rod NOT at positions 00, THEN immediately scram the reactor AND enter DGP 02-03. The ACCUMULATOR TROUBLE light comes in at 1000 psig on the each individual CRD, therefore the student must know the accumulator alarm setpoint of 1000 psig is sufficient accumulator pressure to still be able to scram the reactor. CFR: 41.7 Safety Function: 1 PRA: No Level: High Pedigree: Bank History: 16-1 NRC, 18-1 Cert Explanations: A. Incorrect - Plausible because this would be correct if the CRD was not at 00 B. Incorrect - This is plausible because there are additional requirements based on 2 or more inoperable accumulators in Mode 2. C. Correct - With an accumulator trouble light illuminated for a CRD not at 00, an immediate RX scram is required. D. Incorrect - Plausible because there is allowance under certain conditions for a 20 minute time period to restore CRD pressure before a scram is required. **REQUIRED REFERENCES: None.**

None

6	ID: 22266	Points: 1.00

Unit 2 was operating at near rated power when a secondary containment isolation occurred, with the following conditions:

- All Reactor Building vent fans tripped.
- 2/3 A SBGT train automatically started.
- The NSO reported that the 2/3 A SBGT flow is 5300 CFM on FI 7540-13 on the 923-5 panel.

The SBGT system is operating ___(1)___ the required flowrate and indicated secondary containment D/P would be ___(2)___ than expected, for this condition.

A. (1) below
(2) less negative

B. (1) below
(2) more negative

C. (1) above
(2) less negative

D. (1) above
(2) more negative

Answer: D

Answer Explanation

With 5300 CFM being supplied through the operating SBGT train one must know that the normal operating band is 3600-4400 SCFM based upon 10% tolerance and 300 SCFM flow through the Standby train (this makes the actual band 3900-4700 SCFM). Therefore with high flow, the DP would be expected to be higher than the normal -.25 inches vacuum that should be able to be maintained by SBGT SR 3.6.4.1.3 states this as the proper band.

Question 6 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	3.00
System ID:	22266
User-Defined ID:	22266
Cross Reference Number:	
Topic:	06 - 295035.A2.01
Num Field 1:	
Num Field 2:	
Text Field:	

Comments:	Objective: DRE261LN001.12 Reference: DOS 7500-02. TS 3.6.4.1 K/A: 295035.A2.01 3.8 / 3.9 K/A: Ability to determine and/or interpret the following as they apply to SECONDARY CONTAINMENT HIGH DIFFERENTIAL PRESSURE: Secondary containment pressure. Safety Function: 9 CFR: 41.8 to 41.10 PRA: No Level: High Pedigree: Bank History: 2013 Cert Explanations: A. Incorrect - Normal system flow required by Tech Spec's is 3600 to 4400 scfm. The stem of the question states that flow is 5300. Even with the 300 cfm ambient flow this above normal flow. This would cause containment DP to be more negative. Plausible because must determine the normal flow rate for the system, if flow were below normal, secondary containment pressure would rise and DP would be less negative. B. Incorrect - Normal system flow required by Tech Spec's is 3600 to 4400 scfm. The stem of the question states that flow is 5300. Even with the 300 cfm ambient flow this above normal flow. Part 2 is correct. Plausible because must determine the normal flow rate for the system. Also plausible because part 2 of question is correct. C. Incorrect - Part 1 is correct. The high flow condition would cause containment DP to be more negative. Plausible because Part 1 is correct. Part 2 because must determine the normal flow rate for the system and the impact additional flow through the system would have on secondary containment DP. D. Correct - With 5300 CFM being supplied through the operating SBGT train one must know that the normal operating band is 3600-4400 SCFM based upon 10% tolerance and 300 SCFM flow, secondary containment pressure would lower and the DP would be expected to be higher than the normal -.25 inches vacuum that should be able to be maintained by SBGT SR 3.6.4.1.3 states this as the proper band. **REQUIRED REFERENCES: None.**

Question 6 Table-Item Links

General Question Data - Site Ownership

Dresden

General Question Data - Ops Program

Reactor Operator

7	ID: 13816	Points: 1.00

Given the following set of conditions:

- Unit 2 is in Mode 3 with Recirc loop temperatures 310°F and steady.
- 2A and 2B Recirc Pumps are running.
- 2A and 2B Shutdown Cooling (SDC) Pumps are running in the COOLING mode with their discharge valves 15% open.
- 2C SDC Pump is running to the Fuel Pool.
- 2A and 2B RBCCW pumps are operating on Unit 2.
- RPV level is at +30 inches.

If MO 2-3704, RBCCW OUTLET VLV, is timed open for an additional 4 seconds, the effect would be: _______.

A. Recirculation loop temperature increases

B. Recirculation loop temperature decreases

C. Recirculation loop temperature remains steady

D. BOTH RBCCW pumps trip on low discharge pressure

Answer: B

Answer Explanation

Per DOP 1000-03 lining up the system with the MO 2-3704 RBCCW OUTLET VLV opened for 16 seconds would provide maximum cooling and a Recirc loop temperature decrease.

Question 7 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	4
Difficulty:	3.00
System ID:	13816
User-Defined ID:	13816
Cross Reference Number:	
Topic:	07 - 205000.A1.03
Num Field 1:	
Num Field 2:	
Text Field:	

Comments:	Objective: DRE205001.08 Reference: DOP 1000-03 K/A: 205000.A1.03 3.3 / 3.3 K/A: Ability to predict and/or monitor changes in parameters associated with operating the SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE) controls including: Recirculation loop temperatures. Safety Function: 4 CFR: 41.5/45.5 PRA: No Level: High Pedigree: Bank History: 2007 NRC Explanations: A. Incorrect - Per DOP 1000-03, the 2-3704, RBCCW OUTLET VLV would be initially opened 16 seconds. Opening the MO 2-3704 RBCCW OUTLET VLV for an additional 4 seconds, which is allowed since the 2C SDC pump is aligned to the fuel pool, would result in additional cooling, therefore a recirc loop temperature decrease would occur. Plausible because for the given stem conditions this would be correct if the valve were closed for 4 seconds, instead of opening. The candidate must recognize that the additional 4 seconds open would cause Recirc loop temp to decrease. B. Correct - Per DOP 1000-03, the 2-3704, RBCCW OUTLET VLV would be initially opened 16 seconds. Opening the MO 2-3704 RBCCW OUTLET VLV for an additional 4 seconds would provide additional cooling and a Recirc loop temperature decrease. C. Incorrect - Per DOP 1000-03, the 2-3704, RBCCW OUTLET VLV would be initially opened 16 seconds. Opening the MO 2-3704 RBCCW OUTLET VLV for an additional 4 seconds would provide additional cooling and a Recirc loop temperature decrease. Plausible because for the given stem condition, recirc loop temperatures would remain steady if MO 2-3704 RBCCW OUTLET VLV was NOT opened an additional amount. D. Incorrect - The RBCCW pumps would not trip with these conditions. Plausible because if the 2-3704 vlv was opened too far the pumps could trip on low suction pressure. **REQUIRED REFERENCES: None.**

Question 7 Table-Item Links

General Question Data - Site Ownership

Dresden

General Question Data - Ops Program

Reactor Operator

8 **ID: 14030** **Points: 1.00**

If the reactor mode switch is in **RUN**, which one of the following conditions will cause a **DIRECT** trip of one RPS trip system (i.e., will cause a half scram)?

A. Reactor power at 10% with MSIVs 1C & 2D less than 90% open.

B. Reactor power at 10% with Turbine Stop Valves 3 & 4 less than 90% open.

C. Reactor power at 45% with MSIVs 1A & 1D less than 90% open.

D. Reactor power at 45% with Turbine Stop Valves 2 & 3 less than 90% open.

Answer: A

Answer Explanation

MSIVs C and D do not meet the 5 alive concept, so at this power a 1/2 scram would result.

Question 8 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	4.00
System ID:	14030
User-Defined ID:	14030
Cross Reference Number:	
Topic:	08 - 212000.K5.02
Num Field 1:	0.00
Num Field 2:	0.00
Text Field:	
Comments:	Objective: 212LN001.06 Reference: 12E-2464, 2465, 2466; DAN 902(3)-5 D-14 K/A: 212000.K5.02 3.3/3.4 K/A: Knowledge of the operational implications of the following concepts as they apply to REACTOR PROTECTION SYSTEM: Specific logic arrangements. Safety Function: 7 CFR: 41.5/45.3 PRA: No Level: High Pedigree: Bank History: N/A Explanations: A. Correct - MSIVs C and D do not meet the 5 alive concept, so at this power a 1/2 scram would result. B. Incorrect - TSV 3 and 4 do not add up to 5, but at 10% power, Turbine Stop Valves would not cause a 1/2 scram (38.5% bypass). Plausible because TSV do not add up to 5 (5 alive) C. Incorrect - MSIVs A and D meet the "5" alive requirement (no half scram). Plausible because candidate must determine for this power level which combinations are OK. D. Incorrect - TSV 2 and 3 meet the "5" alive requirement (no half scram). Plausible because candidate must determine for this power level which combinations are OK. **REQUIRED REFERENCES: NONE**

Question 8 Table-Item Links

General Question Data - Site Ownership

Dresden

General Question Data - Ops Program

Reactor Operator

9	ID: 13309	Points: 1.00

A reactor startup is in progress on Unit 3, when a fire completely de-energizes Unit 3 24/48 VDC Bus 3A.

Which of the following have lost power?

1) Stack Gas Rad Monitors
2) SRM channels 21 and 22
3) IRM channels 11,12,13,14
4) RBCCW Rad Monitor

A. 1,2 **ONLY**

B. 3,4 **ONLY**

C. 1,2,3 **ONLY**

D. 1,2,4 **ONLY**

Answer: C

Answer Explanation

2A powers IRM channels 11, 12, 13, 14, SRM channels 21 and 22, as well as the Stack Gas Rad Monitors.

Question 9 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	2
Difficulty:	2.00
System ID:	13309
User-Defined ID:	13309
Cross Reference Number:	
Topic:	09 - 215003.K2.01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	Objective: DRE215LN003.02 Reference: DOP 6900-03, DOA 6900-01 K/A: 215003.K2.01 2.5 / 2.7 K/A: Knowledge of electrical power supplies to the following: IRM channels/detectors. Safety Function: 7 CFR: 41.7 PRA: No Level: Memory Pedigree: New History: N/A Justification for Memory Level: Lesson Plan DRE263LN003, Objective 2, covers knowledge of the inter-relationships between the 24-48 VDC system and related systems, such as the loads fed from 24-48 VDC. This knowledge is required to be known from memory. Explanations: A. Incorrect - The loss of power to 24/48 VDC Bus 2A causes a loss of the Stack Gas Rad Monitors primary power supply as well as the SRMs 21 and 22 but IRMs 11,12,13,14 as well. Plausible because both 1 and 2 are correct just not complete. B. Incorrect - Channels 11,12,13,14 will be deenergized with the loss of U2 24/48 VDC Bus 2A. RBCCW Rad Monitor is powered from 24/48 VDC Bus 2B. Plausible because 3 is correct and 4 is powered from the other division of 24/48 VDC. C. Correct - 2A powers IRM channels 11, 12, 13, 14, SRM channels 21 and 22, as well as the Stack Gas Rad Monitors. D. Incorrect - 2A powers SRM channels 21, 22, as well as the Stack Gas Rad Monitors, but not the RBCCW Rad Monitor. Plausible because 1 and 2 are correct and 4 is powered from the other division of 24/48 VDC. **REQUIRED REFERENCES: None.**

Question 9 Table-Item Links

General Question Data - Site Ownership

Dresden

General Question Data - Ops Program

Reactor Operator

10 **ID: 22555** **Points: 1.00**

Unit 3 is operating at near rated power when the following sequence of events occur:

- 05:15:00, Spurious Reactor scram combined with a loss of all high pressure feed.
- 05:16:30, A small Main Steam Line leak occurs in the X-Area, causing a HIGH temperature alarm.
- 05:22:00, RPV water level decreased to -60 inches and is trending down at a rate of 5 inches per minute.
- 05:22:30, Annunciator 903-3 H-13, LPCI/CS PP AT PRESS, is received.

What is the **EARLIEST** time that all ADS Valves will begin to open?

A. 05:18:30

B. 05:24:00

C. 05:30:30

D. 05:31:00

Answer: C

Answer Explanation

The 8.5 minute timer started at 05:22.00, At 05:30:30 the ADS permissive is met and the valves begin to reposition.

Question 10 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	4
Difficulty:	3.00
System ID:	22555
User-Defined ID:	22555
Cross Reference Number:	
Topic:	10 - 218000.K5.01
Num Field 1:	0.00
Num Field 2:	0.00
Text Field:	
Comments:	Objective: DRE218LN001.06 References: DAN 902(3)-3 B-13; UFSAR 7.3.1.4 K/A: 218000.K5.01 3.8 / 3.8 K/A: Knowledge of the operational implications of the following concepts as they apply to AUTOMATIC DEPRESSURIZATION SYSTEM: ADS logic operation. Safety Function: 3 CFR: 41.5/45.3 Level: High Pedigree: Bank History: 2009 NRC Explanations: For a leak outside the drywell, ADS will provide high pressure systems a chance to recover level by waiting 8.5 minutes (2 minutes if leak INSIDE the drywell). After 8.5 minutes from -59 inches RPV level (time 05:30:30) the ADS valves will begin opening, since the permissives are met (RPV water level <-59 inches, LPCI/CS pump at pressure alarm). The distractor times are based on 2 minutes or on 8.5 minutes from LPCI pump starting. A. Incorrect - Plausible because with a loss of high pressure feed level will not recover and will slowly lower to -59 depending on leak size. This would be correct if the leak were in the DW because the 2 minute timer would have started. B. Incorrect - The leak outside the DW will not start the 2 minute timer. Plausible because if level had dropped to -59 on the scram the 8.5 minute timer would have timed out. C. Correct - The 8.5 minute timer started at 05:22.00, At 05:30:30 the ADS permissive is met and the valves begin to reposition. D. Incorrect - The blowdown would have already started on the 8.5 minute timer. Plausible because this alarm is when low pressure systems can be used to raise level. **REQUIRED REFERENCES: None.**

Question 10 Table-Item Links

General Question Data - Site Ownership

Dresden

General Question Data - Ops Program

Reactor Operator

11	ID: 14663	Points: 1.00

Both Units were operating at near rated power, with the following conditions:

- 'A' SBGT system was in PRI.
- 'B' SBGT system was in STBY.

Then Unit 2 experienced a transient causing RPV water level to drop to -70 inches, before slowly trending up.

An NSO reported that the SBGT system operated as designed.

Then one (1) minute later the NSO reported that the 2/3A SBGT Heater indication changed from RED to GREEN.

Two (2) minutes later what would be the expected **MINIMUM** SBGT system flow and lineup?

A. 0 scfm, with neither system operating.

B. 3800 scfm, being supplied from the 2/3 'B' SBGT system.

C. 4100 scfm, being supplied from the 2/3 'A' SBGT system.

D. 4100 scfm, being supplied from the 2/3 'B' SBGT system.

Answer: D

Answer Explanation

When the operating ‘A’ train trips (due to heater trip) and the standby train ‘B’ has its control switch in the STANDBY position, then the ‘B’ train will auto start. After the auto start of the ‘B’ train, the normal expected system flow is 3900 – 4100 scfm.

Question 11 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	4
Difficulty:	3.00
System ID:	14663
User-Defined ID:	14663
Cross Reference Number:	
Topic:	11 - 261000.A3.01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	Objective: DRE261LN001.06 Reference: DAN 923-5 A-6, DOP 7500-01 K/A: 261000.A3.01 3.2 / 3.3 K/A: Ability to monitor automatic operations of the STANDBY GAS TREATMENT SYSTEM including: System flow. Safety Function: 9 CFR: 41.7/45.7 Level: High Pedigree: Bank History: 2010 NRC Explanations: A. Incorrect - SBGT has an auto start feature for a standby train on a failure of the primary system. The heater lights changing from Red to Green indicates a failure of the primary system. Plausible because the candidate must determine the failure based on light indication and understand the auto start logic. B. Incorrect - The A system will trip on the failure of the heater and the B train will auto start. Plausible because the candidate must determine the failure based on light indication, understand the auto start logic, and recognize with 300 scfm ambient air passing through the standby unit what the normal indicated flow should be. C. Incorrect - The A system will trip on the failure of the heater and the B train will auto start. Plausible because part 1 is correct and the candidate must determine the failure based on light indication and understand the auto start logic. D. Correct - When the operating ‘A’ train trips (due to heater trip) and the standby train ‘B’ has its control switch in the STANDBY position, then the ‘B’ train will auto start. After the auto start of the ‘B’ train, the normal expected system flow is 3900 – 4100 scfm. **REQUIRED REFERENCES: None.**

Question 11 Table-Item Links

General Question Data - Site Ownership

Dresden

General Question Data - Ops Program

Reactor Operator

12	ID: 22569	Points: 1.00

Given the following for the Unit 2 250 VDC :

- 5 Volt POSITIVE ground is present.
- 15 Volt NEGATIVE ground is present.

What will ground detection recorder, 2-8350-250, display if the NEGATIVE button is depressed?

A. Positive 5 VOLTS

B. Positive 10 VOLTS

C. Negative 10 VOLTS

D. Negative 15 VOLTS

Answer: D

Answer Explanation

When depressing the POSITIVE button, the negative reference leg is isolated, displaying only the positive ground. When depressing the NEGATIVE button, the positive reference leg is isolated, displaying only a negative ground.

Question 12 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	4
Difficulty:	3.00
System ID:	22569
User-Defined ID:	22569
Cross Reference Number:	
Topic:	12 - 263000.K1.04
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	Objective: DRE263LN001.02 Reference: DOP 6900-04 K/A: 263000.K1.04 2.6 / 2.9 K/A: Knowledge of the physical connections and/or cause effect relationships between D.C. ELECTRICAL DISTRIBUTION and the following: Ground detection. Safety Function: 6 CFR: 41.2 to 41.9/45.7 to 45.8 Level: High Pedigree: Bank History: 2009 NRC Explanations: A. Incorrect - When depressing the NEGATIVE button, the positive reference leg is isolated, displaying only a negative ground. Plausible because the candidate must understand which reference leg is isolated when pushing the buttons. B. Incorrect - When depressing the NEGATIVE button, the positive reference leg is isolated, displaying only a negative ground. Plausible misconception that the grounds are cumulative. C. Incorrect - When depressing the NEGATIVE button, the positive reference leg is isolated, displaying only a negative ground. Must understand that grounds are not cumulative. Plausible misconception that the grounds are cumulative and if the positive ground was subtracted from the negative ground this would be correct. D. Correct - When depressing the POSITIVE button, the negative reference leg is isolated, displaying only the positive ground. When depressing the NEGATIVE button, the positive reference leg is isolated, displaying only a negative ground. **REQUIRED REFERENCES: None.**

Question 12 Table-Item Links

General Question Data - Site Ownership

Dresden

General Question Data - Ops Program

Reactor Operator

13 **ID: 23490** **Points: 1.00**

A prerequisite of DOS 6620-01, SBO 2 DIESEL GENERATOR SURVEILLANCE AND MARGIN/FULL LOAD TEST/FULL LOAD REJECT TEST, is to ensure *that the bus selected for this surveillance is powered from either Transformers TR 21 or TR 22 AND the associated Emergency Diesel Generator is NOT paralleled to this bus*.

What is the operational concern for this prerequisite?

A. A trip of either Diesel Generator while in this lineup, would cause the Bus to de-energize.

B. Large circulating currents, due to voltage differences, may result in an auto trip of both Diesel Generators.

C. A neutral ground condition, due to voltage differences, may result in an auto trip of both Diesel Generators.

D. A neutral voltage condition, due to phase angle differences, may result in an auto trip of both Diesel Generators.

Answer: B

Answer Explanation

A difference in voltages between generators will cause a circulating current to flow that could exceed breaker load limits.

Question 13 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	3.00
System ID:	23490
User-Defined ID:	23490
Cross Reference Number:	
Topic:	13 - 264000.K5.05
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	Objective: DRE264LN001.10 Reference: BWR Fundamentals Ch. 5 Motors and Generators, DOS 6620-01 K/A: 264000.K5.05 3.4 / 3.4 K/A: Knowledge of the operational implications of the following concepts as they apply to EMERGENCY GENERATORS (DIESEL/JET): Paralleling A.C. power sources. CFR: 41.5 Safety Function: 6 PRA: Yes Level: Memory Pedigree: Bank History: 2012 NRC, 2013 Cert Explanations: A. Incorrect - Upon a trip of either diesel generator, the bus would remain energized powered from the other diesel generator. Plausible because depending on the trip or fault both Diesel Generators could trip. B. Correct - A difference in voltages between generators will cause a circulating current to flow that could exceed breaker load limits. C. Incorrect - A neutral ground condition would be caused by a mechanical fault in the generator or associated wiring. Plausible because some trips would cause a loss of both Diesel Generators. D. Incorrect - Large circulating currents may cause a neutral voltage condition, but with the two diesel generators operating in parallel the EDGs would be in synch, therefore the phase angles would be matched. Plausible because a difference in phase angles would cause high circulating currents. **REQUIRED REFERENCES: None.**

Question 13 Table-Item Links

General Question Data - Site Ownership

Dresden

General Question Data - Ops Program

Reactor Operator

14 **ID: 27560** **Points: 1.00**

Unit 2 is at 60% power when a failure to scram occurs due to electrical failures.

- ARI has failed to function
- Operators are pulling RPS fuses to insert rods during an electrical ATWS
- The indicating light for scram solenoid groups A1, A2, and A3 are EXTINGUISHED
- The indication light for the remaining scram solenoid groups are LIT
- RPS Trip System B de-energizes as a result of a trip of its MG Set

Based on these conditions ______ of the control rods will insert into the core.

A. 0%

B. 25%

C. 75%

D. 100%

Answer: C

Answer Explanation

With B RPS subsystem de-energized and 3 solenoid groups in the A RPS subsystem de-energized the scram solenoids for approximately 75% of the control rods are de-energized, therefore 25% of the control rods are NOT scrammed and 75% are inserted.

Question 14 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	3.00
System ID:	27560
User-Defined ID:	27560
Cross Reference Number:	
Topic:	14 - 212000.K1.06
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	Objective: DRE212LN001.11 Reference: 12E-2467 Sht.1, 2, 3; 12E-2465 Sht. 1 K/A: 212000 K1.06 3.5/3.6 K/A: Knowledge of the physical connections and/or cause-effect relationships between REACTOR PROTECTION SYSTEM and the following: Control rod drive hydraulic system. Safety Function: 7 CFR: 41.2 to 41.9/45.7 to 45.8 Level: High Pedigree: Bank History: N/A Explanations: A. Incorrect - With B RPS subsystem de-energized, and 3 solenoid groups in the A RPS subsystem de-energized, Control Rod movement WILL occur. Plausible misconception that the A RPS MG set is associated with the A RPS trip system. In that case B RPS trip system is still energized and no rods would move. B. Incorrect - With B RPS subsystem de-energized, and 3 solenoid groups in the A RPS subsystem de-energized, the scram solenoids for approximately 75% of the control rods are de-energized, therefore 25% of the control rods are NOT scrammed and 75% are inserted. Plausible because this is the inverse of what will happen Plausible misconception that the trip lights are energize to scram, and this would cause 25% of the rod to enter the core. C. Correct - With B RPS subsystem de-energized and 3 solenoid groups in the A RPS subsystem de-energized the scram solenoids for approximately 75% of the control rods are de-energized, therefore 25% of the control rods are NOT scrammed and 75% are inserted. D. Incorrect - All rods in will not occur with A4 solenoid group still energized. Not all control rods will be inserted. Plausible misconception that the RPS logic will cause a full scram with the RPS B trip system de-energized, and 3 of 4 A RPS trip system channels de-energized. RPS is a one out of two taken twice trip logic so they may conclude that 3 of 4 lights being out could meet the logic trip setpoint. **Required References: None**

15 **ID: 23862** **Points: 1.00**

Unit 2 was operating at near rated power, with the following set of conditions:

- The FWRVs are controlling in AUTO.
- RPV water level setpoint is + 30 inches.
- 3 ELEMENT is "in-control" on the FWLC system.

Then a transient occurred causing a loss of the steam flow signal to the Bailey Control System.

How will RPV water level respond and why?

A. RPV water level will stay at its current level;
the FWRVs have locked up AS-IS.

B. RPV water level will be controlled at +30 inches;
the FWLC system has entered SINGLE ELEMENT control.

C. RPV water level will decrease until manual control is taken;
the FWLC system has failed due to BAD QUALITY

D. RPV water level will decrease for 45 seconds, then return to +30 inches;
the FWLC system has entered SETPOINT SETDOWN.

Answer: B

Answer Explanation

With the Bailey feedwater level control system operating in 3-element control, and experiencing a loss one of its input signals (steam or feed flow), then it will transfer to single element.

Question 15 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	3.00
System ID:	23862
User-Defined ID:	23862
Cross Reference Number:	
Topic:	15 - 259002.K1.02
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	Objective: DRE259LN002.06 Reference: DAN 902(3)-5 G-8, DOA 0600-01. DIS 0600-06 K/A: 259002.K1.02 3.2 / 3.3 K/A: Knowledge of the physical connections and/or cause-effect relationships between REACTOR WATER LEVEL CONTROL SYSTEM and the following: Main steam flow. Safety Function: 2 CFR: 41.2 to 41.9/45.7 to 45.8 PRA: No Level: High Pedigree: Bank History: 2010 NRC Explanations: With the Bailey feedwater level control system operating in 3-element control, and experiencing a loss one of its input signals (steam or feed flow), then it will transfer to single element. A. Incorrect - The FWRVs would only lock up as-is, if there is a loss of Instrument Air to them. Plausible because the reg valves will lock up at 65 psig air pressure. B. Correct - With the Bailey feedwater level control system operating in 3-element control, and experiencing a loss one of its input signals (steam or feed flow), then it will transfer to single element. C. Incorrect - Water level will not decrease, as the FWLC system is being controlled in single element. This would be correct if the FWLC system did not swap to single element. D. Incorrect - Setpoint setdown is entered only if two of the level inputs are lost. Plausible because this would be correct for a loss of 2 or more inputs. **REQUIRED REFERENCES: None.**

Question 15 Table-Item Links

General Question Data - Site Ownership

Dresden

General Question Data - Ops Program

Reactor Operator

16 **ID: 27562** **Points: 1.00**

A LOCA has occurred on U2.

- LPCI system initiated 40 seconds ago
- RPV level is -159 inches and rising slowly
- LPCI is injecting into the RPV at maximum system flow
- LPCI Heat Exchanger Bypass Valves (1501-11A&B) are open

The LPCI HX bypass valves will:

A. NOT close under any circumstances with an initiation signal present.

B. Close if the valve control switch is placed in Pull-to-Lock (PTL).

C. AUTOMATICALLY close if the 316A & B switches are in AUTO.

D. Close if the valve control switch is placed in CLOSE.

Answer: D

Answer Explanation

The LPCI HX bypass valves are interlocked open for 30 seconds following an initiation signal. Thereafter, the valves may be closed after 30 seconds have passed, by taking the control switch to the close position.

Question 16 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	3.00
System ID:	27562
User-Defined ID:	27562
Cross Reference Number:	
Topic:	16 - 203000 A4.04
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	Objective: 203LN001-11 Reference: DOP 1500-02, DOP 1500-03 K/A: 203000 A4.04 3.6/3.6 K/A: Ability to manually operate and/or monitor in the control room: Heat exchanger cooling flow. Safety Function: 2 CFR: 41.7/45.5 to 45.8 Level: Memory Pedigree: Bank History: N/A Explanations: A. Incorrect - The LPCI HX bypass valves can be closed manually 30 seconds after the ECCS initiation signal. Plausibility: Plausible because some components are prevented from operating following an ECCS signal. For example, the LPCI injection valve on the non-selected loop cannot be opened once LPCI loop select has actuated. B. Incorrect - The LPCI HX Bypass Valves do not have a PTL feature. Plausible because many other valves have a PTL feature that allows overriding automatic functions. For example, the 2(3)-1301-3 valves, and the 2(3)-1501-22A/B valves. C. Incorrect - The LPCI HX bypass valves have to be manually closed, there is no auto close signal. Plausibility: Plausible because other equipment will auto position to the desired position with an ECCS signal present. For example, the LPCI injection valves in the selected loop will auto open when appropriate conditions are met. This is also plausible because similar switches (318A and 318B) have an AUTO position, and the associated CCSW pumps will auto trip with an ECCS signal if they are in AUTO. D. Correct. The LPCI HX bypass valves are interlocked open for 30 seconds following an initiation signal. Thereafter, the valves may be closed after 30 seconds have passed, by taking the control switch to the close position. **Required References: NONE**

None

17	ID: 27563	Points: 1.00

HPCI has auto started and is injecting.

- HPCI room temperature is 140F
- 2/3 A & B CST levels are 6 feet
- Reactor Pressure is 150 psig
- HPCI Booster Pump Suction Pressure is 5 psig
- Drywell Pressure is 2.3 psig

What is the expected HPCI system response?

A. HPCI turbine **TRIPS**.

B. HPCI steam supply isolation valves **CLOSE**.

C. HPCI pump suction **TRANSFERS** to the Torus.

D. HPCI turbine exhaust vacuum breakers **OPEN**.

Answer: C

Answer Explanation

Low CST Level causes auto swap of suction to the torus. Nominal value for the transfer is 8 feet (tech spec value of 11.1 for CST 2/3A and 7.5 feet for CST 2/3B). HPCI takes suction from the CST's, which are normally cross connected. If level in Tank 2/3A is less than 11.9 feet (+/- 2 inches); **AND** level in Tank 2/3B is less than 8.3 feet (+/- 2 inches), the HPCI suction valves will re-align to take suction from the Torus.

Question 17 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	4
Difficulty:	3.00
System ID:	27563
User-Defined ID:	27563
Cross Reference Number:	
Topic:	17 - 206000.A3.08
Num Field 1:	
Num Field 2:	
Text Field:	

Comments:	Objective: DRE206LN001.06 Reference: DOP 2300-03; DAN 902(3)-6 A-5 & A-6; DAN 902(3)-3 A-12, A-9, C-7 K/A: 206000 A3.08 3.7/3.6 K/A: Ability to monitor automatic operation of the HIGH PRESSURE COOLANT INJECTION SYSTEM including: Condensate storage tank level. Safety Function: 2 CFR: 41.7/45.7 Level: High Pedigree: Bank History: N/A Explanations: A. Incorrect: The conditions noted in the stem will not result in a HPCI turbine trip. HPCI suction pressure is still above the low suction pressure trip setpoint. Plausible because if HPCI suction pressure was lower, less than 9.4 inHg vacuum, the HPCI turbine would trip. B. Incorrect: HPCI auto isolation signal, causing the HPCI steam isolation valves to close, would not occur for the conditions listed in the stem. Plausibility: This is plausible because the HPCI steam isolation valves would close on an auto isolation signal if HPCI area temperature were higher (173F) or reactor pressure were lower (106 psig). C. Correct - Low CST Level causes auto swap of suction to the torus. Nominal value for the transfer is 8 feet (tech spec value of 11.1 for CST 2/3A and 7.5 feet for CST 2/3B). HPCI takes suction from the CST's, which are normally cross connected. If level in Tank 2/3A is less than 11.9 feet (+/- 2 inches); **AND** level in Tank 2/3B is less than 8.3 feet (+/- 2 inches), the HPCI suction valves will re-align to take suction from the Torus. D. Incorrect: The conditions noted in the stem indicate that steam is being expelled to the torus. The HPCI turbine exhaust vacuum breakers would be closed until steam is no longer being admitted, and then could open to as the steam condensed. Plausible because the function of these breakers is to open and they can open once steam is no longer being admitted to the torus through the HPCI exhaust line. **Required References: NONE**

None

18	ID: 26908	Points: 1.00

11 LPRMs assigned to APRM 6 must be bypassed.

If these LPRMs are bypassed, what effect (if any) does this have on **RMCS**?

A. RMCS is **NOT** affected.

B. A rod out block is generated.

C. A rod select block is generated.

D. A rod select block **AND** a rod out block are generated

Answer: B

Answer Explanation

RMCS receives a rod out block due to APRM 6 having less than 50% of assigned LPRMs in service

Question 18 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	4.00
System ID:	26908
User-Defined ID:	26908
Cross Reference Number:	
Topic:	18 - 215005.K4.01
Num Field 1:	
Num Field 2:	
Text Field:	

Comments:	Objective: DRE215LN005.10 Reference: 12E-2411; 12E-2472; DAN 902(3)-5 D-13, C-12; DOA 0700-03 K/A: 215005.K4.01 3.7/3.7 K/A: Knowledge of AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM design feature and/or interlocks which provide for the following: Rod withdrawal blocks. CFR: 41.7 Safety Function: 7 Level: Memory Pedigree: Bank History: N/A Explanations: A - Incorrect. RMCS receives a rod out block. Plausible because must determine that 11 is more than 50% of LPRMs assigned to APRM 6. B - Correct. RMCS receives a rod out block due to APRM 6 having less that 50% of assigned LPRMs in service C - Incorrect. A rod select block is not received. The following generate a select block: Rod Out Notch >2 seconds (if the "drive out" directional control valve 122 remains energized for >2 seconds), Continuous Rod Out **and** Rod Block >2 seconds (during a continuous rod withdrawal sequence, a Select Block will be generated whenever a Rod Out Block exists for >2 seconds), and De-Energizing Select Relays (if a problem with RPIS occurs then RMCS will not allow control rod movement). Plausible because a rod select block will also stop movement but is not caused by an INOP APRM. D - Incorrect. A rod select block is not received. The following generate a select block: Rod Out Notch >2 seconds (if the "drive out" directional control valve 122 remains energized for >2 seconds), Continuous Rod Out **and** Rod Block >2 seconds (during a continuous rod withdrawal sequence, a Select Block will be generated whenever a Rod Out Block exists for >2 seconds), and De-Energizing Select Relays (if a problem with RPIS occurs then RMCS will not allow control rod movement). Plausible because a rod select block will also stop movement but is not caused by an INOP APRM. Part 2 is correct. **REQUIRED REFERENCE: None.**

None

19 **ID: 23589** **Points: 1.00**

A transient has occurred on Unit 2 and the following conditions exist:

- Bus 24-1 tripped on Overcurrent
- No operator actions have been taken to address Bus 24-1 tripping
- Drywell pressure is 2.6 psig and slowly increasing
- All available Torus Cooling and Torus Sprays were initiated from the control room
- All other equipment is operating as required by plant conditions

Which of the following actions will occur if Bus 29 is cross-tied to Bus 28 (using the appropriate procedural steps), with the rest of the plant in its current configuration?

A. 2A Fuel Zone RWL indicator begins reading on scale.

B. 2A Core Spray Pump Minimum flow valve will open then close.

C. The 20A Torus to Reactor Building Vacuum Breaker will go close.

D. Unit 2 Medium Range “A” RWL indicator begins reading on scale.

Answer: D

Answer Explanation

The MR A indicator is re-energized when ATS Panel 2202-73B receives power from MCC 29-1, when Bus 29 is energized. All the other actuations in the distracters are powered from MCC 28-1 which never lost power.

Justification: The ATS modules can receive either AC or DC power, not both (no alternate supply). Upon loss of either AC or DC it is important to understand what modules and therefore what functions / indications have been lost.

Question 19 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	5
Difficulty:	3.00
System ID:	23589
User-Defined ID:	23589
Cross Reference Number:	
Topic:	19 - 216000.K2.01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	Objective: 216LN002.12 References: DOP 6800-05, DEOP 0010-00, Operator Aid #210 K/A: 216000.K2.01 2.8 / 2.8 K/A: Knowledge of electrical power supplies to the following: Analog trip system: Plant-specific. Safety Function: 7 CFR: 41.7 Level: High Pedigree: Bank History: N/A Explanations: A. Incorrect - 2A Fuel Zone is powered by 28-1 via the Analog Trip System and has not been lost. Recirc is still running therefore FZ is not on scale. Plausible because ATS modules have AC or DC power. Upon a loss of AC it is important to understand what modules and functions have been lost. Also this would be correct if recirc was tripped. B. Incorrect - 2A Core Spray MIN flow valve is powered from 28-1 via the Analog Trip System and not be affected. Plausible because this would be correct if 28-1 was lost. C. Incorrect - 20A Torus to RB Vacuum Breaker is powered from 28-1 via the Analog Trip System and would not close. Plausible because ATS modules have AC or DC power. Upon a loss of AC it is important to understand what modules and functions have been lost. Also this would be correct if 28-1 was lost. D. Correct - The MR A indicator is re-energized when the Analog Trip System Panel 2202-73B receives power from MCC 29-1, when Bus 29 is energized. All the other actuations in the distracters are powered from MCC 28-1 which never lost power. Justification: The ATS modules can receive either AC or DC power, not both (no alternate supply). Upon loss of either AC or DC it is important to understand what modules and therefore what functions / indications have been lost. **Required References: NONE**

Question 19 Table-Item Links

General Question Data - Site Ownership

Dresden

General Question Data - Ops Program

Reactor Operator

20	ID: 24189	Points: 1.00

A hydraulic ATWS has occurred on Unit 2.

The Unit Supervisor has directed you to perform repeated scram resets per DEOP 0500-05.

Which of the following describes the **MINIMUM** electrical safety precautions required to perform this task per SA-AA-129, ELECTRICAL SAFETY ?

All metal removed, safety glasses and ...

A. electrical safety coat

B. long sleeve flame resistant outer garment and leather gloves

C. electrical safety coat, face shield and rubber insulating gloves

D. long sleeve flame resistant outer garment and rubber insulating gloves

Answer: D

Answer Explanation

SA-AA-129 states that for the voltage in the area being worked in for DEOP 500-5 the minimum PPE is all metal removed, appropriate long sleeve outer garment (Flame Resistant clothing meets this criteria), safety glasses and rubber insulating gloves.

Question 20 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	3.00
System ID:	24189
User-Defined ID:	24189
Cross Reference Number:	
Topic:	20 - Generic.2.1.26
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	Objective: 295L105 Reference: SA-AA-129: SA-AA-116 K/A: Generic 2.1.26 3.4 / 3.6 K/A: Knowledge of industrial safety procedures (such as rotating equipment, electrical, high temperature, high pressure, caustic, chlorine, oxygen and hydrogen). Safety Function: N/A CFR: 41.10 PRA: No Level: Memory Pedigree: Bank History: ILT 11-1 NRC Explanations: A. Incorrect - SA-AA-129 states that for the voltage in the area being worked in for DEOP 500-5 the minimum PPE is all metal removed, appropriate long sleeve outer garment (Flame Resistant clothing meets this criteria), safety glasses and rubber insulating gloves. Plausible because it is a partially correct answer. B. Incorrect - SA-AA-129 states that for the voltage in the area being worked in for DEOP 500-5 the minimum PPE is all metal removed, appropriate long sleeve outer garment (Flame Resistant clothing meets this criteria), safety glasses and rubber insulating gloves. Plausible because it is a partially correct answer, and protective leather gloves are used for some higher voltage applications. C. Incorrect - SA-AA-129 states that for the voltage in the area being worked in for DEOP 500-5 the minimum PPE is all metal removed, appropriate long sleeve outer garment (Flame Resistant clothing meets this criteria), safety glasses and rubber insulating gloves. Plausible because it is a partially correct answer. Face shields are required for some higher voltage applications. D. Correct - SA-AA-129 states that for the voltage in the area being worked in for DEOP 500-5 the minimum PPE is all metal removed, appropriate long sleeve outer garment (Flame Resistant clothing meets this criteria), safety glasses and rubber insulating gloves. **REQUIRED REFERENCES: None.**

Question 20 Table-Item Links

General Question Data - Site Ownership

Dresden

General Question Data - Ops Program

Reactor Operator

Pseudo Objectives

295L105

21 **ID: 12939** **Points: 1.00**

You are about to take the shift as a Unit 2 NSO.

The last time you were on shift was seven days ago.

What is the **MINIMUM** time that are you **REQUIRED** to read the Control Room logs back to, prior to completing relief?

A. 1 day.

B. 3 days.

C. 4 days.

D. 7 days.

Answer: C

Answer Explanation

Per OP-AA-112-101 An on-coming Reactor Operator is required to read the Control Room logs through the last previous date on shift, or the preceding four days, whichever is less.

Question 21 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	3.00
System ID:	12939
User-Defined ID:	12939
Cross Reference Number:	
Topic:	21 - Generic.2.1.03
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	Objective: 29900LK022 Reference: OP-AA-112-101 K/A: Generic.2.1.3 3.7/3.9 K/A: Knowledge of shift or short-term relief turnover practices. Safety Function: N/A CFR: 41.10 Level: Memory Pedigree: Bank History: 2008 NRC, 16-1 NRC Explanations: A. Incorrect - Per OP-AA-112-101 An on-coming Reactor Operator is required to read the Control Room logs through the last previous date on shift, or the preceding four days, whichever is less. Plausible because one requirement for Reactor Operators during shift turnover is to tour the Main Control boards and discuss abnormal events occurring within the last 24 hours. B. Incorrect - Per OP-AA-112-101 An on-coming Reactor Operator is required to read the Control Room logs through the last previous date on shift, or the preceding four days, whichever is less. Plausible because three days is the period that an operator would normally be off when rotating from nights to days in the 12 hour shift schedule. C. Correct - Per OP-AA-112-101 An on-coming Reactor Operator is required to read the Control Room logs through the last previous date on shift, or the preceding four days, whichever is less. D. Incorrect - Per OP-AA-112-101 An on-coming Reactor Operator is required to read the Control Room logs through the last previous date on shift, or the preceding four days, whichever is less. Plausible misconception 7 days would be assumed as the minimum days, since it is the amount of time the individual was not on shift. Also, 7 days is the maximum time off in the normal schedule when working the 12 hour shift schedule. **REQUIRED REFERENCES: None.**

Question 21 Table-Item Links

General Question Data - Site Ownership

Dresden

General Question Data - Ops Program

Reactor Operator

22 | **ID: 27566** | **Points: 1.00**

Unit 2 was operating at near rated power when an instrument failure requires an in-plant manipulation which will affect reactivity.

Which of the following describes the **MINIMUM** requirement to perform this evolution?

Communication between the Control Room and ______, in the plant.

A. ANY SRO **ONLY**

B. ANY SRO with a qualified QNE

C. an ACTIVE LICENSED Operator **ONLY**

D. an ACTIVE LICENSED Operator with no restriction which would prohibit solo operations

Answer: D

Answer Explanation

IF local manual Recirc ASD operation is to be performed AND the Unit is NOT in cold shutdown, THEN the Operator at the PDC ASD HMI must be an individual whose NRC license is active AND either their license is unrestricted, states that the only restriction is another person capable of summoning assistance, OR is in view of another qualified individual, based on the specific wording on their license.

Question 22 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	3.00
System ID:	27566
User-Defined ID:	27566
Cross Reference Number:	
Topic:	22 - Generic 2.1.8
Num Field 1:	
Num Field 2:	
Text Field:	

Comments:	Objective: 20200LN001.08 Reference: DOP 0202-16, HR-AA-07-101 K/A: Generic 2.1.8 3.4 / 4.1 K/A: Ability to coordinate personnel activities outside the control room. Safety Function: N/A CFR: 41.10/45.5, 45.12, 45.13 Level: Memory Pedigree: Bank History: 2008 NRC, 2011 Cert Explanations: A. Incorrect - Local Operation of equipment that affects reactivity requires a Active License (RO or SRO) with no restrictions that would prohibit solo operations (Per HR-AA-07-101, “No Solo” Operation: License restriction that prohibits solo operation in the Main Control Room or other specified controlled areas). Plausible because SRO can direct operations in the field that affect reactivity but must have an active license and no restrictions that would prohibit solo operations . B. Incorrect - Local Operation of equipment that affects reactivity requires a Active License, but with the additional caveat of no restrictions that would prohibit solo operations (Per HR-AA-07-101, “No Solo” Operation: License restriction that prohibits solo operation in the Main Control Room or other specified controlled areas). QNE = Qualified Nuclear Engineer. Plausible because the answer is partially correct. Missing the Active License and no solo restrictions. C.Incorrect - Local Operation of equipment that affects reactivity requires a Active License (RO or SRO), but with the additional caveat of no restrictions that would prohibit solo operations (Per HR-AA-07-101, “No Solo” Operation: License restriction that prohibits solo operation in the Main Control Room or other specified controlled areas). Plausible because the answer is partially correct. Missing the no solo restrictions. D. Correct - IF local manual Recirc Adjustable Speed Drive (ASD) operation is to be performed AND the Unit is NOT in cold shutdown, THEN the Operator at the Power Distribution Center (PDC) ASD Human Machine Interface (HMI) must be an individual whose NRC license is active AND either their license is unrestricted, states that the only restriction is another person capable of summoning assistance, OR is in view of another qualified individual, based on the specific wording on their license. **Required References: NONE**

None

23	ID: 27567	Points: 1.00

The Operations department is working 8 hour shifts.

You are performing APPENDIX A on U2 for SHIFT 2.

To ensure required Tech Spec, TRM, and ODCM required surveillance intervals are met, complete the required surveillance checks per this Appendix by ___(1)___. You must notify the Unit Supervisor ___(2)___ this requirement is **NOT** met.

A. (1) 1100
(2) IF

B. (1) 1100
(2) BEFORE

C. (1) 1500
(2) IF

D. (1) 1500
(2) BEFORE

Answer: B

Answer Explanation

Per UNIT DAILY SURVEILLANCE LOG ATTACHMENT A EIGHT HOUR SHIFTS, To ensure required Tech Spec, TRM, and ODCM required surveillance intervals are met, complete the required surveillance checks per this Appendix within the first half of the operating shift. Notify the Unit Supervisor **BEFORE** this requirement is **NOT** met.

Question 23 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	4.00
System ID:	27567
User-Defined ID:	27567
Cross Reference Number:	
Topic:	23 - Generic 2.2.12
Num Field 1:	
Num Field 2:	
Text Field:	

Comments:	Objective: 299LN001.4 Reference: APPENDIX A K/A: Generic 2.2.12 3.7/4.1 K/A: Knowledge of surveillance procedures. Safety Function: N/A CFR: 41.10/45.13 PRA: No Level: Memory Pedigree: New History: N/A Explanations: A. Incorrect - T.S. TRM, and ODCM surveillance must be performed in the first half of the shift. In addition, per APPENDIX A General instructions you must notify the US BEFORE this requirement is not met. Plausible because part 1 is correct. Part 2 is plausible because there are many surveillance that are not reported until the time has not been met. B. Correct - Per UNIT DAILY SURVEILLANCE LOG ATTACHMENT A EIGHT HOUR SHIFTS, To ensure required Tech Spec, TRM, and ODCM required surveillance intervals are met, complete the required surveillance checks per this Appendix within the first half of the operating shift. Notify the Unit Supervisor **BEFORE** this requirement is **NOT** met. C. Incorrect - Per the APPENDIX A General Instructions the surveillance must be performed in the first half of the shift. This is to prevent exceeding the 8 hour requirement if the previous shift performs in the first half of shift and the next shift does the second half. Plausible because the EMERGENCY SYSTEM CHECKLIST of APPENDIX A is laid out with signatures 1x per 8 hour shift. Part 2 is plausible because there are many surveillance that are not reported until the time has not been met. D. Incorrect - Per the APPENDIX A General Instructions the surveillance must be performed in the first half of the shift. This is to prevent exceeding the 8 hour requirement if the previous shift performs in the first half of shift and the next shift does the second half. Plausible because the EMERGENCY SYSTEM CHECKLIST of APPENDIX A is laid out with signatures 1x per 8 hour shift. Part 2 is correct. **Required References: None**

None

24	ID: 13420	Points: 1.00

Given the following conditions:

- Unit 2 is in coastdown (end of cycle) preparing to go into a refueling outage.
- In an effort to obtain the maximum amount of power, PRESSURE SET on the 902-7 panel is being raised.
- You notice that reactor pressure is 1007 psig.

Per Technical Specifications, what is the **FIRST** action that must be taken and what is the reason for that action?

Within___(1)___minutes reduce steam dome pressure to less than 1005 to ensure ___(2)___.

A. (1) 15
(2) a wide margin to non-ductile failure of the reactor vessel

B. (1) 30
(2) a wide margin to non-ductile failure of the reactor vessel

C. (1) 15
(2) the plant is operated within the assumptions of the overpressure analysis

D. (1) 30
(2) the plant is operated within the assumptions of the overpressure analysis.

Answer: C

Answer Explanation

The time limit is 15 minutes but the reason is to ensure the plant is operated within the assumption of the overpressure analysis.

Question 24 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	3.00
System ID:	13420
User-Defined ID:	13420
Cross Reference Number:	
Topic:	24 - Generic.2.2.39
Num Field 1:	0.00
Num Field 2:	0.00
Text Field:	
Comments:	Objective: 299LN001.04 Reference: Tech Spec 3.4.10: B 3.4.10 K/A: Generic.2.2.39 3.9 / 4.5 K/A: Knowledge of less than or equal to one hour Technical Specification action statements for systems. Safety Function: N/A CFR: 41.7, 41.10, 43.2, 45.13 Level: Memory Pedigree: New History: N/A Explanations: A. Incorrect - The time limit is 15 minutes but the reason is to ensure the plant is operated within the assumption of the overpressure analysis. Plausible because Part 1 is correct, part 2 is 30 minutes for brittle fracture. B. Incorrect - The time limit is 15 minutes but the reason is to ensure the plant is operated within the assumption of the overpressure analysis. Plausible because Part 1 is the time limit for T.S 3.4.9 , part 2 is 30 minutes for brittle fracture. C. Correct - The time limit is 15 minutes but the reason is to ensure the plant is operated within the assumption of the overpressure analysis. D. Incorrect - The time limit is 15 minutes but the reason is to ensure the plant is operated within the assumption of the overpressure analysis. Plausible because Part 1 is the time limit for T.S 3.4.9 , part 2 is correct. Justification for Memory: RO's are required to know tech spec actions of less than 1 hour from memory. The focus is on Reactor pressure and not on any specific systems and actions. Purely a design / fundamental question.In each lesson plan objective 7a clearly states a requirement to from memory, associated with any LCO with a duration of 1 hour or less. **Required References: None**

Question 24 Table-Item Links

General Question Data - Site Ownership

Dresden

General Question Data - Ops Program

Reactor Operator

25	**ID: 22219**	**Points: 1.00**

An accident has occurred at the station and you have volunteered to perform an evolution to **SAVE A LIFE**. The dose rate in the area you will be entering is 50 Rem/hr.

What is the **MAXIMUM** time you can spend in the area performing your task without violating TEDE Radiation Exposure Limits per RP-AA-203, EXPOSURE CONTROL AND AUTHORIZATION?

A. 2.4 minutes

B. 6 minutes

C. 12 minutes

D. 30 minutes

Answer: D

Answer Explanation

The exposure limit for protecting valuable property is 25REM TEDE. Based on 50 REM/Hr in the area, the stay time would be 30 minutes.

Question 25 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	2.00
System ID:	22219
User-Defined ID:	22219
Cross Reference Number:	
Topic:	25 - Generic.2.3.04
Num Field 1:	0.00
Num Field 2:	0.00
Text Field:	2008 Cert
Comments:	Objective: 29900LK208 Reference: RP-AA-203, EP-AA-113 K/A: Generic.2.3.4 3.2 / 3.7 K/A: Knowledge of radiation exposure limits under normal or emergency conditions. Safety Function: N/A CFR: 41.12/43.4/45.10 Level: High Pedigree: Bank History: 2014 NRC Explanations: A. Incorrect - The exposure limit for saving a life is 25 REM TEDE. Plausible because this would exceed 2 REM which is the Station TEDE limit. B. Incorrect - The exposure limit for saving a life is 25 REM TEDE. Plausible because this would exceed 5 REM which is the Federal TEDE limit. C. Incorrect - The exposure limit for saving a life is 25 REM TEDE. Plausible because this would exceed 10 REM which is the limit to protect valuable property. D. Correct - The exposure limit for saving a life is 25 REM TEDE. Based on a dose rate of 50 REM/HR the maximum stay time to save a life would be 30 minutes. **REQUIRED REFERENCES: None.**

<u>Question 25 Table-Item Links</u>

<u>General Question Data - Site Ownership</u>

Dresden

<u>General Question Data - Ops Program</u>

Reactor Operator

26	**ID: 24224**	**Points: 1.00**

The Reactor Water Cleanup pump room was recently surveyed.

- General area radiation of 200 mrem/hr
- Smearable contamination of 400 dpm/100cm^2 (beta-gamma)

How should the area be posted IAW NISP-RP-004, RADIOLOGICAL POSTINGS, AND LABELING?

A. "Caution - High Radiation Area" **ONLY**.

B. "Caution - Locked High Radiation Area" **ONLY**.

C. "Caution - High Radiation Area" **AND** "Caution - Contaminated Area"

D. "Caution - Locked High Radiation Area" **AND** "Caution - Contaminated Area"

Answer: A

Answer Explanation

A high rad area is an area that could result in reception of deep dose rate equivalent that meets or exceeds 80 mrem/hr at 30 cm. Reactor Operators are required to know this information (NGET)

Question 26 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	3.00
System ID:	24224
User-Defined ID:	24224
Cross Reference Number:	
Topic:	26 - Generic 2.3.7
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	Objective: ACAD 00-007.10, NGET Reference: RP-AA-376, NISP-RP-004 K/A: Generic 2.3.7 3.5 / 3.6 K/A: Ability to comply with radiation work permit requirements during normal or abnormal conditions. CFR: 41.12 Safety Function: N/A Level: Memory Pedigree: Bank CFR: 41.12 History: 2014 NRC Justification for Memory: Reactor Operators are required to know this information (NGET) Explanation: A. Correct - A high rad area is an area that could result in reception of deep dose rate equivalent meets or exceeds 80 mrem/hr but less than 800 mrem/hr at 30 cm. B. Incorrect - Locked High Rad area is required for areas meets or exceeds 800 mrem/hr. Plausible because High Radiation limit is exceeded. Must know what level causes a LOCKED Hi Rad area. C. Incorrect - The first part of the answer is correct. The second part is incorrect because contaminated areas are areas in which contamination levels meet or exceed 1000 dpm/100cm^2. Plausible because the first part of the answer is correct, and 400 dpm/100cm^2 is above the value of 300 dpm which used by the government. D. Incorrect - Locked High Rad area is required for areas which meet or exceed 800 mrem/hr. Contaminated areas are areas in which contamination levels meet or exceed 1000 dpm/100cm^2. Plausible because rad levels exceed 80 mrem/hr and operators must know what level causes a LOCKED Hi Rad area, and because 400 dpm/100cm^2 is above the value of 300 dpm used by the government. **REQUIRED REFERENCES: NONE**

Question 26 Table-Item Links

General Question Data - Site Ownership

Dresden

General Question Data - Ops Program

Reactor Operator

27 **ID: 9639** **Points: 1.00**

Events at the station have occurred which involve actual core degradation with a loss of containment integrity. Releases can be reasonably expected to exceed EPA Protective Action Guidelines exposure levels offsite for more than the immediate site area.

In accordance with Station EP procedures, ___(1)___ is the **HIGHEST** expected event classification and Protective Action Recommendations (PARs) ___(2)___ required.

A. (1) Site Area Emergency
(2) would be

B. (1) Site Area Emergency
(2) would **NOT** be

C. (1) General Emergency
(2) would be

D. (1) General Emergency
(2) would **NOT** be

Answer: C

Answer Explanation

General Emergency: Events are in progress or have occurred which involve actual or IMMINENT substantial core degradation or melting with potential for loss of containment integrity. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area. Definition from EP-AA-1004 Addendum 3. PARs are required for a General Emergency classification.

Question 27 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	2
Difficulty:	3.00
System ID:	9639
User-Defined ID:	9639
Cross Reference Number:	
Topic:	27 - 295038.K2.05
Num Field 1:	0.00
Num Field 2:	0.00
Text Field:	

Comments:	Objective: 29501LK018 Reference: EP-AA-1004 Addendum 3; EP-AA-111 K/A: 295038.K2.05 3.7 / 4.7 K/A: Knowledge of the interrelations between HIGH OFF-SITE RELEASE RATE and the following: Site Emergency Plan. Safety Function: 9 CFR: 41.7/45.8 Level: Memory Pedigree Bank History: N/A Explanation: This information can be found in EP-AA-1004, definition section. A. Incorrect - Site Area Emergency is not a high enough classification with releases exceeding EPA Protective Action Guidelines off-site the event will be classified as a General Emergency. PARs are required for a General Emergency classification. Plausible due to PARs not being required for ALL declarations only General Emergency. B. Incorrect - Site Area Emergency is not a high enough classification with releases exceeding EPA Protective Action Guidelines off-site the event will be classified as a General Emergency. Second part is incorrect due to PARs being required for a General Emergency classification. Plausible due to PARs not being required for ALL declarations only General Emergency. C. Correct - General Emergency: Events are in progress or have occurred which involve actual or IMMINENT substantial core degradation or melting with potential for loss of containment integrity. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area. Definition from EP-AA-1004 Addendum 3. PARs are required for a General Emergency classification. D. Incorrect - First part is correct: General Emergency: Events are in progress or have occurred which involve actual or IMMINENT substantial core degradation or melting with potential for loss of containment integrity. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area. Definition from EP-AA-1004 Addendum 3. Second part is incorrect due to PARs being required for a General Emergency classification. Plausible due to PARs not being required for ALL declarations only General Emergency. **REQUIRED REFERENCES: None**

Question 27 Table-Item Links

General Question Data - Site Ownership

Dresden

General Question Data - Ops Program

Reactor Operator

28	ID: 27571	Points: 1.00

Unit 2 was operating at near full power.

- The reactor was scrammed due to a LOCA
- The SRO directed entry into DEOP 0100
- A blowdown per DEOP 0400-02, Emergency Depressurization was performed

10 Minutes Later:

- RPV pressure is 80 psig
- 2A Core Spray pump is the ONLY available injection source, and is injecting at 4500 gpm
- Level is being maintained between the Minimum Steam Cooling Reactor Water Level (MSCRWL) AND the Minimum Zero Injection Reactor Water Level (MZRWL)

Per the definition of **ADEQUATE CORE COOLING**:

What core cooling method is **CURRENTLY** meeting the “Adequate Core Cooling” requirements, if any?

A. **NOT** being met

B. Spray Cooling **ONLY**

C. Steam Cooling **ONLY**

D. Spray Cooling **AND** Steam Cooling

Answer: A

Answer Explanation

Per DEOP 0010-00, Key Definitions: Adequate Core Cooling can be met by Submergence, Steam Cooling or Spray Cooling. We do not have submergence due to level below TAF. Adequate spray cooling is provided when design spray flow requirements are satisfied and RPV water level is at or above the elevation of the jet pump suctions, which is at 2/3 core height (Core Spray flow > 4750 gpm AND reactor water level >-191 inches). In this case, we do not meet the flow requirements for spray cooling. With RPV water level below MSCRWL, but above MZIRWL, adequate core cooling is assured if only if there is no injection. This is because, with injection, water at the core inlet is subcooled. Some of the energy produced by the core must then be expended in raising the temperature of the liquid to saturation and less steam will be produced to cool the uncovered portions of the core. The listed injection flow from core spray prevents us from meeting the steam cooling requirements when below MSCRWL.

Question 28 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	5
Difficulty:	4.00
System ID:	27571
User-Defined ID:	27571
Cross Reference Number:	
Topic:	28 - Generic 2.4.17
Num Field 1:	
Num Field 2:	
Text Field:	

Comments:	Objective: 29500SE015 Reference: DEOP 0010-00, EOP-DEOP TB K/A: Generic 2.4.17 3.9 / 4.3 K/A: Knowledge of EOP terms and definitions. Safety Function: N/A CFR: 41.10/45.13 PRA: No Level: High Pedigree: New History: N/A Explanation: A. Correct - Per DEOP 0010-00, Key Definitions: Adequate Core Cooling can be met by Submergence, Steam Cooling or Spray Cooling. We do not have submergence due to level below TAF. Adequate spray cooling is provided when design spray flow requirements are satisfied and RPV water level is at or above the elevation of the jet pump suctions, which is at 2/3 core height (Core Spray flow > 4750 gpm AND reactor water level >-191 inches). In this case, we do not meet the flow requirements for spray cooling. With RPV water level below MSCRWL, but above MZIRWL, adequate core cooling is assured if only if there is no injection. This is because, with injection, water at the core inlet is subcooled. Some of the energy produced by the core must then be expended in raising the temperature of the liquid to saturation and less steam will be produced to cool the uncovered portions of the core. The listed injection flow from core spray prevents us from meeting the steam cooling requirements when below MSCRWL. B. Incorrect - With RPV water level below MZIRWL, and above 2/3 core height, adequate core cooling by spray cooling is assured if Core Spray injection flow is at or above the design value of 4750 gpm. This flow is too low to meet the spray cooling requirements. Plausibility - This is plausible because core spray flow capacity is listed as 4500 gpm in TSG-2. However, this is not the design flow rated needed to meet the spray cooling requirements. If level is above 2/3 core height, and design core spray flow equal to or greater than 4750 exists, then spray cooling will ensure adequate core cooling C. Incorrect - With RPV water level below MSCRWL, but above MZIRWL, adequate core cooling is assured only if there is no injection. This is because, with injection, water at the core inlet is subcooled. Some of the energy produced by the core must then be expended in raising the temperature of the liquid to saturation and less steam will be produced to cool the uncovered portions of the core. The listed injection flow from core spray prevents us from meeting the steam cooling requirements when below MSCRWL. Plausibility - This is plausible because adequate core cooling would exist if no injection were occurring, with RPV level above MZIRWL; or if injection were occurring and RPV level was above MSCRWL. D. Incorrect - With RPV water level below MSCRWL, but above MZIRWL, adequate core cooling is assured only if there is no injection. This is because, with injection, water at the core inlet is subcooled. Some of the energy produced by the core must then be expended in raising the temperature of the liquid to saturation and less steam will be produced to cool the uncovered portions of the core. The listed injection flow from core spray prevents us from meeting the steam cooling requirements when below MSCRWL. Additionally, spray cooling is not met, even though we are above 2/3 core height, because we do not meet the required flow of 4750 gpm Plausibility - This is plausible because adequate core cooling using steam cooling would exist if no injection were occurring, with RPV level above MZIRWL; or if injection were occurring and RPV level was above MSCRWL. Additionally, adequate core cooling by spray cooling would exist if core spray flow were at 4750 gpm or higher. **Required References: None**

29 **ID: 12918** **Points: 1.00**

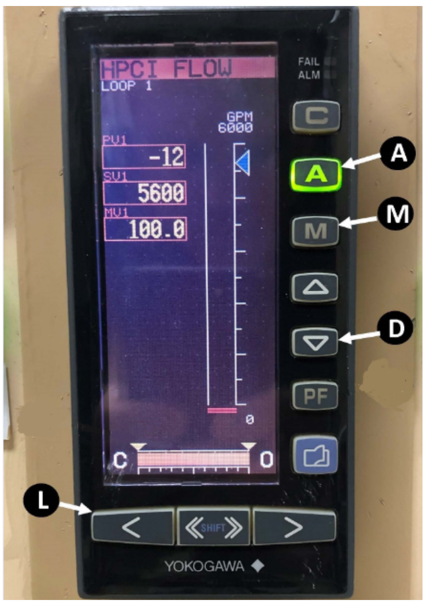
Unit 3 has just scrammed with the following conditions:

- Drywell pressure is 2.8 psig and steady.
- The 903-5 panel NSO reports that HPCI is **NOT** needed for RPV water level control, but **MAY** be needed at a later time.

Which of the following actions on the HPCI flow controller will place HPCI in the configuration specified by DOP 2300-04 hardcard attachment A, HPCI CONTROL/SHUTDOWN?

Depress pushbutton _____(1)_____.

Press and hold pushbutton _____(2)_____.



A. (1) "M"
(2) "D" until the vertical LED bar is at "0" gpm on the vertical scale.

B. (1) "A"
(2) "D" until the vertical LED bar is at "0" gpm on the vertical scale.

C. (1) "M"
(2) "L" until the horizontal LED bar is at "C" on the horizontal scale.

D. (1) "A"
(2) “L”until the horizontal LED bar is at "C" on the horizontal scale.

Answer: C

Answer Explanation

According to DOP 2300-04, Attachment 1, IF HPCI may be needed for injecting at a later time, THEN place the HPCI flow controller in MANUAL and reduce output to ZERO.

Question 29 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	3.00
System ID:	12918
User-Defined ID:	12918
Cross Reference Number:	
Topic:	29 - 206000.A2.16 - Print in Color
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	Objective: DRE206LN001.11 Reference: DOP 2300-04 K/A: 206000.A2.16 4.0 / 4.1 K/A: Ability to predict the impacts of the following on the HIGH PRESSURE COOLANT INJECTION SYSTEM; and based on those predictions, use procedures to correct, control or mitigate the consequences of those abnormal conditions or operations: High drywell pressure. Safety Function: 3 CFR 41.5/45.6 Level: Memory Pedigree: Bank History: 2007 NRC Explanations: A. Incorrect - If the controller is placed in manual by depressing the M pushbutton changing the setpoint using the D pushbutton would not affect flow. It only changes the setpoint for when the controller is in auto. Plausible because part 1 is correct, part 2 the candidate must understand that changing the setpoint in manual will not affect flow, and that the hardcard requires the HPCI controller to be in manual and output reduced to zero. B. Incorrect - If the controller is in auto and setpoint is taken to 0 gpm flow will eventually drop to zero. However, this is not correct per the Hard Card. Plausible because this method would also lower flow to zero. C. Correct - According to DOP 2300-04, Attachment 1, IF HPCI may be needed for injecting at a later time, THEN place the HPCI flow controller in MANUAL and reduce output to ZERO. D. Incorrect - The controller would still be in auto, therefore using the open/close arrows would not change flow. Plausible because the the student must understand that the open/close arrows only work in manual. **REQUIRED REFERENCES: None.**

Question 29 Table-Item Links

General Question Data - Site Ownership

Dresden

General Question Data - Ops Program

Reactor Operator

30 **ID: 9844** **Points: 1.00**

Unit 2 was operating at rated power when annunciator 902-8 H-11, ANNUN DC PWR FAILURE, illuminated on the 902-8 panel **ONLY**.

This was caused due a loss of power from 125VDC ___(1)___ and the required Operator actions is to check fuses ___(2)___.

A. (1) Div 1;
(2) in the AEER

B. (1) Div 1;
(2) in the back of the 902-8 panel

C. (1) Div 2;
(2) in the AEER

D. (1) Div 2;
(2) in the back of the 902-8 panel

Answer: A

Answer Explanation

All panel annunciators are powered from loss Div 1 125VDC. With a loss of the 902-8 panel only, the Operator action is to check/replace fuses in the AEER, inside panel 902-34.

Question 30 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	4.00
System ID:	9844
User-Defined ID:	9844
Cross Reference Number:	
Topic:	30 - Generic.2.4.32
Num Field 1:	0.00
Num Field 2:	0.00
Text Field:	
Comments:	Objective: 26302LK001 Reference: DOA 6900-02, DAN 902(3)-8 H-11, DOP 6900-06 K/A: Generic.2.4.32 3.6 / 4.0 K/A: Knowledge of operator response to loss of all annunciators. K/A Justification: Loss of all annunciators on the panel would be due to a loss of power (ie fuse, breaker, battery). Operator must have knowledge of power supplies and what actions to restore annunciators. CFR: 41.10 / 43.5 / 45.13 Safety Function: N/A PRA: Yes Pedigree: Bank Level: High History: 11-1 NRC Explanations: A. Correct - All panel annunciators are powered from loss Div 1 125VDC. With a loss of the 902-8 panel only, the Operator action is to check/replace fuses in the AEER, inside panel 902-34. B. Incorrect - Inside the back of the 902-8 panel is the reset switch for the Bus 28 UV, not the fuses for the 125VDC annunciator feed. Plausible because part 1 is correct. Part 2 is plausible because there are UV resets in the back panel for 902-8. C. Incorrect - The 902-8 panel annunciator feed is not supplied by 125VDC Div 2. Plausible because 2B-2 is the reserve feed for 902-8 panel annunciators and part 2 is correct. D. Incorrect - The 902-8 panel annunciator feed is not supplied by 125VDC Div 2, and also the fuses are not located inside the back of the panel. Plausible because 2B-2 is the reserve feed for 902-8 panel annunciators. Part 2 is plausible because there are UV resets in the back panel for 902-8. **REQUIRED REFERENCES: None.**

Question 30 Table-Item Links

Dresden Procedures

DOA 6900-003

General Question Data - Site Ownership

Dresden

General Question Data - Ops Program

Reactor Operator

31	ID: 9608	Points: 1.00

A transient has occurred on Unit 2 producing the following indications:

- Drywell pressure is 1.6 psig and rising slowly.
- Drywell Temperature is 170 degrees and rising slowly.
- Bus 24-1 tripped on overcurrent.
- Bus 29 is de-energized.

What actions should be taken to address the increasing drywell temperature?

A. Spray the Torus.

B. Start another RBCCW Pump.

C. Energize Bus 29 from Bus 28 to restart Drywell Coolers.

D. Bypass interlocks per DEOP 0500-02 and start all available Drywell Coolers.

Answer: C

Answer Explanation

No interlocks are preventing starting of drywell coolers, only the loss of power to Bus 29. DEOP 0200-01 has you start all available Drywell coolers, and to do that you must re-energize Bus 29 from Bus 28.

Question 31 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	3.00
System ID:	9608
User-Defined ID:	9608
Cross Reference Number:	295012
Topic:	31 - 295028.A1.02
Num Field 1:	0.00
Num Field 2:	0.00
Text Field:	
Comments:	Objective: 29502LK014 References: DGA 12, DEOP 200-1 K/A: 295028.A1.02 3.9/3.8 K/A: Ability to operate and/or monitor the following as they apply to HIGH DRYWELL TEMPERATURE: drywell ventilation system. Safety Function: 5 CFR: 41.7/45.6 PRA: No Level: High Pedigree: Bank History: N/A Explanations: A. Starting Torus Sprays will not reduce Drywell Temperature. Plausible because DEOP 200-1 entry conditions have been met, but drywell pressure has not exceeded 2 psig. B. Starting an additional RBCCW pump would not provide sufficient cooling and no procedural bases. Plausible because additional RBCCW flow would increase cooling to the Drywell Coolers that were still running on Bus 28. C. Correct - No interlocks are preventing starting of drywell coolers, only the loss of power to Bus 29. DEOP 0200-01 has you start all available Drywell coolers, and to do that you must re-energize Bus 29 from Bus 28. D. There are currently no interlocks preventing the restart of Drywell coolers. Plausible because if drywell pressure had exceeded 2 psig. Then DEOP 200-1 provides guidance to bypass interlocks per DEOP 500-2 and restart Drywell coolers. **Required References: None**

Question 31 Table-Item Links

Dresden Procedures

DEOP 0200-001

General Question Data - Site Ownership

Dresden

32	ID: 27576	Points: 1.00

Torus level was normal when a large unisolable leak developed on the bottom of the Torus.

Which of the following describes the expected change in Drywell to Torus dP and the validity of the safety analysis?

Drywell to Torus dP will increase until Torus level reaches ___(1)___ and then equalize.

Safety analysis assumptions are valid until Torus level reaches ___(2)___.
.

A. (1) 11 feet
(2) 11 feet 6.5 inches

B. (1) 11 feet
(2) 14 feet 6.5 inches

C. (1) 12 feet
(2) 11 feet 6.5 inches

D. (1) 12 feet
(2) 14 feet 6.5 inches

Answer: B

Answer Explanation

Downcomers are 3.67 to 4 feet below the surface of the Torus. This means the downcomers will uncovered at ~11 feet, at which point Drywell to Torus DP will stop increasing. T.S. 3.6.2.2 states that safety analysis conditions are not met if Torus water level is outside of the limits.

Question 32 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	4
Difficulty:	4.00
System ID:	27576
User-Defined ID:	27576
Cross Reference Number:	
Topic:	32 - 295030 A2.04
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	Objective: 223LN001.01 References: DEOP 200-1, T.S. 3.6.2.2, EOP-DEOP TB, TSG-3, DOP 1600-16 K/A: 295030 A2.04 3.5 / 3.7 K/A: Ability to determine and/or interpret the following as they apply to LOW SUPPRESSION POOL WATER LEVEL: Drywell/suppression chamber differential pressure. Safety Function: 5 CFR: 41.1043.5/45.13 PRA: No Level: High Pedigree: Bank History: N/A Explanation: A. Incorrect - 11 feet is the point at which the downcomers are uncovered and DP will equalize. The safety analysis is based on 14 feet 6.5 inches in the T.S.. Plausible because part 1 is correct and Part 2 is the level at which HPCI exhaust is uncovered and nearing the point where downcomers are uncovered requiring an emergency depressurization. B. Correct - Downcomers are 3.67 to 4 feet below the surface of the Torus. This means the downcomers will uncovered at ~11 feet, at which point Drywell to Torus DP will stop increasing. T.S. 3.6.2.2 states that safety analysis conditions are not met if Torus water level is outside of the limits. C. Incorrect - 12 feet is for the HPCI exhaust line and would not cause DP to equalize. The safety analysis is based on 14 feet 6.5 inches in the T.S.. Plausible because HPCI exhaust is uncovered and nearing the point where downcomers are uncovered requiring an emergency depressurization. D. Incorrect - 12 feet is for the HPCI exhaust line and would not cause DP to equalize. Plausible because HPCI exhaust is uncovered and nearing the point where downcomers are uncovered requiring an emergency depressurization. Part 2 is correct. **Required References: NONE**

None

33	**ID: 24074**	**Points: 1.00**

Unit 2 was operating at near rated power when a transient occurred.

- Drywell pressure is 3.0 psig and steady.
- Reactor pressure is 950 psig and steady.
- RWCU blowdown rate is 200 gpm and steady.
- RWCU recirculation and blowdown modes are being used to control RPV pressure.
- RPV water level is 12 inches and lowering at rate of 1 inch/minute with all available systems operating.

Concerning RWCU operation, which of the following actions are required?

A. Secure blowdown **AND** recirculation modes.

B. Secure blowdown **AND** maintain recirculation mode.

C. Maintain existing blowdown **AND** recirculation modes.

D. Maintain existing blowdown **AND** secure recirculation mode.

Answer: B

Answer Explanation

Securing blowdown will decrease the amount of water leaving the reactor, which mitigates the lowering RPV water level. DEOP 100 allows installing jumpers to maintain RWCU Recirc mode for pressure control.

Question 33 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	4.00
System ID:	24074
User-Defined ID:	24074
Cross Reference Number:	
Topic:	33 - 295009.A2.03
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	Objective: 204LN001.08 Reference: DEOP 0100, DOP 1200-02, DGP 02-03 K/A: 295009.A2.03 2.9 / 2.9 K/A: Ability to determine and/or interpret the following as they apply to LOW REACTOR WATER LEVEL: Reactor water cleanup blowdown rate. CFR: 41.7 Safety Function: 2 PRA: No Level: High Pedigree: Bank History: 11-1 NRC Explanation: A. Incorrect - because securing recirc mode would secure the current means of pressure control. Plausible because securing blowdown will decrease the amount of water leaving the reactor, which mitigates the lowering RPV water level. B. Correct - Securing blowdown will decrease the amount of water leaving the reactor, which mitigates the lowering RPV water level. DEOP 100 allows installing jumpers to maintain RWCU Recirc mode for pressure control. C. Incorrect - Maintaining both modes in operation would not mitigate the given conditions but is plausible because it would be allowed by procedure. D. Incorrect - Plausible because securing recirc mode while maintaining blowdown mode would maintain pressure control, but it would not mitigate the given conditions. **REQUIRED REFERENCES: None.**

Question 33 Table-Item Links

General Question Data - Site Ownership

Dresden

General Question Data - Ops Program

Reactor Operator

34 **ID: 27577** **Points: 1.00**

Unit 2 is at rated power with a normal electrical line-up.

- The U2 EDG is being synchronized to Bus 24-1, per DOS 6600-01, DIESEL GENERATOR SURVEILLANCE TESTS.

In order to line up the U2 EDG to Bus 24-1, the operator would ensure that:

1) **INCOMING** voltage is slightly higher than **RUNNING** voltage to prevent_____(1)_____ .

2) To minimize the potential for motorizing the EDG, the EDG output breaker is closed when the synchroscope is at approximately the twelve (12) o'clock position, rotating approximately one (1) revolution every 30 seconds in the _____(2)_____ direction.

A. (1) inductive power loading on the EDG
(2) fast

B. (1) inductive power loading on the EDG
(2) slow

C. (1) capacitive power loading on the EDG
(2) fast

D. (1) capacitive power loading on the EDG
(2) slow

Answer: C

Answer Explanation

(1) Per DOS 6600-01, INCOMING voltage should be slightly higher than RUNNING voltage to allow the oncoming generator to generate reactive power thus not weakening the generator field
(2) Per DOS 6600-01, the output breaker should be closed when the synchroscope is rotating one revolution in approximately 30 seconds in the fast direction, and incoming voltage is slightly higher than running voltage. The closure is performed when the breaker is in synch with the bus it is being placed on, which occurs when the synchroscope is at approximately the 12 o'clock position.

Question 34 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	3.00
System ID:	27577
User-Defined ID:	27577
Cross Reference Number:	
Topic:	34 - 262001.A4.02
Num Field 1:	
Num Field 2:	
Text Field:	

Comments:	Objective: 262LN003.11 References: DOS 6600-01 K/A: 262001.A4.02 3.4/3.4 K/A: Ability to manually operate and/or monitor in the control room: Synchroscope, including understanding of running and incoming voltages: AC Electrical distribution. Safety Function: 6 CFR: 41.7/45.5 to 45.8 PRA: Yes Level: Memory Pedigree: New History: N/A Explanation: A. INCORRECT - (1) Per DOS 6600-01, INCOMING voltage should be slightly higher than RUNNING voltage because the oncoming generator will generate reactive power rather than draw reactive power which could weaken the generator field. (2) The second part is correct. B. INCORRECT - (1) Per DOS 6600-01, INCOMING voltage should be slightly higher than RUNNING voltage because the oncoming generator will generate reactive power rather than draw reactive power which could weaken the generator field. (2) If the synchroscope were rotating in the SLOW direction, then Bus 24-1 frequency would be higher than the EDG frequency thus the EDG would not pick up real load when its output breaker closes. C. Correct - (1) Per DOS 6600-01, INCOMING voltage should be slightly higher than RUNNING voltage to allow the oncoming generator to generate reactive power thus not weakening the generator field (2) Per DOS 6600-01, the output breaker should be closed when the synchroscope is rotating one revolution in approximately 30 seconds in the fast direction, and incoming voltage is slightly higher than running voltage. The closure is performed when the breaker is in synch with the bus it is being placed on, which occurs when the synchroscope is at approximately the 12 o'clock position. D. INCORRECT (1) The first part is correct (2) If the synchroscope were rotating in the SLOW direction, then Bus 24-1 frequency would be higher than the EDG frequency thus the EDG would not pick up real load when its output breaker closes. **Required References: NONE**

None

35	ID: 23825	Points: 1.00

Unit 2 was in a refuel outage with all four of the RPS Shorting Links **INSTALLED** in the 902-15 and -17 Panels, when SRM 21 spiked to a full scale indication.

What response, if any, is expected from the RPS system under these conditions?

A. No RPS actuation.

B. 1/2 scram on RPS channel A **ONLY**.

C. 1/2 scram on RPS channel B **ONLY**.

D. Full scram.

Answer: A

Answer Explanation

With shorting links installed, the normal logic (an RPS trip from any neutron monitoring instrument-SRM,IRM, or APRM-results in a full scram) is bypassed.

Question 35 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	3.00
System ID:	23825
User-Defined ID:	23825
Cross Reference Number:	
Topic:	35 - 215004.K3.01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	Objective: DRE212LN001.06 Reference: DOP 0700-01, DAN 902(3)-5 B-12, 12E-2465 Sh. 2 K/A: 215004.K3.01 3.4 / 3.4 K/A: Knowledge of the effect that loss or malfunction of the SOURCE RANGE MONITOR (SRM) SYSTEM will have on following: RPS Safety Function: 7 CFR: 41.7/45.4 PRA: No Level: Memory Pedigree: Bank History: 2010 NRC Explanation: A. Correct - With shorting links installed, the normal logic (an RPS trip from any neutron monitoring instrument-SRM,IRM, or APRM-results in a full scram) is bypassed. B. Incorrect - With shorting links installed the normal logic is bypassed. Plausible because if shorting links were not installed this would be the correct answer. SRM 21 is part of the A system. C. Incorrect - With shorting links installed the normal logic is bypassed. Plausible because if shorting links were not installed this would cause a RPS B 1/2 scram in addition to a RPS A 1/2 scram. D. Incorrect - With shorting links installed the normal logic is bypassed. Plausible because if shorting links were not installed any 1 SRM Hi Hi would cause a full scram. **REQUIRED REFERENCES: None.**

Question 35 Table-Item Links

General Question Data - Site Ownership

Dresden

General Question Data - Ops Program

Reactor Operator

36 **ID: 23492** **Points: 1.00**

Unit 2 was operating at near rated power when the following timeline occurred:

- 01:30:00 - Drywell pressure is 1.3 psig
- 01:30:05 - RPV pressure is 1073 psig.
- 01:30:10 - Drywell pressure is 1.6 psig
- 01:30:15 - RPV pressure is 1093 psig.
- 01:30:25 - RPV pressure is 1078 psig.
- 01:30:35 - RPV pressure is 1025 psig.

The reason for the SCRAM was high ___(1)___ pressure; and the Isolation Condenser ___(2)___ initiated to control RPV pressure.

A. (1) RPV
(2) automatically

B. (1) RPV
(2) must be manually

C. (1) Drywell
(2) automatically

D. (1) Drywell
(2) must be manually

Answer: A

Answer Explanation

The Scram would occur with Reactor Pressure greater than 1033 psig. When pressure rises above 1070 psig and is sustained for greater than 15 seconds the Isolation Condenser will auto start.

Question 36 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	3.00
System ID:	23492
User-Defined ID:	23492
Cross Reference Number:	
Topic:	36 - 295025 K2.02
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	Objective: DRE207LN001.06 Reference: DAN 902(3)-5 C-13, D-11, H-5; DAN 902(3)-4 A-15;DOP 1300-02 K/A: 295025 K2.02 3.9 / 4.0 K/A: Knowledge of the interrelations between HIGH REACTOR PRESSURE and the following: isolation condenser. Safety Function: 3 CFR: 41.7/45.8 PRA: No Level: High Pedigree: Bank History: 2013 Cert Explanation: A. Correct - The scram was caused by High Reactor pressure BEFORE the required scram for Drywell pressure exceeding 1.5 psig. The IC initiates when RPV pressure is sustained above setpoint (1047 to 1063) for a nominal time of 15 seconds. RPV pressure went above the initiation setpoint at time 01:30:05, then stayed above the setpoint for > 20 seconds, which would cause the IC to initiate automatically. B. Incorrect - The scram would be caused by High Reactor pressure prior to Drywell pressure Iso Condenser would have auto started at 1:30:20 when pressure was greater than 1070 for greater than 15 seconds. Plausible because part 1 is correct and part 2 because pressure dropped back below 1070. The candidate must recognize the time frame above 1070. C. Incorrect - High Reactor pressure would occur prior to Drywell pressure exceeding 1.5 psig, part 2 is correct. Plausible because DW pressure would exceed 1.5 psig less than 5 seconds after RPV high pressure signal and part 2 is correct. D. Incorrect - High Reactor pressure would occur prior to Drywell pressure exceeding 1.5 psig, IC would start automatically. Plausible because DW pressure would exceed 1.5 psig less than 5 seconds after RPV high pressure signal. Part 2 because pressure dropped back below 1070. The candidate must recognize the time frame above 1070. **REQUIRED REFERENCES: None.**

Question 36 Table-Item Links

General Question Data - Site Ownership

Dresden

General Question Data - Ops Program

Reactor Operator

37 **ID: 27406** **Points: 1.00**

Unit 2 was operating at rated power when a LOCA occurred.

- RPV level has been restored above TAF
- **ONLY** 2A Core Spray pump is injecting

Annunciator 902-3 E-5, 2A CORE SPRAY HDR DP HI, is in alarm.

What abnormal condition is the cause of this alarm, if any, and what action is required, if any, to maintain RPV level above TAF?

A. This alarm is expected.
No actions are required.

B. Neither Core Spray System is intact.
Inject with LPCI system.

C. 2A Core Spray header is not intact.
Inject with 2B Core Spray system.

D. RPV pressure has exceeded the shutoff head of LP ECCS pumps.
Inject with High Pressure injections source.

Answer: C

Answer Explanation

With a leak in the 2A Core Spray system, DAN 902(3)-3 E-5 directs the operator to inject with 2B Core Spray system.

Question 37 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	4.00
System ID:	27406
User-Defined ID:	27406
Cross Reference Number:	
Topic:	37 - 209001.A4.01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	Objective: 209LN001.12a Reference: DAN 902(3)-3 E-5 K/A: 209001.A4.01 3.8 / 3.6 K/A: Ability to manually operate and/or monitor in the control room: Core spray pump. CFR: 41.7/45.5 to 45.8 Safety Function: 2 Pedigree: Bank History: 16-1 NRC Level: High Explanation: A. Incorrect - This alarm indicates a break in the 2A Core Spray header. Plausible because this alarm may actuate during shutdown and is considered expected under shutdown conditions. B. Incorrect - Core Spray system is comprised of 2 independent subsystems, each with their own spray header. Plausible because LPCI (another LP ECCS system) is crosstied. C. Correct - With a leak in the 2A Core Spray system, DAN 902(3)-3 E-5 directs the operator to inject with 2B Core Spray system. D. Incorrect - This alarm is based on differential pressure between the core shroud and the header. Plausible because the sensor is driven by differential pressure. **REQUIRED REFERENCES: None.**

None

38	**ID: 27602**	**Points: 1.00**

Unit 2 and 3 are operating at full power, when the following events occurred:

- A fire occurs in the 902-8 panel
- The SRO directed entry into DOA 0010-10 FIRE / EXPLOSION, and DOA 5750-04, SMOKE, NOXIOUS FUMES OR AIRBORNE CONTAMINANTS IN THE CONTROL ROOM
- The control room is slowly filling with smoke
- Control room operators have donned SCBAs and activated the breathing air system

Per DOA 0010-10, and DOA 5750-04:

- The crew **MUST** place the Control Room HVAC system in the___(1)___ mode of operation
- The crew **MUST ENTER** procedure___(2)___.

A. (1) Isolation
(2) DSSP 0100-CR, HOT SHUTDOWN PROCEDURE - CONTROL ROOM EVACUATION

B. (1) Isolation
(2) DSSP 0010-01, DETERMINING SAFE SHUTDOWN PATHS FOR EXTENSIVE PLANT DAMAGE

C. (1) Smoke Purge
(2) DSSP 0100-CR, HOT SHUTDOWN PROCEDURE - CONTROL ROOM EVACUATION

D. (1) Smoke Purge
(2) DSSP 0010-01, DETERMINING SAFE SHUTDOWN PATHS FOR EXTENSIVE PLANT DAMAGE

Answer: D

Answer Explanation

(1) with a fire in the main control room, DOA 5750-04 directs placing the control room HVAC system in Smoke Purge mode of operation. (2) The control room conditions given do NOT mandate evacuation. DOA 0010-10 requires entry into DSSP 0010-01 but neither DOA **REQUIRES** entry into DSSP 0100-CR. Entry into DSSP 0100-CR would be **REQUIRED** only if it became impossible to operate the plant from within the main control room, or if conditions in the main control room indicated that control of the plant was being lost because of fire damage.

Question 38 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	4
Difficulty:	4.00
System ID:	27602
User-Defined ID:	27602
Cross Reference Number:	
Topic:	38 - 295016 K2.03
Num Field 1:	
Num Field 2:	
Text Field:	

Comments:	Objective: DRE288LN003 Obj 2 Reference: DOA 0010-10, DOA 5750-04, DSSP 0100-CR, DSSP 0010-01 K/A: 295016.AK2.03 2.9 / 3.1 K/A: Knowledge of the interrelations between CONTROL ROOM ABANDONMENT and the following: Control room HVAC Safety Function: 8 CFR: 41.7/45.8 PRA: No Level: High Pedigree: New History: N/A Explanations: A. INCORRECT - (1) with a fire in the main control room, DOA 5750-04 requires placing the control room HVAC system in Smoke Purge mode of operation. (2) The control room conditions given do NOT mandate evacuation. DOA 0010-10 requires entry into DSSP 0010-01, but neither DOA requires entry into DSSP 0100-CR. Entry into DSSP 0100-CR would be REQUIRED only if it became impossible to operate the plant from within the main control room, or if conditions in the main control room indicated that control of the plant was being lost because of fire damage. Plausibility: This is plausible because (1) if the fire was in a location outside the main control room and the "B" HVAC room, then the procedure would require placing the control room HVAC system in Isolation mode. (2) DOA 5750 gives guidance to review DSSP 0100-CR, if required. However, the operator needs to recognize that entry into DSSP 0100-CR and evacuation of the Main Control Room is a significant action that is NOT taken unless conditions require it. B. INCORRECT - (1) with a fire in the main control room, DOA 5750-04 requires placing the control room HVAC system in Smoke Purge mode of operation. (2) The second part of this answer is correct. Plausibility: This is plausible because (1) if the fire was in a location outside the main control room and the "B" HVAC room, then the procedure would require placing the control room HVAC system in Isolation Mode. (2) The second part of this answer is correct. C. INCORRECT - (1) The first part of the answer is correct. (2) The control room conditions given do NOT mandate evacuation. DOA 0010-10 requires entry into DSSP 0010-01, but neither DOA requires entry into DSSP 0100-CR. Entry into DSSP 0100-CR would be REQUIRED only if it became impossible to operate the plant from within the main control room, or if conditions in the main control room indicated that control of the plant was being lost because of fire damage Plausibility: This is plausible because (1) The first part of the answer is correct. (2) DOA 5750 gives guidance to review DSSP 0100-CR, if required. However, the operator needs to recognize that entry into DSSP 0100-CR and evacuation of the Main Control Room is a significant action that is NOT taken unless conditions require it. D. CORRECT - (1) with a fire in the main control room, DOA 5750-04 directs placing the control room HVAC system in Smoke Purge mode of operation. (2) The control room conditions given do NOT mandate evacuation. DOA 0010-10 requires entry into DSSP 0010-01, but neither DOA REQUIRES entry into DSSP 0100-CR. Entry into DSSP 0100-CR would be REQUIRED only if it became impossible to operate the plant from within the main control room, or if conditions in the main control room indicated that control of the plant was being lost because of fire damage. **Required references: None**

39 **ID: 27580** **Points: 1.00**

Given the following trends on Unit 2, when is DEOP 200-1, PRIMARY CONTAINMENT CONTROL, entry **FIRST** required?

TIME	DRYWELL PRESSURE	DRYWELL TEMPERATURE
00:00	1.35	141 F
00:02	1.50	146 F
00:04	1.65	151 F
00:06	1.80	156 F

A. 00:02

B. 00:04

C. 00:08

D. 00:10

Answer: C

Answer Explanation

At time 00:08 using the current trend Drywell temperature will exceed 160 degrees which is a DEOP 200-1 entry condition.

Question 39 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	4
Difficulty:	3.00
System ID:	27580
User-Defined ID:	27580
Cross Reference Number:	
Topic:	39 - 295012 G 2.4.1
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	Objective: 29502LK001 Reference: DEOP 0200-01 K/A: 295012 G.2.4.1 4.6 / 4.8 K/A: Knowledge of EOP entry conditions and immediate actions steps. High Drywell Temperature. Safety Function: 5 CFR: 41.10/43.5/45.13 PRA: No Level: High Pedigree: New History: N/A Explanation: A. Incorrect - DW pressure is less than DEOP entry condition of 2 psig and DW temperature is less than DEOP entry condition 160 F. Plausible because DW pressure is above the threshold for SCRAM per OP-DR-103-102-1002 (1.5 psig). B. Incorrect - DW pressure is less than DEOP entry condition of 2 psig and DW temperature is less than DEOP entry condition 160 F. Plausible because SCRAM threshold for DW pressure has been met as well as Tech Spec limit for DW temperature (150 F). C. Correct - At time 00:08 using the current trend Drywell temperature will exceed 160 degrees which is a DEOP 200-1 entry condition. D. Incorrect - Both DW pressure and DW temperature DEOP entry conditions have been met. But question ask for the FIRST entry. Plausible because this is the time at which DW pressure entry condition has been met. **REQUIRED REFERENCES: NONE**

None

40	ID: 9625	Points: 1.00

With both units at full power, a transient has occurred resulting in a reading of 8 mR/hr on the Unit 2 Reactor Building Vent Radiation monitors.

How should the crew respond to this condition?

A. Verify **ONLY** Unit 2 Reactor Building Vent has isolated

B. Verify Unit 2 and 3 Reactor Building Vent is running
Verify SBGT has initiated

C. Verify Unit 2 and 3 Reactor Building Vent has isolated
Verify SBGT has initiated

D. Verify Unit 2 Reactor Building Vent is isolated
Verify Unit 3 Reactor Building Vent is running
Verify SBGT has initiated

Answer: C

Answer Explanation

RB ventilation is a shared system. When radiation levels of greater than 4mr are sensed, both U2 & U3 RB vent systems isolate. This also is a initiation signal for an auto start of SBGT.

Question 40 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	4.00
System ID:	9625
User-Defined ID:	9625
Cross Reference Number:	LIH
Topic:	40 - 295033 A1.03
Num Field 1:	0.00
Num Field 2:	0.00
Text Field:	
Comments:	Objective: 29502LP017 Reference: DEOP 0300-01; DAN 902(3)-3 A-3, F-14 K/A: 295033.A1.03 3.8 / 3.8 K/A: Ability to operate and/or monitor the following as they apply to HIGH SECONDARY CONTAINMENT AREA RADIATION LEVELS: Secondary containment ventilation. Safety Function: 9 CFR: 41.7/45.6 PRA: No Level: High Pedigree: Bank History: N/A Explanation: A. Incorrect - Unit 2 RB ventilation has isolated, but in addition Unit 3 RB ventilation has isolated and SBGT has auto started on rad greater than 4 mr. Plausible because it is a partially correct answer, it does isolate. B. Incorrect - Unit 2 and Unit 3 RB ventilation would be isolated. Plausible because signal does impact both units, but Rad of greater than 4 mr has been exceeded causing both units to isolate. C. Correct - RB ventilation is a shared system. When radiation levels of greater than 4mr are sensed, both U2 & U3 RB vent systems isolate. This also is a initiation signal for an auto start of SBGT. D. Incorrect - Unit 3 would also be isolated RB ventilation is a shared system between units. An alarm on either unit for Hi rad affects both units. Plausible because U2 RB vent would be isolated as well as SBGT initiated. **Required References: None**

Question 40 Table-Item Links

Dresden Procedures

DEOP 0300-001

General Question Data - Site Ownership

Dresden

41 **ID: 24041** **Points: 1.00**

Unit 2 is operating at full power with the following conditions:

- 2A and 2B RBCCW pumps and heat exchangers are in service.
- 2/3 RBCCW pump and heat exchanger are lined up to Unit 3.

A transient occurs resulting in the following conditions:

- Reactor scram due to high drywell pressure
- Drywell pressure is 7.3 psig and climbing at a rate of 1.0 psig per minute.
- Bus 24-1 was de-energized and is now powered from the U2 EDG

One (1) Minute later:

- 2A RBCCW pump tripped on overcurrent.

What action(s) is(are) required regarding the Unit 2 RBCCW system?

A. Isolate RBCCW to the drywell.

B. Immediately start 2B RBCCW pump.

C. Remove 2A RBCCW heat exchanger from service.

D. Reset the under-voltage on Bus 24-1, then start 2B RBCCW pump.

Answer: A

Answer Explanation

Given the conditions in the stem, all RBCCW pumps on Unit 2 have been lost. Bus 24-1 is the power supply to RBCCW pump 2B. RBCCW Pump 2B is interlocked to trip on a combination of Core Spray System initiation (+2 psig in the drywell) and loss of normal power to Bus 24-1.DOA 3700-01, LOSS OF COOLING BY RBCCW, immediate actions state: If a LOCA has occurred and drywell pressure is greater than 2 psig, then isolate the drywell RBCCW.

Question 41 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	4
Difficulty:	3.00
System ID:	24041
User-Defined ID:	24041
Cross Reference Number:	
Topic:	41 - 400000.K6.04
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	Objective: 208LN001.12 Reference: DOA 3700-01, DAN 923-1 C-1 K/A: 400000.K6.04 3.0 / 3.1 K/A: Knowledge of the effect that a loss or malfunction of the following will have on the CCWS: Pumps CFR: 41.7 Safety Function: 8 PRA: No Level: High Pedigree: Bank History: 2015 Cert Explanation: A. Correct - Given the conditions in the stem, all RBCCW pumps on Unit 2 have been lost. Bus 24-1 is the power supply to RBCCW pump 2B. RBCCW Pump 2B is interlocked to trip on a combination of Core Spray System initiation (+2 psig in the drywell) and loss of normal power to Bus 24-1.DOA 3700-01, LOSS OF COOLING BY RBCCW, immediate actions state: If a LOCA has occurred and drywell pressure is greater than 2 psig, then isolate the drywell RBCCW. B. Incorrect - The 2B RBCCW can not be restarted under these conditions until leads are lifted, per DEOP 0500-02, that prevent the 2B RBCCW pump from starting. Plausible because DEOP 500-2 does allow this evolution just not with the 2 psig in the Drywell C. Incorrect - Plausible because If a high drywell pressure signal was not present, the 2B RBCCW pump would be running and the examinee may determine removing the 2A RBCCW heat exchanger from service to prevent overloading the 2B RBCCW pump would be the correct action, common misconception. D. Incorrect - On a loss of Bus 24-1, the undervoltage relay is required to be reset to start loads powered on Bus 29, this action would not allow the 2B RBCCW pump to start. Plausible because this is a common misconception for this undervoltage relay. **REQUIRED REFERENCES: None.**

Question 41 Table-Item Links

General Question Data - Site Ownership

Dresden

General Question Data - Ops Program

Reactor Operator

42	ID: 6012	Points: 1.00

Unit 2 is at full power with 2B LPCI pump OOS for maintenance.

Then the following occurs:

- Unit 2 has a loss of offsite power.
- A leak develops in the drywell and pressure is +2.1 psig and rising.
- Annunciator 902-8 H-5, U2/3 DIESEL GEN DIFFERENTIAL FAULT, alarms.

What ECCS pumps are running?

A. 2A LPCI Pump and the 2A CS Pump

B. 2C and 2D LPCI Pumps and 2B CS Pump

C. 2A and 2C LPCI Pumps and the 2B CS Pump

D. 2C and 2D LPCI Pumps and the 2A CS Pump

Answer: B

Answer Explanation

After a loss of offsite power with a LOCA signal all pumps will attempt to start. The loss of Bus 23-1 prevents 2A and 2B from starting. 2C and 2D are powered from 24-1.

Question 42 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	2.00
System ID:	6012
User-Defined ID:	6012
Cross Reference Number:	LC
Topic:	42 - 264000 K3.01
Num Field 1:	0.00
Num Field 2:	0.00
Text Field:	
Comments:	Objective: 203LN001-12 Reference: DOP 6500-11, DOP 6500-13, DAN 902(3)-8 H-5, DOS 6600-01 K/A: 264000 K3.01 4.2 / 4.4 K/A: Knowledge of the effect that a loss or malfunction of the EMERGENCY GENERATORS (DIESEL/JET) will have on the following: Emergency core cooling systems. Safety Function: 6 CFR: 41.7/45.4 PRA: Yes Level: High Pedigree: Bank History: N/A Explanations: A. Incorrect - With the OVERCURRENT on Bus 23-1 Division I ECCS pumps are not available. Plausible because must determine power supplies to each of the ECCS pumps and the 2 B LPCI being OOS makes the answer more valid. B. Correct - After a loss of offsite power with a LOCA signal all pumps will attempt to start. The loss of Bus 23-1 prevents 2A and 2B from starting. 2C and 2D are powered from 24-1. C. Incorrect - 2C LPCI pump and 2B CS pumps will be running, but 2A LPCI is not available due to the overcurrent. Plausible because 2 of the 3 pumps are correct and the third must determine divisional power supplies. D. Incorrect - 2C and 2D LPCI pumps will be running but 2A core spray is not available due to the overcurrent on Bus 23-1. Plausible because 2 of the 3 pumps are correct and the third must determine divisional power supplies. Having 2B LPCI pump OOS makes the answer more valid. **Required References: None**

Question 42 Table-Item Links

General Question Data - Site Ownership

Dresden

43 **ID: 27478** **Points: 1.00**

Both Units were operating at 100% power.

Fire header pressure has reached 92 psig and is dropping at a rate of 4 psig/min.

If the trend is never arrested and no operator action is taken, the **EARLIEST** time at which the U1 DFP will have auto started is....

A. 3 minutes later

B. 4 minutes later

C. 5 minutes later

D. 6 minutes later

Answer: C

Answer Explanation

The Unit 1 DFP will auto start at 75 psig. At the current rate this setpoint will not be met until the 5 minute mark.

Question 43 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	4.00
System ID:	27478
User-Defined ID:	27478
Cross Reference Number:	
Topic:	43 - 286000.A3.02
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	Objective: DRE286LN001.11 Reference: DOA 3900-01, DAN XL3 82-30, DAN 901-2 H-8 K/A: 286000.A3.02 3.1/3.2 K/A: Ability to monitor automatic operations of the FIRE PROTECTION SYSTEM including: Fire main pressure. CFR: 41.7 Safety Function: 8 PRA: No Pedigree: New Level: High Explanations: A. Incorrect - At the 3 minute point, fire main pressure would be 80 psig, The U1 DFP does not start until 75 psig. Plausible because the 2/3 DFP will start at this pressure. B. Incorrect - At the 4 minute point, fire main pressure would be 76 psig. The U1 DFP will not auto start until header pressure drops to 75 psig. Plausible because the 2/3 DFP will have auto started and within 1 psig of U1 DFP auto start. C. Correct - The Unit 1 DFP will auto start at 75 psig. At the current rate this setpoint will not be met until the 5 minute mark. D. Incorrect - At the 6 minute point, fire main pressure would be 68 psig. The U2/3 and U1 DFP's will have started earlier. Plausible because at this time the U1 screen wash pumps will also have started at 70 psig. **REQUIRED REFERENCES: None.**

None

44 **ID: 27606** **Points: 1.00**

Unit 2 is operating at 85% power.

- The U2 NSO is directed to commence blowdown withe RWCU system per DOP 1200-02, RWCU SYSTEM BLOWDOWN as part of a Post Maintenance Test.

When establishing RWCU blowdown flow to the Main Condenser, it is expected that indicated CORE THERMAL POWER will ____(1)____.

Opening valves 2-1201-11, BLOWDN TO CONDR, AND 2-1201-12, BLOWDN TO RW VLV SIMULTANEOUSLY is not allowed by the procedure. This is because this configuration could cause a _____(2)_____.

A. (1) increase
(2) pressure spike and system isolation

B. (1) increase
(2) drop in vacuum in the main condenser

C. (1) decrease
(2) pressure spike and system isolation

D. (1) decrease
(2) drop in vacuum in the main condenser

Answer: B

Answer Explanation

Per DOP 1200-03, establishing blowdown to the main condenser at power will cause indicated core power to increase. Per DOP 1200-03, Opening MO 2(3)-1201-11, BLOWDN TO CONDR, AND MO 2(3)-1201-12, BLOWDN TO RW VLV, simultaneously will provide an open pathway between Radwaste and the Main Condenser. This will cause a drop in vacuum, if established, in the Main Condenser.

Question 44 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	3.00
System ID:	27606
User-Defined ID:	27606
Cross Reference Number:	
Topic:	44 - 204000. K1.06
Num Field 1:	
Num Field 2:	
Text Field:	

Comments:	Objective: DRE204LN001.02 Reference: DOP 1200-03 K/A: 20400 K1.06 K/A: Knowledge of the physical connections and/or cause-effect relationships between REACTOR WATER CLEANUP SYSTEM and the following: Main condenser. Safety Function: 2 CFR: 41.2 to 41.9/45.7 to 45.8 PRA: No Level: Memory Pedigree: New History: N/A Explanation: A. Incorrect - (1) The first part of the answer is correct. (2) Per DOP 1200-03, Opening MO 2(3)-1201-11, BLOWDN TO CONDR, AND MO 2(3)-1201-12, BLOWDN TO RW VLV, simultaneously will provide an open pathway between Radwaste and the Main Condenser. This will cause a drop in vacuum, if established, in the Main Condenser. Plausible because (1) The first part of the answer is correct. (2) Per DOP 1200-03, at low power conditions adjusting RWCU System flow could cause a pressure spike and system isolation. Opening both valves would reasonably be expected to cause an increase in flow, but the unit is operating at high power. B. Correct. (1) Per DOP 1200-03, establishing blowdown to the main condenser at power will cause indicated core power to increase. (2) Per DOP 1200-03, Opening MO 2(3)-1201-11, BLOWDN TO CONDR, AND MO 2(3)-1201-12, BLOWDN TO RW VLV, simultaneously will provide an open pathway between Radwaste and the Main Condenser. This will cause a drop in vacuum, if established, in the Main Condenser. C. Incorrect - (1) Per DOP 1200-03, establishing blowdown to the main condenser at power will cause indicated core power to increase. (2) Per DOP 1900-03, Opening MO 2(3)-1201-11, BLOWDN TO CONDR, AND MO 2(3)-1201-12, BLOWDN TO RW VLV, simultaneously will provide an open pathway between Radwaste and the Main Condenser. This will cause a drop in vacuum, if established, in the Main Condenser. Plausible because (1) When performing blowdown to the main condenser it invalidates the heat balance assumptions, and causes calculated core thermal power to increase during RWCU blowdown with no change in actual reactor power. The student must understand the impact of establishing blowdown on the heat balance equation OR must be familiar enough with the procedural guidance to know that indicated Core Thermal Power is expected to go up. (2) Per DOP 1200-03, at low power conditions adjusting RWCU System flow could cause pressure spikes and system isolations. Opening both valves would reasonably be expected to cause an increase in flow, but the unit is operating at high power. D. Incorrect - (1) Per DOP 1200-03, establishing blowdown to the main condenser at power will cause indicated core power to increase. (2) The second part of the answer is correct. Plausible because (1) When performing blowdown to the main condenser it invalidates the heat balance assumptions, and causes calculated core thermal power to increase during RWCU blowdown with no change in actual reactor power.. The student must understand the impact of establishing blowdown on the heat balance equation, OR must be familiar enough with the procedural guidance to know that indicated Core Thermal Power is expected to go up. (2) The second part of the answer is correct. **Required Reference: None**

45 **ID: 7564** **Points: 1.00**

With the reactor at 100% power, the gland seal steam FEED valve fails closed.

What action must be taken in order to maintain adequate gland seal steam system operation?

A. No action required.

B. The seal steam unloading valves must be CLOSED.

C. The seal steam bypass feed valve must be throttled OPEN.

D. The seal steam bypass feed valve must be throttled CLOSED.

Answer: A

Answer Explanation

At 100% power, the sealing steam requirements are met by the steam leaving the high pressure turbine, i.e., system is self sealing - doesn't require supply from the main steam system through the steam seal feed valves (normal or bypass). Since the valve is already closed for the given conditions, there is no affect on the plant.

Question 45 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	2
Difficulty:	3.00
System ID:	7564
User-Defined ID:	7564
Cross Reference Number:	O
Topic:	45 - 245000 K5.06
Num Field 1:	0.00
Num Field 2:	0.00
Text Field:	
Comments:	Objective: DRE245LN001.12 Reference: DOP 5600-02 K/A: 245000 K5.06 2.5/2.6 K/A: Knowledge of the operational implications of the following concepts as they apply to MAIN TURBINE GENERATOR AND AUXILIARY SYSTEMS: Turbine shaft sealing. Safety Function: 4 CFR: 41.5/45.3 PRA: No Level: High Pedigree: Bank History: N/A Explanation: A. Correct - At 100% power, the sealing steam requirements are met by the steam leaving the high pressure turbine, i.e., system is self sealing - doesn't require supply from the main steam system through the steam seal feed valves (normal or bypass). Since the valve is already closed for the given conditions, there is no affect on the plant. B. Incorrect - With power at 100% the steam unloading will be open and required to stay open to unload excess turbine steam to prevent overpressurization and lifting of the 2-3027-700 and 2-3016-700 relief valves. Plausible because at low power/pressure the unloading valve would have to be closed to maintain seal pressure. C. Incorrect - With power at 100% the bypass steam feed valve would not be needed to maintain seal pressure. Plausible because at low power/pressure with a failure of the gland seal steam feed vlv the bypass feed vlv would have to be opened to maintain seal pressure. D. Incorrect - With power at 100% the bypass feed valve would not have to be repositioned with loss of the gland seal steam feed valve failed closed. Feed is being provided by the main turbine. Plausible because at low power/pressure the bypass feed vlv is throttled to maintain between 5.5 psig and 10 psig seal pressure. **Required Reference: None**

Question 45 Table-Item Links

General Question Data - Site Ownership

Dresden

46 **ID: 24061** **Points: 1.00**

Unit 2 is operating at full power.

- CRD exercising is in progress on rod L7.
- Rod L7 is moved to position 48.
- RPIS indication for rod L7 indicated '48', then RED '--', then went out.

What is this an indication of and what is (are) the NSO required actions?

A. Rod overtravel; attempt to recouple the rod

B. Rod overtravel; bypass the RWM and drive the rod to 00

C. Failed open RPIS reed switch; enter a substitute position

D. Failed open RPIS reed switch; bypass the RWM and drive the rod to 00

Answer: A

Answer Explanation

DOA 0300-05 actions for uncoupled rod with reactor power greater than 10% and a rod overtravel are to attempt to recouple the rod.

Question 46 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	3.00
System ID:	24061
User-Defined ID:	24061
Cross Reference Number:	
Topic:	46 - 201003.A4.02
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	Objective: 201LN002.08 Reference: DOA 0300-05; DOA 0300-06; DAN 902(3)-5 E-3 K/A: 201003.A4.02 3.5 / 3.5 K/A: Ability to manually operate and/or monitor in the control room: CRD mechanism position. CFR: 41.7/45.5 to 45.8 Safety Function: 1 PRA: No Level: High Pedigree: Bank History: 2012 NRC Explanation: A. Correct, DOA 0300-05 actions for uncoupled rod with reactor power greater than 10% B. Incorrect - DOA 0300-05 actions for uncoupled rod with reactor power greater than 10% is to attempt to recouple rod. Plausible because this is the action for an uncoupled rod if reactor power less than 10% C. Incorrect - Open reed switches would cause RPS indication to be blank. Plausible because current indication is blank and part 2 is correct for loss of RPIS. D. Incorrect - Open reed switches would cause RPS indication to be blank. Plausible because current indication is blank and part 2 is correct for a loss of RPIS and can not be moved to a position that has good RPIS indication. **REQUIRED REFERENCES: None.**

Question 46 Table-Item Links

General Question Data - Site Ownership

Dresden

General Question Data - Ops Program

Reactor Operator

47	**ID: 8116**	**Points: 1.00**

Annunciator 902-5 B-3, ROD WORTH MIN BLOCK, is illuminated.

The Rod Worth Minimizer for control rod F-06 is displaying a rod position of ?? in RED.

This indicates that the rod:

A. has a loss of RPIS.

B. has an alternate limit assigned.

C. has been taken OOS in the RWM.

D. has a substitute position assigned.

Answer: A

Answer Explanation

Per DAN 902(3)-5 B-3 Unknown control rod position will cause a ROD WORTH MIN BLOCK. DOP 0400-02 a control rod with an invalid position will displayed on the RWM TOUCHSCREEN with two question marks displayed in Red.

Question 47 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	2.00
System ID:	8116
User-Defined ID:	8116
Cross Reference Number:	LI
Topic:	47 - 201006 G.2.4.46
Num Field 1:	0.00
Num Field 2:	0.00
Text Field:	
Comments:	Objective: 201LN006-03 Reference: DOP 0400-02: DAN 902(3)-5 B-3 K/A: 201006 G.2.04.46 4.2 / 4.2 K/A: Ability to verify that the alarms are consistent with the plant conditions: Rod Worth Minimizer. Safety Function: 7 CFR: 41.10/43.5/45.3/45.12 PRA: No Level: Memory Pedigree: New History: N/A Explanation: A. Correct - Per DAN 902(3)-5 B-3 Unknown control rod position will cause a ROD WORTH MIN BLOCK. DOP 0400-02 a control rod with an invalid position will displayed on the RWM TOUCHSCREEN with two question marks displayed in Red. B. Incorrect - alternate limits are displayed in white because position is known. Plausible because RWM indications are not common knowledge for each of the 10 RWM Block symptoms. C. Incorrect - rods taken OOS are shown in CYAN. Plausible because RWM indications are not common knowledge for each of the 10 RWM Block symptoms. D. Incorrect - rods with a substitute position assigned will be displayed in YELLOW. Plausible because RWM indications are not common knowledge for each of the 10 RWM Block symptoms. **Required References: None**

Question 47 Table-Item Links

General Question Data - Site Ownership

Dresden

48 **ID: 26928** **Points: 1.00**

Unit 2 is operating at 50% Reactor power.

A disturbance on the grid results in the following indications:

- "ALARM S1 P2 POWER LOAD UNBALANCE OCCURRED" is in ALARM on the DEHC HMI
- 902-7 H-5, POWER LOAD UNBALANCE TRIP, is in ALARM.

Which one of the following describes how the DEHC system **INITIALLY** responds to these conditions?

A. The Bypass Valves OPEN **ONLY**

B. Turbine Control Valves CLOSE slowly to reduce stator amps

C. The Turbine Stop Valves CLOSE and the Bypass Valves OPEN, **ONLY**

D. The Turbine Control Valves **AND** Intercept Valves FAST CLOSE, and the Bypass Valves OPEN

Answer: D

Answer Explanation

The Power Load Unbalance function of DEHC is activated by the electrical load reject condition and will cause the TCVs to Fast Close. The load error and closing error will also cause the Intercept valves to fast close. The resultant rise in pressure will cause the Bypass Valves to open to control reactor pressure.

Question 48 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	4.00
System ID:	26928
User-Defined ID:	26928
Cross Reference Number:	
Topic:	48 - 239001 K4.07
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	Objective: 245LN001.08 References: DAN 902(3)-7 H-5, DAN 902(3)-5 A-13, DOP 5650-15, DOS 5600-15, DOA 5600-01 K/A: 239001 K4.07 3.7/3.7 K/A: Knowledge of MAIN AND REHEAT STEAM SYSTEM design feature(s) and/or interlocks which provide for the following: Over pressure control. Safety Function: 4 CFR: 41.7 PRA: No Level: High Pedigree: Bank History: 15-1 CERT Justification for High Level: The student needs to recognize that the fast closure of the Control valves will occur, and that the load and closing errors will also cause a fast closure of the intercept valves. Additionally, when these close, the rising RPV pressure will cause the Bypass valves to open. Explanations: A. Incorrect - the bypass valves will open but the Turbine Control vlvs and Intercept valves close as well. Plausible because the bypass valves will open but at this power that is not enough capacity. B. Incorrect - the Turbine Control vlvs and Intercept valves FAST close. Plausible because the Control and Intercept valves do close must know that it is a fast closure. C. Incorrect - the Turbine Stop valves do not close initially. Plausible because the bypass valves do open and the Stop vlv will close just not initially. D. Correct - The Power Load Unbalance function of DEHC is activated by the electrical load reject condition and will cause the TCVs to Fast Close. The load error and closing error will also cause the Intercept valves to fast close. The resultant rise in pressure will cause the Bypass Valves to open to control reactor pressure. **Required References: NONE**

None

49	ID: 27581	Points: 1.00

Unit 2 is at rated power.

The feed flow element for 2B reactor feed pump (2B PP FLOW FI 2-640-24B) is **SLOWLY** failing down but is **NOT** bad quality.

Reactor water level would be ___(1)___ and DOA 0600-01, TRANSIENT LEVEL CONTROL, actions must be taken to ___(2)___.

A. (1) lowering
(2) place level control in manual and match steam and feed flow

B. (1) lowering
(2) place level control in Single Element by depressing the 1-ELEM pushbutton

C. (1) rising
(2) place level control in manual and match steam and feed flow

D. (1) rising
(2) place level control in Single Element by depressing the 1-ELEM pushbutton

Answer: D

Answer Explanation

With the 2B reactor feed pump flow failing down, the bailey system will not see bad quality until it is downscale. This will cause a mismatch in steam and feed flow causing an increase in feed and rising water level. Per DOA 0600-01 the corrective action is to place the feed system in Single Element control. This would happen automatically if feed flow was bad quality.

Question 49 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	4
Difficulty:	4.00
System ID:	27581
User-Defined ID:	27481
Cross Reference Number:	
Topic:	49 - 259002 A2.02
Num Field 1:	
Num Field 2:	
Text Field:	

Comments:	Objective: DRE259LN002.08 Reference: DOA 0600-01, DAN 902(3)-5 G-8 K/A: 259002 A2.02 3.3/3.4 K/A: Ability to predict the impacts of the following on the REACTOR WATER LEVEL CONTROL SYSTEM and based on those predictions use procedure to correct, control, or mitigate the consequences of those abnormal conditions or operations: Loss of any number of reactor feedwater flow inputs. Safety Function: 2 CFR: 41.5/45.7 PRA: No Level: High Pedigree: New History: N/A Explanations: A. Incorrect - The bailey system would see a decrease in feed flow and open the FRV causing level to rise. The system has not failed and manual control is not required and with inaccurate feed flow matching would not be the corrective action. Plausible because with indicated feed flow decreasing it could be seen as reactor water level lowering. Part two is plausible because this would be the corrective action if the reactor water level control system was failing. B. Incorrect - The bailey system would see a decrease in feed flow and open the FRV causing level to rise. Plausible because with indicated feed flow decreasing it could be seen as reactor water level lowering. Part two is correct. C. Incorrect - The first part is correct. The system has not failed and manual control is not required and with inaccurate feed flow matching would not be the corrective action. Plausible because part 1 is correct. Part two is plausible because this would be the corrective action if the reactor water level control system was failing. D. Correct - With the 2B reactor feed pump flow failing down, the bailey system will not see bad quality until it is downscale. This will cause a mismatch in steam and feed flow causing an increase in feed and rising water level. Per DOA 0600-01 the corrective action is to place the feed system in Single Element control. This would happen automatically if feed flow was bad quality. **Required References: NONE**

None

50 **ID: 27018** **Points: 1.00**

Unit 2 is operating at rated power with the ESS UPS inverter out of service, when Bus 25 spuriously tripped.

When power is restored to Bus 25, the 120 VAC ESS Bus is powered from___(1)___and___(2)___ back to the static switch.

A. (1) Bus 29
(2) will AUTOMATICALLY transfer

B. (1) Bus 29
(2) must be MANUALLY transferred

C. (1) MCC 28-2
(2) will AUTOMATICALLY transfer

D. (1) MCC 28-2
(2) must be MANUALLY transferred

Answer: D

Answer Explanation

Power to the ESS Bus through the static switch has been interrupted. This will cause the POWER seeking ABT to transfer ESS Bus to Reserve power provided by MCC 28-2. The ABT must be manually transferred back to the static switch.

Question 50 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	4.00
System ID:	27018
User-Defined ID:	27018
Cross Reference Number:	
Topic:	50 - 262002.G2.1.29
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	Objective: DRE262LN005.06 Reference: DOP 6800-01 K/A: 262002 G.2.1.29 4.1 / 4.0 K/A: Knowledge of how to conduct system lineups, such as valves, breakers switches, etc.: Uninterruptable Power Supply. Safety Function: 6 CFR: 41.7 PRA: No Level: High Pedigree: Bank History: NRC 15-1 Explanation: A. Incorrect - Bus 29 normally provides power to the Unit 2 ESS Bus but with the ESS inverter OOS Bus 29 will not provide power. Plausible because (1) power supplies to ESS Bus and arrangement in circuit is commonly confused. (2) Other ABTs on site are automatic. B. Incorrect - Bus 29 normally provides power to the Unit 2 ESS Bus but with the ESS inverter OOS Bus 29 will not provide power. Plausible because (1) power supplies to ESS Bus and arrangement in circuit is commonly confused. (2) Part 2 of answer is correct. C. Incorrect - Power to the ESS Bus through the static switch has been interrupted. This will cause the POWER seeking ABT to transfer ESS Bus to Reserve power provided by MCC 28-2. In this case it must be manually transferred back. Plausible because (1) part one is correct, and (2) it must known that this is a power seeking ABT. Other ABTs on site are automatic. D. Correct - Power to the ESS Bus through the static switch has been interrupted. This will cause the POWER seeking ABT to transfer ESS Bus to Reserve power provided by MCC 28-2. The ABT must be manually transferred back to the static switch. **Required References: None.**

None

51	ID: 27383	Points: 1.00

Unit 2 was operating at rated power when LT 2-0263-58A, U2 RX LOW WTR SCRAM & LOW LOW WTR ISOL, failed downscale.

What is the expected response?

A. MR A RPV Level Instrument fails downscale **ONLY**.

B. PCIS Group I, II, and III isolation annunciators will alarm **ONLY** (no isolation).

C. PCIS Group I, II, and III isolation annunciators will alarm **AND** PCIS Group I, II, and III isolations will occur.

D. MR A RPV Level Instrument fails downscale **AND** PCIS Group I, II, and III isolation annunciators will alarm (no isolation).

Answer: D

Answer Explanation

Any sensor reaching setpoint will result in alarm, but trip logic is one out of two twice. LT 2-0263-58A is the input to A MR level instrument. One sensor failing downscale will not impair the ability of the system to cause protective actions.

Question 51 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	4.00
System ID:	27383
User-Defined ID:	27383
Cross Reference Number:	
Topic:	51 - 223002.K6.04
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	Objective: 223LN005.06 Reference: DAN 902(3)-5 B-10, B-13, E-5, D-4, D-5;12E-6822; DEOP 0010-00; DIS 0500-02 K/A: 223002.K6.04 3.3 / 3.5 K/A: Knowledge of the effect that a loss or malfunction of the following will have on the PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF: Nuclear boiler instrumentation. CFR: 41.7/45.7 Safety Function: 5 PRA: No Pedigree: Bank History: 16-1 NRC Level: High Explanation: A. Incorrect - This condition will result however it is not the only plant response. Plausible because this will occur. B. Incorrect - This condition will result however it is not the only plant response. Plausible because this will occur. C. Incorrect - No actuations will occur. Additional failures must exist. Plausible because the annunciators will occur, must determine trip logic is one out of two twice. D. Correct - Any sensor reaching setpoint will result in alarm, but trip logic is one out of two twice. LT 2-0263-58A is the input to A MR level instrument. One sensor failing downscale will not impair the ability of the system to cause protective actions. **REQUIRED REFERENCES: 12E-6822**

None

52 **ID: 27607** **Points: 1.00**

Per EOP-DEOP TB, Dresden Nuclear Power Station DEOP Technical Bases, operators must scram the reactor BEFORE Torus bulk temperature reaches ___(1)___ F.

This action is REQUIRED to ensure that ____(2)____.

A. (1) 95
(2) sufficient net positive suction head is maintained for LPCI pumps for Torus Cooling

B. (1) 95
(2) Boron Injection Initiation Temperature (BIIT) is **NOT** exceeded before the reactor is shutdown

C. (1) 110
(2) sufficient net positive suction head is maintained for LPCI pumps for Torus Cooling

D. (1) 110
(2) Boron Injection Initiation Temperature (BIIT) is **NOT** exceeded before the reactor is shutdown

Answer: D

Answer Explanation

EOP-DEOP TB, Dresden Nuclear Power Station DEOP Technical Bases, states that, while in DEOP 0200-01, PRIMARY CONTAINMENT, a manual reactor scram is performed and DEOP 100 is entered, if not already in use, “before” torus temperature reaches 110°F. Per the PSTG-APPENDIX A, DERIVATION OF THE DRESDEN 2/3 PLANT SPECIFIC TECHNICAL GUIDELINES, This ensures that RPV Control will be entered “before” torus temperature reaches the Boron Injection Initiation Temperature (110 F), and is consistent with the Technical Specification scram requirement. An explicit direction to “Scram” the reactor has been added immediately before the transfer to RPV Control.

Question 52 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	3.00
System ID:	27607
User-Defined ID:	27607
Cross Reference Number:	
Topic:	52 - 295026 K3.05
Num Field 1:	
Num Field 2:	
Text Field:	

Comments:	Objective: 29502LK011 Reference: EOP-DEOP TB, PSTG APPENDIX A K/A: 295026.K3.05 3.9 / 4.5 K/A: Knowledge of the reasons for the following responses as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE: Reactor Scram. Safety Function: 3 CFR: 41.5/45.6 PRA: No Level: Memory Pedigree: New History: N/A Explanation: A. INCORRECT - (1) EOP-DEOP TB, Dresden Nuclear Power Station DEOP Technical Bases, states that, while in DEOP 0200-01, PRIMARY CONTAINMENT, a manual reactor scram is performed and DEOP 100 is entered, if not already in use, "before" torus temperature reaches 110°F. (2) Per the PSTG-APPENDIX A, DERIVATION OF THE DRESDEN 2/3 PLANT SPECIFIC TECHNICAL GUIDELINES, This ensures that RPV Control will be entered "before" torus temperature reaches the Boron Injection Initiation Temperature (110 F), and is consistent with the Technical Specification scram requirement. An explicit direction to "Scram" the reactor has been added immediately before the transfer to RPV Control. Plausibility - (1) 95 F the action level where torus cooling must be initiated in DEOP 0200-01. (2) NPSH for the LPCI pumps is directly related to temperature of the torus water. Additionally, DEOP 0200-01 directs starting torus cooling at 95.F, so maintaining NPSH would be important for the LPCI pumps. B. INCORRECT - (1) EOP-DEOP TB, Dresden Nuclear Power Station DEOP Technical Bases¸ states that, while in DEOP 0200-01, PRIMARY CONTAINMENT, a manual reactor scram is performed and DEOP 100 is entered, if not already in use, "before" torus temperature reaches 110°F. (2) This portion of the answer is correct. Plausibility - (1) 95 F the action level where torus cooling must be initiated in DEOP 0200-01. . (2) This portion of the answer is correct. C. INCORRECT - (1) The first part of the answer is correct. (2) Per the PSTG-APPENDIX A, DERIVATION OF THE DRESDEN 2/3 PLANT SPECIFIC TECHNICAL GUIDELINES, This ensures that RPV Control will be entered "before" torus temperature reaches the Boron Injection Initiation Temperature (110 F), and is consistent with the Technical Specification scram requirement. An explicit direction to "Scram" the reactor has been added immediately before the transfer to RPV Control. Plausible because (1) The first part of the answer is correct (2) NPSH for the LPCI pumps is directly related to temperature of the torus water. Additionally, DEOP 0200-01 directs starting torus cooling at 95.F, so maintaining NPSH would be important for the LPCI pumps. D. CORRECT - (1) EOP-DEOP TB, Dresden Nuclear Power Station DEOP Technical Bases, states that, while in DEOP 0200-01, PRIMARY CONTAINMENT, a manual reactor scram is performed and DEOP 100 is entered, if not already in use, "before" torus temperature reaches 110°F. (2) Per the PSTG-APPENDIX A, DERIVATION OF THE DRESDEN 2/3 PLANT SPECIFIC TECHNICAL GUIDELINES, scramming at 110 F ensures that RPV Control will be entered "before" torus temperature reaches the Boron Injection Initiation Temperature (110 F), and is consistent with the Technical Specification scram requirement. An explicit direction to "Scram" the reactor has been added immediately before the transfer to RPV Control. **Required Reference: None**

53 **ID: 27582** **Points: 1.00**

U2 is in a refuel outage with the gates installed.

Level in the fuel pool starts to lower and the cause cannot be determined.

As radiation levels rise, exceeding the setpoint for ___(1)___ will cause SBGT to auto start.

Per DFP 0850-01, SLOW OR RAPID WATER LEVEL LOSS IN FUEL POOL-REACTOR CAVITY, ___(2)___is the **MOST PREFERRED** source for adding water to the U2 fuel pool.

A. (1) REFUEL FLOOR RAD HI
(2) Contaminated Condensate per DOP 1900-03, REACTOR CAVITY, DRYER-SEPARATOR STORAGE PIT AND FUEL POOL LEVEL CONTROL.

B. (1) REFUEL FLOOR RAD HI
(2) Clean Demin Water using Clean Demin Pumps per DOP 1900-03, REACTOR CAVITY, DRYER-SEPARATOR STORAGE PIT AND FUEL POOL LEVEL CONTROL.

C. (1) RX BLDG FUEL POOL A/B RAD HI
(2) Contaminated Condensate per DOP 1900-03, REACTOR CAVITY, DRYER-SEPARATOR STORAGE PIT AND FUEL POOL LEVEL CONTROL.

D. (1) RX BLDG FUEL POOL A/B RAD HI
(2) Clean Demin Water using Clean Demin Pumps per DOP 1900-03, REACTOR CAVITY, DRYER-SEPARATOR STORAGE PIT AND FUEL POOL LEVEL CONTROL.

Answer: C

Answer Explanation

As fuel pool level continues to drop, radiation levels will increase. When Fuel Pool radiation levels reach 45 mr Rx Building ventilation will trip and isolate and SBGT will auto start. The preferred method to add water to the fuel pool per DFP 0850-01 is Contaminated Condensate following DOP 1900-03.

Question 53 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	5
Difficulty:	3.00
System ID:	27582
User-Defined ID:	27582
Cross Reference Number:	
Topic:	53 - 272000 A2.09
Num Field 1:	
Num Field 2:	
Text Field:	

Comments:	Objective: DRE272LN002.06, DRE233LN001.03 References: DFP 0850-01; DAN 902(3)-3 A-3, B-1, C-16, E-16 K/A: 272000 A2.09 3.1/3.3 K/A: Ability to predict the impacts of the following on the RADIATION MONITORING SYSTEM; and based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Low fuel pool level. Safety Function: 7 CFR: 41.5/45.6 PRA: No Level: Memory Pedigree: New History: N/A Justification for Memory level: DRE233LN001.03j - Discuss the operation of the following components under normal operating conditions including support required from other plant systems and local indications, operations or override capability: Skimmer Surge Tanks Explanations: A. Incorrect - Refuel floor high rad has a setpoint of 100mr versus fuel pool rad hi of 45 mr. Refuel floor high rad does not cause a SBGT auto start. Plausible because refuel floor and fuel pool radiation detectors are easily confused for setpoints and actuations. Part 2 is correct. B. Incorrect - Refuel floor high rad has a setpoint of 100mr versus fuel pool rad hi of 45 mr. Refuel floor high rad does not cause a SBGT auto start. Plausible because refuel floor and fuel pool radiation detectors are easily confused for setpoints and actuations. Part 2 is plausible because it is the second preferred make up source. C. Correct - As fuel pool level continues to drop, radiation levels will increase. When Fuel Pool radiation levels reach 45 mr Rx Building ventilation will trip and isolate and SBGT will auto start. The preferred method to add water to the fuel pool per DFP 0850-01 is Contaminated Condensate following DOP 1900-03. D. Incorrect - The first half is correct but clean demin is the second choice of M/U per DFP 0850-01. Plausible because part 1 is correct and part 2 is the second preferred make up source. **Required References: NONE**

None

54 **ID: 7404** **Points: 1.00**

Unit 2 is operating at 100% power with the following plant conditions:

- H2/FW ratio set controller setpoint adjusted to achieve 20 scfm hydrogen flow.

Then a leak develops in the piping of the H2 Addition System downstream of the H2 Flow Control Valve, resulting in the following conditions:

- H2 flow makes a step change to 94 scfm and stabilizes.
- The 'AREA H2 CONC HI' annunciator alarms.
- Hydrogen pressure remains unchanged.

ONE MINUTE LATER:

Which one of the following statements describes the status of the system response to the leak?

A. The H2 flow control valve has closed

B. The H2 excess flow check valves have closed

C. The excess oxygen setpoint has switched to 10%

D. The H2 **AND** O2 solenoid operating valves have closed

Answer: A

Answer Explanation

Receipt of the AREA H2 CONC HI alarm trips the hydrogen addition system, thereby closing the H2 FCV - correct. The O2 flow control valve does not close (Note that the solenoids will close on low flow after the FCV closes BUT, the question asks for the initial response.)

Question 54 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	4.00
System ID:	7404
User-Defined ID:	7404
Cross Reference Number:	LIH
Topic:	54 - 271000 A1.13
Num Field 1:	0.00
Num Field 2:	0.00
Text Field:	
Comments:	Objective: DRE281LN001.06 Reference: DAN 902-65 A-5, DOP 3390-01, DOP 3900-04 K/A: 271000 A1.13 3.2/3.7 K/A: Ability to predict and/or monitor changes in parameters associated with operation the OFFGAS SYSTEM controls including: Hydrogen gas concentration. Safety Function: 9 CFR: 41.5/45.5 PRA: No Level: High Pedigree: Bank History: N/A Explanation: A. Correct - Receipt of the AREA H2 CONC HI alarm trips the hydrogen addition system, thereby closing the H2 FCV. (Note that the solenoids will close on low flow after the FCV closes BUT, the question asks for the initial response.) B. Incorrect - The value stated in the stem, 94 scfm, is below the setpoint of the excess flow check valve. Plausible because excess flow check valves will close on high flow but per DOP 3390-01 the setpoint is not exceeded (600scfm). C. Incorrect - The setpoint can be set down to 10% manually; the only auto switching is UP from 10% to 21% IF there had been a step change in the H2 concentration. Plausible because excess oxygen setpoint can be manually changed to 10%. D. Incorrect - The O2 and H2 solenoid operated isolation valves close, however the O2 isolation valve does not close for 5 minutes. Therefore, one minute into the event, the O2 solenoid isolation solenoids are still open. Plausible because both isolation valves close, however the O2 isolation valve does not close until 5 minutes have passed. **Required Reference: NONE**

Question 54 Table-Item Links

General Question Data - Site Ownership

Dresden

55 **ID: 27608** **Points: 1.00**

Unit 2 is operating at rated power.

Bearing lube oil pressure on the 2A RFP is 34 psig and trending down.

If pressure continues to drop, the Aux Oil Pump would normally start at ___(1)___.

If the Aux Oil pump FAILS TO START, and pressure continues to drop, the 2A RFP will ___(2)___.

A. (1) 15 psig
(2) trip on low bearing oil pressure

B. (1) 15 psig
(2) continue to run until destruction

C. (1) 20 psig
(2) trip on low bearing oil pressure

D. (1) 20 psig
(2) continue to run until destruction

Answer: C

Answer Explanation

Per DAN 902(3)-6 H-7, 2A RFP BEARING OIL PRESSURE LOW, the Aux Oil pump starts at a RFP bearing oil pressure of 20 psig. Per DAN 902(3)-6 F-7 RFP TRIP if bearing oil pressure drops to 15 psig at trip of the pump will occur.

Question 55 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	3.00
System ID:	27608
User-Defined ID:	27608
Cross Reference Number:	
Topic:	55 - 259001 K6.09
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	Objective: DRE 259LN001.06 References: DANs 902(3)-6 H-7, G-7, F-7; DOP 3200-02 K/A: 259001.K6.09 2.8/2.9 K/A: Knowledge of the effect that loss or malfunction of the following will have on the REACTOR FEEDWATER SYSTEM: Reactor feedwater pump lube oil system. Safety Function: 2 CFR: 41.7/45.7 PRA: No Level: Memory Pedigree: New History: N/A Explanations: A. Incorrect - (1) This is the pressure at which the RFP trips on low bearing oil pressure. (2) The second part of the answer is correct. Plausible because (1) The student must recognize the aux oil pump and RFP setpoints. (2) The second part of the answer is correct. . B. Incorrect – (1) This is the pressure at which the RFP trips on low bearing oil pressure. (2) The RFP would trip on low bearing lube oil pressure. Plausible because (1) The student must recognize the aux oil pump and RFP setpoints. (2) Other pumps, such as the condensate booster pumps, do NOT have a low pressure bearing oil trip. C. Correct - (1) Per DAN 902(3)-6 H-7, 2A RFP BEARING OIL PRESSURE LOW, the Aux Oil pump starts at a RFP bearing oil pressure of 20 psig. (2) Per DAN 902(3)-6 F-7 RFP TRIP if bearing oil pressure drops to 15 psig at trip of the pump will occur. D. Incorrect – (1) The first part of the answer is correct. (2) The RFP would trip on low bearing lube oil pressure. Plausible because (1) The student must recognize the aux oil pump and RFP setpoints. (2) Other pumps, such as the condensate booster pumps, do NOT have a low pressure bearing oil trip. **Required References: NONE**

None

56 **ID: 27585** **Points: 1.00**

A Scram has occurred on Unit 3.

All 4 LPCI pumps are being used for injection.

Two minutes later:

- Annunciator 923-5 G-6 U3 LPCI/CS PP AREA TEMP HI, alarms.
- East corner room temp is 135 degrees and rising indicated on the 2253-24 panel.
- The East corner room cooler will not start.

Which LPCI pumps are currently injecting on U3, if any?

A. No pumps

B. 3A and 3B ONLY

C. 3C and 3D ONLY

D. All 4 pumps

Answer: D

Answer Explanation

All 4 pumps that were injecting into the vessel will continue to inject. There is not a high temperature trip on the LPCI pumps needed for injection. The action is to request and engineering evaluation.

Question 56 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	4
Difficulty:	3.00
System ID:	27585
User-Defined ID:	27585
Cross Reference Number:	
Topic:	56 - 203000 K6.08
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	Objective: 203LN001.08 Reference: DAN 923-5 G-6, DAN 902(3)-3 B-6 K/A: 203000 K6.08 2.9/3.1 K/A: Knowledge of the effect that a loss or malfunction of the following will have on the RHR/LPCI: INJECTIONS MODE: ECCS room cooling. Safety Function: 2 CFR: 41.7/45.7 PRA: Yes Level: Memory Pedigree: New History: N/A Explanations: A. Incorrect - All pumps that were running would not trip. Plausible because the alarm for LPCI/CS pp area temp hi is a shared alarm and may have a shared trip. B. Incorrect - 3A and 3B are running but so are 3C and 3D. Plausible because these two pumps are located in the east corner room but will not trip on high temperature. The action is to request an engineering evaluation. C. Incorrect - 3C and 3D are running but so are 3A and 3B. Plausible because it must be determined which pumps are located in the east corner room and may be affected. D. Correct - All 4 pumps that were injecting into the vessel will continue to inject. There is not a high temperature trip on the LPCI pumps needed for injection. The action is to request an engineering evaluation. **Required references: None**

None

57	ID: 13842	Points: 1.00

Unit 2 was operating at near rated power when an Instrument Air system leak occurred. An NSO reported the following timeline:

- 04:44 Inst Air Header pressure is 91 psig.
- 04:46 Inst Air Header pressure is 89 psig.
- 04:49 Inst Air Header pressure is 86 psig.

Without operator action, at time 04:59, what automatic action(s) will have occurred?

A. Both U2 Inst Air dryer bypasses AUTOMATICALLY opened and remained open.

B. Both U2 Inst Air dryer bypasses AUTOMATICALLY opened, then re-closed when header pressure is restored.

C. The U2 Service Air to Inst Air crosstie valve AUTOMATICALLY opened and will have to be MANUALLY re-closed when header pressure is restored.

D. The U2 Service Air to Inst Air crosstie valve AUTOMATICALLY opened, then will AUTOMATICALLY re-close when header pressure is restored.

Answer: C

Answer Explanation

With Instrument Air header pressure dropping at a rate of 1.0 psig/minute, at 04:59 (15 minutes from transient) the header pressure will be 76 psig (started at 91 psig). The AO backup from the Unit 2 Service Air System opens at 85 psig and dropping and will remain open until MANUALLY reset (not automatic).

Question 57 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	3.00
System ID:	13842
User-Defined ID:	13842
Cross Reference Number:	
Topic:	57 - 300000.K4.01
Num Field 1:	0.00
Num Field 2:	0.00
Text Field:	
Comments:	Objective: 278LN001.06 Reference: DOA 4700-01; DAN 923-1 E-4, E-6, F-4 K/A: 300000.K4.01 2.8/2.9 K/A: Knowledge of (INSTRUMENT AIR SYSTEM) design feature(s) and/or interlocks which provide for the following: Manual/automatic transfer of control. CFR: 41.7 Safety Function: 8 PRA: No Level: Memory Pedigree: Bank History: 2015 Cert Explanation: A. Incorrect - The dryer bypasses will not open until header pressure reaches 60 psig. Plausible because the dryer bypass valves will open AUTOMATICALLY when the setpoint is reached. Must determine rate of trend and setpoint. B. Incorrect- The dryer bypasses will not open until header pressure reaches 60 psig. Plausible because the dryer bypass valves will open AUTOMATICALLY when the setpoint is reached. Must determine rate of trend and setpoint. Pressure is above reset setpoint but must be done MANUALLY. C. Correct - With Instrument Air header pressure dropping at a rate of 1.0 psig/minute, at 04:59 (15 minutes from transient) the header pressure will be 76 psig (started at 91 psig). The AO backup from the Unit 2 Service Air System opens at 85 psig and dropping and will remain open until MANUALLY reset (not automatic). D. Incorrect - With Instrument Air header pressure dropping at a rate of 1.0 psig/minute, at 04:59 (15 minutes from transient) the header pressure will be 76 psig (started at 91 psig). The AO backup from the Unit 2 Service Air System opens at 85 psig and dropping. It does not close AUTOMATICALLY. Plausible because part 1 is correct. Part 2 is plausible because we have a number of systems that auto reset when setpoints are met. **REQUIRED REFERENCES: None.**

Question 57 Table-Item Links

General Question Data - Site Ownership

Dresden

General Question Data - Ops Program

Reactor Operator

58 **ID: 27126** **Points: 1.00**

Unit 3 is in Mode 5 with fuel moves in progress.

SRM counts rise steadily over a 5 minute period in the quadrant containing the fuel moves.

How are fuel moves affected?

A. Fuel moves may continue. No limitation on grapple operation.

B. Fuel moves may continue. The grapple may be lowered, but **NOT** raised.

C. Fuel moves may continue. The grapple may be raised, but **NOT** lowered.

D. Stop **ALL** fuel moves. Do **NOT** attempt to raise or lower the grapple.

Answer: D

Answer Explanation

Per DOA 0800-03, INADVERTANT CRITICALITY DURING FUEL MOVES, conditions in the stem should be interpreted as criticality. Immediate actions require operators to suspend fuel moves and NEITHER raise nor lower the grapple.

Question 58 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	4.00
System ID:	27126
User-Defined ID:	27126
Cross Reference Number:	
Topic:	58 - 295023.G2.4.31
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	Objective: DRE272LN002.06 Reference: DOA 0800-03, DFP 0800-01 K/A: 295023 G.2.4.31 4.2/4.1 K/A: Knowledge of annunciator alarms, indications, or response procedures: Refueling Accidents. Safety Function: 8 CFR: 41.10/45.3 PRA: Yes Level: Memory Pedigree: Bank History: ILT 14-1 NRC Explanation: A. Incorrect - Per DOA 0800-03, criticality has occurred and fuel moves must be suspended. Plausible because in cases where SRM's spike during moves then return to normal levels, there would be no limitation on fuel grapple operation. B. Incorrect - Per DOA 0800-03, criticality has occurred and fuel moves must be suspended. Plausible because candidate must recognize an criticality and stop all moves. C. Incorrect - Per DOA 0800-03, criticality has occurred and fuel moves must be suspended. Plausible because candidate must recognize ant criticality and stop all moves. D. Correct - Per DOA 0800-03, conditions in the stem should be interpreted as criticality. Immediate actions of DOA 0800-03 require operators to suspend fuel moves and NEITHER raise nor lower the grapple. **Required references: None**

None

59 **ID: 27586** **Points: 1.00**

Unit 2 is at rated power.

RBCCW is in the **PREFERRED** system lineup.

On a trip of a RBCCW pump, the standby pump is started to prevent....

A. pump runout of single pump.

B. damage to Recirc pump seals and bearings due to loss of cooling.

C. RWCU system isolation on high temperature out of RWCU Non-Regen Hx.

D. an un-isolated leakage path from the Primary containment to the Reactor Building.

Answer: A

Answer Explanation

Per DOP 3700-02 limitations and actions, operating one (1) RBCCW pump AND two (2) heat exchangers can cause pump run out. The preferred system lineup consists of two (2) RBCCW pumps AND two (2) RBCCW heat exchangers.

Question 59 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	3.00
System ID:	27586
User-Defined ID:	27586
Cross Reference Number:	
Topic:	59 - 295018 AK3.04
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	Objective: DRE208LN001.12 References: DOA 3700-01, DOP 3700-02 K/A: 295018 AK3.04 3.3/3.3 K/A: Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER: Starting standby pump. Safety Function: 8 CFR: 41.5/45.6 PRA: No Level: Memory Pedigree: New History: N/A Explanations: A. Correct - Per DOP 3700-02 limitations and actions, operating one (1) RBCCW pump <u>AND</u> two (2) heat exchangers can cause pump run out. The preferred system lineup consists of two (2) RBCCW pumps <u>AND</u> two (2) RBCCW heat exchangers. B. Incorrect - there is not a complete loss of RBCCW. Plausible because this would be correct with a completed loss of RBCCW. Must recognize that the preferred lineup system lineup consists of two (2) RBCCW pumps <u>AND</u> two (2) RBCCW heat exchangers. C. Incorrect - the RWCU Non Regen Hx still has cooling flow with 1 RBCCW pump running. Plausible because this would be correct with a completed loss of RBCCW. Must recognize that the preferred lineup system lineup consists of two (2) RBCCW pumps <u>AND</u> two (2) RBCCW heat exchangers. D. Incorrect - DW pressure is not greater than 2 psig. Plausible because a loss of all RBCCW pumps with a leak in the DW could result in an un-isolated leakage path from the Primary containment to the Reactor Building via the RBCCW head tank vent. Must recognize that the preferred lineup system lineup consists of two (2) RBCCW pumps <u>AND</u> two (2) RBCCW heat exchangers. **Required references: None**

None

60	**ID: 23827**	**Points: 1.00**

Unit 2 was operating at near rated power when the Turbine tripped.

- The Generator has NOT tripped.
- The Generator Field Breaker is Open.

The team is required to open Generator OCBs ___(1)___ from the ___(2)___ panel.

A. (1) After 90 seconds:
(2) 902-8

B. (1) IMMEDIATELY;
(2) 923-2

C. (1) IMMEDIATELY:
(2) 902-8

D. (1) After 90 seconds:
(2) 923-2

Answer: B

Answer Explanation

IF Generator fails to trip AND Field Breaker opens, THEN IMMEDIATELY open the Generator OCBs from Panel 923-2.

Question 60 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	3.00
System ID:	23827
User-Defined ID:	23827
Cross Reference Number:	
Topic:	60 - 295005 G.2.1.20
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	Objective: 24501LK017 Reference: DOA 5600-01, DOP 6400-13, DGP 02-01 K/A: 295005.G.2.1.20 4.6 / 4.6 K/A: Main Turbine Generator Trip: Ability to interpret and execute procedure steps. Safety Function: 3 CFR: 41.10/43.5/45.12 PRA: No Level: Memory Pedigree: New History: N/A Explanation: A. Incorrect - (1) with the generator still on line the OCBs must be opened on the 923-2 panel. (2) This action should be taken immediately. Plausible because (1) OCBs 1-2 and 1-7 are operated from the 902-8 panel under normal circumstances, and (2) the operate is directed to wait 90 seconds if the field breaker remains closed. B. Correct - with the generator still on line and the Field Breaker open then the OCBs should be IMMEDIATELY opened from the 923-2 panel per DOA 5600-01 immediate operator action 3. C. Incorrect IF Generator fails to trip AND Field Breaker opens, THEN IMMEDIATELY open the Generator OCBs from Panel 923-2. Plausible (1) because part 1 is correct and (2) OCBs 1-2 and 1-7 are operated from the 902-8 panel under normal circumstances. D. Incorrect - IF Generator fails to trip AND Field Breaker opens, THEN IMMEDIATELY open the Generator OCBs from Panel 923-2. Plausible (1) because if the Field Breaker remains closed part 1 would be correct. (2) Part 2 is correct. **REQUIRED REFERENCES: None.**

Question 60 Table-Item Links

General Question Data - Site Ownership

Dresden

General Question Data - Ops Program

Reactor Operator

61	ID: 7302	Points: 1.00

An ATWS has occurred on Unit 2.

- Unit 2 Instrument Bus has been lost.
- The US has directed injection of SBLC.

The SBLC pumps are started from ____(1)____, and pump discharge pressure and tank level are monitored at ____(2)____ to verify proper operation of the SBLC system.

A. (1) MCR panel 902-5
(2) MCR panel 902-5

B. (1) MCR panel 902-5
(2) Reactor Building 589 elevation locally at the pumps.

C. (1) Reactor Building 589 elevation locally at the pumps.
(2) MCR panel 902-5

D. (1) Reactor Building 589 elevation locally at the pumps.
(2) Reactor Building 589 elevation locally at the pumps.

Answer: B

Answer Explanation

Both pumps still have control power so they can be started from the MCR panel 902-5 panel. This will also fire the squib valves to allow injection. The loss of instrument bus causes a loss of SBLC panel indications in the main control room, so level and discharge pressure must be monitored locally.

Question 61 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	3.00
System ID:	7302
User-Defined ID:	7302
Cross Reference Number:	LI
Topic:	61 - 211000 G 2.1.30
Num Field 1:	0.00
Num Field 2:	0.00
Text Field:	
Comments:	Objective: DRE211LN001.04 Reference: DOP 1100-02, DOA 6800-01, 12E-2460, DOA 6900-T1 K/A: 211000 Generic 2.1.30 4.4/4.0 K/A: Ability to locate and operate components, including local controls: Standby Liquid Control. CFR: 41.7/45.7 Safety Function: 1 PRA: No Level: High Pedigree: New History: N/A Explanations: A. Loss of instrument bus causes loss of 902-5 panel indications. Plausible because part 1 is correct and part 2 would be correct if instrument bus was not lost. B. Correct - Both pumps still have control power so they can be started from the MCR panel 902-5 panel. This will also fire the squib valves to allow injection. The loss of instrument bus causes a loss of SBLC panel indications in the main control room, so level and discharge pressure must be monitored locally. C. SBLC pumps can be started from the main control room. If started locally the squib vlvs will not fire. Plausible because there are control switches at the pumps that are used for starting the pumps for surveillance testing. Part 2 is correct. D. SBLC pumps can be started from the main control room. If started locally the squib vlvs will not fire. Plausible because there are control switches at the pumps that are used for starting the pumps for surveillance testing. Part 2 is plausible because that is the normal location for monitoring, but is not available because those instruments are powered from the instrument bus. **Required Reference: NONE**

Question 61 Table-Item Links

General Question Data - Site Ownership

Dresden

*** Questions Without K&A Links ***

Active Questions without K&A Links

Pseudo Objectives

262LC03-06

62 **ID: 10414** **Points: 1.00**

Given the following:

- Unit 2 is operating at 100% power.
- Generator voltage control is in automatic.
- Due to a Grid disturbance the TSO calls directing the Blue bus voltage to be raised by two (2) kV.

The NSO accomplishes this by placing the___(1)___switch to **RAISE** to pickup an additional___(2)___MVARS.

A. (1) Governor Switch
(2) 50-60

B. (1) Governor Switch
(2) 100-120

C. (1) Voltage Regulator setpoint adjust
(2) 50-60

D. (1) Voltage Regulator setpoint adjust
(2) 100-120

Answer: D

Answer Explanation

The generator is in automatic, the voltage adjust switch would be used. The second part of the question examines what relationship exists between 345 KV bus voltage and MVARs. This is listed in DOP 6400-08 as 50-60 MVAR for every 1 KV. Since the TSO requested a 2 KV change, picking up an additional 100-120 MVARs is appropriate.

Question 62 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	5
Difficulty:	3.00
System ID:	10414
User-Defined ID:	10414
Cross Reference Number:	
Topic:	62 - 700000 AA1.03
Num Field 1:	0.00
Num Field 2:	0.00
Text Field:	
Comments:	Objective: DRE245LN003.08 References: DOP 6400-08 K/A: 700000 AA1.03 3.8/3.7 K/A: Ability to operate and/or monitor the following as they apply to GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES: Voltage regulator controls. Safety Function: 6 CFR: 41.5/41.10/45.5,45.7,45.8 PRA: No Level: Memory Pedigree: Bank History: No Explanations: A. Incorrect - With the Generator voltage control in auto the setpoint adjust switch would be used. Plausible because Governor switch would be correct for manual operation. Part 2 is plausible because 50-60 MVARs would be correct for a 1 KV increase. B. Incorrect - With the Generator voltage control in auto the setpoint adjust switch would be used. Plausible because the governor switch would be used for manual control. Part 2 is correct. C. Incorrect - The stem ask for a 2 KV increase at 50-60 MVARS per KV this is only 12 KV. Plausible because part 1 is correct. Part 2 would be correct for a 1 KV increase. D. Correct - The generator is in automatic, the voltage adjust switch would be used. The second part of the question examines what relationship exists between 345 KV bus voltage and MVARs. This is listed in DOP 6400-08 as 50-60 MVAR for every 1 KV. Since the TSO requested a 2 KV change, picking up an additional 100-120 MVARs is appropriate. **Required Reference: None**

Question 62 Table-Item Links

Licenced Operator Annual Exam Bank

LORT Exam Bank Question

SYSTEMS

General Question Data - Site Ownership

Dresden

63 **ID: 27590** **Points: 1.00**

Regarding the hierarchy between the Dresden Abnormal Operating Procedures (DOA) and the Dresden Emergency Procedures (DEOP), which of the following is correct?

DOAs ___(1)___ contain steps **DIRECTING** entry into a DEOP.

Once the DEOPs have been entered, any subsequent DOA entry conditions ___(2)___ **REQUIRE** entry into the associated DOA.

A. (1) may
(2) will

B. (1) may
(2) will **NOT**

C. (1) will **NEVER**
(2) will

D. (1) will **NEVER**
(2) will **NOT**

Answer: A

Answer Explanation

(1) A number of DOA's do have specific steps directing entry into the appropriate DEOP procedure. These include DOA 0600-01, DOA 0040-02, DOA 0010-04, DOA 0800-03, DOA 1900-01. (2) Although DEOPS have been entered, the actions of DOAs must still be taken, and any entry conditions to a DOA would still require entry and execution of that procedure.

Question 63 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	2
Difficulty:	2.00
System ID:	27590
User-Defined ID:	27590
Cross Reference Number:	
Topic:	63 - Generic 2.4.8
Num Field 1:	
Num Field 2:	
Text Field:	

Comments:	Objective: 29501LK011 Reference: EOP-DEOP TB K/A: Generic 2.4.8 3.8/4.5 K/A: Knowledge of how abnormal operating procedures are used in conjunction with EOPs. Safety Function: N/A CFR: 41.10/43.5/45.13 PRA: N/A Level: Memory Pedigree: New History: None Explanations: A. Correct - (1) A number of DOA's do have specific steps directing entry into the appropriate DEOP procedure. These include DOA 0600-01, DOA 0040-02, DOA 0010-04, DOA 0800-03, DOA 1900-01. (2) Although DEOPS have been entered, the actions of DOAs must still be taken, and any entry conditions to a DOA would still require entry and execution of that procedure. B. Incorrect - (1) The first part of the answer is correct. (2) Although DEOPS have been entered, the actions of DOAs must still be taken, and any entry conditions to a DOA would still require entry and execution of that procedure. Plausibility because (1) The first part of the answer is correct. (2) There are other examples of higher tier procedures that preclude entry into a lower tier procedure. e.g. when the Severe Accident Guidelines (SAGs) are entered from the DEOPs, they do not allow re-entry into the lower tier DEOP procedures. C. Incorrect - (1) Although most DOAs do not, a number of DOA's DO have specific steps directing entry into the appropriate DEOP procedure. These include DOA 0600-01, DOA 0040-02, DOA 0010-04, DOA 0800-03, DOA 1900-01. (2) The second part of the answer is correct. Plausible because (1) most DOA procedures do not contain any steps specifically directing entry into a DEOP. Exceptions to this include DOA 0600-01, DOA 0040-02, DOA 0010-04, DOA 0800-03, DOA 1900-01. (2) The second part of the answer is correct. D. Incorrect - (1) Although most DOAs do not, a number of DOA's DO have specific steps directing entry into the appropriate DEOP procedure. These include DOA 0600-01, DOA 0040-02, DOA 0010-04, DOA 0800-03, DOA 1900-01. (2) Although DEOPS have been entered, the actions of DOAs must still be taken, and any entry conditions to a DOA would still require entry and execution of that procedure. Plausible because (1) most DOA procedures do not contain any steps specifically directing entry into a DEOP. Exceptions to this include DOA 0600-01, DOA 0040-02, DOA 0010-04, DOA 0800-03, DOA 1900-01. (2) There are other examples of higher tier procedures that preclude entry into a lower tier procedure. e.g. when the SAGS are entered from the DEOPs, they do not allow re-entry into the lower tier DEOP procedures. **Required references: None**

None

64	**ID: 27246**	**Points: 1.00**

Unit 2 is operating at rated power with the Unit 2/3 250V DC Battery Charger in service from U2 and Unit 2 125V DC Battery Charger in service.

125 VDC and 250 VDC voltages are dropping.

This is caused by a loss of ______.

A. MCC 28-2

B. MCC 28-3

C. MCC 29-2

D. MCC 39-2

Answer: C

Answer Explanation

Unit 2/3 250V DC charger is powered from MCC 29-2. This is also the power supply to U2 125 VDC battery charger. With a loss of power to the battery charger and battery loads voltage will lower.

Question 64 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	1.00
System ID:	27246
User-Defined ID:	27246
Cross Reference Number:	
Topic:	64 - 295004.A2.01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	Objective: DRE263LN001.02 Reference: DOP 6700-18, DOP 6700-19, DOP 6900-02, DOP 6900-01, DOA 6700-T1 K/A: 295004.A2.01 3.2/3.6 K/A: Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF DC POWER: Cause of partial or complete loss of DC power. CFR: 41.10/43.5/45.13 Safety Function: 6 PRA: Yes Pedigree: New Level: Memory History: N/A Explanation: A. Incorrect - Plausible because MCC 28-2 is the power supply to 2A 125 VDC battery charger and a common misconception of 250 VDC power supply. B. Incorrect - Plausible because MCC 28-3 is the power supply to U2 250 VDC battery charger and 28-3 is a common misconception for 125 VDC power supply. C. Correct - Unit 2/3 250V DC charger is powered from MCC 29-2. This is also the power supply to U2 125 VDC battery charger. With a loss of power to the battery charger and battery loads, voltage will lower. D - Incorrect. Plausible because MCC 39-2 is the backup power supply to U2/3 battery charger but 125V DC system is also being affected. Unlike 250 VDC, 125VDC does not cross between units. **Required References: None.**

None

65 **ID: 13041** **Points: 1.00**

Unit 2 is in SHUTDOWN at 0910, with the following conditions:

- 2A and 2B Recirc Pumps are in operation at minimum speed.
- 2A and 2B Shutdown Cooling (SDC) loops are controlling cool down rate.
- RPV water temperature is currently 305°F.

At 0930, the following annunciators are received:

- 902-4 H-4, 2A RECIRC LOOP WATER TEMP HI
- 902-4 H-8, 2B RECIRC LOOP WATER TEMP HI

At 0945, the NSO reports that the Recirc Loop temperatures indicate 350°F on the Recirc Loop Temperature recorder.

SDC flow will ___(1)___ and will require___(2)___.

A. (1) be lost;
(2) securing both recirc pumps

B. (1) be lost;
(2) increasing RWCU flow to maximize heat removal rate

C. (1) still be established;
(2) securing both recirc pumps

D. (1) still be established;
(2) increasing RWCU flow to maximize heat removal rate

Answer: B

Answer Explanation

With recirc loop temperatures above 345 °F, the SDC pumps will trip, causing loss of flow. Per the stem, we are above 345 °F. Additionally, when RPV temperature reaches a point where the saturation pressure is at approximately 100 psig, the SDC loops will isolate. The Recirc pumps are required to continue running to ensure proper core flow, With the loss of SDC, to continue cooldown (per the DOA), RWCU flow is required to be maximized.

Question 65 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	4
Difficulty:	3.00
System ID:	13041
User-Defined ID:	13041
Cross Reference Number:	
Topic:	65 - 295021.A2.02
Num Field 1:	
Num Field 2:	
Text Field:	

Comments:	Objective: DRE205LN001.12 Reference: DAN 902(3)-4 H-4, H-8; DOA 1000-01 K/A: 295021.A2.02 3.4 / 3.4 K/A: Ability to determine and/or interpret the following as they apply to LOSS OF SHUTDOWN COOLING: RHR/shutdown cooling system flow. Safety Function: 4 CFR: 41.10/43.5/45.13 PRA: No Level: High Pedigree: Bank History: 2007 NRC Explanations: A. Incorrect - SDC flow will be lost due to a pump trip and system isolation. Recirc will continue to provide core flow and prevent stratification. Plausible because Part 1 is correct and part 2 is plausible because recirc pumps add heat. The candidate may believe that securing the recirc pumps will reduce heat. B. Correct - With recirc loop temperatures above 345 °F, the SDC pumps will trip, causing loss of flow. Per the stem, we are above 345 °F. Additionally, when RPV temperature reaches a point where the saturation pressure is at approximately 100 psig, the SDC loops will isolate. The Recirc pumps are required to continue running to ensure proper core flow, With the loss of SDC, to continue cooldown (per the DOA), RWCU flow is required to be maximized. C. Incorrect - SDC pumps trip at 345 degrees and system isolates on Rx pressure of >100psig. This correlates to ~340 degrees F. Plausible because must know the setpoint for pump trips and system isolation versus annunciator setpoint of 330 degrees F. The candidate may believe that securing the recirc pumps will reduce heat. B. Correct - With recirc loop temperatures above 345 °F, the SDC pumps will trip, causing loss of flow. . D. Incorrect - SDC pumps trip at 345 degrees and system isolates on Rx pressure of >100psig. This correlates to ~340 degrees F. Plausible because must know the setpoint for pump trips and system isolation versus annunciator setpoint of 330 degrees F. Part 2 is correct. **REQUIRED REFERENCES: None.**

Question 65 Table-Item Links

General Question Data - Site Ownership

Dresden

General Question Data - Ops Program

Senior Reactor Operator

66	ID: 24131	Points: 1.00

Unit 2 was operating at rated power when, U2 125 VDC power was lost to Division I of the Isolation Condenser initiation and isolation logic.

What indications would the NSO expect to see two (2) minutes after the above transient?

A. An Isolation Condenser initiation has occurred.

B. ALL Isolation Condenser valves have remained as is.

C. ALL Isolation Condenser isolation valves, EXCEPT 2-1301-2, closed.

D. ALL Isolation Condenser isolation valves on the 902-3 panel closed.

Answer: D

Answer Explanation

Isolation Condenser Group 5 isolation circuits are de-energized to actuate. The power supplies are 2A-1 and 2B-1 for U2. Los of either power supply will cause a complete isolation.

Question 66 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	3.00
System ID:	24131
User-Defined ID:	24131
Cross Reference Number:	
Topic:	66 - 207000.A3.05
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	Objective: DRE207LN001.11 Reference: DOA 6900-02 K/A: 207000.A3.05 3.6/3.8 K/A: Ability to monitor automatic operations of the ISOLATION (EMERGENCY) CONDENSER including: System lineup. CFR: 41.7/45.7 Safety Function: 4 PRA: Yes Level: Memory Pedigree: Bank History: 2009 NRC, 2015 Cert Explanations: A. Incorrect - IC will isolate on a loss of either 125VDC 2A-1 or 2B1 not initiate. Plausible because a group 4 is energize to actuate group 5 is de-energize to actuate. B. Incorrect - IC will isolate on a loss of either 125VDC 2A-1 or 2B-1. Plausible because IC is one of only a few systems that will actuate on any loss of power, when the other is still available. C. Incorrect - All IC will be closed on an isolation signal. Plausible because the 2-1301-2 valve is the only normally opened valve that is DC powered. With a loss of 125VDC a misconception that this is the DC power supply to this valve. D. Correct - Isolation Condenser Group 5 isolation circuits are de-energized to actuate. The power supplies are 2A-1 and 2B-1 for U2. Los of either power supply will cause a complete isolation. **REQUIRED REFERENCES: None.**

Question 66 Table-Item Links

General Question Data - Site Ownership

Dresden

General Question Data - Ops Program

Reactor Operator

67	ID: 27593	Points: 1.00

A Scram has occurred on U2.

Torus water level is trending up at the following rate.

0900	15.0 feet
0915	16.0 feet
0930	17.0 feet

If the trend continues, at what time is an EMERGENCY RPV DEPRESSURIZATION **REQUIRED**, and why?

A. 0945
SRV operation with suppression pool water level above the SRV Tail Pipe Level Limit could damage the SRV discharge lines.

B. 1000
SRV operation with suppression pool water level above the SRV Tail Pipe Level Limit could damage the SRV discharge lines.

C. 0945
To prevent submergence of the Torus to Reactor Building vacuum breakers to allow for removal of noncondensibles into the drywell and equalize pressures.

D. 1000
To prevent submergence of the Torus to Reactor Building vacuum breakers to allow for removal of noncondensibles into the drywell and equalize pressures.

Answer: B

Answer Explanation

Based on the current trend torus level will be above 18.5 feet at time 1000. SRV operation with suppression pool water level above the SRV Tail Pipe Level Limit (STPLL) could damage the SRV discharge lines. This, in turn, could lead to containment failure from direct pressurization and damage to equipment inside the containment (ECCS piping, RPV water level instrument runs, wetwell-to-dry well vacuum breakers, etc.) from pipe-whip and jet-impingement loads. The RPV is therefore not permitted to remain at pressure if suppression pool water level and RPV pressure cannot be restored and maintained below the STPLL.

Question 67 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	4
Difficulty:	3.00
System ID:	27593
User-Defined ID:	27593
Cross Reference Number:	
Topic:	67 - 295029 EK3.01
Num Field 1:	
Num Field 2:	
Text Field:	

Comments:	Objective: 29502LK100 References: DEOP 200-1, DEOP 0010-00, EOP-DEOP TB, EPG B-7-56 K/A: 295029 EK3.01 3.5/3.9 K/A; Knowledge of the reasons for the following responses as they apply to HIGH SUPPRESSION POOL WATER LEVEL: Emergency depressurization. Safety Function: 5 CFR: 41.5/45.6 PRA: No Level: High Pedigree: New History: N/A Explanations: A. Incorrect - Per the EPGs and definition section of DEOP 0010, the blowdown is not required until the 18.5 feet limit is exceeded. Plausible because a number of DEOP actions use BEFORE the limits are met. Part 2 is correct. B. Correct - Based on the current trend torus level will be above 18.5 feet at time 1000 SRV operation with suppression pool water level above the SRV Tail Pipe Level Limit (STPLL) could damage the SRV discharge lines. This, in turn, could lead to containment failure from direct pressurization and damage to equipment inside the containment (ECCS piping, RPV water level instrument runs, wetwell-to-dry well vacuum breakers, etc.) from pipe-whip and jet-impingement loads. The RPV is therefore not permitted to remain at pressure if suppression pool water level and RPV pressure cannot be restored and maintained below the STPLL. C. Incorrect - Per the EPGs and definition section of DEOP 0010, the blowdown is not required until the 18.5 feet limit is exceeded. Plausible because a number of DEOP actions use BEFORE the limits are met. Part 2 is to ensure Drywell Sprays can be used and not an issue until 27 feet. D. Incorrect - Part 1 is correct but submergence of vacuum breakers would not be an issue at 18.5 feet. Plausible because Part 1 is correct and Part 2 is to ensure Drywell Sprays can be used and not an issue until 27 feet. **Required References: NONE**

None

68 **ID: 27595** **Points: 1.00**

Unit 2 was operating at near full power when a SCRAM occurred.

- All Rods did NOT go in
- DEOP 0400-05 was entered

Which of the following conditions would allow the **OPERATORS** to make the determination that the Reactor will **REMAIN** shut down under **ALL** conditions?

A. **ALL** IRM's are DECREASING on range 5, with **NO** boron injected.

B. The position of TWO control rods is UNKNOWN. **ALL** other control rods are at Notch Position 00

C. Control rod H-1 remains withdrawn at notch position 06. **ALL** other control rods are at Notch Position 04

D. Control rod H-8 remains withdrawn at notch position 48. **ALL** other control rods are at Notch Position 00

Answer: D

Answer Explanation

In order to ensure that the reactor will stay shutdown under all conditions, only a single rod can be withdrawn, or all rods must be at notch position 04 or less, or a Qualified Nuclear Engineer must determine that the reactor will stay shutdown under all conditions. In this case, a single rod is withdrawn, so this ensures the reactor will remain shutdown under ALL conditions.

Question 68 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	2.00
System ID:	27595
User-Defined ID:	27595
Cross Reference Number:	

Topic:	68 - 295006 AK1.02
Num Field 1:	
Num Field 2:	
Text Field:	

Comments:	Objective: 29501LK020 Reference: DEOP 0100, EOP-DEOP TB K/A: 295006 AK 1.02 3.4/3.7 K/A: Knowledge of the operational implications of the following concepts as they apply to SCRAM: Shutdown margin. Safety Function: 1 CFR: 41.8 to 41.10 PRA: No Level: Memory Pedigree: New History: N/A Explanations: A. Incorrect – Although DEOP 0400-05 has a step allowing continuing to depressurize the reactor once ALL IRM's are decreasing below Range 7, with NO boron injected, this does NOT mean that the reactor will remain shutdown under all conditions. Subsequent addition of positive reactivity due to moderator temperature coefficient, pressure coefficient, xenon decay, etc. may result in a re-criticality. Plausible because decreasing IRMs below range 7 indicate that the reactor is currently shutdown. Additionally, DEOP 0400-05 allows a plant cooldown in an ATWS if IRMs decreasing below IRM range 7 with NO boron injected. B. Incorrect - In order to ensure that the reactor will stay shutdown under all conditions, only a single rod can be withdrawn, or all rods must be at notch position 04 or less, or a Qualified Nuclear Engineer must determine that the reactor will stay shutdown under all conditions. With two rod positions unknown, a determination that the reactor will stay shutdown under all conditions cannot be made by the operators, since you cannot confirm that these conditions are met. Plausible because both rods may be fully inserted. The operator may not recognize that the inability to confirm rod position means that it is NOT conservative to declare the reactor shutdown under all conditions. C. Incorrect - In order to ensure that the reactor will stay shutdown under all conditions, only a single rod can be withdrawn, or all rods must be at notch position 04 or less, or a Qualified Nuclear Engineer must determine that the reactor will stay shutdown under all conditions. With multiple rods out, and one or more at greater than notch position 04, a determination that the reactor will stay shutdown under all conditions cannot be made by the operators. Plausible because this would be true of all rods were at 04 or less. Additionally, the only rod that is at greater than notch position is a peripheral rod. The operator may believe that a peripheral rod would have minimal impact, and would be a more conservative answer than a case where a rod of a higher rod worth was withdrawn. D. Correct - In order to ensure that the reactor will stay shutdown under all conditions, only a single rod can be withdrawn, or all rods must be at notch position 04 or less, or a Qualified Nuclear Engineer must determine that the reactor will stay shutdown under all conditions. In this case, a single rod is withdrawn, so this ensures the reactor will remain shutdown under ALL conditions. **Required References: NONE**

69	ID: 5916	Points: 1.00

Given the following conditions:

- An automatic scram occurred on Unit 2
- The control rods did not fully insert and reactor power is at 10%
- Containment parameters are degrading and DEOP 200 -1 values for an emergency depressurization will be reached within 15 minutes if trends are not changed

Which one of the following choices completes the statement below regarding the use of the Bypass Valves under these conditions?

Opening all of the Bypass Valves to rapidly depressurize the RPV should __________ .

A. be performed in anticipation of an emergency blowdown

B. be performed to reduce reactor power below IRM range 7

C. **NOT** be performed to avoid removal of boron from the RPV

D. **NOT** be performed to avoid adding significant positive reactivity

Answer: D

Answer Explanation

With the reactor at power under an ATWS condition, rapid depressurization would add significant positive reactivity (moderator temperature reduction) complicating the power control actions in progress per DEOP 400-5. For this reason, the allowance to use the BPVs to rapidly depressurize the RPV in anticipation of a blowdown are provided in DEOP 100 pressure leg, and are NOT provided in any leg of DEOP 400-5.

Question 69 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	4.00
System ID:	5916
User-Defined ID:	5916
Cross Reference Number:	LIH
Topic:	69 - 295015 AK1.02
Num Field 1:	0.00
Num Field 2:	0.00
Text Field:	
Comments:	Objective: 29501LK031 Reference: DEOP 100 and 400-5 K/A: 295015 AK1.02 3.9 / 4.1 K/A: Knowledge of the operational implications of the following concepts as they apply to INCOMPLETE SCRAM: Cooldown effects on reactor power. Safety Function: 1 CFR: 41.8 to 41.10 PRA: No Level: High Pedigree: Bank History: NRC 2012 Explanations: A. Incorrect - Anticipating Blowdown is not allowed with the conditions in the stem. Plausible because this action is allowed in DEOP 100 but not with some rods not fully inserted. B. Incorrect - Rapid depressurization would cause power to go up not down. Plausible because common misconception that reducing pressure would reduce power. Range 7 called out in DEOP 400-5 for exiting the power leg. C. Incorrect - A rapid depressurization would not remove boron (steam removed, not boron). Plausible because removing inventory from the reactor could cause a reduction in boron concentration. The impact is minimal via steam via bypass vlvs. D. Correct - With the reactor at power under an ATWS condition, rapid depressurization would add significant positive reactivity (moderator temperature reduction). **Required References: NONE**

Question 69 Table-Item Links

Dresden Procedures

DEOP 100, Rev 10
DEOP 0400-005

General Question Data - Site Ownership

Dresden

70	ID: 27596	Points: 1.00

Unit 2 was operating at 100% power when a level transient occurs.

00:00 Annunciator 902-5 F-8, RPV LVL LO, illuminates
00:05 Annunciator 902-5 B-13, CHANNEL A/B RPV LVL LO, illuminates.

Which of the following systems have received an isolation signal? ___(1)___

1. RWCU
2. MSL Drains
3. Iso Condenser Vents
4. TIPS

00:30 Reactor water level returns to normal.

What actions are to be taken? ___(2)___

A. (1) 2,3
(2) Do **NOT** reset Group Isolations until actions of DANs for initiating conditions have been completed

B. (1) 1,4
(2) Do **NOT** reset Group Isolations until actions of DANs for initiating conditions have been completed

C. (1) 2,3
(2) **IMMEDIATELY** reset group isolations so that the tripped systems can be restarted.

D. (1) 1,4
(2) **IMMEDIATELY** reset group isolations so that the tripped systems can be restarted.

Answer: B

Answer Explanation

Based on the annunciators level has dropped below the Rx Scram and Group 2 and 3 setpoint of +8 inches reactor water. This will cause a trip of RWCU, SDC, and TIPS in addition to other systems. Per the DANS for Group 2 and 3 isolations, the isolations should not be reset until actions required by DANs for the initiating conditions have been completed.

Question 70 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	3.00
System ID:	27596
User-Defined ID:	27596
Cross Reference Number:	
Topic:	70 - 295031 G2.2.44
Num Field 1:	
Num Field 2:	
Text Field:	

Comments:	Objective: DRE223LN001.06 Reference: DAN's 902(3)-5 B-13, D-4, D-5, E-5 K/A: 295031 G2.2.44 4.2 / 4.0 K/A: Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions. Reactor Low Water Level. Safety Function: 2 CFR: 41.5/43.5/45.12 PRA: No Level: High Pedigree: New History: N/A Explanations: A. Incorrect - Iso Condenser Vents isolate on a Group 1 signal. Based on the stem that threshold has not been met. Plausible because the candidate must know that the annunciators in the stem would not meet Group 1 thresholds. Two of the three systems are correct and the third is a GR1 isolation. Part 2 is correct. B. Correct - Based on the annunciators level has dropped below the Rx Scram and Group 2 and 3 setpoint of +8 inches reactor water. This will cause a trip of RWCU, SDC, and TIPS in addition to other systems. Per the DANS for Group 2 and 3 isolations, the isolations should not be reset until actions required by DANs for the initiating conditions have been completed. C. Incorrect - Iso Condenser Vents isolate on a Group 1 signal. Based on the stem that threshold has not been met. Plausible because the candidate must know that the annunciators in the stem would not meet Group 1 thresholds. Two of the three systems are correct and the third is a GR1 isolation. Part 2 is plausible because with level returning to normal the GR 2 and GR 3 isolations could be reset except for guidance from the DANS to NOT do IMMEDIATELY. D. Incorrect - Iso Condenser Vents isolate on a Group 1 signal. Based on the stem that threshold has not been met. Plausible because the candidate must know that the annunciators in the stem would not meet Group 1 thresholds. Two of the three systems are correct and the third is a GR1 isolation. Part 2 is plausible because with level returning to normal the GR 2 and GR 3 isolations could be reset except for guidance from the DANS to NOT do IMMEDIATELY. **Required References: NONE**

None

71 **ID: 27597** **Points: 1.00**

A Reactor Scram has occurred on U2.

- MCC 28-1 has tripped on overcurrent
- Drywell Pressure is 5 psig and rising slowly
- Reactor Pressure is 875 and rising slowly

If Torus Sprays are started from the Main Control Room:

___(1)___ Loop(s) of Torus Sprays are running, and Drywell pressure will **ONLY** be reduced if ___(2)___ .

A. (1) One
(2) 95% of all noncondensible gases have been transferred from the drywell to the torus.

B. (1) Two
(2) 95% of all noncondensible gases have been transferred from the drywell to the torus.

C. (1) One
(2) steam is bypassing the torus suppression function and entering the torus air space directly.

D. (1) Two
(2) steam is bypassing the torus suppression function and entering the torus air space directly.

Answer: C

Answer Explanation

With the loss of MCC 28-1, the 2-1501-19A and 20A valves do not have power so one loop of torus sprays are not available. Per the DEOP 200-1 technical bases torus sprays may reduce primary containment pressure through convective cooling of the torus atmosphere if steam is bypassing the torus and entering the torus directly. Initiation of torus sprays may thus obviate the need for drywell sprays.

Question 71 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	4
Difficulty:	3.00
System ID:	27597
User-Defined ID:	27597
Cross Reference Number:	
Topic:	71 - 230000 K3.03
Num Field 1:	
Num Field 2:	
Text Field:	

Comments:	Objective: DRE29502LK066 References: DOP 1500-01, EPGs bases 4-11, EOP-DEOP TB, DOA 6700-T1 K/A: 230000 K3.03 3.4/3.6 K/A: Knowledge of the effect that a loss or malfunction of the RHR/LPCI TORUS/SUPPRESSION POOL SPRAY MODE will have on the following: Drywell pressure. Safety Function: 5 CFR: 41.7/45.4 PRA: Yes Level: High Pedigree: New HIstory: N/A Explanations: A. Incorrect - (1) The first part of the answer is correct. (2) Spraying the torus with 95% of non condensible gases transferred to the torus will not lower Drywell pressure. Plausible because (1) part 1 is correct. (2) Part 2 is plausible because DEOP 200-1 requires starting torus sprays,and because the point where 95% of non-condensible gases being transferred to the torus is where drywell sprays must be initiated. B. Incorrect - (1) With the loss of MCC 28-1, the 2-1501-19A and 20A valves do not have power so one loop of torus sprays are not available. (2) Spraying the torus with 95% of non condensible gases transferred to the torus will not lower Drywell pressure. . Plausible because (1) all LPCI pumps still have power and will have started. (2) Part 2 is plausible because DEOP 200-1 requires starting torus sprays,and because the point where 95% of non-condensible gases being transferred to the torus is where drywell sprays must be initiated. . C. Correct - (1) With the loss of MCC 28-1, the 2-1501-19A and 20A valves do not have power so one loop of torus sprays are not available. (2) Per the DEOP 200-1 technical bases torus sprays may reduce primary containment pressure through convective cooling of the torus atmosphere if steam is bypassing the torus and entering the torus directly. Initiation of torus sprays may thus obviate the need for drywell sprays. D. Incorrect - (1) With the loss of MCC 28-1, the 2-1501-19A and 20A valves do not have power so one loop of torus sprays are not available. (2) Part two is correct. Plausible because all LPCI pumps still have power and will have started. Part 2 is correct. **Required References: NONE**

None

72 **ID: 23024** **Points: 1.00**

Unit 2 was operating at near rated power, when a transient occurred, causing the following events to occur sequentially:

- Drywell pressure reached 2.5 psig.
- TR-86 sudden pressure relay (SPR) activated.

Per DGA-12, PARTIAL OR COMPLETE LOSS OF AC POWER, what are the recommended power supplies to the following buses?

(assume backfeeding, where appropriate)

power: from:	Bus 23	Bus 23-1	Bus 24	Bus 24-1
A.	Bus 23-1	2/3 EDG	U2 SBO	U2 EDG
B.	Bus 23-1	2/3 EDG	U2 SBO	Bus 24
C.	U2 SBO	2/3 EDG	Bus 24-1	Bus 24
D.	U2 SBO	Bus 23	Bus 24-1	U2 EDG

Answer: A

Answer Explanation

With Drywell pressure reaching 2.5 psig, a scram occurred, which caused a loss of Div I power. When TR-86 experienced the SPR event, Div II power is lost. This is a complete loss of AC power to the unit. Per DGA-12 table 1, the preferred power supplies would be to power: Bus 23 from Bus 23-1 (backfeed), Bus 23-1 from EDG 2/3, Bus 24 from the U2 SBO, Bus 24-1 from the U2 EDG.

Question 72 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	4
Difficulty:	3.00
System ID:	23024
User-Defined ID:	23024
Cross Reference Number:	
Topic:	72 - 295003.A1.03
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	Objective: DRE262LN001.08 Reference: DGA-12 table 1 K/A: 295003.A1.03 4.4 / 4.4 K/A: Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER: Systems necessary to assure safe plant shutdown. Safety Function: 6 CFR; 41.7/45.6 PRA: Yes Level: High Pedigree: Bank History: 2008 NRC Explanations: A. Correct - With Drywell pressure reaching 2.5 psig, a scram occurred, which caused a loss of Div I power. When TR-86 experienced the SPR event, Div II power is lost. This is a complete loss of AC power to the unit. Per DGA-12 table 1, the preferred power supplies would be to power: Bus 23 from Bus 23-1 (backfeed), Bus 23-1 from EDG 2/3, Bus 24 from the U2 SBO, Bus 24-1 from the U2 EDG. B. Incorrect - Three of the 4 are correct Bus 24-1 would be fed from U2 EDG. Plausible because this would be correct with a failure of the U2 EDG. C. Incorrect - With no EDG failures this would not be the preferred lineup. Plausible because this would be correct with a failure of the 2/3 EDG. D. Incorrect - With no EDG failures this would not be the preferred lineup. Plausible because this would be correct for duel unit loss of AC power. **REQUIRED REFERENCES: None.**

None

73 **ID: 27598** **Points: 1.00**

Unit 3 is operating at full power when 3B Electromatic Relief Valve (ERV) spuriously opens.

Assuming NO operator action, core thermal power would be expected to decrease initially and then stabilize at a power level which is ___(1)___ the **ORIGINAL** power level.

The 3B ERV Decay Heat Removal (DHR) capacity is ___(2)___ MWth.

A. (1) **NEAR**
(2) 112.5

B. (1) **NEAR**
(2) 139.5

C. (1) approximately the DHR capacity of the 3B ERV **BELOW**
(2) 112.5

D. (1) approximately the DHR capacity of the 3B ERV **BELOW**
(2) 139.5

Answer: B

Answer Explanation

(1) Per the UFSAR, section 15.6.1, the spurious operation of an ERV is expected to result in power stabilizing at a power level near the initial power. (2) The DHR capacity of the 3B ERV is equal to 139.5 MWth.

Question 73 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	3.00
System ID:	27598
User-Defined ID:	27598
Cross Reference Number:	
Topic:	73 - 239002 A1.06
Num Field 1:	
Num Field 2:	
Text Field:	

Comments:	Objective: DRE239LN001.8 Reference: UFSAR 15.1.4, UFSAR 15.6.1, DGP 02-01, DGP 02-03, DOA 0250-01, DOA 1000-01 K/A: 239002 A1.06 3.7 / 3.8 K/A: Ability to predict and/or monitor changes in parameters associated with operating the RELIEF/SAFETY VALVES controls including: Reactor power. Safety Function: 3 CFR: 41.5/45.5 PRA: No Level: High Pedigree: New History: N/A Explanations: A. Incorrect - (1) The first part of the answer is correct. (2) The DHR capacity of the 3B ERV is equal to 139.5 MWth. Plausible because (1) The first part of the answer is correct. (2) The answer is the DHR capacity for the Turbine Bypass Valves, which is a similar system used for DHR. B. Correct – (1) Per the UFSAR, section 15.6.1, the spurious operation of an ERV is expected to result in power stabilizing at a power level near the initial power. (2) The DHR capacity of the 3B ERV is equal to 139.5 MWth. C. Incorrect – (1) The spurious operation of an ERV is expected to result in power stabilizing at a power level near the initial power. (2) The DHR capacity of the 3B ERV is equal to 139.5 MWth. Plausible because (1) turbine generator output in MWe is going to decrease due to the open ERV. The candidate must recognize that the EHC system will adjust to make power level remain approximately the same, even though generator output goes down. (2) The answer is the DHR capacity for the Turbine Bypass Valves, which is a similar system used for DHR. D. Incorrect – (1) The spurious operation of an ERV is expected to result in power stabilizing at a power level near the initial power. (2) The second part of the answer is correct. Plausible because (1) turbine generator output in MWe is going to decrease due to the open ERV. The candidate must recognize that the EHC system will adjust to make power level remain approximately the same, even though generator output goes down. (2) the second part of the answer is correct. **Required References: None**

None

74 **ID: 23017** **Points: 1.00**

To perform repairs to the 2A ASD system, Unit 2 secured the 2A Recirc pump and entered single loop operations with only the 2B Recirc pump in operation at 42% of rated speed.

What indication change would confirm **REVERSE** flow through the 2A loop and why?

A. A decrease in idle loop Jet Pump Flow;
the core flow measurement instrumentation can **NOT** differentiate between reverse flow and forward flow.

B. An increase in idle loop Jet Pump Flow;
the core flow measurement instrumentation can **NOT** differentiate between reverse flow and forward flow.

C. A decrease in idle loop Jet Pump Flow;
the drive flow manifold does **NOT** distribute flow equally which causes decreased resistance between the jet pumps.

D. An increase in idle loop Jet Pump Flow;
the drive flow manifold does **NOT** distribute flow equally which causes decreased resistance between the jet pumps.

Answer: B

Answer Explanation

At Recirc pump speeds > 40% reverse flow through an inactive loop can be verified by an increase in idle loop Jet Pump Flow. This increase is caused because the jet pump flow indicators work on differential pressure and cannot differentiate between reverse flow and forward flow.

Question 74 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	3.00
System ID:	23017
User-Defined ID:	23017
Cross Reference Number:	
Topic:	74 - 295001.K3.06
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	Objective: DRE202LN001.12 Reference: DGP 03-03 K/A: 295001.K3.06 2.9 / 3.0 K/A: Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION: Core flow indication. Safety Function: 1 + 4 CFR: 41.5/45.6 PRA: No Level: Memory Pedigree: Bank History: 2009 Cert Explanations: A. Incorrect - Reverse flow would not be a cause for a decrease in idle loop Jet Pump flow. Plausible because if the amount of reverse flow was not seen as total loop flow this would be correct. Part 2 is correct. B. Correct - At Recirc pump speeds > 40% reverse flow through an inactive loop can be verified by an increase in idle loop Jet Pump Flow. This increase is caused because the jet pump flow indicators work on differential pressure and cannot differentiate between reverse flow and forward flow. C. Incorrect - Reverse flow would not be a cause for a decrease in idle loop Jet Pump flow. Plausible because if the amount of reverse flow was not seen as total loop flow this would be correct. Part 2 is plausible because the amount of drive flow with the pump secured would be different. D. Incorrect - The reason for the increase in flow is that the flow measurement instrument can not differentiate direction of flow. Plausible because part 1 is correct and Part 2 because the amount of drive flow with the pump secured would be different. **REQUIRED REFERENCES: None.**

None

75	ID: 12531	Points: 1.00

Given the following conditions:

- Unit 2 is operating at 55% power.

While **SINGLE NOTCHING** a control rod from notch position 18 to 20, the RMCS timer PLC freezes with the "rod out" logic made up.

The operator would expect annunciator 902-5 D-3, TIMER MALFUNCTION ROD SELECT BLOCK to come in and....

A. annunciator 902-5 A-3, ROD DRIFT to alarm **ONLY**.

B. the control rod to continuously withdraw to position 48.

C. a Rod Block Monitor rod block to stop control rod motion.

D. a Rod Select Block to occur after approximately 2 to 3 seconds.

Answer: D

Answer Explanation

Per DAN 902-5 D-3 TIMER MALFUNCTION ROD SEL BLOCK, once the RMCS timer begins its sequence, all "normal" rod blocks are essentially bypassed. The logic assumes the timer will complete its sequence. The 115 (rod out) contacts are monitored by the 130 relay. If they remain closed for more than 3 seconds (and a continuous withdraw signal is not present) the 130 relay will energize. The 130 relay de-energizes the rod out bus causing rod motion to stop (Rod Select Block).

Question 75 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	2
Difficulty:	3.00
System ID:	12531
User-Defined ID:	12531
Cross Reference Number:	LI
Topic:	75 - 201002 A3.01
Num Field 1:	0.00
Num Field 2:	0.00
Text Field:	

Comments:	Objective: DRE201LN002.06 Reference: DAN 902(3)-5 D-3 K/A: 201002 A3.01 3.2/3.1 K/A: Ability to monitor automatic operation of the REACTOR MANUAL CONTROL SYSTEM including: control rod block actuation. Safety Function: 1 CFR: 41.7/45.7 PRA: No Level: Memory Pedigree: Bank History: N/A Explanations: A. Incorrect - The TIMER MALFUNCTION ROD SEL BLOCK, would come in, but would also cause a Rod Select Block to occur. Plausible because if the Rod Select Block did not occur a Rod Drift would be expected. B. Incorrect - The logic assumes the timer will complete its sequence. The 115 (rod out) contacts are monitored by the 130 relay. If they remain closed for more than 3 seconds (and a continuous withdraw signal is not present) the 130 relay will energize. The 130 relay de-energizes the rod out bus causing rod motion to stop (Rod Select Block). Plausible because student must recognize without continuous rod out signal the RMCS logic times out and prevents continuous withdraw (rod drift). C. Incorrect - The TIMER MALFUNCTION ROD SEL BLOCK, would come in, but would also cause a Rod Select Block to occur. A RBM rod block only occurs on RBM downscale or RBM HI/INOP. Plausible because with a rod moving without a continuous signal there is a ROD OUT BLOCK just not caused by RBM D. Correct - Per DAN 902-5 D-3 TIMER MALFUNCTION ROD SEL BLOCK, once the RMCS timer begins its sequence, all "normal" rod blocks are essentially bypassed. The logic assumes the timer will complete its sequence. The 115 (rod out) contacts are monitored by the 130 relay. If they remain closed for more than 3 seconds (and a continuous withdraw signal is not present) the 130 relay will energize. The 130 relay de-energizes the rod out bus causing rod motion to stop (Rod Select Block). **Required References: None**

<u>Question 75 Table-Item Links</u>

<u>Licenced Operator Annual Exam Bank</u>

NRC Exam

<u>General Question Data - Site Ownership</u>

Dresden

76	ID: 27611	Points: 1.00

Unit 2 is cooling down in accordance with DEOP 100 following a SCRAM from full power:

- Both Reactor Recirc Pumps are running
- 2A SDC pump is lined up to the RPV
- RPV pressure is at 95 psig and slowly lowering
- RPV temperature is at 335 F and slowly lowering
- RPV level is in auto at 30 inches

THEN A TRANSIENT OCCURRED:

- RPV level is -10 inches and steady
- RPV temperature 341 F and rising slowly
- RPV Pressure is 103 psig and rising slowly

At this time, to address the rise in RPV pressure and temperature, the SRO will give the direction to place:

A. SDC back in operation.

B. RWCU in Recirculation Mode

C. RWCU in the Blowdown Mode.

D. HPCI in the Pressure Control Mode.

Answer: B

Answer Explanation

DEOP 100 allows the Group 3 Isolations to by bypassed for operating RWCU in this mode of operation.

Question 76 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	5
Difficulty:	3.00
System ID:	27611
User-Defined ID:	27611
Cross Reference Number:	
Topic:	76 - 295021 G2.4.6 (2)
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	Objective: 29501LK046 References: DEOP 0100, DOA 1000-01, EOP-DEOP TB K/A: 295021 G2.4.6 4.7 K/A: Knowledge of EOP mitigation strategies: Loss of Shutdown Cooling. Safety Function: 4 PRA: No CFR: 43.5 Level: High Pedigree: New History: None **SRO Only Criteria: 10CFR55.43(b)(5) - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.** Explanation: A. Incorrect - SDC is not available due to the Group 3 isolation and there is no allowance to bypass the Group 3 isolation for SDC operation. Plausible as this is the normal method to decay heat at this pressure and temperature. B. Correct - DEOP 100 allows the Group 3 Isolations to by bypassed for operating RWCU in this mode of operation. C. Incorrect - RWCU is not available due to the Group 3 isolation that is in and there is no allowance to bypass the Group 3 isolation for this mode of operation. Plausible because RWCU in Blowdown Mode would be able to remove enough pressure to counteract the loss of SDC. D. Incorrect - HPCI cannot be started due to the <106 psig in RPV interlock and there is no allowance to bypass the low pressure isolation. Plausible because HPCI would be able to remove enough pressure to counteract the loss of SDC. **REQUIRED REFERENCES: None**

None

77	ID: 27570	Points: 1.00

Per the basis section of TS 3.5.3, IC System:

1) The LOWEST allowable level in the shell side of the Isolation Condenser is ___(1)___ feet.

2) The basis for this water level is to ensure the Isolation Condenser has sufficient inventory to remove the required decay heat for ___(2)___ minutes.

A. (1) 3
(2) 20

B. (1) 3
(2) 32

C. (1) 6
(2) 20

D. (1) 6
(2) 32

Answer: C

Answer Explanation

Per TS 3.5.3 basis, the minimum IC shell side level must be at or higher than 6 feet, and the IC is designed to provide the required Decay Heat Removal for 20 minutes.

Question 77 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	5
Difficulty:	3.00
System ID:	27570
User-Defined ID:	27570
Cross Reference Number:	
Topic:	77 - 207000 G.2.2.38
Num Field 1:	
Num Field 2:	
Text Field:	

Comments:	Objective: DRE207LN001.07c References: T.S. Bases 3.5.3, DSSP 100-CR K/A: 207000 G.2.2.38 -- / 4.5 K/A: Knowledge of conditions and limitations in the facility license: Isolation Condenser. Safety Function: 4 CFR: 41.7, 41.10, 43.1, 45.13 PRA: Yes Level: Memory Pedigree: New History: None **SRO Only Criteria: 10CFR55.43(b)(2) - Facility operating limitations in the technical specifications and their bases.** Explanation: A. Incorrect - Minimum allowable IC level is 6 feet, per the basis section of TS 3.5.3 (2) The second part of the answer is correct. Plausibility – (1) Per TS 3.5.3 basis, three feet is the amount of level above the tube bundles that exists when shell side level is at 6 feet. (2) The second part of the answer is correct B. Incorrect - Minimum allowable IC level is 6 feet, per the basis section of TS 3.5.3 (2) Per the basis section of TS 3.5.3, with the IC at minimum level of 6 feet, the isolation condenser contains sufficient inventory to remove the design decay heat load for at least 20 minutes. Plausibility – (1) Per TS 3.5.3 basis, three feet is the amount of level above the tube bundles that exists when shell side level is at 6 feet. (2) 32 minutes is the time which Decay Heat Removal MUST be initiated using the Isolation Condenser in DSSP 0100-CR. C. Correct - Per TS 3.5.3 basis, the minimum IC shell side level must be at or higher than 6 feet, and the IC is designed to provide the required Decay Heat Removal for 20 minutes. D. Incorrect - The first part of the answer is correct (2) Per the basis section of TS 3.5.3, with the IC at minimum level of 6 feet, the isolation condenser contains sufficient inventory to remove the design decay heat load for at least 20 minutes. Plausibility – (1) The first part of the answer is correct. (2) 32 minutes is the time which Decay Heat Removal MUST be initiated using the IC in DSSP 0100-CR. **Required Reference: None**

None

78	ID: 27609	Points: 1.00

Unit 2 was at full power when a LOCA occurred:

- A release is in progress
- The SAMG's have NOT been entered
- Torus Bottom pressure is 45 psig and slowly rising
- Field Survey teams have reported the following gamma dose rates, which are expected to remain at this level for the next 90 minutes:
    - 8 mRem/hr at the 345 KV switchyard
    - 12 mRem/hr at the Lift Station
    - 15 mRem/hr at the Training Building parking lot
    - 18 mRem/hr at the Pre-Access Facility
- The Shift Manager has determined that primary containment pressure reduction is **REQUIRED** in order to **REDUCE THE EXPECTED OFFSITE DOSE**, per the override in DEOP 0200-01, PRIMARY CONTAINMENT CONTROL.

Based on the **CURRENT** conditions, the Unit Supervisor should direct **ENTERING** DEOP 0500-04, CONTAINMENT VENTING _____(1)_____.

Per the guidance in OP-AA-103-102-1002, STRATEGIES FOR SUCCESSFUL TRANSIENT MITIGATION, venting containment to **REDUCE TOTAL OFFSITE DOSE** containment pressure should be lowered to ___(2)_____ psig.

A. 1) **ONLY**
(2) 0

B. (1) **ONLY**
(2) **NO** lower than 10

C. (1) **AND** DEOP 0300-02, RADIOACTIVITY RELEASE CONTROL
(2) 0

D. (1) **AND** DEOP 0300-02, RADIOACTIVITY RELEASE CONTROL
(2) **NO** lower than 10

Answer: D

Answer Explanation

Of the areas listed, only the lift station is outside of the site-boundary (off-site). It has an expected dose above 10 mRem/hr, based on Field Team reports, and is expected to last for more than 60 minutes.. Therefore, it would meet the EAL ALERT condition for RA1, and entry into DEOP 0300-02, Radioactivity control is required. OP-DR-103-102-1002, STRATEGIES FOR SUCCESSFUL TRANSIENT MITIGATION, requires venting no lower than approximately 10 psig when venting to reduce total offsite dose per the override in DEOP 0200-01. This is done to ensure adequate NPSH for ECCS when in an accident condition.

Question 78 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	27609
User-Defined ID:	27609
Cross Reference Number:	

Topic:	78 - 295038.EA2.01
Num Field 1:	
Num Field 2:	
Text Field:	

Comments:

Objective: 29502LK103
Reference: EP-AA-1000, ODCM, DEOP 300-2, EP-AA-1004 Addendum 3, DEOP 0500-04, OP-AA-103-102-1002
K/A: 295038.EA2.01 3.3 / 4.3
K/A: Ability to determine and/or interpret the following as they apply to HIGH OFF-SITE RELEASE RATE: Off-site
Safety Function: 9
CFR: 41.10, 43.5, 45.13
Level: High
Pedigree: New
History: N/A

SRO Only Criteria: 10CFR55.43(b)(5) – Assessment of facility conditions and selection of appropriate procedures during normal, abnormal and emergency conditions. 10CFR55.43(b)(4) - Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions.

Explanation:

A. INCORRECT – (1) Of the areas listed, only the lift station is outside of the site-boundary (off-site). It has an expected dose above 10 mRem/hr, based on Field Team reports, and is expected to last for more than 60 minutes.. Therefore, it would meet the EAL ALERT condition for RA1, and entry into DEOP 0300-02, Radioactivity control is required. (2) OP-DR-103-102-1002, STRATEGIES FOR SUCCESSFUL TRANSIENT MITIGATION, requires venting to no lower than approximately 10 psig when venting to reduce total offsite dose per the override in DEOP 0200-01. This is done to ensure adequate NPSH for ECCS when in an accident condition.
Plausible because (1) Of the areas listed, only the lift station is OFFSITE. The students may believe that all the areas are onsite, which is a common misconception. (2) When venting in DEOP 0500-04, Attachment 4 to control H2 in the drywell, pressure is intentionally reduced all the way to zero psig. Additionally, DEOP 0200-01 gives guidance to stop drywell sprays and torus sprays before reaching 0 psig..

B. INCORRECT - (1) Of the areas listed, only the lift station is outside of the site-boundary (off-site). It has an expected dose above 10 mRem/hr, based on Field Team reports, and is expected to last for more than 60 minutes.. Therefore, it would meet the EAL ALERT condition for RA1, and entry into DEOP 0300-02, Radioactivity control is required.. (2) The second part of the answer is correct.
Plausible because (1) Of the areas listed, only the lift station is OFFSITE. The students may believe that all the areas are onsite, which is a common misconception. (2) The second part of the answer is correct.

C. INCORRECT – (1) The first part of the answer is correct (2) OP-DR-103-102-1002, STRATEGIES FOR SUCCESSFUL TRANSIENT MITIGATION, requires venting to no lower than approximately 10 psig when venting to reduce total offsite dose per the override in DEOP 0200-01. This is done to ensure adequate NPSH for ECCS when in an accident condition.
Plausible because(1) The first part of the answer is correct (2) When venting in DEOP 0500-04, Attachment 4 to control H2 in the drywell, pressure is intentionally reduced all the way to zero psig. Additionally, DEOP 0200-01 gives guidance to stop drywell sprays and torus sprays before reaching 0 psig.

D. CORRECT – (1) Of the areas listed, only the lift station is outside of the site-boundary (off-site). It has an expected dose above 10 mRem/hr, based on Field Team reports, and is expected to last for more than 60 minutes.. Therefore, it would meet the EAL ALERT condition for RA1, and entry into DEOP 0300-02, Radioactivity control is required. (2) OP-DR-103-102-1002, STRATEGIES FOR SUCCESSFUL TRANSIENT MITIGATION, requires venting no lower than approximately 10 psig when venting to reduce total offsite dose per the override in DEOP 0200-01. This is done to ensure adequate NPSH for ECCS when in an accident condition.

Required References: EP-AA-1004 Addendum 3 and ODCM Figure 1-2

79 **ID: 23899** **Points: 1.00**

Which of the following events are required to be reported to the Plant Manager, Station Security, and the Senior Resident Inspector per OP-AA-106-101, SIGNIFICANT EVENT REPORTING?

A. Initiation of a Prompt Investigation.

B. An unplanned shutdown or load reduction.

C. A report of a suspicious and malicious activity directed at plant safety.

D. A significant breakdown of plant radiological or environmental controls.

Answer: C

Answer Explanation

Per OP-AA-106-101 if there is suspicion of activity directed at plant safety or security the following individuals must be notified: Plant Manager, Station Security, and Senior Resident Inspector.

Question 79 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	3.00
System ID:	23899
User-Defined ID:	23899
Cross Reference Number:	
Topic:	79 - Generic.2.4.30
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	Objective: 29900LK152 Reference: OP-AA-106-101 K/A: Generic.2.4.30 -- / 4.1 K/A: 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. Safety Function: N/A CFR: 41.10/43.5/45.11 PRA: No Level: Memory Pedigree: Bank History: 2010 NRC **SRO Only Criteria: 10CFR55.43(b)(5) - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.** Explanation: A. Incorrect - A prompt investigation requires notification of Plant Manager and Senior Resident Inspector but not Station Security. Plausible because 2 of 3 are correct and an investigation could include security. B. Incorrect - An unplanned shutdown or load reduction requires notification of Plant Manager and Senior Resident Inspector but not Station Security. Plausible because 2 of 3 are correct and security may be needed to control access to certain areas during a shutdown. C. Correct - Per OP-AA-106-101 if there is suspicion of activity directed at plant safety or security the following individuals must be notified: Plant Manager, Station Security, and Senior Resident Inspector. D. Incorrect - A significant breakdown of plant radiological or environmental controls.requires notification of Plant Manager and Senior Resident Inspector but not Station Security. Plausible because 2 of 3 are correct and security may be needed to limit access to affected areas. **REQUIRED REFERENCES: None.**

Question 79 Table-Item Links

General Question Data - Site Ownership

Dresden

General Question Data - Ops Program

Senior Reactor Operator

80 **ID: 13375** **Points: 1.00**

A systems engineer brings a "Special Procedure" for the Control Rod Drive system to the WEC.

How can it be determined if the "Special Procedure" contains any un-reviewed safety questions?

A. The documentation of a 50.59 screening being conducted on the special procedure is included with the special procedure.

B. Any previously un-reviewed safety question will be listed on the procedural approval form, included with the special procedure.

C. The documentation of procedure OP-AA-108-110, EVALUATION OF SPECIAL TESTS OR EVOLUTIONS is included with the special procedure.

D. Any previously un-reviewed safety questions will be listed on DAP 09-09 form 09-09A, CONTENTS OF SPECIAL PROCEDURE CHECKLIST, included with the special procedure.

Answer: A

Answer Explanation

If it has been determined from the applicability review that 10 CFR 50.59 is applicable to the proposed activity, then prepare a 50.59 screening. This provides complete responses, with justification, to each 50.59 screening question.

Question 80 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	3.00
System ID:	13375
User-Defined ID:	13375
Cross Reference Number:	
Topic:	80 - Generic.2.2.07
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	Objective: 29500SE033 Reference: LS-AA-104 page 5, DAP 09-09 K/A: Generic.2.2.07 2.9 / 3.6 K/A: Knowledge of the process for conducting special or infrequent tests. Safety Function: N/A CFR: 41.10/43.3/45.13 PRA: No Level: Memory Pedigree: Bank History: 2011 Cert **SRO Criteria: 10CFR55.43(b)(3) - Facility licensee procedures required to obtain authority for design and operating changes in the facility.** Explanations: A. Correct - Per the above reference, if it has been determined from the applicability review that 10 CFR 50.59 is applicable to the proposed activity, then prepare a 50.59 screening. This provides complete responses, with justification, to each 50.59 screening question. B. Incorrect - AD-AA 101- F-01, Document Site Approval Form, provides a checklist for type and routing list but does not list un-reviewed safety questions. Plausible because it does identify whether or not a 10CFR50.59 or 10CFR72.48 are required but do not list the resolved or unresolved safety issues. C. Incorrect - OP-AA-108-110, EVALUATION OF SPECIAL TESTS OR EVOLUTIONS, provides a mechanism of evaluation infrequently performed complex tests, procedures, or plant evolutions to implement special administrative and management controls. Plausible because Attachment 2 is used to determine what potential risks are involved, but not the resolutions. D. Incorrect - DAP Form 09-09A confirms the unique requirement for a Special Procedure is satisfied. It does not document un-reviewed safety items. Plausible because the form must be completed for each special procedure to be performed. **REQUIRED REFERENCES: None.**

Question 80 Table-Item Links

General Question Data - Site Ownership

Dresden

General Question Data - Ops Program

Senior Reactor Operator

81 **ID: 24016** **Points: 1.00**

Unit 2 was operating at near rated power, with the SBGT system control switches in the following positions:

- 2/3A - PRI
- 2/3B - STBY

A transient occurs causing RPV water level trend down to -15 inches.

Two minutes later:

- Aux NSO reports that the 2/3A and 2/3 B SBGT are de-energized.
- RPV water level has recovered to 10 inches.

The Unit Supervisor will direct entering __(1)___ and take action to ___(2)___

A. (1) DOA 7500-01, STANDBY GAS TREATMENT SYSTEM FAN TRIP
(2) place 2/3A SBGT to OFF **AND** then 2/3B SBGT to START.

B. (1) DOA 5750-01, VENTILATION SYSTEM FAILURE
(2) place 2/3A SBGT to OFF **AND** then 2/3B SBGT to START.

C. (1) DOA 7500-01, STANDBY GAS TREATMENT SYSTEM FAN TRIP
(2) Restart Reactor Building ventilation supply and exhaust fans.

D. (1) DOA 5750-01, VENTILATION SYSTEM FAILURE
(2) Restart Reactor Building ventilation supply and exhaust fans.

Answer: A

Answer Explanation

When RPV water level decreased to < 6 inches, an auto start signal is given to the SBGT train which is selected to PRI (2/3A). Twenty seconds after the auto start signal, if the PRI train's heater is not operating, that train will trip and the STBY train (2/3B) will auto start. The SRO will direct entry into DOA 7500-01 subsequent operator actions to move the SBGT SELECT to A OFF and then to the START 2/3B SBGT.

Question 81 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	3.00
System ID:	24016
User-Defined ID:	24016
Cross Reference Number:	
Topic:	81 - 261000.A2.10
Num Field 1:	
Num Field 2:	
Text Field:	

Comments:	Objective: DRE261LN001.08 Reference: DAN 902(3)-5 E-5, DOA 7500-01, T.S. Bases 3.6.4.3 K/A: 261000.A2.10 3.1 / 3.2 K/A: Ability to (a) predict the impacts of the following on the STANDBY GAS TREATMENT SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Low reactor water level: Plant-Specific. CFR: 41.5 / 45.6 Safety Function: 9 PRA: No Level: High Pedigree: New History: N/A **SRO Only Criteria: 10CFR55.43(b)(5) - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.** Explanation: A. Correct - When RPV water level decreased to < 6 inches, an auto start signal is given to the SBGT train which is selected to PRI (2/3A). Twenty seconds after the auto start signal, if the PRI train's heater is not operating, that train will trip and the STBY train (2/3B) will auto start. The SRO will direct entry into DOA 7500-01 subsequent operator actions to move the SBGT SELECT to A OFF and then to START 2/3B SBGT. B. Incorrect - DOA 5750-01 only addresses building ventilation systems with a group II signal present, not SBGT. Plausible because the DOA for Ventilation system failure discusses the auto start of SBGT on a Gr 2 signal. The signal has cleared but guidance is not given to restart RB ventilation, part 2 is correct. C. Incorrect - The 2/3B train should have auto started. Plausible because part 1 is correct. Part 2 is plausible because the Group 2 condition has cleared, but guidance is not provided to restart RB ventilation with the conditions listed in the stem. D. Incorrect - DOA 5750-01 only addresses building ventilation systems with a group II signal present, not SBGT. Plausible because the DOA for Ventilation system failure discusses the auto start of SBGT on a Gr 2 signal. The signal has cleared but guidance is not given to restart RB ventilation **REQUIRED REFERENCES: None.**

<u>Question 81 Table-Item Links</u>

<u>General Question Data - Site Ownership</u>

Dresden

<u>General Question Data - Ops Program</u>

Senior Reactor Operator

82 **ID: 23760** **Points: 1.00**

Given the following:

- Unit 2 is operating at 85% power.
- APRM 2 is OOS.
- NSO notices APRM 6 reading 118%.
- Backpanel indications agree with the front panel recorder.
- Flow converters indicate 87% core flow.

The US will review Technical Specifications and TRM's, then direct bypassing APRM 6 and _______.

A. being in MODE 2 within 8 hours

B. resetting the Control Rod Block **ONLY**

C. resetting the Half Scram on RPS B **ONLY**

D. resetting Half Scram on RPS B **AND** the Control Rod Block

Answer: D

Answer Explanation

Per TRM 3.3.a., Rod Block setting is < .56W + 55.4 which equals 104.12 for the given conditions. The Scram setting is < .56W + 67.4 which equals 116.12 for the given conditions. With 2 APRMs operable in both channels, APRM 6 is bypassed and since we are above both Rod Block and Scram setpoints DAN 902-5 gives guidance to reset the 1/2 scram; DOP 0500-7 is the procedure used to reset the 1/2 scram and DAN 902-5 A-6 directs reseting the rod block.

Question 82 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	2
Difficulty:	3.00
System ID:	23760
User-Defined ID:	23760
Cross Reference Number:	
Topic:	82 - 215005 A2.04
Num Field 1:	2004
Num Field 2:	
Text Field:	

Comments:	Objective: 212LN001-06 References: DAN 902(3)-5 A-6, C-12, D-13; T.S. 3.3.1.1; TRM 3.3.a K/A: 215005 A2.04 3.8/3.9 K/A: Ability to predict the impacts of the following on the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM;and based on those predictions, use procedures to correct, control or mitigate the consequences of those abnormal conditions or operations: SCRAM trip signals. Safety Function: 7 CFR: 43.5 PRA: No Level: High Pedigree: New History: N/A **SRO Only Criteria: 10CFR55.43(b)(5) - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.** Explanations: A. Incorrect - With 1 APRM in each channel INOP, we still have the required number of APRMs available. Plausible because this would be correct if both INOP APRMs were in the same channel. B. Incorrect - After doing the math, both the Rod Block and SCRAM setpoints have been exceeded and will need to be bypassed after the APRM is bypassed. Plausible because 118% would not exceed the Scram setpoint for 100% power (T.S. 122%). C. Incorrect - After doing the math, both the Rod Block and SCRAM setpoints have been exceeded and will need to be bypassed after the APRM is bypassed. Plausible because must determine that APRMs Hi also cause Rod Blocks D. Correct - Per TRM 3.3.a., Rod Block setting is < .56W + 55.4 which equals 104.12 for the given conditions. The Scram setting is < .56W + 67.4 which equals 116.12 for the given conditions. With 2 APRMs operable in both channels, APRM 6 is bypassed and since we are above both Rod Block and Scram setpoints DAN 902-5 D-13 gives guidance to reset the 1/2 scram; DOP 0500-7 is the procedure used to reset the 1/2 scram and DAN 902-5 A-6 directs reseting the rod block. **Required Reference: T.S.3.3.1.1 and TRM 3.3.a**

<u>Question 82 Table-Item Links</u>

<u>General Question Data - Site Ownership</u>

Dresden

83	ID: 27573	Points: 1.00

Unit 2 was operating at near rated power and Unit 3 is in Mode 5 for a refuel outage, with the following conditions:

- Transformer 32 is O.O.S.
- Unit 3 is being powered by backfeeding TR-3 and TR-31.

Concerning operability of the Auxiliary Power System, which unit (if any) is/are in a Technical Specification action statement, if any?

A. **NEITHER** unit.

B. Unit 2 **ONLY**

C. Unit 3 **ONLY**

D. **BOTH** units.

Answer: A

Answer Explanation

Neither Unit requires an action. Per TS 3.8.1, Unit 2 requires two qualified circuits between the offsite transmission network and the onsite Class 1E AC Electrical Power distribution system, as well as one qualified circuit between the offsite transmission network and the opposite unit's Division 2 onsite class 1E AC Electrical Power distribution subsystem to power the SBGT system. These ACE sources are met by TR-22 and by TR-31 (which is being backfed). Per TS 3.8.2, Unit 3 requires only one AC source, and this is can be met by either the TR-22 or TR-31 AC feeds. Therefore, the AC source requirements of TS 3.8.1 and 3.8.2 are met.

Question 83 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	5
Difficulty:	4.00
System ID:	27573
User-Defined ID:	27573
Cross Reference Number:	
Topic:	83 - 262001 G2.2.37
Num Field 1:	
Num Field 2:	
Text Field:	

Comments:	Objective: DRE262LN001.07 Reference: TS B.3.8.1, T.S. B.3.8.2 K/A: 262001.G.2.2.37 -- / 4.6 K/A: A.C. Electrical Distribution: Ability to determine operability and/or availability of safety related equipment. CFR: 41.7 /43.5 / 45.12 Safety Function: 6 PRA: Yes Level: High Pedigree: Bank History: N/A **SRO Only Criteria: 10CFR55.43(b)(2) - Facility operating limitations in the technical specifications and their bases.** Explanation: A. Correct - Neither Unit requires an action. Per TS 3.8.1, Unit 2 requires two qualified circuits between the offsite transmission network and the onsite Class 1E AC Electrical Power distribution system, as well as one qualified circuit between the offsite transmission network and the opposite unit's Division 2 onsite class 1E AC Electrical Power distribution subsystem to power the SBGT system. These ACE sources are met by TR-22 and by TR-31 (which is being backfed). Per TS 3.8.2, Unit 3 requires only one AC source, and this is can be met by either the TR-22 or TR-31 AC feeds. Therefore, the AC source requirements of TS 3.8.1 and 3.8.2 are met. B. Incorrect - Unit 2 does not need an action. It's TS 3.8.1 AC power needs are met by TR-22 and TR-31, and the associated Class 1E AC Electrical Power Distribution System. Plausible because if backfeeding TR-31 did not cause the transformer to meet the requirements of a qualified circuit, then Unit 2 would be in an LCO, since it would not have two qualified AC sources from the offsite transmission network. C. Incorrect - Unit 3 does not need an action. It's TS 3.8.2 AC power needs are met by TR-22 and TR-31 (either one), and the associated Class 1E AC Electrical Power Distribution System Plausible because the TS basis for TS 3.8.1 states that using TR-21 on backfeed is an acceptable substitute for TR-32, but does not mention TR-31 (since the Unit is online). Therefore, if the student believes that TS 3.8.2 applies to both units, that would imply that Unit 3 lacks the second qualified source. D. Incorrect - Both units do not need an action. Unit 2 does not need an action. It's TS 3.8.1 AC power needs are met by TR-22 and TR-31, and the associated Class 1E AC Electrical Power Distribution System. Unit 3 does not need an action. It's TS 3.8.2 AC power needs are met by TR-22 and TR-31 (either one), and the associated Class 1E AC Electrical Power Distribution System Plausible because the if the student believes that TS 3.8.2 applies to both units, and does not recognize that backfeeding a unit aux transformer meets the requirements for a qualified circuit, then both units would lack one qualified source. **REQUIRED REFERENCES: None.**

84	ID: 13864	Points: 1.00

Which of the following activities would **REQUIRE** a Plant PA announcement?

A change in ________.

1. Fire Risk
2. Online Risk
3. Reactor Mode
4. Shutdown Risk

A. 1 and 2

B. 3 and 4

C. 2 and 3

D. 1 and 4

Answer: C

Answer Explanation

Per OP-DR-108-117-1001, PROTECTED PATHWAY EQUIPMENT AND PATHWAY POLICY, Change in on line risk will be announced via Plant Public Address (PA) system and updated on monitors in the PAF. Also, per OP-AA-104-101, changes in Reactor Mode require a PA announcement.

Question 84 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	4.00
System ID:	13864
User-Defined ID:	13864
Cross Reference Number:	

Topic:	84 - Generic 2.1.14
Num Field 1:	0.00
Num Field 2:	0.00
Text Field:	

Comments:	Objective: 29501LK093 Reference: OP-AA-104-101, OP-DR-201-012-1001, OP-DR-108-117-1001 K/A: Generic 2.1.14 3.1 / 3.1 K/A: Knowledge of criteria or conditions that require plant-wide announcements, such as pump starts, reactor trips, mode changes, etc. Safety Function: N/A CFR: 41.10/43.5/45.12 PRA: N/A Level: Memory Pedigree: New History: N/A **SRO Only Criteria: 10CFR55.43(b)(5) - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.** Explanation: A. INCORRECT - Per OP-DR-201-012-1001, DRESDEN ON-LINE FIRE RISK MANAGEMENT, Changes in Fire Risk do not require a PA announcement. They will be briefed with the operations crew and the fire brigade. The second part, changes in online risk, is correct. Plausible because the student must be familiar with the administrative requirements of the Fire Risk procedure, including requirements for communications that must occur. The second part of the answer is correct. B: CORRECT - Per OP-DR-108-117-1001, PROTECTED PATHWAY EQUIPMENT AND PATHWAY POLICY, Change in on line risk will be announced via Plant Public Address (PA) system and updated on monitors in the PAF. Also, per OP-AA-104-101, changes in Reactor Mode require a PA announcement. C. INCORRECT - The first part of the answer, changes in online risk, is correct. Per OP-DR-108-117-1001, PROTECTED PATHWAY EQUIPMENT AND PATHWAY POLICY, Changes in Shutdown risk do not require a PA announcement. They will be communicated via daily communication memos distributed at the Main Access Facility and updated on monitors in the PAF. Plausible because the first part of the answer is correct. The student The student must be familiar with the administrative requirements of the Shutdown Risk procedure. It is similar to the online risk, but does not require a plant PA announcement for changes. D INCORRECT - Per OP-DR-201-012-1001, DRESDEN ON-LINE FIRE RISK MANAGEMENT, Changes in Fire Risk do not require a PA announcement. They will be briefed with the operations crew and the fire brigade. Per OP-DR-108-117-1001, PROTECTED PATHWAY EQUIPMENT AND PATHWAY POLICY, Changes in Shutdown risk do not require a PA announcement. They will be communicated via daily communication memos distributed at the Main Access Facility and updated on monitors in the PAF. Plausible because the student must be familiar with the administrative requirements of the Fire Risk procedure, including requirements for communications that must occur. The student must also be familiar with the administrative requirements of the Shutdown Risk procedure. It is similar to the online risk, but does not require a plant PA announcement for changes. **REQUIRED REFERENCES: None.**

Question 84 Table-Item Links

General Question Data - Site Ownership

Dresden

General Question Data - Ops Program

Senior Reactor Operator

85 **ID: 24112** **Points: 1.00**

Unit 2 has a core offload in progress in compliance with an approved spiral reload sequence. All other control rods in core cells containing one or more fuel assemblies are fully inserted.

Which of the following is **REQUIRED** to be performed prior to removing control rod XX-YY from the core for blade replacement?

A. Shorting links **NOT** installed.

B. Ensure that the control cell for control rod XX-YY is defueled.

C. Three SRMs are operable with at least one in an adjacent quadrant.

D. Ensure that the REFUEL INTERLOCKS for control rod XX-YY have been defeated.

Answer: B

Answer Explanation

Per DFP 0800-16 prerequisites, the control cell for control rod XX-YY must be defueled prior to removing the control rod.

Question 85 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	5
Difficulty:	4.00
System ID:	24112
User-Defined ID:	24112
Cross Reference Number:	
Topic:	85 - Generic.2.1.42
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	Objective: 29800LK084 Reference: DFP 0800-16, DGP 04-1, T.S. 3.10.5 K/A: Generic.2.1.42 -- / 3.4 K/A: Knowledge of new and spent fuel movement procedures. Safety Function: 8 CFR: 41.10, 43.7, 45.13 PRA: No Level: Memory Pedigree: Bank History: 2012 NRC **SRO Only Criteria: 10CFR55.43(b)(7) - Fuel handling facilities and procedures.** Explanation: A. Incorrect - Shorting Links are normally installed during refuel operations. Plausible because the shorting links provide full SCRAM capability to SRM operations. B. Correct - Per DFP 0800-16 prerequisites, the control cell for control rod XX-YY must be defueled prior to removing the control rod. C. Incorrect - Per T.S. 3.3.1.2, only 1 SRM s required to be in operation when spiral core offload is in progress. Plausible because 3 SRM's required in other Modes of operation. D. Incorrect - Refueling interlocks are not required to be defeated to lift the rod out of the top core. Plausible because required to withdraw the rod out the bottom for testing. **REQUIRED REFERENCES: None.**

Question 85 Table-Item Links

General Question Data - Site Ownership

Dresden

General Question Data - Ops Program

Senior Reactor Operator

86 **ID: 14705** **Points: 1.00**

Which of the following provide the bases to maintain Torus water level above -4.5 inches, in Modes 1, 2, and 3?

Ensures a sufficient amount of water . . . . .

A. with the Minimum CST Volume, to ensure Long-Term Cooling is available for the Design Basis Accident.

B. would be available to adequately condense the steam from the relief valve quenchers **ONLY**.

C. with the Minimum CST Volume, in the event of a LOCA to permit recirculation cooling flow to the core.

D. would be available to adequately condense the steam from the relief valve quenchers, downcomer lines, or HPCI turbine exhaust line.

Answer: D

Answer Explanation

Answer is per the EOP bases which requires compliance with T.S. and bases. CST is not part of bases. If the suppression pool water level is too low, an insufficient amount of water would be available to adequately condense the steam from the relief valve quenchers, downcomer lines, or HPCI turbine exhaust line.

Question 86 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	4
Difficulty:	3.00
System ID:	14705
User-Defined ID:	14705
Cross Reference Number:	
Topic:	86 - 295030.G2.4.18
Num Field 1:	
Num Field 2:	
Text Field:	2009 CERT
Comments:	Objective: DRE223LN001.07 Reference: T.S.Bases 3.6.2.2 , EPG B-7-44, EOP-DEOP TB K/A: 295030.G.2.4.18 3.3 / 4.0 K/A: Low Suppression Pool Water Level: Knowledge of specific bases for EOPs. Safety Function: 5 CFR: 41.10/43.1/45.13 PRA: No Level: Memory Pedigree: Bank History: 2012 NRC **SRO Only Criteria: 10CFR55.43(b)(2) - Facility operating limitations in the technical specifications and their bases.** Explanation: A. Incorrect - The EOPs and T.S. bases do not take credit for the CST for a design basis accident. Plausible because CST water can be used to supplement injection for HPCI. B. Incorrect - Partially correct Must also quench downcomer lines and HPCI exhaust. Plausible because answer is partially correct. Must also understand the additional sources of steam. C. Incorrect - The EOPs and T.S. bases do not take credit for the CST for a design basis accident. Low pressure injection systems would not be impacted in Mode 1,2,3. Plausible because CST water can be used to supplement injection for HPCI. D. Correct - Per the EOP bases which requires compliance with T.S. and bases. If the suppression pool water level is too low, an insufficient amount of water would be available to adequately condense the steam from the relief valve quenchers, downcomer lines, or HPCI turbine exhaust line. **REQUIRED REFERENCES: None.**

87 **ID: 27574** **Points: 1.00**

U2 was at rated power with FCL at 100%.

- 'A' Recirc pump trips
- CRAM rods are inserted to reduce power to 38%
- 'B' Recirc speed is currently at 50%
- 'B' Loop flow indicates 84%
- 'A' Loop flow indicates 5%

The US directs entry into ___(1)___ to determine actual core flow of ___(2)___.

A. (1) DOA 0202-01, RECIRCULATION PUMP TRIP - ONE OR BOTH PUMPS,
(2) 38.8%

B. (1) DOA 0500-01, INADVERTENT ENTRY INTO THE UNSTABLE POWER/FLOW REGION,
(2) 38.8%

C. (1) DOA 0202-01, RECIRCULATION PUMP TRIP - ONE OR BOTH PUMPS,
(2) 43.5%

D. (1) DOA 0500-01, INADVERTENT ENTRY INTO THE UNSTABLE POWER/FLOW REGION,
(2) 43.5%

Answer: A

Answer Explanation

With a trip of the A Recirc pump the US must direct entry into DOA 0202-01. The first subsequent action directs entry into DGP 03-03, SINGLE RECIRCULATION LOOP OPERATION. Determines on U2 Inactive loop flow is Reverse Flow. Uses the calculation. WTslo =(0.49) [(% loop flow active) - (0.95) (% loop flow inactive)] to come up with 38.8%.

Question 87 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	5
Difficulty:	3.00
System ID:	27574
User-Defined ID:	27574
Cross Reference Number:	
Topic:	87 - 295001 A2.03
Num Field 1:	
Num Field 2:	
Text Field:	

Comments:	Objective: 20200LK015 Reference: DGP 03-03, DOA 0202-01 K/A: 295001 A2.03 K/A: Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION: Actual core flow. Safety Function: 1 CFR: 41.10/43.5/45.13 PRA: No Level: High Pedigree: New History: None **SRO Only Criteria: 10CFR55.43(b)(5) - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.** Explanation: A. Correct - With a trip of the A Recirc pump the US must direct entry into DOA 0202-01. The first subsequent action directs entry into DGP 03-03, SINGLE RECIRCULATION LOOP OPERATION. Determines on U2 Inactive loop flow is Reverse Flow. Uses the calculation. WTslo =(0.49) [(% loop flow active) - (0.95) (% loop flow inactive)] to come up with 38.8%. B. Incorrect - DOA 0500-1 would be entered to determine if scram is required. It does not provide direction for actual flow calculation. Plausible because the the DOA would be entered. Part 2 the power to flow map would not be accurate due to reverse flow. C. Incorrect - US must direct entry into DOA 0202-01. The first subsequent action directs entry into DGP 03-03, SINGLE RECIRCULATION LOOP OPERATION. Determines on U2 Inactive loop flow is Reverse Flow 38.8%. Plausible because the first part is correct and if candidate uses calc for Forward flow 43.5% is correct. D. Incorrect - DOA 0500-1 would be entered to determine if scram is required. It does not provide direction for actual flow calculation. Plausible because if candidate uses calc for Forward flow 43.5% is correct. The power to flow map would not be accurate with these conditions, **REQUIRED REFERENCE: DGP 03-03**

None

88 **ID: 13719** **Points: 1.00**

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

- Torus water level is 30 feet.
- Drywell pressure is 1.31 psig and increasing 0.1 psig/10 minutes.
- Current Iodine-131 sample is 2.0 X 10^{-8} uCi/cc.
- Current Beta/Gamma (total particulate) is 5.0 X 10^{-7} uCi/cc.
- Radiation protection is unavailable to perform an off-site dose calculation.

The Unit Supervisor is required to direct the Operating team to vent the Drywell to the ___(1)___ system in accordance with ___(2)___ , to reduce Drywell pressure.

A. (1) SBGT;
(2) DEOP 500-4, CONTAINMENT VENTING

B. (1) SBGT;
(2) DOP 1600-01, NORMAL PRESSURE CONTROL OF THE DRYWELL

C. (1) Rx Building Vent;
(2) DEOP 500-4, CONTAINMENT VENTING

D. (1) Rx Building Vent;
(2) DOP 1600-01, NORMAL PRESSURE CONTROL OF THE DRYWELL

Answer: B

Answer Explanation

SBGT is the preferred vent path to get the filtering and hold up of the SBGT system directed per the DOP 1600-01, Normal Pressure Control of the Drywell.

Question 88 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	3.00
System ID:	13719
User-Defined ID:	13719
Cross Reference Number:	
Topic:	88 - Generic.2.3.11
Num Field 1:	
Num Field 2:	
Text Field:	

Comments:	Objective: 22301LP006 Reference: DOP 1600-01, DOP 1600-05, DEOP 200-1, DEOP 500-4 K/A: Generic.2.3.11 3.8 / 4.3 K/A: Ability to control radiation releases. Safety Function: 9 CFR: 43.4 PRA: No Level: High Pedigree: Bank History: 15-1 Cert **SRO Only Criteria: 10CFR55.43(b)(5) - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.** Explanation: Per the limitations and actions of DOP 1600-05, with the given radiation levels, the Drywell is required to be vented to the SBGT system (because of the first box in DEOP 200-1 under Primary Containment Pressure rising - need to vent to stay below 2.0 psig). This is directed per DEOP 200-1. A. Incorrect - While SBGT is correct for the vent path, the threshold for DEOP 500-4 is not met. Plausible because part 1 is correct. Part 2 would be correct if venting to stay below Primary Containment pressure limit B. Correct - SBGT is the preferred vent path to get the filtering and hold up of the SBGT system directed per the DOP 1600-01, Normal Pressure Control of the Drywell. C. Incorrect - Current Iodine -131 sample results as well as Beta/Gamma are above the limits for venting via Rx Building Vent. Plausible because candidate must know the limits for containment samples in DOP 1600-05. Part 2 would be correct if venting to stay below Primary Containment pressure limit D. Incorrect - Current Iodine -131 sample results as well as Beta/Gamma are above the limits for venting via Rx Building Vent. Plausible because candidate must know the limits for containment samples in DOP 1600-05. Part 2 is correct. **REQUIRED REFERENCES: DEOP charts, with the entry conditions blanked out. DOP 1600-05**

Question 88 Table-Item Links

General Question Data - Site Ownership

Dresden

General Question Data - Ops Program

Senior Reactor Operator

89 **ID: 13368** **Points: 1.00**

During conduct of a surveillance test of HPCI, the HPCI room temperature reaches 155ºF with the room cooler in operation.

What is the status of HPCI and why?

A. HPCI is operable, since the HPCI room cooler fan is running and cooling water is available.

B. HPCI is operable, since the HPCI room temperature has **NOT** exceeded the DEOP 300-1 entry level condition.

C. HPCI is NOT operable, due to the lack of high temperature qualification of instruments in the HPCI room.

D. HPCI is NOT operable, due to the lack of high temperature qualification of pipe supports and hangers in the HPCI room.

Answer: A

Answer Explanation

HPCI is operable in spite of exceeding EQ temps. Plausible because EQ limit of 127 degrees F with room coolers on, but HPCI is exempt from EQ requirements of 10CFR50.49 with room cooler running per UFSAR 3.11.1.4.2.

UFSAR 3.11.1.4.2 Mild Post-Accident Areas

A mild environment, by definition, meets all of the following criteria:

A. Temperature equal to or lower than 120°F;

B. Total radiation equal to or lower than 5×10^4 rads; and

C. Pressure no higher than that of all plant locations other than the drywell, the pressure suppression pool, and the main steam tunnel. A LOCA or HELB results only in minor changes in pressure in the mild environment areas.

Areas where the temperature does not exceed 120°F, due to a DBA, are considered mild temperature areas (except for HPCI Rooms where this temperature is 127°F with room coolers and 134°F without room coolers) and as such do not expose equipment required to perform safety related functions in response to a DBA to immediate or prolonged high-stress conditions during a DBA. The maximum service temperature represents no significant change from the normal temperature for equipment located in these areas. For all equipment located in these areas, the temperature is the result of normal plant operation; the loss of the heating, ventilation, and air conditioning (HVAC) system; or operation of equipment required for post-accident plant recovery. It is not the result of direct exposure to a LOCA or HELB environment. In all cases, the increase in temperature from the normal temperature to the maximum 120°F is gradual. The resulting applied stresses on the equipment are relatively low and well within the maximum stress level capability of the equipment, which is conservatively designed, fabricated, installed, and maintained. In some cases, the temperature during normal plant operation may slightly exceed 120°F. Operability of similar equipment in such mild temperature environments has been demonstrated by many years of experience in the utility industry. This operating experience does not indicate that a common-mode failure of safety-related equipment resulting from mild temperature environments is a problem. Furthermore, 10 CFR 50.49 does not require qualification to mild environments. Hence, no additional evaluations or documentation are necessary to ensure that this equipment performs its safety function.

Question 89 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	5
Difficulty:	3.00
System ID:	13368
User-Defined ID:	13368
Cross Reference Number:	
Topic:	89 - 295032.A2.02
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	Objective: 20600LK010 Reference: UFSAR 9.4.6 and 3.11.4, DOS 2300-03, DEOP 300-1, DAN 923-5 H-1 K/A: 295032.A2.02 -- / 3.5 K/A: Ability to determine and/or interpret the following as they apply to HIGH SECONDARY CONTAINMENT AREA TEMPERATURE: Equipment operability. Safety Function: 2 CFR: 41.10/43.5/45.13 PRA: Yes Level: Memory Pedigree: Bank History: 2013 Cert **SRO Only Criteria: 10CFR55.43(b)(1) - Conditions and limitations in the facility license.** Explanation: A. Correct - HPCI is operable in spite of exceeding EQ temps. Plausible because EQ limit of 127 degrees F with room coolers on, but HPCI is exempt from EQ requirements of 10CFR50.49 with room cooler running. B. Incorrect - HPCI is operable even though temperature HAS exceed DEOP 300-1 entry condition for Reactor Building Area Temperatures. Plausible because area temp of 150 degrees has exceeded DEOP 300-1 entry condition and and EQ limit of 127 degrees F with room coolers on, but HPCI is exempt from EQ requirements of 10CFR50.49 with room cooler running. C. Incorrect - HPCI is operable in spite of exceeding EQ temps. Plausible because EQ limit of 127 degrees F with room coolers on, but HPCI is exempt from EQ requirements of 10CFR50.49 with room cooler running. D. Incorrect - HPCI is operable in spite of exceeding EQ temps. Plausible because EQ limit of 127 degrees F with room coolers on, but HPCI is exempt from EQ requirements of 10CFR50.49 with room cooler running. **REQUIRED REFERENCES: None.**

90 **ID: 27575** **Points: 1.00**

Unit 2 is operating at near full power when a LOCA occurred on Unit 2.

- A manual SCRAM was inserted
- A spurious Group 1 occurred
- All rods did NOT go in
- Reactor power was initially 43%
- The NSO has taken all required actions for the DGP 02-03 SCRAM hard card
- Valves 2-1201-1, RX OUTLET ISOL and 2-1201-2, INLET ISOL failed to close on a GP 3 isolation, and could not be closed manually from the MCR
- RWCU Pump Room temperature is 183 F and rising slowly

10 minutes later:

- RWCU Pump Room temperature is 200 F and rising slowly
- Reactor power is 9% and slowly lowering
- Torus Bottom pressure is 15 psig and slowly rising
- Torus temperature is 155 F and rising
- RPV pressure is 900 psig and steady
- RPV level is being maintained between -170 and -185 inches with feedwater per DEOP 0400-05

The **HIGHEST** Emergency Action Level (EAL) currently met is _____________.

A. ALERT, due to FA1 or MA3

B. SITE AREA EMERGENCY, due to FS1

C. SITE AREA EMERGENCY, due to MS3

D. GENERAL EMERGENCY, due to FG1

Answer: B

Answer Explanation

With an unisolable direct downstream path to the environment, due to the failure of RWCU to isolate, and with indications of a line break in the RWCU pump room, we have a loss of RC and CT. Therefore, the correct answer is FS1. Although FA1 and MA3 also apply, they are not the highest EAL declarations. FG1 does not apply since we have a loss of RC and CT, but do not have a loss of FC. MS3 does not apply because we are able to restore and maintain RPV level above MSCRWL, and have not exceeded HCTL.

Question 90 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	6
Difficulty:	4.00
System ID:	27575
User-Defined ID:	27575
Cross Reference Number:	
Topic:	90 - Generic 2.4.38
Num Field 1:	
Num Field 2:	
Text Field:	

Comments:	Objective: 29501LP032 Reference: EP-AA-1004 Addendum 3 K/A: Generic 2.4.38 -- / 4.4 K/A: Ability to take actions called for in the facility emergency plan, including supporting or acting as emergency coordinator is required. Safety Function: N/A CFR: 41.10, 43.5, 45.11 PRA: No Level: High Pedigree: New History: N/A **SRO Only Criteria: 10CFR55.43(b)(5) - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.** Explanation: A. Incorrect - Although FA1 and MA3 are both applicable, they are not the highest EAL that we currently meet. That is a Site Area Emergency under EAL FS1. Plausible because both conditions do apply, and if RWCU did not have indications of a line break in the RWCU pump room, leading to a higher EAL, this would be the correct answer. B. Correct - With an unisolable direct downstream path to the environment, due to the failure of RWCU to isolate, and with indications of a line break in the RWCU pump room, we have a loss of RC and CT. Therefore, the correct answer is FS1. Although FA1 and MA3 also apply, they are not the highest EAL declarations. FG1 does not apply since we have a loss of RC and CT, but do not have a loss of FC. MS3 does not apply because we are able to restore and maintain RPV level above MSCRWL, and have not exceeded HCTL. C. Incorrect - MS3 does not apply since we have not exceeded HCTL, and water level can be restored and maintained above MSCRWL. Although level is being maintained below TAF per DEOP 0400-05, it can be restored and maintained above MSCRWL. Plausible because if we were exceeding HCTL, or if water level could not be maintained above MSCRWL, MS3 this would be correct. D. Incorrect - FG1 does not apply since we do not have a loss or potential loss of FC. Although we are maintaining level below TAF, the basis of FC2 states that we would address this EAL condition under MS3, not FG1. Plausible because a loss of RCS and a loss of CT have occurred, and since water level is being maintained below TAF but above MSCRWL per DEOP 0400-05, the student may believe that a potential loss of FC would be applicable. Also, if we were not in DEOP 0400-05, and we were maintaining RPV level below TAF, then FG1 would be correct. **Required References: EP-AA-1004 Addendum 3**

None

91 **ID: 27578** **Points: 1.00**

An Annunciator in the main control room has been spuriously activating. The Unit Supervisor has determined that this annunciator should be disabled to minimize operator distraction in the Main Control Room.

- The action to disable the annunciator is NOT in support of a maintenance activity or troubleshooting
- A review by the NSO shows that there currently is no operating procedure guidance containing actions for disabling this annunciator

Per CC-AA-112, TEMPORARY CONFIGURATION CHANGES, this is an example of a _____(1)______.

This activity _____(2)______ require a Temporary Configuration Change Package (TCCP) to be issued to allow disabling the annunciator.

A. (1) Controlled Exclusion
(2) **WILL**

B. (1) Controlled Exclusion
(2) **WILL NOT**

C. (1) Temporary Configuration Change (TCC)
(2) **WILL**

D. (1) Temporary Configuration Change (TCC)
(2) **WILL NOT**

Answer: C

Answer Explanation

Per CC-AA-112, this is a TCC, and a TCCP will be required in order to disable the annunciator. The stem of the question shows that it is not an MR90, it does not meet the requirements for a Controlled Exclusion, and it is not Procedurally Controlled TCC (PCTCC).

Question 91 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	5
Difficulty:	4.00
System ID:	27578
User-Defined ID:	27578
Cross Reference Number:	
Topic:	91 - Generic 2.2.43
Num Field 1:	
Num Field 2:	
Text Field:	

Comments:	Objective: 29900LK147 Reference: CC-AA-112 K/A: Generic 2.2.43 -- / 3.3 K/A: Knowledge of the process used to track inoperable alarms CFR: 41.10 / 43.5 / 45.13 Safety Function: N/A PRA: No Level: High Pedigree: New History: N/A **SRO Only Criteria: 10CFR55.43(b)(3) - Facility licensee procedures required to obtain authority for design and operating changes in the facility.** Explanation: A. Incorrect - Per CC-AA-112, disabling an annunciator is not considered a Controlled Exclusion. The second part of the answer is correct. Plausible because many types of configuration changes fall under Controlled Exclusions, and the student may believe that a TCCP would still be required in order to document them, since CC-AA-112 requires that the change be controlled per the appropriate station procedure, even for Controlled Exclusions. B. Incorrect - Per CC-AA-112, disabling an annunciator is not considered a Controlled Exclusion. The second part of the answer is incorrect, since a TCCP will be required. Plausible because many types of configuration changes fall under Controlled Exclusions, and they would not require a TCCP to be issued in order to perform them. C. Incorrect - Per CC-AA-112, this is a TCC, and will require a TCCP to be developed in order to implement it. The stem of the question shows that it is not an MR90, it does not meet the requirements for a Controlled Exclusion, and it is not Procedurally Controlled TCC (PCTCC). Plausible because this is a TCC. Some types of TCC do not require a TCCP to be issued in order to perform them (for example, actions being done under a Clearance Order). D. Correct - Per CC-AA-112, this is a TCC, and a TCCP will be required in order to disable the annunciator. The stem of the question shows that it is not an MR90, it does not meet the requirements for a Controlled Exclusion, and it is not Procedurally Controlled TCC (PCTCC). **REQUIRED REFERENCES: NONE**

None

92 **ID: 27584** **Points: 1.00**

The CREV System is designed to maintain a habitable environment in the Control Room Envelope for a ___(1)___ continuous occupancy after a DBA without exceeding ___(2)___ total effective dose equivalent.

A. (1) 7 days
(2) 10 rem

B. (1) 7 days
(2) 5 rem

C. (1) 30 days
(2) 10 rem

D. (1) 30 day
(2) 5 rem

Answer: D

Answer Explanation

Per Tech Spec Bases 3.7.4 the CREV System is designed to maintain a habitable environment in the Control Room Envelope for a 30 continuous occupancy after a DBA without exceeding 5 rem total effective dose equivalent.

Question 92 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	3.00
System ID:	27584
User-Defined ID:	27584
Cross Reference Number:	
Topic:	92 - 290003 G2.2.25
Num Field 1:	
Num Field 2:	
Text Field:	

Comments:	Objective: DRE288LN003.7 References: T.S. Bases 3.7.4, UFSAR 6.4.2.5, UFSAR Table 15.6-14 K/A: 290003 G.2.2.25 -- / 4.2 K/A: Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits: Control Room Ventilation. Safety Function: 9 CFR: 41.5, 41.7/43.2 PRA: No Level: Memory Pedigree: New History: N/A **SRO Only Criteria: 10CFR55.43(b)(2) Facility operating limitations in the technical specificatons and their bases.** Explanations: A. Incorrect - Per Tech Spec Bases 3.7.4 the CREV System is designed to maintain a habitable environment in the Control Room Envelope for a 30 continuous occupancy after a DBA without exceeding 5 rem total effective dose equivalent. Plausible because 7 days is the Tech Spec LCO with CREVs Inop and 10 rem is the emergency limit when protecting valuable property. B. Incorrect - Per Tech Spec Bases 3.7.4 the CREV System is designed to maintain a habitable environment in the Control Room Envelope for a 30 continuous occupancy after a DBA without exceeding 5 rem total effective dose equivalent. Plausible because 7 days is the T.S. LCO with CREVs Inop and Part 2 is correct. C. Per Tech Spec Bases 3.7.4 the CREV System is designed to maintain a habitable environment in the Control Room Envelope for a 30 continuous occupancy after a DBA without exceeding 5 rem total effective dose equivalent. Plausible because part 1 is correct and 10 rem is the emergency limit when protecting valuable property. D. Correct - Per Tech Spec Bases 3.7.4 the CREV System is designed to maintain a habitable environment in the Control Room Envelope for a 30 continuous occupancy after a DBA without exceeding 5 rem total effective dose equivalent. **Required References: None**

None

93	ID: 27604	Points: 1.00

Unit 2 is in a refueling outage with the Mode Switch in **STARTUP**

- Control rod M-6 is selected and is being withdrawn
- The refuel bridge is over the refuel pool with the grapple **FULLY UP** and **UNLOADED**
- The “over the core” limit switches have **NOT** been disabled
- Fuel Handlers are working on the refuel floor when they accidently activate one of the “over the core” limit switches

Further travel of the refuel bridge towards the core ___(1)___ be blocked, and a Rod Block ___(2)___ be generated.

A. (1) will
(2) will

B. (1) will
(2) will **NOT**

C. (1) will **NOT**
(2) will

D. (1) will **NOT**
(2) will **NOT**

Answer: A

Answer Explanation

With the reactor mode switch in STARTUP, the over the core limit switches enabled and at least one of them activated, movement of the refuel bridge toward the core would be blocked With the reactor mode switch in STARTUP, the over the core limit switches enabled and at least one of them activated, a Rod Block would be generated.

Question 93 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	27604
User-Defined ID:	27604
Cross Reference Number:	
Topic:	93 - 234000 A3.02
Num Field 1:	
Num Field 2:	
Text Field:	

<table>
<tr>
<td>Comments:</td>
<td>
Objective: 201F-03-03
References: DOS 0800-01 Step 75
K/A: 234000 A3.02 3.1 / 3.7
K/A: Ability to monitor automatic operations of the FUEL HANDLING EQUIPMENT including, Interlock operations.
Safety Function: 8
CFR: 41.7/45.7
PRA: No
Level: High
Pedigree: New
History: N/A

SRO Criteria 10 CFR 55.43.(b).7 - Fuel handling facility and procedures

A. CORRECT – (1) With the reactor mode switch in STARTUP, the over the core limit switches enabled and at least one of them activated, movement of the refuel bridge toward the core would be blocked (2) With the reactor mode switch in STARTUP, the over the core limit switches enabled and at least one of them activated, a Rod Block would be generated. .
B. INCORRECT – (1) The first part is correct (2) With the reactor mode switch in STARTUP, the over the core limit switches enabled and at least one of them activated, a Rod Block would be generated..
Plausibility (1) The first part of the answer is correct (2) This answer would be true if the mode switch were in REFUEL.
C. INCORRECT – (1) With the reactor mode switch in STARTUP, the over the core limit switches enabled and at least one of them activated, movement of the refuel bridge toward the core would be blocked (2) The second part of the answer is correct.
Plausibility (1) The answer would be correct if the mode switch were in REFUEL. (2) The second half of the answer is correct.
D. INCORRECT - (1) With the reactor mode switch in STARTUP, the over the core limit switches enabled and at least one of them activated, movement of the refuel bridge toward the core would be blocked (2) With the reactor mode switch in STARTUP, the over the core limit switches enabled and at least one of them activated, a Rod Block would be generated. .
Plausibility (1) This answer would be true if the mode switch were in REFUEL. (2) This answer would be true if the mode switch were in REFUEL.

Required References: NONE
</td>
</tr>
</table>

None

94	**ID: 27036**	**Points: 1.00**

Unit 2 was operating at 30% power when a transient occurred, resulting in the following timeline events:

- 11:00 Annunciator 902-7 H-3, TURB VACUUM LO alarmed at 24.0 inHg vac.
- 11:10 Annunciator 902-5 F-5, CONDR VACUUM LO alarmed at 23.0 inHg vac.

Assuming vacuum continues to degrade at the same rate, what is the **LATEST** time the action below must be taken by, in order to prevent an automatic SCRAM?

A. At 11:15, trip the Main Turbine per DOA 3300-02, LOSS OF CONDENSER VACUUM

B. At 11:15, insert a manual scram per OP-DR-103-102-1002, STRATEGIES FOR SUCCESSFUL TRANSIENT MITIGATION

C. At 11:40, trip the Main Turbine per DOA 3300-02, LOSS OF CONDENSER VACUUM

D. At 11:40, insert a manual scram per OP-DR-103-102-1002, STRATEGIES FOR SUCCESSFUL TRANSIENT MITIGATION

Answer: B

Answer Explanation

OP-DR-103-102-1002 directs setting a scram contingency based on condenser vacuum of 22.5 inHg. Based on the alarms in the stem and the time values, the candidate must determine the rate of change for condenser vacuum is 1 inHg every 10 minutes. Based on this knowledge the candidate must recall and apply the OP-DR-103-102-1002 scram contingency requirements.

Question 94 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	4.00
System ID:	27036
User-Defined ID:	27036
Cross Reference Number:	
Topic:	94 - 295002 A2.01
Num Field 1:	
Num Field 2:	
Text Field:	

Comments:	Objective: DRE245LN001.10 Reference: DOA 5600-01, DAN 902(3)-5 F-5, DAN 902(3)-7 H-3, OP-DR-103-102-1002 K/A: 295002 A2.01 --- / 3.1 K/A: Ability to determine and/or interpret the following as they apply LOSS OF MAIN CONDENSER VACUUM: Condenser vacuum/absolute pressure. CFR: 41.10/43.5/45.13 Safety Function: 3 PRA: No Level: High Pedigree: Bank History: 15-1 NRC **SRO Only Criteria: 10CFR55.43(b)(5) - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal and emergency situation.Conditions and limitations in the facility license.** Explanation: A. Incorrect - Although a reactor scram via turbine trip will not occur due to 38.5% scram being bypassed, a RX scram is still going to occur based on low vacuum. DOA 3300-02 directs inserting a manual scram if loss of vacuum is imminent. The time is correct to reach OP-DR-103-102-1002 low vacuum scram contingency. B. Correct - OP-DR-103-102-1002 directs setting a scram contingency based on condenser vacuum of 22.5 inHg. Based on the alarms in the stem and the time values, the candidate must determine the rate of change for condenser vacuum is 1 inHg every 10 minutes. Based on this knowledge the candidate must recall and apply the OP-DR-103-102-1002 scram contingency requirements. C. Incorrect - Although a reactor scram via turbine trip will not occur due to 38.5% scram being bypassed, a RX scram is still going to occur based on low vacuum. DOA 3300-02 directs inserting a manual scram if loss of vacuum is imminent. Time is incorrect. Plausible because this time corresponds to the time based on rate of change to turbine trip on low vacuum (20 inHg) D. Incorrect - Plausible because this time corresponds to the time based on rate of change to turbine trip on low vacuum (20 inHg) **REQUIRED REFERENCES: None.**

None

95	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

Answer Explanation

The average (bulk) Torus temperature is 110.9ºF (which is above the DEOP 200-1 entry condition of 95ºF). The required actions for this is to start all available Torus Cooling and also when average Torus temperature is above 110ºF, 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.

Question 95 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	4
Difficulty:	4.00
System ID:	22439
User-Defined ID:	22439
Cross Reference Number:	
Topic:	95 - 295013 G2.1.07
Num Field 1:	
Num Field 2:	
Text Field:	

Comments:	Objective: 29502LK011 Reference: DEOP 200-1, DAN 902(3)-4 A-18 K/A: 295013.G2.1.07 --/4.7 K/A: High Suppression Pool Temperature: Ability to evaluate plant performance and make operational judgements based on operating characteristics, reactor behavior, and instrument interpretation. Safety Function: 5 CFR: 41.5/43.5/45.12/45.13 PRA: No Level: High Pedigree: Bank History: 18-1 NRC, 09-1 NRC **SRO Only Criteria: 10CFR55.43(b)(5) - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal and emergency situation.Conditions and limitations in the facility license.** Explanation: A. Incorrect - The torus bulk temperature exceeds the DEOP 200-1 entry condition. Plausible because part one is correct. Part two is plausible because 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^{o}F (which is above the DEOP 200-1 entry condition of 95^{o}F). The required actions for this is to start all available Torus Cooling and also when average Torus temperature is above 110^{o}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. **REQUIRED REFERENCES: DEOP 200-1 with the entry conditions blanked out.**

Question 95 Table-Item Links

General Question Data - Site Ownership

Dresden

General Question Data - Ops Program

Senior Reactor Operator

96 **ID: 27588** **Points: 1.00**

Unit 2 is at 40% power.

DOS 0250-02, FULL CLOSURE TIMING AND EXERCISING OF MAIN STEAM ISOLATION VALVES, is being performed.

Inboard MSIV AO 2-0203-1A has a stroke time of 5.2 seconds.

This requires entry into a ___(1)___ hour clock,

The bases for the PCIS function of MSIVs is to ___(2)___ during an accident.

A. (1) 4
(2) limit fission product release

B. (1) 4
(2) prevent excessive RPV cooldown

C. (1) 8
(2) limit fission product release

D. (1) 8
(2) prevent excessive RPV cooldown

Answer: C

Answer Explanation

Per DOS 0250-02 with the unit on-line MSIV timing must be 3-5 seconds. If Acceptance Criteria is not met then T.S. 3.6.1.3 requires an 8 hour clock for MSIV's. Per the bases of SR 3.6.1.3.6 the isolation time test ensures that the MSIV will isolate in a time period that does not exceed the times assumed in the DBA and transient analyses. This ensure calculated radiological consequences remain within 10CFR 50.67 limits in a DBA LOCA.

Question 96 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	5
Difficulty:	4.00
System ID:	27588
User-Defined ID:	27588
Cross Reference Number:	
Topic:	96 - 223002 G2.1.32
Num Field 1:	
Num Field 2:	
Text Field:	

Comments:	Objective: DRE239LN001.07 Reference: DOS 0250-02, T.S. 3.6.1.3, T.S. Bases 3.6.1.3 K/A: 223002 G2.1.32 -- / 4.0 K/A: Ability to explain and apply system limits and precautions: Primary Containment Isolation/Nuclear Steam Supply Shutoff. Safety Function: 5 CFR: 41.10/43.2/45.12 PRA: Yes Level: High Pedigree: New History: N/A **SRO Only Criteria: 10CFR55.43(b)(2) Facility operating limitations in the technical specifications and their bases.** Explanations: A. Incorrect – (1) MSIV operability is an 8 hour clock. (2) The second part of the answer is correct. Plausible because (1) All other PCIV's have a 4 hour clock. (2) The second part of the answer is correct. B. Incorrect – (1) MSIV operability is an 8 hour clock. Plausible because (1) All other PCIV's have a 4 hour clock. (2) Closing MSIVs is a strategy used in OP-DR-103-102-1002,STRATEGIES FOR SUCCESSFUL TRANSIENT MITIGATION, to prevent excessive cooldown. C. Correct - Per DOS 0250-02 with the unit on-line MSIV timing must be 3-5 seconds. If Acceptance Criteria is not met then T.S. 3.6.1.3 requires an 8 hour clock for MSIV's. Per the Background of SR 3.6.1.3 The function of the PCIVs, in combination with other accident mitigation systems, is to limit fission product release during and following postulated Design Basis Accidents (DBAs) to within limits. D. Incorrect – (1) The first part of the answer is correct. (2) Per the Background of SR 3.6.1.3 The function of the PCIVs, in combination with other accident mitigation systems, is to limit fission product release during and following postulated Design Basis Accidents (DBAs) to within limits. Plausible because (1) the first part of the answer is correct. (2) Closing MSIVs is a strategy used in OP-DR-103-102-1002,STRATEGIES FOR SUCCESSFUL TRANSIENT MITIGATION, to prevent excessive cooldown. **Reference Provided: TS 3.6.1.3**

None

97 **ID: 27591** **Points: 1.00**

Unit 2 was operating at full power when a LOCA occurred in the Drywell. The SRO directed entry into DEOP 0400-02, EMERGENCY DEPRESSURIZATION:

Five minutes later:

- 2B ERV did NOT open
- Torus Bottom Pressure is 40 psig and slowly lowering
- DW pressure is 32 psig and slowly lowering
- RPV pressure is 105 psig and slowly lowering

Based on the conditions listed above, the RPV___(1)___considered depressurized

The SRO should ___(2)___.

A. (1) is
(2) **DIRECT** entry into DEOP 0500-07, ALTERNATE EMERGENCY DEPRESSURIZATION SYSTEMS

B. (1) is **NOT**
(2) **DIRECT** entry into DEOP 0500-07, ALTERNATE EMERGENCY DEPRESSURIZATION SYSTEMS

C. (1) is
(2) wait until the Shutdown Cooling interlock clears and further cooldown is required

D. (1) is **NOT**
(2) wait until the Shutdown Cooling Interlock clears and further cooldown is required

Answer: B

Answer Explanation

RPV pressure minus drywell pressure is not less than 67 psid, therefore the RPV is not depressurized. With the RPV not depressurized, and one ERV failing to open, DEOP 0400-02 requires that DEOP 0500-07 be entered to utilize additional alternate emergency depressurization systems to depressurize the RPV.

Question 97 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	4.00
System ID:	27591
User-Defined ID:	27591
Cross Reference Number:	
Topic:	97 - 295025 A2.05
Num Field 1:	
Num Field 2:	
Text Field:	

Comments:	
	Objective: 29502LK030 Reference: DEOP 0400-02, EOP-DEOP TB K/A: 295025 A2.05 3.4/3.6 K/A: Ability to determine and/or interpret the following as they apply to HIGH REACTOR PRESSURE: Decay Heat Generation. Safety Function: 3 CFR: 41.10/43.5/45.13 PRA: Yes Level: High Pedigree: New History: N/A **SRO Only Criteria: 10CFR55.43(b)(5) - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal and emergency situation.Conditions and limitations in the facility license.** K/A Justification: Although reactor pressure is not what would normally be considered a “HIGH” pressure condition, it is too high with respect to the requirements of DEOP 0400-02, with one ERV failed closed. In that case, RPV pressure must be less than 67 psid above above DW Pressure to meet the intended condition (i.e. depressurized). Explanations: A. Incorrect – (1) RPV pressure minus drywell pressure is not less than 67 psid. Therefore, the reactor is not considered depressurized. (2) the second part of the answer is correct. Plausibility: (1) RPV pressure minus DW pressure is <67 psid. If the SRO calculates the wrong value, they would believe that the RPV is depressurized. (2) The second part of the distractor is correct. B. Correct – (1) RPV pressure minus drywell pressure is not less than 67 psid, therefore the RPV is not depressurized. (2) With the RPV not depressurized, and one ERV failing to open, DEOP 0400-02 requires that DEOP 0500-07 be entered to utilize additional alternate emergency depressurization systems to depressurize the RPV. C. Incorrect – (1) RPV pressure minus drywell pressure is not less than 67 psid. Therefore, the reactor is not considered depressurized. (2) with the reactor not fully depressurized, the SRO should DIRECT entry into DEOP 0500-07 to depressurize the RPV. Plausibility: (1) RPV pressure minus DW pressure is <67 psid, but it is required that RPV pressure minus Torus bottom pressure be less than 67 psid. If the SRO calculates the wrong value, they would believe that the RPV is depressurized. (2) If the RPV was considered depressurized, then waiting until the shutdown cooling interlock clears (which occurs automatically when pressure drops low enough), and further cooldown is required would be a correct response. D. Incorrect – (1) The first part of the answer is correct. 2) With the reactor not fully depressurized, the SRO should DIRECT entry into DEOP 0500-07 to depressurize the RPV. Plausibility: (1) the first part of this distractor is correct. (2) DEOP 0100 directs waiting for the shutdown cooling interlock to clear (which occurs automatically when pressure drops low enough), and further cooldown being required prior to continuing with cooldown. The alternate depressurization systems used for this, in the case where SDC does not work, include many of the systems operated by DEOP 0500-07, and these could be used regardless of how many ERVs failed to open. **Required References: NONE**

98 **ID: 27592** **Points: 1.00**

Unit 2 was operating at 550 MWe.

THEN:

- Severe vibrations were reported coming from the Main Turbine
- The reactor was scrammed
- Turbine TRIP pushbuttons on the 902-7 panel were **UNSUCCESSFUL**
- Reverse power trip did not occur after reactor scram
- Breaker I-2, U2 250 VDC REACTOR BUILDING MCC #2B (MAIN FEED BKR), on 250 VDC MCC #3 trips open during the transient

The SRO will direct ___(1)___ to isolate the steam supply to the Main Turbine, and to control RPV pressure with ___(2)___?

A. (1) placing BOTH EHC pumps in PTL
(2) ADS valves

B. (1) placing BOTH EHC pumps in PTL
(2) Isolation Condenser

C. (1) shut the MSIVs and MSL drains
(2) ADS valves

D. (1) shut the MSIVs and MSL drains
(2) Isolation Condenser

Answer: A

Answer Explanation

(1) Per DOA 5600-01, if the 902-7 panel pushbuttons are not successful, and reverse power does not trip the turbine the EHC pumps should be placed in PTL.
(2) Per DOA 5600-01 immediate actions pressure control should be transitioned to IC or HPCI, but with I-2 on 250 VDC MCC #3 tripped open 250 VDC MCC 2A has no power which powers both IC and HPCI valves/pumps that are required for operation; the Unit Supervisor will then have to direct the ADS valves for pressure control IAW DEOP 100 pressure leg.

Question 98 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	4
Difficulty:	3.00
System ID:	27592
User-Defined ID:	27592
Cross Reference Number:	
Topic:	98 - 295005 G2.4.11
Num Field 1:	
Num Field 2:	
Text Field:	

Comments:	Objective: DRE245LN001.08 References: DOA 6000-01, DOA 5600-01, DGP 02-01, DGP 02-03 K/A: 295005 G2.4.11 4.0/4.2 K/A: Knowledge of abnormal condition procedures: Main Turbine Generator Trip. Safety Function: 3 CFR: 41.10/43.5/45.13 PRA: No Level: High Pedigree: New History: N/A Justification for SRO Only: By adding the DC breaker trip, the question moves beyond the immediate operator action and requires the SRO to direct another method to control pressure. **SRO Only Criteria: 10CFR55.43(b)(5) - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal and emergency situation. Conditions and limitations in the facility license.** Explanations: A. Correct - (1) Per DOA 5600-01, if the 902-7 panel pushbuttons are not successful, and reverse power does not trip the turbine the EHC pumps should be placed in PTL. (2) Per DOA 5600-01 immediate actions pressure control should be transitioned to IC or HPCI, but with I-2 on 250 VDC MCC #3 tripped open 250 VDC MCC 2A has no power which powers both IC and HPCI valves/pumps that are required for operation; the Unit Supervisor will then have to direct the ADS valves for pressure control IAW DEOP 100 pressure leg. B. Incorrect - (1) The first part is correct. (2) DOA 5600-01 directs the use of HPCI or Isolation condenser for the conditions listed in the stem, but no power to the 2-1301-3 the IC is not available. Plausible because (1) the first part of the answer is correct. (2) use of IC for pressure control is the next step in DOA 5600-01, but with no power to the 2-1301-3 the IC is not available. C. Incorrect - (1) Closing the MSIVs and MSL drains would isolate the steam supply to the Main Turbine but DOA 5600-01 directs keeping the MSIVs open and to maintain Main Condenser vacuum and steam seal pressure (2) the second part of the question is correct. Plausible because (1) These are the actions to stop an uncontrolled cooldown but without the bypass valves being available DOA 5600-01 directs maintaining Main Condenser vacuum and steam seal pressure for the acceptance of house loads (2) the second part of the question is correct. D. Incorrect - (1) Per DOA 5600-01, if the 902-7 panel pushbuttons are not successful, and reverse power does not trip the turbine the EHC pumps should be secured. (2) the second part of the question is correct Plausible because (1) These are the actions to stop an uncontrolled cooldown but without the bypass valves being available DOA 5600-01 directs maintaining Main Condenser vacuum and steam seal pressure for the acceptance of house loads (2) use of IC for pressure control is the next step in DOA 5600-01, but with no power to the 2-1301-3 the IC is not available. **Required References: NONE**

99 **ID: 27594** **Points: 1.00**

Unit 2 is operating at 80% power, with Reactor Recirc in the Master Manual mode.

- Reactor power and the 2B Recirc Pump ASD SPEED % indication began to lower without operator action
- The Unit Supervisor directed entry into DOA 0202-03, REACTOR RECIRCULATION SYSTEM FLOW CONTROL FAILURE
- The NSO momentarily placed 2B ASD SPEED HOLD switch 2-202-60-302B to HOLD at Panel 902-4

Following the transient, the following conditions exist.

- Jet Pump loop flow mismatch was determined to be 7.4%
- Total Core Flow the 902-5 panel is 69.5 Mlbm

Based on the above, **ENTRY** into TS 3.4.1 ___(1)___ required.

The Unit Supervisor will set a post SCRAM contingency to momentarily select **RESET** at ASD 2B SPEED HOLD/RESET switch on Panel 902-4 **AND** to ___(2)___.

A. (1) is
(2) **VERIFY** the recirc pump runs back to minimum speed or **TRIP** the associated recirc pump

B. (1) is
(2) **VERIFY** the recirc pump runs back to minimum speed **ONLY**

C. (1) is **NOT**
(2) **VERIFY** the recirc pump runs back to minimum speed or **TRIP** the associated recirc pump

D. (1) is **NOT**
(2) **VERIFY** the recirc pump runs back to minimum speed **ONLY**

Answer: A

Answer Explanation

(1) With greater than a 5% mismatch between the recirc pump flows, and flow above 70% of rated flow (i.e. greater than 68.6 Mlbm), entry into TS 3.4.1 is required. (2) Per DOP 0202-16, the Unit Supervisor would set a post SCRAM contingency to reset the speed hold for the affected pump, verify that the pump runs back to minimum AND to trip the affected pump if it fails to runback to 30%.

Question 99 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	5
Difficulty:	4.00
System ID:	27594
User-Defined ID:	27594
Cross Reference Number:	
Topic:	99 - 202002 A2.04
Num Field 1:	
Num Field 2:	
Text Field:	

Comments:	Objective: 202LN001.7 Reference: DOP 0202-16, TS 3.4.1, DGP 02-03, DAN 902(3)-4 C-1, DTS 81-57 K/A: 202002 A2.04 3.0/3.2 K/A: Ability to (a) predict the impacts of the following on the RECIRCULATION FLOW CONTROL SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Recirculation pump speed mismatch between loops: Plant-Specific CFR: 41.5/43.2/43.5/45.6 Safety Function 1 PRA: No Level: High Pedigree: New History: N/A **SRO Only Criteria: 10CFR55.43(b)(2) Facility Operating Limitations in the technical specifications and their bases. SRO level due to 10 CFR 55:43 criteria (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal and emergency situations.** Explanations: A. Correct – (1) With greater than a 5% mismatch between the recirc pump flows, and flow above 70% of rated flow (i.e. greater than 68.6 Mlbm), entry into TS 3.4.1 is required. (2) Per DOP 0202-16, the Unit Supervisor would set a post SCRAM contingency to reset the speed hold for the affected pump, verify that the pump runs back to minimum AND to trip the affected pump if it fails to runback to 30%. B. Incorrect - (1) This portion of the answer is correct (2) Per DOP 0202-16, the Unit Supervisor would set a post SCRAM contingency to reset the speed hold for the affected pump, verify that the pump runs back to minimum AND to trip the affected pump if it fails to runback to 30%.. Plausible because (1) This portion of the answer is correct. (2) Other portions of DOP 0202-16 allow performing either tripping of a recirc pump OR resetting the speed hold switch to allow a runback signal to 68% to occur. In this case, the contingency is for post SCRAM actions, and the contingency would contain both actions. C. Incorrect – (1) With greater than a 5% mismatch between the recirc pumps, and flow above 70% of rated flow (i.e. greater than 68.6 Mlbm), entry into TS 3.4.1 is required. (2) This portion of the answer is correct. Plausible because (1) if the candidate does not recognize that the requirement is 70% of RATED flow (i.e. 70% of 98 Mlbm), then they will believe the higher limits of TS 3.4.1 for flows less than 70% of rated flow. If Total Core Flow were less than 68.6 Mlbm, TS entry would not be required. (2) This portion of the answer is correct. D. Incorrect – (1) With greater than a 5% mismatch between the recirc pumps, and flow above 70% of rated flow (i.e. greater than 68.6 Mlbm), entry into TS 3.4.1 is required. (2) Per DOP 0202-16, the Unit Supervisor would set a post SCRAM contingency to reset the speed hold for the affected pump, verify that the pump runs back to minimum AND to trip the affected pump if it fails to runback to 30%. Plausibly because (1)) if the candidate does not recognize that the requirement is 70% of RATED flow (i.e. 70% of 98 Mlbm), then they will believe the higher limits of TS 3.4.1 for flows less than 70% of rated flow. If Total Core Flow were less than 68.6 Mlbm, TS entry would not be required. (2) Other actions in DOP 0202-16 allow performing either tripping of a recirc pump OR resetting the speed hold switch to allow a runback signal to 68% to occur. In this case, the contingency is for post SCRAM actions, and the contingency would contain both actions **Required Reference: T.S. 3.4.1**

100 **ID: 27603** **Points: 1.00**

Unit 2 was operating at 20% power when a transient occurred, resulting in the following sequence of events:

- Annunciator 902-7 G-3, TURB BYP VLV OPEN, alarmed.
- RPV water level is 22 inches and trending down.
- Main Steam line pressure is 800 psig.
- An automatic scram occurred, reactor power is 4%.
- RPV water level is 10 inches and trending down.

The Reactor scrammed on___(1)___.

The Unit Supervisor is required to direct entering___(2)___.

A. (1) RPV water level;
(2) DGP 2-3, DEOP 100, DEOP 400-5

B. (1) Main Steam Isolation Valves closure;
(2) DGP 2-3, DEOP 100, DEOP 400-5

C. (1) RPV water level;
(2) DGP 2-3, AND DEOP 400-5 **ONLY**

D. (1) Main Steam Isolation Valves closure;
(2) DGP 2-3, AND DEOP 400-5 **ONLY**

Answer: B

Answer Explanation

When MSL pressure dropped below 827 psig and with the mode switch is in RUN, a scram will occur due to 9.5% Main Steam Isolation Valves closure. DGP 2-3 is entered for the SCRAM and in the contingency section requires entry into DEOP 100 and DEOP 400-5 with all rods not in.

Question 100 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	5
Difficulty:	3.00
System ID:	27603
User-Defined ID:	22694
Cross Reference Number:	
Topic:	100 - 295006.A2.06
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	Objective: DRE245LN001.06 Reference: DOA 5650-05; DAN 902(3)-5 D-4; T.S 3.3.1.1., 3.3.6.1; DGP 02-03 K/A: 295006.A2.06 3.5 / 3.8 K/A: Ability to determine and/or interpret the following as the apply to SCRAM: cause of reactor SCRAM. Safety Function: 1 CFR: 41.10/43.5/45.13 PRA: No Level: High Pedigree: New History: N/A **SRO Only Criteria: 10CFR55.43(b)(5) - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.** Explanation: A. Incorrect - Reactor water level is below normal but has not met the scram DEOP 100 threshold. Part 1 is plausible because the reactor water level is below the low level setpoint. Part 2 is correct. B. Correct - When MSL pressure dropped below 827 psig and with the mode switch is in RUN, a scram will occur due to 9.5% Main Steam Isolation Valves closure. DGP 2-3 is entered for the SCRAM and in the contingency section requires entry into DEOP 100 and DEOP 400-5 with all rods not in. C. Incorrect - Reactor water level is below normal but has not met the scram threshold. Plausible because the reactor water level is below the low level setpoint but has not reached the threshold for entry into DEOP 100. D. Incorrect - The reactor did Scram on MSIV closure but DEOP 100 entry conditions have not been met. Plausible because Part 1 is correct. Part 2 is partially correct but DEOP 100 should not be entered at this point. **REQUIRED REFERENCES: None.**

Question 100 Table-Item Links

General Question Data - Site Ownership

Dresden

General Question Data - Ops Program

Senior Reactor Operator