

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

1

ID: 24229

Points: 1.00

Unit 2 was at rated power when a control power fuse blew inside a Control Room Panel. The Shift Manager has declared that the event did NOT create an emergency situation. A replacement fuse has been located and determined to be like-for like.

IAW CC-AA-206, Fuse Control, which of the following is correct?

- A. The Operator may NOT install the fuse.
- B. ONLY EMD may install the fuse with NO further engineering evaluation.
- C. The Operator may install the fuse with NO further engineering evaluation.
- D. The Operator may install the fuse ONLY after the fuse is evaluated by the Fuse Engineer.

Answer: C

Answer Explanation

Since the plant was at rated power when a control room fuse blew and the new fuse is like-for like with the old fuse IAW the reference, Operations may install the fuse with no further engineering evaluation required.

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

Question 1 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:	24229
User-Defined ID:	24229
Cross Reference Number:	
Topic:	01 - Generic 2.2.14
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	<p>Objective: 29900LK081 Reference: CC-AA-206 K/A: Generic 2.2.14 3.9 / 4.3 K/A: Knowledge of the process for controlling equipment configuration or status. CFR: 41.10 Safety Function: N/A Level: Memory Pedigree: Bank History: 14-1 NRC, 18-1 NRC</p> <p>Explanation:</p> <p>A. Incorrect - This is plausible because the maintenance department is assigned the responsibility to install fuses per Work Orders and Work Requests, but this does not preclude operations department from replacing fuses in plant equipment.</p> <p>B. Incorrect - This is plausible because a maintenance/engineering evaluation is required to verify fuses not assigned an EPN in black box systems (AVR, DEHC). Control room panel fuses are assigned EPNs and identified in station drawings.</p> <p>C. Correct - Since the plant was at rated power when a control room fuse blew and the new fuse is like-for like with the old fuse IAW the reference, Operations may install the fuse with no further engineering evaluation required.</p> <p>D. Incorrect - This is plausible because this would be correct for any non-emergency situation if the fuse were not like for like.</p> <p>REQUIRED REFERENCES: None</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

2

ID: 27452

Points: 1.00

What is the Technical Specification **LIMIT** for Average Drywell Air Temperature in MODE 1?

- A. $\leq 105^{\circ}\text{F}$
- B. $\leq 110^{\circ}\text{F}$
- C. $\leq 150^{\circ}\text{F}$
- D. $\leq 281^{\circ}\text{F}$

Answer: C

Answer Explanation

This is the TS limit for Drywell Air Temperature.

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:	27452
User-Defined ID:	27452
Cross Reference Number:	
Topic:	02 - Generic 2.2.38
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	<p>Objective: 223LN001.07.a Reference: TS 3.6.1.5; Appendix A K/A: G.2.2.38 3.6 / 4.5 K/A: Knowledge of conditions and limitations in the facility license. CFR: 41.7, 41.10, 43.1 Safety Function: 5 Pedigree: Bank Level: Memory History: 18-1 NRC</p> <p>Explanation:</p> <p>A. Incorrect - This is the TS limit for Torus water temperature when performing testing that adds heat to the Torus.</p> <p>B. Incorrect - 110 is plausible since this is the Torus temperature that would require boron injection during an ATWS.</p> <p>C. Correct - This is the TS limit for Drywell Air Temperature.</p> <p>D. Incorrect - 281 is plausible since this is the primary containment limit per DEOP 200-1.</p> <p>REQUIRED REFERENCES: None</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

3

ID: 27956

Points: 1.00

DSSP 0100-CR, Control Room Evacuation, is in progress.

The 2-1301-1, U2 ISOLATION CONDENSER RX OUTLET ISOL VLV, has gone closed.

How would the operator re-open the 2-1301-1 valve?

Proceed to ____ (1) ____ and open the valve at the ____ (2) ____.

- A. (1) Reactor Building 3rd Floor
(2) 2202-76, Unit 2 Isolation Condenser Valves Control Panel
- B. (1) Reactor Building 3rd Floor
(2) MO 2-1301-1, U2 ISOL COND RX OUTLET ISOL VLV Pushbutton Control Station
- C. (1) 2/3 EDG Room
(2) 2202-76, Unit 2 Isolation Condenser Valves Control Panel
- D. (1) 2/3 EDG Room
(2) MO 2-1301-1, U2 ISOL COND RX OUTLET ISOL VLV Pushbutton Control Station

Answer: C

Answer Explanation

Per DSSP 0100-CR, the 2-1301-01 is operated from the 2202-76, U2 Isolation Condenser Valves Control Panel located in the 2/3 EDG Room.

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

Question 3 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:	27956
User-Defined ID:	349050
Cross Reference Number:	LI
Topic:	03 - 295016.A1.09
Num Field 1:	0.00
Num Field 2:	0.00
Text Field:	
Comments:	<p>Objective: DRE07LN001.05 Reference: DSSP 0100-CR K/A: 295016.A1.09 4.0 / -- K/A: Ability to operate and/or monitor the following as they apply to CONTROL ROOM ABANDONMENT: Isolation/emergency condenser(s): Plant-Specific CFR: 41.7 / 45.6 Level: Memory Safety Function: 7 Perigee: Bank History: 03-1 NRC</p> <p>Explanation: A. Incorrect - The 2-1301-01 valve is operated from the 2/3 EDG Room. Plausible because the 2-1301-3 is operated from the 3rd Floor of the Reactor Building. B. Incorrect - The 2-1301-01 valve is operated from the 2/3 EDG Room and does not have a local pushbutton control station. Plausible because the 2-1301-3 is operated from the 3rd Floor of the Reactor Building and other Isolation Condenser valves have a local pushbutton station. C. Correct - Per DSSP 0100-CR, the 2-1301-01 is operated from the 2202-76, U2 Isolation Condenser Valves Control Panel located in the 2/3 EDG Room. D. Incorrect - The 2-1301-01 valve does not have a local pushbutton station. Plausible because other Isolation Condenser valves have a local pushbutton station.</p> <p>REQUIRED REFERENCES: None</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

4

ID: 27898

Points: 1.00

Units 2 and 3 are operating at 100% power.

The 2/3 RBCCW pump is OOS for motor replacement.

- 2A RBCCW pump has tripped.
- 2B RBCCW pump amps are rising **SLOWLY**

What actions are required?

- A. Valve **IN** RBCCW to the 2/3 RBCCW heat exchanger.
- B. Valve **OUT** RBCCW to the 2A RBCCW heat exchanger.
- C. Insert a manual scram **AND** trip Recirc pumps within 1 minute **ONLY**
- D. Insert a manual scram **AND** trip Recirc pumps within 1 minute **AND** isolate RBCCW to the Drywell.

Answer: B

Answer Explanation

If RBCCW system is degrading with 2 heat exchangers in service, the correct action is to remove 1 RBCCW heat exchanger from service.

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

Question 4 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:	27898
User-Defined ID:	27898
Cross Reference Number:	
Topic:	04 - 295018.A2.04
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	<p><QQ 26983(1411)><<Objective: DRE208LN001.08 Reference: DOA 3700-01, DOP 3700-02 K/A: 295018.A2.04 2.9/2.9 K/A: Partial or Complete Loss of Component Cooling Water: Ability to determine and/or interpret the following as they apply to Partial or Total Loss of CCW: System flow CFR: 41.10 Safety Function: 8 Level: High Pedigree: Bank History: 18-1 NRC</p> <p>Explanation:</p> <p>A. Incorrect - RBCCW to 2A Heat Exchanger needs to be valved out and operate with one heat exchanger. Plausible if only one heat exchanger was in service and a TCV or RBCCW Heat Exchanger was not functioning.</p> <p>B. Correct - This action is dependent on determining that the RBCCW system is cavitating due to having too much flow through a single pump, with 2 heat exchangers in service, the correct action is to remove 1 RBCCW heat exchanger from service.</p> <p>C. Incorrect - Operating with one heat exchanger instead of two will restore the RBCCW system. Plausible if RBCCW flow was lost.</p> <p>D. Incorrect - RBCCW to 2A Heat Exchanger needs to be valved out and operate with one heat exchanger. Plausible if LOCA had occurred concurrent with a loss of RBCCW.</p> <p>Note: K/A Justification. The K/A is being evaluated due to the fact that the candidate must identify that the cause for the indications is the fact that with 2 Hx in service and only 1 pump running the system has too much flow for the one running RBCCW pump, and the pump will cavitate and require the actions in the answer to be taken.</p> <p>REQUIRED REFERENCES: None.>></p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

5

ID: 27985

Points: 1.00

Unit 2 is operating at rated power.

- Annunciator DAN 923-1 E-6, 2B INST AIR DRYER TROUBLE, alarms.
- Pressure downstream of 2B IA dryer is 55 psig.

AO 2-4799-587, 2B INST AIR DRYER AO BYP, will be ____ (1) ____ open(ed).

Per DAN 923-1 E-6 2B INST AIR DRYER TROUBLE when the dryer trouble has been corrected, the bypass valve will/must be ____ (2) ____ reset.

- A. (1) Manually
(2) Manually
- B. (1) Manually
(2) Automatically
- C. (1) Automatically
(2) Manually
- D. (1) Automatically
(2) Automatically

Answer: C

Answer Explanation

Per DAN 923-1 E-6 2B INST AIR DRYER TROUBLE, If <60 psig downstream of 2B IA Dryer, Then AO 2-4799-587, 2B INST AIR DRYER AO BYP, opens. When dryer trouble has been correct, Then close AO 2-4799-587, 2B INST AIR DRYER AO BYP by locally depressing AUTO/BYPASS ALARM/MANUAL RESET.

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

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:	3.00
System ID:	27985
User-Defined ID:	27985
Cross Reference Number:	
Topic:	05 - 300000.A2.01
Num Field 1:	0.00
Num Field 2:	0.00
Text Field:	
Comments:	<p>Objective: DRE278LN001.09 Reference: DAN 923-1 E-6 K/A: 300000.A2.01 2.9/-- K/A: Ability to predict the impacts of the following on the instrument air; and based on those predictions, use procedures to correct, control or mitigate the consequences of those abnormal conditions or operations: Air dryer and filter malfunctions.</p> <p>Safety Function: 8 CFR: 41.5 PRA: No Level: Memory Pedigree: New History: N/A</p> <p>Explanations:</p> <p>A. Incorrect - the bypass valve opens automatically when pressure is less than 60 psig downstream of the air dryer. Plausible many systems have to manually bypassed when low flow is present. Part 2 is correct.</p> <p>B. Incorrect - the bypass valve opens automatically when pressure is less than 60 psig downstream of the air dryer. Plausible many systems have to manually bypassed when low flow is present. Dresden has systems that reset automatically when setpoint thresholds are met.</p> <p>C. Correct - Per DAN 923-1 E-6 2B INST AIR DRYER TROUBLE, If <60 psig downstream of 2B IA Dryer, Then AO 2-4799-587, 2B INST AIR DRYER AO BYP, opens. When dryer trouble has been corrected, Then close AO 2-4799-587, 2B INST AIR DRYER AO BYP by locally depressing AUTO/BYPASS ALARM/MANUAL RESET.</p> <p>D. Incorrect - the bypass valve must be manually reset after condition clears. Plausible because part 1 is correct. Part 2 is plausible because Dresden has systems that reset automatically when setpoint thresholds are met.</p> <p>REQUIRED REFERENCES: None.</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

6

ID: 27936

Points: 1.00

An electrical ATWS has occurred on Unit 2.

The NSO will ____ (1) ____.

The **MINIMUM** electrical safety precautions required to perform this task are ____ (2) ____.

- A. (1) de-energize scram solenoids
(2) all metal removed, safety glasses, long sleeve shirt, and rubber gloves
- B. (1) de-energize scram solenoids
(2) safety glasses, long sleeve electrical safety coat, and rubber gloves
- C. (1) perform repeated scrams/resets
(2) all metal removed, safety glasses, long sleeve shirt, and rubber gloves
- D. (1) perform repeated scrams/resets
(2) safety glasses, long sleeve electrical safety coat, and rubber gloves

Answer: A

Answer Explanation

Per DEOP 400-5 - Scram solenoid fuses are the correct path for CRD insertion. Per SA-AA-129 - the voltage in the area being worked in for DEOP 500-5 the minimum PPE is all metal removed, long sleeve arc-rated FR shirt and FR pants or FR coverall, safety glasses and rubber gloves.

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

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:	4.00
System ID:	27936
User-Defined ID:	26971
Cross Reference Number:	
Topic:	06 - Generic 2.1.26
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	<p>Objective: 295L105 Reference: SA-AA-129, DEOP 400-5, DEOP 500-05 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). CFR: 41.10 PRA: No Level: High Safety Function: N/A Pedigree: New History: None</p> <p>Explanation:</p> <p>A. Correct - Per DEOP 400-5 - Scram solenoid fuses are the correct path for CRD insertion. Per SA-AA-129 - the voltage in the area being worked in for DEOP 500-5 the minimum PPE is all metal removed, long sleeve arc-rated FR shirt and FR pants or FR coverall, safety glasses and rubber gloves.</p> <p>B. Incorrect - Need to remove all metal. Plausible because pulling scram solenoid fuses is the correct action. .</p> <p>C. Incorrect - Repeated scrams/resets is not the correct action in an electrical ATWS. Plausible because the PPE is correct.</p> <p>D. Incorrect - Repeated scrams/resets is not the correct action in an electrical ATWS and all metal needs to be removed. Plausible because the NSO must determine the correct action for an electrical ATWS and determine the minimum PPE.</p> <p>REQUIRED REFERENCES: None.</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

7

ID: 27950

Points: 1.00

A call was received from an industrial plant informing the site that a toxic gas release has occurred and impact to Dresden is expected in 5 minutes.

Per DOA 0010-12, TOXIC GAS/CHEMICAL RELEASE FROM NEARBY CHEMICAL FACILITY, which of the following actions may be taken from memory?

- A. Place CRM AIR FLTR UNIT BOOSTER FANS to ON.
- B. Align Main Control Room Ventilation to PURGE MODE.
- C. Don MSA Air Packs AND activate the breathing air system.
- D. Isolate Main Control Room by placing CRM ISOL switch to ISOLATE.

Answer: D

Answer Explanation

DOA 0010-12 directs placing CRM ISOL switch to isolate given the conditions of the stem. This isolates the MCR HVAC from outside air.

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

Question 7 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:	27950
User-Defined ID:	27787
Cross Reference Number:	
Topic:	07 - 290003.K5.01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	<p>Objective: 29501LK078 Reference: DOA 0010-12 K/A: 290003.K.5.01 3.2/3.5 K/A: Knowledge of the operational implications of the following concepts as they apply to CONTROL ROOM HVAC: Airborne contamination (e.g. radiological, toxic gas, smoke) control. Control Room Ventilation. CFR: 41.5 Safety Function: 9 Level: Memory Pedigree: Bank History: 15-1 NRC</p> <p>Explanation:</p> <p>A. Incorrect - Plausible because this would be the correct action if there was toxic gas or smoke in the MCR. This maximizes the flow of outside air - not the desired outcome in this situation.</p> <p>B. Incorrect - Plausible because donning MSA packs is an immediate operator action for toxic gas in the MCR. This DOA is not executed with the conditions in the stem</p> <p>C. Correct - DOA 0010-12 directs placing CRM ISOL switch to isolate given the conditions of the stem. This isolates the MCR HVAC from outside air.</p> <p>D. Incorrect - Plausible because DOA 0010-12 directs placing CRM AIR FLTR UNIT BOOSTER FANS to OFF at 923-5 panel vs ON.</p> <p>REQUIRED REFERENCES: None.</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

8

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 (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.

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

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:	3.00
System ID:	24061
User-Defined ID:	24061
Cross Reference Number:	
Topic:	08 - 201003.A4.02
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	<p>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, 19-1 NRC</p> <p>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.</p> <p>REQUIRED REFERENCES: None.</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

9

ID: 27902

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 not arrested and no operator action is taken, the **EARLIEST** time at which the U1 Diesel Fire Pump 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.

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

Question 9 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:	27902
User-Defined ID:	27902
Cross Reference Number:	
Topic:	09 - 286000.A3.04
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	<p>Objective: DRE286LN001.11 Reference: DOA 3900-01, DAN XL3 82-30, DAN 901-2 H-8 K/A: 286000.A3.04 3.2/3.3 K/A: Ability to monitor automatic operations of the FIRE PROTECTION SYSTEM including: System initiation. CFR: 41.7 Safety Function: 8 Pedigree: New Level: High</p> <p>Explanation:</p> <p>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.</p> <p>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.</p> <p>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.</p> <p>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.</p> <p>REQUIRED REFERENCES: None.</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

10

ID: 27911

Points: 1.00

Unit 2 was operating at near rated power, when U2 125 VDC 2A-1 DIST PANEL de-energized.

Which of the following loads will have lost control power?

- A. U2 'B' EHC Pump.
- B. U2 'B' RBCCW Pump.
- C. U2 'B' Circulating Water Pump.
- D. U2 'C' Circulating Water Pump.

Answer: C

Answer Explanation

With 2A-1 Dist Panel becoming de-energized, Bus 23 loses its main source of control power (control power indications). 'B' Circ Water pump is powered from Bus 23 and would have lost indications.

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

Question 10 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:	27911
User-Defined ID:	27911
Cross Reference Number:	
Topic:	10 - 295004.K2.03
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	<p>Objective: DRE263LN002.12 Reference: DAN 902-8 F-1, DOP 6900-06, DOA 6900-T1, 12E-2303 Sh 2, 12E-2304, 12E-2305 K/A: 295004.K2.03 3.3/3.3 K/A: Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF D.C. POWER and the following: D.C. bus loads CFR: 41.7 Safety Function: 6 PRA: Yes Level: Memory Pedigree: Bank History: 2007 NRC, 2010 Cert, 2016 NRC</p> <p>Explanation: A. Incorrect - "B" EHC pump is powered from Bus 27. Control power to Bus 27 is 2B-2. Plausible because other components have 2A-1 for control power. B. Incorrect - "B" RBCCW pump is powered from Bus 24-1. Control power to Bus 24-1 is 2B-1. Plausible because other 2A RBCCW Pump control power is 2A-1. C. Correct - With 2A-1 Dist Panel becoming de-energized, Bus 23 loses its main source of control power (control power indications). 'B' Circ Water pump is powered from Bus 23 and would have lost indications. D. Incorrect - "C" Circ water pump is powered from Bus 24. Control power to Bus 24 is 2B-1. Plausible because reserve control power is 2A-1.</p> <p>REQUIRED REFERENCES: None.</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

11

ID: 27912

Points: 1.00

An accident has occurred at the station and you have volunteered to perform an evolution to protect valuable property. The dose rate in the area you will be entering is 30 Rem/hr.

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

- A. 2 minutes
- B. 9 minutes
- C. 19 minutes
- D. 49 minutes

Answer: C

Answer Explanation

The exposure limit for protecting valuable property is 10 TEDE. Based on 30 REM in the area, the stay time would be 20 minutes.

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

Question 11 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:	27912
User-Defined ID:	27912
Cross Reference Number:	
Topic:	11 - Generic 2.3.04
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	<p>Objective: 29900LK208 Reference: RP-AA-203, EP-AA-113 K/A: Generic.2.3.04 3.2 / 3.7 K/A: Knowledge of radiation exposure limits under normal or emergency conditions. CFR: 41.12 / 43.4 / 45.10 Level: High Pedigree: Bank History: 2010 NRC</p> <p>Explanation:</p> <p>A. Incorrect - 2 minutes would be correct if the candidate assumed the company yearly limit of 2 Rem.</p> <p>B. Incorrect - 9 minutes would be correct if the candidate assumed the federal yearly limit of 5 Rem.</p> <p>C. Correct - The exposure limit for protecting valuable property is 10 TEDE. Based on 30 REM in the area, the stay time would be 20 minutes.</p> <p>D. Incorrect - 49 minutes would be correct if the candidate assumed life saving instead of valuable property.</p> <p>REQUIRED REFERENCE: None</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

12

ID: 27914

Points: 1.00

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

The last time you were on shift was 3 days ago.

What is the **MINIMUM** number of days of previous log entries you are REQUIRED to review PRIOR TO completing relief?

- A. 1
- B. 2
- C. 3
- D. 4

Answer: C

Answer Explanation

OP-AA-112-101 requires Reactor Operator log review through the last previous date on shift, or the preceding four days, whichever is less, therefore 3 days of logs are the minimum required to be reviewed.

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

Question 12 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:	27914
User-Defined ID:	27914
Cross Reference Number:	
Topic:	12 - Generic 2.1.03
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	<p>Objective: 29900LK011 Reference: OP-AA-112-101, OP-DR-201-012-1001 K/A: Generic 2.1.03 3.7 / 3.9 K/A: Knowledge of shift or short-term relief turnover practices. CFR: 41.10 / 45.13 Safety Function: N/A Level: Memory Pedigree: New History: N/A</p> <p>Explanation:</p> <p>A. Incorrect - 1 day of log review is less than the minimum required. Plausible because during most turnovers only a single day's review of logs is required, since they were on shift the previous day. The candidate must recognize that for absences of greater than one day, they must review the logs for the period missed, up to a total of 4 days worth of logs.</p> <p>B. Incorrect - 2 days of log review is less than the minimum required. Plausible because a review of plant equipment and implementation of Risk Management Actions is required by the Unit Supervisor for equipment that will be inoperable for 48 hours (two days) or longer, per OP-DR-201-012-1001.</p> <p>C. Correct - OP-AA-112-101 requires Reactor Operator log review through the last previous date on shift, or the preceding four days, whichever is less, therefore 3 days of logs are the minimum required to be reviewed.</p> <p>D. Incorrect - The 4 day distractor is the maximum number of days of log entries required to be reviewed, but is not the minimum number required for the given conditions. Plausible because 4 days is the maximum number of days of log entries required to be reviewed.</p> <p>REQUIRED REFERENCES: None</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

13

ID: 27915

Points: 1.00

Unit 3 was operating at near rated power when a spurious Group I isolation occurred.

A lightning strike causes a fault and subsequent trip of the following breakers:

- 345KV BT 8-9 CB.
- 345KV BT 8-15 CB.

With **NO** operator action, what will the electrical lineup be one minute after the lightning strike?

- A. Bus 31 is energized
Bus 35 is energized from Bus 36
Bus 34-1 energized from the Unit 3 EDG
- B. Bus 31 is NOT energized
Bus 36 is energized from Bus 34
Bus 33-1 is energized from the Unit 2/3 EDG
- C. Bus 32 is NOT energized
Bus 39 is energized from Bus 34-1
Bus 34-1 is energized from the Unit 3 EDG
- D. Bus 32 is energized
Bus 35 is energized from Bus 33
Bus 33-1 is energized from the Unit 2/3 EDG

Answer: C

Answer Explanation

With a Group I isolation, a reactor scram will occur. The subsequent loss of the breakers in the 345 yard (off site feed to TR-32) will cause a LOOP. With the LOOP, the EDGs pick up and power their respective dash buses. The dash buses power their respective Reactor Building 480V buses. With no operator action, there are NO buses powered in the Turbine Building.

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

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:	27915
User-Defined ID:	27915
Cross Reference Number:	
Topic:	13 - 295003.A1.01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	<p>Objective: DRE262LN001.12 Reference: DOP 6400-13 K/A: 295003.A1.01 3.7 / 3.8 K/A: Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER: A.C. electrical distribution system Level: High CFR: 41.7 / 45.6 Safety Function: 6 Pedigree: Bank History: None</p> <p>Explanation: A. Incorrect - With NO operator action, Bus 31 and Bus 35 are Turbine Building Buses and are not powered. Plausible because when power is lost to Bus 35, Bus 36 will close on to Bus 35. B. Incorrect - With NO operator action, Bus 36 is a Turbine Building Bus and is not powered. Plausible because Bus 35 is normally powered from Bus 33. C. Correct - With a Group I isolation, a reactor scram will occur. The subsequent loss of the breakers in the 345 yard (off site feed to TR-32) will cause a LOOP. With the LOOP, the EDGs pick up and power their respective dash buses. The dash buses power their respective Reactor Building 480V buses. With no operator action, there are NO buses powered in the Turbine Building. D. Incorrect - With NO operator action, Bus 32 and Bus 35 are Turbine Building Buses and are not powered. Plausible because Bus 36 is normally powered from Bus 34.</p> <p>REQUIRED REFERENCES: None.</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

14

ID: 27916

Points: 1.00

Both units were operating at near rated power.

- U3 Service Air Compressor is OFF.
- ALL available Instrument Air cross-tie valves are closed.
- 2A and 2B Inst Air Compressors are supplying the U2 Inst air header.
- 3A and 3B Inst Air Compressors are supplying the U3 Inst air header.
- U2 Service Air Compressor is carrying the U3 Service Air header via the Service Air cross-tie manual being full open.

Then the U3 Inst Air header developed a leak in the U3 Turbine building.

How will the plant respond?

- A. The AO backup from the U3 Service Air System will auto close at 87 psi sensed at the U3 Main Air Receiver.
- B. The AO backup from the U3 Service Air System will open at 85 psi sensed at the Unit's Local Air Receiver and stabilize at 85 psi.
- C. The AO dryer bypass valves will open at 70 psi sensed at the outlet of the 3A and 3B Air Dryers and stabilize pressure at 85 psi.
- D. The AO backup from the U3 Service Air System will open at 85 psi sensed at the Unit's Main Air Receiver and U3 Inst air pressure would match service air pressure.

Answer: D

Answer Explanation

The AO backup from the U3 Service Air System will open at 85 psi sensed at the U3 Main Air Receiver

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

Question 14 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:	27916
User-Defined ID:	27916
Cross Reference Number:	
Topic:	14 - 295019.A1.01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	<p>Objective: DRE278LN001.06 Reference: DOA 4700-01, DAN 923-1 G-6, DOS 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 COMPLETE LOSS OF INSTRUMENT AIR: Backup air supply CFR: 41.7 / 45.6 Safety function: 8 Level: Memory Pedigree: Bank</p> <p>Explanation: A. Incorrect - The AO backup from the U3 Service Air System will NOT auto close at 87 psi sensed at the U3 Main Air Receiver just on pressure alone, operator must depress the AO cross-tie valve reset pushbutton locally when pressure is above 87 psi. Plausible because the system has to be above 87 psi to reset the valve. B. Incorrect - The AO Air dryer bypass valves will open but at 60 psi not 70 psi. Plausible because the air dryer bypass valve does open automatically. C. Incorrect - The AO backup from the U3 Service Air System does NOT open from pressure sensed at the Local Receiver. Plausible because at 85 psi the backup from Service Air will open. D. Correct - The AO backup from the U3 Service Air System will open at 85 psi sensed at the U3 Main Air Receiver</p> <p>REQUIRED REFERENCES: None.</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

15

ID: 27969

Points: 1.00

Unit 2 was operating at full power when the following events occurred:

- The D Main steam line breaks in the steam tunnel.
- Loss of the Instrument Bus due to a short circuit.
- An NSO reports the following valves open:
 - ◆ RX RETURN VLV, 2-1201-7
 - ◆ 2B FW REG ISOL, 2-3206B
 - ◆ MSIV, 2-0203-1A

Which of the following describes the status of the PCIS status box in the Safety Parameter Display System (SPDS)?

- A. GREEN
- B. YELLOW
- C. RED
- D. CYAN

Answer: C

Answer Explanation

A steam line break would cause a Group 1 PCIS isolation due to high secondary containment temperature in the steam tunnel. The PCIS status box is RED when a PCIS isolation signal is present and all required PCIS isolation valves are not shut. All MSIVs are required to be shut when a Group 1 signal is received.

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

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:	2.00
System ID:	27969
User-Defined ID:	27969
Cross Reference Number:	
Topic:	15 - Generic 2.1.19
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	<p>Objective: 283LN002.06 Reference: DOP 9950-17 K/A: Generic 2.1.19 K/A: Ability to use plant computers to evaluate system or component status. CFR: 41.10 Safety Function: 5 PRA: No Level: High Pedigree: Bank History: 2012 NRC</p> <p>Explanation:</p> <p>A. Incorrect - The PCIS status box is GREEN when a PCIS isolation signal is present and all required PCIS isolation valves are shut. Plausible because the PCIS box could be GREEN.</p> <p>B. Incorrect - Yellow is a pre-alarm indication on PCIS used by the parameters with analog values on SPDS. Plausible because YELLOW could be a color in relation to PCIS on the SPDS.</p> <p>C. Correct - A steam line break would cause a Group 1 PCIS isolation due to high secondary containment temperature in the steam tunnel. The PCIS status box is RED when a PCIS isolation signal is present and all required PCIS isolation valves are not shut. All MSIVs are required to be shut when a Group 1 signal is received.</p> <p>D. Incorrect - Would be correct if the status of the PCIS valves were undeterminable. A loss of the Instrument Bus would not affect the ability of the SPDS to determine valve position even though some of the PCIS valves would lose remote indication. Plausible because CYAN is a possible status color on the SPDS.</p> <p>REQUIRED REFERENCES: None.</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

16

ID: 13845

Points: 1.00

With the unit operating at near rated power when a transient occurs.

- Reactor Coolant System unidentified leakage is 4 gpm
- Reactor Steam Dome pressure is 1010 psig
- Torus water level is 14 ft 8.5 inches
- Drywell pressure is 1.4 psig

Which parameters have exceeded a Technical Specification ENTRY CONDITION?

- A. Reactor Coolant System unidentified leakage
- B. Reactor Steam Dome pressure
- C. Torus water level
- D. Drywell pressure

Answer: B

Answer Explanation

TS 3.4.10 entry condition is if Reactor Steam Dome pressure >1005 psig

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

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:	13845
User-Defined ID:	13845
Cross Reference Number:	
Topic:	16 - 295025.G.2.2.42
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	<p>Objective: DRE299LN001.04 Reference: Tech Spec 3.4.10 K/A: 295025.G.2.2.42 3.9 / 4.6 K/A: High Reactor Pressure - Ability to recognize system parameters that are entry-level conditions for Technical Specifications. CFR: 41.7 / 41.10 Safety Function: 3 PRA: No Level: Memory Pedigree: Bank History: 2007 NRC, 14-1 Cert</p> <p>Explanation:</p> <p>A. Incorrect - TS 3.4.4 entry condition is if RCS leakage >5gpm. Plausible because there is a tech spec entry condition for RCS leakage.</p> <p>B. Correct - TS 3.4.10 entry condition is if Reactor Steam Dome pressure >1005 psig</p> <p>C. Incorrect - TS 3.6.2.2 entry condition is if Torus water level is not between 14 ft 6.5 inches and 14 ft 10.5 inches. Plausible because there is a tech spec entry condition for torus water level.</p> <p>D. Incorrect - TS 3.6.1.4 entry condition is if Drywell pressure is > 1.5 psig. Plausible because there is a tech spec entry condition for drywell pressure.</p> <p>REQUIRED REFERENCES: None</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

17

ID: 27963

Points: 1.00

U2 was operating at near rated power, when the following events occurred:

- A coolant leak developed inside the Drywell
- Drywell pressure rose to 2.3 psig, and is rising at a rate of 0.2 psig/min
- A storm caused a loss of off-site power to Unit 2
- Actions of DGA 12, LOSS OF OFFSITE POWER, are in progress
- The Unit Supervisor ordered the 2A RBCCW pump re-started for Drywell cooling

To start the 2A RBCCW pump, an Operator must...

- A. take the c/s to CLOSE **ONLY**.
- B. pull fuses in the AEER and take the c/s to CLOSE.
- C. close 4KV Bus 23 & Bus 23-1 TIE GCBs and take the c/s to CLOSE.
- D. close 4KV Bus 24 & Bus 24-1 TIE GCBs and take the c/s to CLOSE.

Answer: C

Answer Explanation

Closing the Bus 23 to Bus 23-1 tie breaker per DEOP 0500-02 allows the 2A RBCCw pump to be restarted under the described conditions.

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

Question 17 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:	27963
User-Defined ID:	27963
Cross Reference Number:	
Topic:	17 - 295010.A1.01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	<p>Objective: 29502LK088 Reference: DEOP 0500-02, DOA 3700-01 K/A: 295010.A01.01 3.4 / -- K/A: Ability to operate and/or monitor the following as they apply to HIGH DRYWELL PRESSURE: Drywell ventilation/cooling CFR: 41.7 PRA: Yes Safety Function: 5 Level: High Pedigree: New History: N/A</p> <p>Explanation: A. Incorrect - Due to the loss of offsite power and the ECCS initiation signal present, the Bus 23 to Bus 23-1 tie breaker must be closed before placing the c/s to CLOSE. Plausible because taking the c/s to CLOSE for normal operation would start 2A RBCCW pump. B. Incorrect - There are no fuses associated with RBCCW pump starts in the AEER. Plausible because DEOP 0500-02 gives direction to lift and tape leads in the AEER to restore RBCCW pumps. C. Correct - Closing the Bus 23 to Bus 23-1 tie breaker per DEOP 0500-02 allows the 2A RBCCw pump to be restarted under the described conditions. D. Incorrect - 2A RBCCW pump is powered from Bus 23-1. Plausible because this would be true for 2B RBCCW Pump.</p> <p>REQUIRED REFERENCES: None.</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

18

ID: 12911

Points: 1.00

A transient has occurred causing the following:

- MO 1501-32A XTIE VLV is closed.
- Div 1 LPCI is lined up for Torus Cooling.
- Div 2 LPCI is injecting into the vessel at 5000 gpm to maintain level.
- Torus Bottom Pressure is 5 psig.
- Torus temperature is 120°F and rising.
- LPCI pump flows and amp readings begin to oscillate.

What action is required to mitigate the issue?

- A. Align 'A' LPCI pump **ONLY** to take suction from the 2/3 CSTs.
- B. Align 'C' LPCI pump **ONLY** to take suction from the 2/3 CSTs.
- C. Align the 'A' **AND** 'D' LPCI pumps to take suction from the 2/3 CSTs.
- D. Align the 'C' **AND** 'D' LPCI pumps to take suction from the 2/3 CSTs.

Answer: B

Answer Explanation

'C' LPCI pp is correct because per procedure only one pump suction can be lined up to the CST

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

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:	3.00
System ID:	12911
User-Defined ID:	12911
Cross Reference Number:	
Topic:	18 - 219000.G.2.1.20
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	<p>Objective: DRE203LN001.08 Reference: UFSAR table 6.3-18, Section 6.3.3, DEOP 200-1 figure X, DEOP 0500-03 K/A: 219000.G.2.1.20 4.6 / 4.6 K/A: RHR/LPCI: Torus/Suppression Pool Cooling Mode, Ability to interpret and execute procedure steps. CFR: 41.10 / 43.5 / 45.12 Level: Memory Safety Function: 5 Pedigree: Bank History: 2003 Cert, 2007 Cert</p> <p>Explanation:</p> <p>A. Incorrect - 'A' LPCI pump is being used for Torus Cooling. Plausible because any LPCI pump can be aligned to the CST.</p> <p>B. Correct - 'C' LPCI pp is correct because per procedure only one pump suction can be lined up to the CST</p> <p>C. Incorrect - Injection with 2 LPCI pumps would add more water, but procedural guidance only allows 1 pp to be lined up from the CSTs. Plausible because any LPCI pump can be aligned to the CST.</p> <p>D. Incorrect - Injection with 2 LPCI pumps would add more water, but procedural guidance only allows 1 pp to be lined up from the CSTs. Plausible because any LPCI pump can be aligned to the CST.</p> <p>REQUIRED REFERENCES: None.</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

19

ID: 27917

Points: 1.00

Unit 3 was operating at near rated power, when Bus 37 experienced an overcurrent condition.

Which of the following Unit 3 Air Compressors will lose its NORMAL power supply?

- 1) 3A IAC
- 2) 3B IAC
- 3) 3C IAC
- 4) U3 SAC

- A. 1 **ONLY**
- B. 2 **ONLY**
- C. 2 and 3 **ONLY**
- D. 2, 3, and 4 **ONLY**

Answer: C

Answer Explanation

The 3B and 3C IAC are powered from Bus 37

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

Question 19 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:	27917
User-Defined ID:	27917
Cross Reference Number:	
Topic:	19 - 300000.K2.01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	<p>Objective: DRE278LN001.06 Reference: 12E-3305, DOP 6700-12 K/A: 300000.K2.01 2.8 / 2.8 K/A: Knowledge of electrical power supplies to the following: Instrument Air Compressor CFR: 41.7 / 45.7 Safety Function: 8 Level: Memory Pedigree: Bank History: None</p> <p>Explanation: A. Incorrect - The 3A IAC is powered from Bus 36. Plausible because Bus 37 powers other air compressors. B. Incorrect - The 3C IAC will also lose power upon a loss of Bus 37. Plausible because 3B IAC is powered from Bus 37. C. Correct - The 3B and 3C IAC are powered from Bus 37. D. Incorrect - The U3 SAC is powered from Bus 36. Plausible because Bus 37 powers other air compressors.</p> <p>REQUIRED REFERENCES: None</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

20

ID: 12467

Points: 1.00

Unit 2 has experienced a leak and is currently executing DOA 0040-01, SLOW LEAK. The leak has continued to degrade and the following conditions now exist:

- RPV water level is 24"
- Drywell pressure is 1.3 psig
- HPCI Room temperature is 100°F
- West LPCI corner room water level is 1"
- West LPCI Pump Area is 2 mrem/hr
- East LPCI Pump Area is 1 mrem/hr

The expected course of action for the above conditions is...

- A. Enter DEOP 300-1, Secondary Containment Control, exit DOA 0040-01 Slow Leak.
- B. Enter DEOP 300-1, Secondary Containment Control, continue to execute DOA 0040-01 Slow Leak.
- C. Enter DEOP 100, RPV Control and 200-1 Primary Containment Control, continue to execute DOA 0040-01 Slow Leak.
- D. Enter DEOP 100, RPV Control and 200-1 Primary Containment Control, exit DOA 0040-01 Slow Leak.

Answer: B

Answer Explanation

The entry conditions for DEOP 0100 and 0200-01 have not been met. LPCI Corner Room Level above 0 inches is an entry condition for DEOP 0300-01. Slow Leak directs the operator to follow DEOP 0300-01. DOAs are performed in parallel with DEOPS, unless otherwise stated.

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

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:	12467
User-Defined ID:	12467
Cross Reference Number:	LIH
Topic:	20 - Generic.2.4.16
Num Field 1:	0.00
Num Field 2:	0.00
Text Field:	
Comments:	<p>Objective: 29502LK049 Reference: DEOP 0300-01 K/A: Generic 2.4.16 3.5 / 4.4 K/A: Knowledge of EOP implementation hierarchy and coordination with other support procedures or guidelines such as, operating procedures, abnormal operating procedures, and severe accident management guidelines. CFR: 41.10 / 43.5 / 45.13 Level: High Safety Function: N/A Pedigree: Bank History: None</p> <p>Explanations:</p> <p>A. Incorrect - The entry conditions for DEOP 0100 and 0200-01 have not been met. LPCI Corner Room Level above 0 inches is an entry condition for DEOP 0300-01. DOA 0040-01 Slow Leak directs the operator to follow DEOP 0300-01.</p> <p>B. Correct - The entry conditions for DEOP 0100 and 0200-01 have not been met. LPCI Corner Room Level above 0 inches is an entry condition for DEOP 0300-01. Slow Leak directs the operator to follow DEOP 0300-01. DOAs are performed in parallel with DEOPS, unless otherwise stated.</p> <p>C. Incorrect - The entry conditions for DEOP 0100 and 0200-01 have not been met and DOA 0040-01 Slow Leak directs the operator to follow DEOP 0300-01. Plausible because the leak has caused RPV LVL LO alarm would be in and conditions getting worse. If level reaches +8 entry into 100 and 200-1 would be required.</p> <p>D. Incorrect - The entry conditions for DEOP 0100 and 0200-01 have not been met and DOA 0040-01 Slow Leak directs the operator to follow DEOP 0300-01. Plausible because the leak has caused RPV LVL LO alarm would be in and conditions getting worse. If level reaches +8 entry into 100 and 200-1 would be required. Part 3 is correct.</p> <p>REQUIRED REFERENCES: None</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

21

ID: 27920

Points: 1.00

Unit 2 cooldown was in progress with the Shutdown Cooling System, when a transient occurred.

Which of the following will stop the cooldown?

- A. RPV level reaches 0 inches.
- B. Drywell Pressure reaches 2.0 psig.
- C. X-Area temperature reaches 200°F.
- D. Drywell radiation levels reach 30 R/hr.

Answer: A

Answer Explanation

When RPV level drops below +8 inches a PCIS group 3 is initiated. SDC MOVs will close and the cooldown will be stopped.

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

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:	27920
User-Defined ID:	27920
Cross Reference Number:	
Topic:	21 - 205000.K4.03
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	<p>Objective: 205LN001.12B Reference: DAN 902(3)-5 D-5, E-5 K/A: 205000.K4.03 3.8/3.8 K/A: Knowledge of SHUTDOWN COOLING SYSTEM/MODE design feature(s) and/or interlocks which provide for the following: Low reactor water level: Plant-Specific CFR: 41.7 Safety Function: 4 Pedigree: Bank History: 16-1 NRC Level: High</p> <p>Explanation:</p> <p>A. Correct - When RPV level drops below +8 inches a PCIS group 3 is initiated. SDC MOVs will close and the cooldown will be stopped.</p> <p>B. Incorrect - This is not a PCIS Group 3 setpoint. Plausible because this is setpoint for a PCIS group II isolation.</p> <p>C. Incorrect - This is not a PCIS Group 3 setpoint. Plausible because this is a setpoint for a Group I PCIS isolation. If the Cooldown was with the bypass valves this would stop the cooldown.</p> <p>D. Incorrect - This is not a PCIS Group 3 setpoint. Plausible because this is a setpoint for a PCIS group II isolation.</p> <p>PCIS Group II and III setpoints and actions are commonly mistaken.</p> <p>REQUIRED REFERENCES: None.</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

22

ID: 27921

Points: 1.00

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

- HPCI failed to inject.
- 2/3 EDG failed to start.
- 3 ADS valves are open.
- 2B Core Spray pump failed to start.
- RPV pressure is 275 psig and lowering.
- RPV water level is -110 inches and rising.

Core cooling is

- A. assured due to the core being submerged.
- B. assured due to the steam cooling effects of the open SRVs.
- C. NOT assured due to ONLY 3 ADS valves open.
- D. NOT assured due to only two LPCI pumps injecting.

Answer: A

Answer Explanation

It must be determined from the conditions of the stem that the only injection source is 2 LPCI pumps in the injection mode with level rising and RPV level is above TAF core cooling is assured via submergence cooling.

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

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:	27921
User-Defined ID:	27921
Cross Reference Number:	
Topic:	22 - 203000.A1.01
Comments:	<p>Objective: 299LN049-2 Reference: TSG Attach L, DEOP 0010-00 K/A: 203000.A1.01 4.2 K/A: Ability to predict and/or monitor changes in parameters associated with operating the RHR/LPCI: Injection Mode controls including: Reactor water level CFR: 41.5 / 45.5 Safety Function: 2 PRA: Yes Level: High Pedigree: Bank History: None</p> <p>Explanation:</p> <p>A. Correct - When RPV level is above TAF core cooling is assured via submergence cooling.</p> <p>B. Incorrect - Minimum Steam Cooling is does not assure core cooling in this situation. Plausible because minimum steam cooling pressure (as present when comparing number of open ERVs to RPV pressure) is applicable if Failure to Scram conditions exist.</p> <p>C. Incorrect - Minimum steam cooling pressure (as present when comparing number of open ERVs to RPV pressure) is not applicable unless Failure to Scram conditions exist. Plausible because other systems that are degraded would not perform their required function.</p> <p>D. Incorrect - The number of running LPCI pumps is not directly connected to the assurance of adequate core cooling. With RPV pressure < 340, the LPCI pumps are allowed to inject. This injection is sufficient to provide core submergence as indicated by RPV level going up. Plausible because other systems that are degraded would not perform their required function.</p> <p>K/A Justification: Based on the conditions in the stem, the only injection source is LPCI which is causing Reactor Water Level to rise.</p> <p>Justification for HIGH order: The candidate is required to assess multiple conditions present and apply knowledge of RPV level milestones</p> <p>REQUIRED REFERENCES: None</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

23

ID: 27922

Points: 1.00

Unit 2 was operating at low power when the Operating Team was directed to place a second FWRV in AUTO from MANUAL.

After verifying the oncoming REG VLV CONTROL STATION is in MAN, the Operator is required to manually open the FWRV to ___(1)___ open, then ___(2)___.

- A. (1) <4%
(2) manually balance the 'A' FWRV and the 'B' FWRV positions and then place the oncoming REG VLV CONTROL STATION in AUTO
- B. (1) >5%
(2) manually balance the 'A' FWRV and the 'B' FWRV positions and then place the oncoming REG VLV CONTROL STATION in AUTO
- C. (1) <4%
(2) place the REG VLV CONTROL STATION in AUTO and verify the positions of the 'A' FWRV and the 'B' FWRV balance
- D. (1) >5%
(2) place the REG VLV CONTROL STATION in AUTO and verify the positions of the 'A' FWRV and the 'B' FWRV balance

Answer: D

Answer Explanation

When transferring a second FWRV from manual to auto, the FWRV is manually opened to >5%, placed in auto and then verified that both FWRVs AUTOMATICALLY adjust to equal positions.

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

Question 23 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:	27922
User-Defined ID:	27922
Cross Reference Number:	
Topic:	23 - 259002.A4.03
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	<p>Objective: DRE259LN002.11 Reference: DOP 0600-06 K/A: 259002.A4.03 3.8 / 3.6 K/A: Ability to manually operate and/or monitor in the control room: All individual component controllers when transferring from manual to automatic modes CFR: 41.7 / 45.5 to 45.8 Safety Function: 2 Level: Memory Pedigree: Bank History: None</p> <p>Explanation:</p> <p>A. Incorrect - The FWRV is manually opened to >5% not <4% and verified that both FWRVs AUTO adjust to equal positions. Plausible because the controller can be set to <4% and other systems are adjusted in manual.</p> <p>B. Incorrect - Both FWRVs are verified to AUTO adjust to equal positions, not manually operated. Plausible because the first part is correct and other systems are adjusted in manual.</p> <p>C. Incorrect - The FWRV is manually opened to >5% not <4%. Plausible because the second part is correct and the controller can be set to <4%.</p> <p>D. Correct - When transferring a second FWRV from manual to auto, the FWRV is manually opened to >5%, placed in auto and then verified that both FWRVs AUTOMATICALLY adjust to equal positions.</p> <p>REQUIRED REFERENCES: None</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

24

ID: 27924

Points: 1.00

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

- Drywell Radiation level increased to 33 R/hr.
- Reactor Water level dropped to and remained steady at 20 inches.

A result of the event will be to receive a ____ (1) ____ Isolation, which will cause ____ (2) ____.

- A. 1) Group 2
 2) an auto start of SBGT
- B. 1) Group 2
 2) the SDC 1, 2, 4, and 5 valves to close
- C. 1) Group 3
 2) an auto start of SBGT
- D. 1) Group 3;
 2) the SDC 1, 2, 4, and 5 valves to close

Answer: A

Answer Explanation

When Radiation level reaches 30 R/hr, a Group 2 Isolation will occur, causing an auto start of the SBGT System.

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

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:	2.00
System ID:	27924
User-Defined ID:	27924
Cross Reference Number:	
Topic:	24 - 261000.K1.08
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	<p>Objective: DRE261LN001.10 Reference: DAN 902(3)-5 E-5 K/A: 261000.K1.08 2.8 / 3.1 K/A: Knowledge of the physical connections and/or cause-effect relationships between STANDBY GAS TREATMENT SYSTEM and the following: Process radiation monitoring system CFR: 41.2 to 41.9 / 45.7 to 45.8 Level: High Safety Function: 9 Pedigree: Bank History: None</p> <p>Explanation: A. Correct - When Radiation level reaches 30 R/hr, a Group 2 Isolation will occur, causing an auto start of the SBTG System. B. Incorrect - A Group 3 will cause a closure of the SDC valves, not a Group 2. Plausible because the first part is correct and if it was a Group 3, the second part would be correct. C. Incorrect - A Group 3 is caused by Reactor Water Level decreasing to 6 inches, not 20 inches. A Group 2 causes an auto start of the SBTG System, not a Group 3. Plausible because the second part is correct and RPV water level is a setpoint for Group 3. D. Incorrect - A Group 3 is caused by Reactor Water Level decreasing to 6 inches, not 20 inches. SDC isolation on a Group 2 is a common misconception. Plausible because a Group 3 causes the SDC isolation.</p> <p>REQUIRED REFERENCES: None</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

25

ID: 27990

Points: 1.00

Per DOS 6600-01, DIESEL GENERATOR SURVEILLANCE TESTS, which set of FAST START data would make U2 Diesel OPERABLE?

- A. 59 Hz and 4000 V in 15 seconds
- B. 58 Hz and 4100 V in 13 seconds
- C. 59 Hz and 4100 V in 13 seconds
- D. 58 Hz and 4000 V in 15 seconds

Answer: C

Answer Explanation

Acceptance Criteria is > 58.8 Hz and > 3952 V in ≤ 13 seconds.

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

Question 25 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:	27990
User-Defined ID:	27990
Cross Reference Number:	
Topic:	25 - 262001.G.2.2.37
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	<p>Objective: DRE209LN001.12 Reference: DAN 902(3)-8 B-3, DAN 902(3)-3 G-16, DOS 6600-01 K/A: 262001.G.2.2.37 3.6 / 4.6 K/A: Ability to determine operability and/or availability of safety related equipment. CFR: 41.7 / 43.5 / 45.12 Safety Function: 6 Level: Memory Pedigree: New History: N/A</p> <p>Explanation: A. Incorrect – Need to reach required voltage and frequency in ≤ 13 seconds. Plausible because the first two parts of the data are correct. B. Incorrect – Frequency is not within spec. Plausible because voltage and time are correct. C. Correct – Acceptance Criteria is > 58.8 Hz and > 3952 V in ≤ 13 seconds. D. Incorrect – Frequency and time are not within spec. Plausible because voltage is correct.</p> <p>REQUIRED REFERENCES: None</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

26

ID: 24224

Points: 1.00

<QQ 680246(1410)><<<QQ 5702(1410)><The Reactor Water Cleanup pump room was recently surveyed.

- General area radiation of 200 mrem/hr
- Smearable contamination of 400 dpm/100cm² (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 100 mrem/hr at 30 cm. Reactor Operators are required to know this information (NGET)

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

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 2.3.07
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	<p>Objective: ACAD 00-007.10, NGET Reference: NISP-RP-004, RA-AA-18 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 History: 14-1 NRC, 18-1 NRC, 19-1 NRC</p> <p>Justification for Memory: Reactor Operators are required to know this information (NGET)</p> <p>Explanation:</p> <p>A. Correct - A high rad area is an area that could result in reception of deep dose rate equivalent meets or exceeds 100 mrem/hr but less than 1000 mrem/hr at 30 cm.</p> <p>B. Incorrect - Locked High Rad area is required for areas meets or exceeds 1000 mrem/hr. Plausible because High Radiation limit is exceeded. Must know what level causes a LOCKED Hi Rad area.</p> <p>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/100cm². Plausible because the first part of the answer is correct, and 400 dpm/100cm² is above the value of 300 dpm which used by the government.</p> <p>D. Incorrect - Locked High Rad area is required for areas which meet or exceed 1000 mrem/hr. Contaminated areas are areas in which contamination levels meet or exceed 1000 dpm/100cm². Plausible because rad levels exceed 100 mrem/hr and operators must know what level causes a LOCKED Hi Rad area, and because 400 dpm/100cm² is above the value of 300 dpm used by the government.</p> <p>REQUIRED REFERENCES: NONE</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

27

ID: 27926

Points: 1.00

Unit 2 was operating at near rated power when a complete loss of U2 125 VDC Division 1 occurred.

The Control Room Bus control switch indication lights will be lost for

- A. Bus 23.
- B. Bus 24.
- C. Bus 26.
- D. Bus 27.

Answer: A

Answer Explanation

With a loss of 125 Vdc Division 1, all division 1 buses will lose remote indication and protection ability. Bus 23 is the only Div 1 bus listed.

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

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:	27926
User-Defined ID:	27926
Cross Reference Number:	
Topic:	27 - 263000.A3.01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	<p>Objective: DRE263LN002.12 Reference: DOA 6900-02, DOA 6900-T1 K/A: 263000.A3.01 3.2 / 3.3 K/A: Ability to monitor automatic operations of the D.C. ELECTRICAL DISTRIBUTION including: Meters, dials, recorders, alarms, and indicating lights CFR: 41.7 / 45.7 Safety Function: 6 Level: Memory Pedigree: Bank History: None</p> <p>Explanation: A. Correct - With a loss of 125 Vdc Division 1, all division 1 buses will lose remote indication and protection ability. Bus 23 is the only Div 1 bus listed. B. Incorrect - Bus 24 indication is supplied by the 125 Vdc 2B-1 Dist Panel. Plausible because Bus 24 reserve control power is supplied from U2 125VDC Div 1. C. Incorrect - Bus 26 indication is supplied by the 125 Vdc 2B-2 Dist Panel. Plausible because Bus 26 reserve control power is supplied from U2 125VDC Div 1. D. Incorrect - Bus 27 indication is supplied by the 125 Vdc 2B-2 Dist Panel. Plausible because Bus 27 reserve control power is supplied from U2 125VDC Div 1.</p> <p>REQUIRED REFERENCES: None</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

28

ID: 27929

Points: 1.00

Unit 2 was operating at near rated power when a transient occurred which requires the injection of SBLC for level control.

The NSO placed the SBLC system control switch to the "2&1" position.

What is the expected SBLC response if the 'B Squib Valve' fails to fire?

- A. 'A' SBLC pump starts and supplies flow to the vessel; 'B' SBLC pump remains off
- B. 'A' SBLC pump starts and supplies flow to the vessel; 'B' SBLC pump starts, then trips on high discharge pressure
- C. Both SBLC pumps start and both pumps supply flow to the vessel
- D. Both SBLC pumps start; 'A' pump supplies flow to the vessel; 'B' pump recirculates flow to the tank through its relief valve

Answer: C

Answer Explanation

Each squib has 40 gpm capacity, and they are downstream of a common discharge header from the pumps. Therefore, both pumps would be running and providing flow to the common discharge header, and through the 'A' squib to the vessel. Note: A failed squib has no impact on pump start circuitry/logic.

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

Question 28 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:	27929
User-Defined ID:	27929
Cross Reference Number:	
Topic:	28 - 211000.K3.01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	<p>Objective: DRE211LN001.03 Reference: M-33, M-364, DAN 902(3)-5 H-6, DOP 1100-02 K/A: 211000.K3.01 4.3 / 4.4 K/A: Knowledge of the effect that a loss or malfunction of the STANDBY LIQUID CONTROL SYSTEM will have on following: †Ability to shutdown the reactor in certain conditions CFR: 41.7 / 45.4 Level: High Safety Function: 1 Pedigree: Bank History: None</p> <p>Explanation:</p> <p>A. Incorrect - The SBLC pumps discharge to a common header therefore both will flow through the 'A' squib valve to the vessel. Plausible because most pumps have individual discharge paths and if the valve doesn't open then there won't be any flow.</p> <p>B. Incorrect - Both pumps will start and flow through the 'A' squib valve into the vessel. There is no high discharge pressure trip for the SBLC pumps. Plausible because if there was no flow path for the 'B' pump then the relief would lift and the flow would be back to the SBLC tank.</p> <p>C. Correct - Each squib has 40 gpm capacity, and they are downstream of a common discharge header from the pumps. Therefore, both pumps would be running and providing flow to the common discharge header, and through the 'A' squib to the vessel. Note: A failed squib has no impact on pump start circuitry/logic.</p> <p>D. Incorrect - The SBLC pumps discharge to a common header therefore both will flow through the 'A' squib valve to the vessel. Plausible because there is a path for the 'B' pump through the 'A' squib and will not lift the relief valve.</p> <p>REQUIRED REFERENCES: None</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

29

ID: 27930

Points: 1.00

Unit 2 is in STARTUP.

- Reactor is subcritical.
- Rod pulls are in progress at the point of 2 doublings.

When a rod is pulled, that does not take the reactor critical, what is the expected response of the Period Meter on panel 902-5?

- A. Remains stable
- B. Deflects up, then returns to infinity
- C. Deflects down, then returns to infinity
- D. Deflects up and settles at a new stable higher reading.

Answer: B

Answer Explanation

With no change in count rate, the meters rest at infinity. During a rod pull, the SRM count rate will increase which will give a positive reactor period (upscale). As the counts settle out at a new value, the change in count rate slows and stops which brings the period reading back to infinity.

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

Question 29 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:	27930
User-Defined ID:	27930
Cross Reference Number:	
Topic:	29 - 215004.A4.01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	<p>Objective: 215LN004.11 Reference: DGP 1-1, DGP 3-4 K/A: 215004.A4.01 3.9 / 3.8 K/A: Ability to manually operate and/or monitor in the control room: SRM count rate and period CFR: 41.7 / 45.5 to 45.8 Level: Memory Safety Function: 7 Pedigree: Bank History: None</p> <p>Explanation:</p> <p>A. Incorrect - The period meter will deflect up to indicate a positive trend. Plausible because the meter will remain steady with no rod movement.</p> <p>B. Correct - With no change in count rate, the meters rest at infinity. During a rod pull, the SRM count rate will increase which will give a positive reactor period (upscale). As the counts settle out at a new value, the change in count rate slows and stops which brings the period reading back to infinity.</p> <p>C. Incorrect - The period meter will deflect up to indicate a positive trend. Plausible because the candidate must understand what the meters is reading and which way it will deflect for a reactivity increase.</p> <p>D. Incorrect - The period meter will deflect up to indicate a positive trend. Plausible because the period meter will deflect up and other meters will settle at new higher stable reading. i.e. SRMs</p> <p>REQUIRED REFERENCES: None</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

30

ID: 23463

Points: 1.00

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

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

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.

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

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:	3.00
System ID:	23463
User-Defined ID:	23463
Topic:	30 - 295024.K1.01
Comments:	<p>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 Level: Memory Pedigree: Bank History: 2013 Cert, 19-1 NRC</p> <p>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.</p> <p>Explanations:</p> <p>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.</p> <p>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.</p> <p>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.</p> <p>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.</p> <p>REQUIRED REFERENCES: DEOP 200-1</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

31

ID: 27967

Points: 1.00

Unit 2 was operating at 100% power.

A LOCA occurred with lowering Suppression Pool water level.

Which of the following are the **FIRST** indications of inadequate NPSH to ECCS pumps?

- 1 Erratic pump discharge pressure
 - 2 Erratic pump amperage indications
 - 3 Sustained LOW pump discharge pressure
- A. 1, 2, and 3
 - B. 1 and 2 ONLY
 - C. 1 and 3 ONLY
 - D. 2 and 3 ONLY

Answer: B

Answer Explanation

With inadequate NPSH to ECCS pumps, pump discharge pressure and pump motor amperage will behave erratically.

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

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:	4.00
System ID:	27967
User-Defined ID:	27967
Cross Reference Number:	
Topic:	31 - 295030.K1.02
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	<p>Objective: 29501LK003 Reference: DEOP 0100, DEOP 0010, OP-DR-103-102-1001, DOP 1500-02 K/A: 295030.K1.02 3.5 / 3.8 K/A: Knowledge of the operational implications of the following concepts as they apply to LOW SUPPRESSION POOL WATER LEVEL: Pump NPSH CFR: 41.8 to 41.10 Safety Function: 5 Level: Memory Pedigree: Modified History: 18-1 NRC Exam</p> <p>Explanation: A. Incorrect - Sustained discharge pressure would be an indication of adequate NPSH. Plausible because erratic pump discharge pressure and pump motor amperage is an indication of inadequate NPSH. B. Correct - With inadequate NPSH to ECCS pumps, pump discharge pressure and pump motor amperage will behave erratically. C. Incorrect - Sustained discharge pressure would be an indication of adequate NPSH. Plausible because erratic discharge pressure would be an indication of inadequate NPSH. D. Incorrect - Sustained discharge pressure would be an indication of adequate NPSH. Plausible because erratic pump amps would be an indication of inadequate NPSH.</p> <p>REQUIRED REFERENCES: None.</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

32

ID: 14647

Points: 1.00

Unit 2 was operating at near rated power, when a Reactor Scram occurred.

If RPV pressure is 455 psig and steady, what level correlates to Top of Active Fuel (TAF) on the 902-3 panel Fuel Zone indicators?

- A. -143 inches
- B. -170 inches
- C. -191 inches
- D. -209 inches

Answer: A

Answer Explanation

-143 inches would be TAF if < 500 psig RPV pressure.

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

Question 32 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:	14647
User-Defined ID:	14647
Cross Reference Number:	
Topic:	32 - 295031.A2.03
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	<p>Objective: DRE216LN001.11 Reference: TSG, attachment L hardcard, DEOP 0100 K/A: 295031.A2.03 4.2 / 4.2 K/A: Ability to determine and/or interpret the following as they apply to REACTOR LOW WATER LEVEL: Reactor pressure CFR: 41.10 / 43.5 / 45.13 Level: Memory Safety Function: 2 Pedigree: Bank History: None</p> <p>Explanation: A. Correct - RWL of -143 inches would be TAF if < 500 psig RPV pressure. B. Incorrect - RWL of -170 inches is TAF with RPV pressure > 500 psig. Plausible because -170 inches would be correct if RPV Pressure was > 500 psig. C. Incorrect - RWL of -191 inches is 2/3 core height with RPV pressure < 500 psig. Plausible because you must blowdown the reactor prior to reaching -191 inches when > 500 psig. D. Incorrect - RWL of -209 inches is 2/3 core height with RPV pressure > 500 psig. Plausible because you must blowdown the reactor prior to reaching -209 inches when > 500 psig.</p> <p>REQUIRED REFERENCES: None.</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

33

ID: 22609

Points: 1.00

With the Reactor experiencing an ATWS condition, COLD SHUTDOWN BORON WEIGHT is defined as the amount of boron necessary to maintain the Reactor shutdown with which of the following additional conditions?

- A. RPV water is at its most reactive temperature.
 All control rods are fully withdrawn
 RPV water level +8 to -59 inches
- B. No voids are present in the reactor core.
 All control rods are fully withdrawn
 RPV water level at high level trip setpoint
- C. 72 hour shutdown Xenon
 All control rods fully withdrawn
 RPV water level +8 to -59 inches
- D. 72 hour shutdown Xenon
 SDC and RWCU systems isolated
 RPV water level at high level trip setpoint

Answer: B

Answer Explanation

Cold Shutdown Boron Weight is defined as the amount of boron necessary to shutdown the reactor and maintain a shutdown condition with the following conditions: RPV water level is assumed at high level trip setpoint, Xenon is at equilibrium, and Recirc system in service.

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

Question 33 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:	22609
User-Defined ID:	22609
Cross Reference Number:	
Topic:	33 - 295037.K3.05
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	<p>Objective: DRE211LN001.01 Reference: EPG B-17-9, DEOP Technical Bases K/A: 295037.K3.05 3.2 / 3.7 K/A: Knowledge of the reasons for the following responses as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN: Cold shutdown boron weight: Plant-Specific. CFR: 41.5 / 45.6 Level: Memory Safety Function: 1 Pedigree: Bank History: Susquehanna 2004 NRC, 2009 NRC</p> <p>Explanation:</p> <p>A. Incorrect - Plausible because part 1 and part 2 are correct. RPV level should be at high level trip setpoint.</p> <p>B. Correct - Cold Shutdown Boron Weight is defined as the amount of boron necessary to shutdown the reactor and maintain a shutdown condition with the following conditions: RPV water level is at high level trip setpoint, no voids are present in the reactor core, and all control rods are fully withdrawn.</p> <p>C. Incorrect - Plausible because Xenon should be at equilibrium, second part is correct, and RPV level should be at high level trip setpoint.</p> <p>D. Incorrect - Plausible because Xenon should be at equilibrium, Recirc should be in service, and third part is correct.</p> <p>REQUIRED REFERENCES: None.</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

34

ID: 27932

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.

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

Question 34 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:	27932
User-Defined ID:	27932
Topic:	34 - 295038.K2.05
Comments:	<p>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: 19-1 NRC</p> <p>Explanation - This information can be found in EP-AA-1004, definition section:</p> <p>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.</p> <p>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.</p> <p>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.</p> <p>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.</p> <p>REQUIRED REFERENCES: None</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

35

ID: 27999

Points: 1.00

U2 is at rated power.

Annunciator 902-7 A-4, TURBINE TRIP, alarms.

- Turbine Stop Valves are still open
- The Turbine TRIP pushbuttons have been depressed
- The local Turbine TRIP pushbuttons at the front standard have been depressed

The Turbine Stop Valves are still open.

What **IMMEDIATE** action(s) are required **FIRST** per DOA 5600-01, TURBINE TRIP?

- A. Scram Reactor and initiate Iso Condenser or HPCI for pressure control.
- B. Direct an operator to close the 2-5699-38 and 2-5699-39 valves at the EHC skid.
- C. Reduce recirc flow and/or insert CRAM rods to reduce core thermal power below 38.5%
- D. Scram Reactor, trip and place both EHC pumps in PTL if Turbine does not trip in approximately 90 seconds.

Answer: D

Answer Explanation

Per DOA 5600-01 if a turbine trip signal is received and the turbine stop valves are not closed, then the turbine trip pushbuttons on the 902-7 panel and the front standard are depressed. If the stop valves are still open then a reactor scram is required. If the turbine does not trip and stop valves are still open tripping and placing both EHC pumps is required.

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

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:	27999
User-Defined ID:	27999
Cross Reference Number:	
Topic:	35 - Generic 2.4.49
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	<p>Objective: 29501LK078 References: DOA 5600-01 K/A: Generic 2.4.49 4.6/4.4 K/A: Ability to perform without reference to procedures those actions that require immediate operation of system components and controls. Safety Function: N/A CFR: 41.10 PRA: No Level: Memory Pedigree: New History: None</p> <p>Explanations: A. Incorrect - This would not be successful at this power level. Plausible because this is a possible immediate action of this procedure for a lower power trip. B. Incorrect - Not required at this point. Plausible because this is the first subsequent action after the appropriate immediate actions have been taken. C. Incorrect - Not an appropriate action due to the Stop Valves being open. Plausible because this is an immediate action of this procedure if the bypass valves are closes. D. Correct - Per DOA 5600-01 if a turbine trip signal is received and the turbine stop valves are not closed, then the turbine trip pushbuttons on the 902-7 panel and the front standard are depressed. If the stop valves are still open then a reactor scram is required. If the turbine does not trip and stop valves are still open tripping and placing both EHC pumps is required.</p> <p>Required References: None</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

36

ID: 27928

Points: 1.00

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

- MCC 39-2 experienced a loss of power

The loss of the MCC will result in a loss of the ____ (1) ____ and the ability to fill the ____ (2) ____.

- A. 1) 2/3 EDG Fuel Oil Transfer Pump
2) 2/3 EDG Day Tank
- B. 1) 2/3 EDG Fuel Oil Transfer Pump
2) U3 EDG Day Tank
- C. 1) U3 EDG Fuel Oil Transfer Pump
2) 2/3 EDG Day Tank
- D. 1) U3 EDG Fuel Oil Transfer Pump
2) U3 EDG Day Tank

Answer: D

Answer Explanation

When Drywell pressure climbed to greater than 2 psig, the Unit will scram and the Unit 3 and 2/3 EDGs will auto start. The running EDGs will begin to lower their subsequent Fuel Day Tank level, which will be refilled by the associated Fuel Oil Transfer Pump. Upon a loss of MCC 39-2, power will be lost to the U3 Fuel Oil Transfer Pump which will cause a loss of the ability to fill the Unit 3 EDG Day Tank.

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

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:	27928
User-Defined ID:	27928
Cross Reference Number:	
Topic:	36 - 264000.K6.02
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	<p>Objective: DRE264LN004.12.a Reference: DOA 6600-01, DOS 6700-05, DAN 902(3)-5 D-11, DOS 6600-14, M-41, DOA 6700-T1 K/A: 264000.K6.02 3.6 / 3.6 K/A: Knowledge of the effect that a loss or malfunction of the following will have on the EMERGENCY GENERATORS (DIESEL/JET): Fuel Oil Pumps CFR: 41.7 / 45.7 Safety Function: 6 PRA: Yes Level: Memory Pedigree: Bank History: None</p> <p>Explanation:</p> <p>A. Incorrect - The 2/3 EDG Fuel Oil Transfer Pump is powered by MCC 28-1 and 38-1, and has not lost power, thus the ability to fill the 2/3 EDG Day Tank is still available. Plausible because 2/3 EDG auxiliaries have multiple power supplies.</p> <p>B. Incorrect - The 2/3 EDG Fuel Oil Transfer Pump is powered by MCC 28-1 and 38-1, and has not lost power. The 2/3 EDG Fuel Oil Transfer Pump does NOT supply the U3 EDG Day Tank. Plausible because part 2 is correct and 2/3 EDG auxiliaries have multiple power supplies.</p> <p>C. Incorrect - Upon a loss of MCC 39-2, power will be lost to the U3 Fuel Oil Transfer Pump which will cause a loss of the ability to fill the Unit 3 EDG Day Tank, NOT the 2/3 EDG Day Tank. Plausible because part 1 is correct and there is still an ability to fill the 2/3 EDG Day Tank.</p> <p>D. Correct - When Drywell pressure climbed to greater than 2 psig, the Unit will scram and the Unit 3 and 2/3 EDGs will auto start. The running EDGs will begin to lower their subsequent Fuel Day Tank level, which will be refilled by the associated Fuel Oil Transfer Pump. Upon a loss of MCC 39-2, power will be lost to the U3 Fuel Oil Transfer Pump which will cause a loss of the ability to fill the Unit 3 EDG Day Tank.</p> <p>REQUIRED REFERENCES: None</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

37

ID: 27976

Points: 1.00

A transient has occurred resulting in a LOOP and Drywell pressure reaching a maximum of +4 psig.

- Drywell pressure is now 1 psig and steady
- RPV level is -18 inches and going down slowly
- RPV pressure is 230 psig and going down slowly
- Core Spray outboard injection valve 3-1402-24A is full Closed
- LPCI is aligned to Max Torus Cooling
- Core Spray inboard injection valve 3-1402-25A is full Open and will not close

To minimize EDG loading, the 3B CS pump had been previously placed in PTL.

What actions are required to raise RPV water level using Core Spray?

- A. Secure the 3C LPCI pump. Start the 3B CS pump to control RPV level.
- B. Open the 3-1402-24A valve, then throttle the 3-1402-24A valve for level control.
- C. Open the 3-1402-24A valve. Close the 3-1402-25A valve and throttle the 3-1402-24A to control injection rate.
- D. Close the 3-1402-25A valve, then open the 3-1402-24A valve. Then reopen the 3-1402-25A valve and throttle the 3-1402-24A valve to control level.

Answer: A

Answer Explanation

Once the 3-1402-25A is closed the 3-1402-24A can be opened. Once the 3-1402-24A is open then the 3-1402-25A can be re-opened for injection. Below 350 psig Reactor pressure, both valves can be opened simultaneously, however, an interlock on these valves requires the 3-1402-25 valve to be closed before the 3-1402-24 valve can be opened. After the 3-1402-24 valve is open, the 3-1402-25 valve can be opened. Example: <350 psig, 3-1402-25 valve open, 3-1402-24 valve closed. In order to get both valves open: CLOSE the 3-1402-25 valve, OPEN the 3-1402-24 valve, then open the 3-1402-25 valve. Because the 3-1402-25A cannot be closed then the interlock will remain in effect. The only way to raise level is to swap pumps.

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

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:	27976
User-Defined ID:	27481
Cross Reference Number:	
Topic:	37 - 209001.K3.01
Comments:	<p>Objective: DRE209LN001.06 Reference: DOP 1400-02 K/A: 209001.K3.01 3.8 / 3.9 K/A: Knowledge of the effect that a loss or malfunction of the LOW PRESSURE CORE SPRAY SYSTEM will have on the following: Reactor water level CFR: 41.7 Safety Function: 2 & 4 Level: High Pedigree: Bank History: 18-1 NRC</p> <p>Explanation:</p> <p>A. Correct - Once the 3-1402-25A is closed the 3-1402-24A can be opened. Once the 3-1402-24A is open then the 3-1402-25A can be re-opened for injection. Below 350 psig Reactor pressure, both valves can be opened simultaneously, however, an interlock on these valves requires the 3-1402-25 valve to be closed before the 3-1402-24 valve can be opened. After the 3-1402-24 valve is open, the 3-1402-25 valve can be opened. Example: <350 psig, 3-1402-25 valve open, 3-1402-24 valve closed. In order to get both valves open: CLOSE the 3-1402-25 valve, OPEN the 3-1402-24 valve, then open the 3-1402-25 valve. Because the 3-1402-25A cannot be closed then the interlock will remain in effect. The only way to raise level is to swap pumps.</p> <p>B. Incorrect - This is plausible because with the 3-1402-25 vlv open, opening the 3-1402-24 vlv would provide a flowpath. An interlock exist that requires the 3-1402-25A must be full closed prior to opening the 3-1402-24A.</p> <p>C. Incorrect - The 3-1402-24A cannot be opened until the 3-1402-25A is closed. The 3-1402-25A does not have a PTL feature. Plausible because the CS pump suction valve has a PTC feature (same function as PTL).</p> <p>D. Incorrect - Plausible because If the 3-1405-25A could be closed this would be the correct answer.</p> <p>Note: the 1402-24A/B valves cannot be throttled from the main control room. Core Spray logic does not seal in and thus not present with DW pressure below 2 psig.</p> <p>REQUIRED REFERENCES: None.</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

38

ID: 27933

Points: 1.00

Unit 2 was operating at near rated power with two (2) Circulating Water pumps running when annunciator 902-7 B-15 "SCREEN WASH CONTROL PANEL TROUBLE" alarms.

An Equipment Operator reported the following:

- A large buildup of debris on the inlet side of the traveling screens.
- There is a 14 inch level difference across the traveling screens.

Then 15 minutes later the following occurred:

- The NSO reports Condenser vacuum trending down at a rate of 0.5 inches Hg per minute.
- The EO reports the level difference is getting worse as more debris is accumulating on the traveling screens.

Condenser vacuum is observed as 22.5 in Hg.

Based on these reports, which of the following actions must **FIRST** be performed, **AND** what is the reason for the action?

- A. Start the standby Circulating Water pump;
to maintain vacuum and CCSW system available.
- B. Start the standby Circulating Water pump;
to protect the Condenser from over pressure and maintain heat sink available.
- C. Enter DGP 2-3, REACTOR SCRAM;
to maintain vacuum and CCSW system available.
- D. Enter DGP 2-3, REACTOR SCRAM;
to protect the Condenser from over pressure and maintain heat sink available.

Answer: D

Answer Explanation

The DOA 4400-01 states that if Condenser vacuum is dropping, then enter DOA 3300-02, which states if a scram is imminent due to loss of condenser vacuum, then enter and perform DGP 02-03. The Tech Spec Bases states that the reason for a low vacuum scram is to protect the main condenser from over pressure and maintain the heat sink available.

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

Question 38 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:	27933
User-Defined ID:	27933
Cross Reference Number:	
Topic:	38 - 295002.K1.03
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	<p>Objective: 27501LK012 Reference: DOA 4400-01, DOA 3300-02, Tech Spec Bases 3.3.1.1, DOA 4400-06, OP-DR-103-102-1002 K/A: 295002.K1.03 4.6 / 4.6 K/A: Knowledge of the operational implications of the following concepts as they apply to LOSS OF MAIN CONDENSER VACUUM: Loss of heat sink CFR: 41.8 to 41.10 Level: High Safety Function: 3 Pedigree: Bank History: 2008 Cert</p> <p>Explanation:</p> <p>A. Incorrect - Starting the 3rd Circ Water pump would make the problem worse by lowering the amount of available water quicker therefore vacuum would lower even quicker. CCSW pumps will be available for longer than the requirement to Scram would occur.</p> <p>B. Incorrect - Starting the 3rd Circ Water pump would make the problem worse by lowering the amount of available water quicker therefore vacuum would lower even quicker. Second part is correct</p> <p>C. Incorrect - First part is correct. CCSW pumps will be available for longer than the requirement to Scram would occur.</p> <p>D. Correct - The DOA 4400-01 states that if Condenser vacuum is dropping, then enter DOA 3300-02, which states if a scram is imminent due to loss of condenser vacuum, then enter and perform DGP 02-03. The Tech Spec Bases states that the reason for a low vacuum scram is to protect the main condenser from over pressure and maintain the heat sink available.</p> <p>REQUIRED REFERENCES: None.</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

39

ID: 27938

Points: 1.00

<QQ 12758(14Unit 2 is in refuel, when a refueling accident causes ALL radiation instruments on the refuel floor to RISE to 35 mrem/hr.

What are the consequences of the radiation level?

- A. The Reactor Building Ventilation system will isolate.
- B. The Reactor Building overhead crane hoist RAISE function is inhibited **ONLY**.
- C. The Reactor Building overhead crane RAISE and LOWER function is inhibited.
- D. The Reactor Building overhead crane hoist LOWER function is inhibited **ONLY**.

Answer: B

Answer Explanation

At 35 mrem/hr on the refuel floor, the RB overhead crane hoist RAISE function is inhibited.

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

Question 39 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:	27938
User-Defined ID:	14660
Cross Reference Number:	
Topic:	39 - 295023.G.2.1.28
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	<p><QQ 12758(14Objective: DRE272LN002.06 Reference: DFP 0850-03, DAN 923-5 A-1, DFP 0800-02, DOS 0800-01, DOA 0800-03, DFP 0800-20 K/A: 295023.G.2.1.28 4.1/4.1 K/A: Refueling Accidents: Knowledge of the purpose and function of major system components and controls. CFR: 41.7 Safety Function: 8 PRA: No Level: Memory Pedigree: Bank History: 2009 Cert, 2013 Cert</p> <p>Explanations: >30 mrem/hr sensed at the cab of the overhead crane inhibits the RAISE (not lower) function. Reactor Building Vent actions occur at nominal setpoint of 45 mrem/hr per DAN 923-5 A-1.</p> <p>A. Incorrect - Reactor Building Vent actions occur at nominal setpoint of 45 mrem/hr per DAN 923-5 A-1. Plausible because it would be true at >100 mrem/hr.</p> <p>B. Correct - At 35 mrem/hr on the refuel floor, the RB overhead crane hoist RAISE function is inhibited.</p> <p>C. Incorrect - The Reactor Building overhead crane LOWER function is not inhibited with refuel floor rad levels at 35 mrem/hr. Plausible because there are interlocks that inhibit crane hoist LOWER functions.</p> <p>D. Incorrect - The Reactor Building overhead crane LOWER function is not inhibited with refuel floor rad levels at 35 mrem/hr. Plausible because there are interlocks that inhibit crane hoist LOWER functions.</p> <p>REQUIRED REFERENCES: None.</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

40

ID: 27949

Points: 1.00

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

- RPV water level is -68 inches.
- Smoke began billowing from the Control Room Ventilation ducts.

Where can the Operator monitor current RPV water level in the Reactor Building?

- A. 2202-5 and 2202-6 instrument racks.
- B. 2202-5 and 2202-7 instrument racks.
- C. 2202-6 and 2202-8 instrument racks.
- D. 2202-7 and 2202-8 instrument racks.

Answer: D

Answer Explanation

The 2202-7 and 2202-8 racks are Fuel Zone reading that are calibrated cold and read from -340 inches to +60 inches.

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

Question 40 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:	27949
User-Defined ID:	27949
Cross Reference Number:	
Topic:	40 - 216000.K4.01
Num Field 1:	0.00
Num Field 2:	0.00
Text Field:	
Comments:	<p>Objective: 299L-03-04 Reference: DSSP 100-CR, DEOP 0010-00, TSG-2 K/A: 216000.K4.01 3.6 / 3.6 K/A: Nuclear Boiler Instrumentation: Knowledge of NUCLEAR BOILER INSTRUMENTATION design feature(s) and/or interlocks which provide for the following: Reading of nuclear boiler parameters outside the control room. CFR: 41.7 Level: Memory Safety Function: 7 Pedigree: Bank History: 2008 NRC</p> <p>Explanation:</p> <p>A. Incorrect - The 2202-5 and 22002-6 racks provide local MR level indication from +60 to -60 inches. Plausible because this would be correct if the level stated was in this band.</p> <p>B. Incorrect - The 2202-7 band is correct but the 2202-5 only reads MR down to -60 inches. Plausible because this would be correct if level stated was greater than -60 inches. Part 2 is correct.</p> <p>C. Incorrect - The 2202-8 band is correct but the 2202-6 only reads MR down to -60 inches. Plausible because this would be correct if level stated was greater than -60 inches. Part 2 is correct.</p> <p>D. Correct - The 2202-7 and 2202-8 racks are Fuel Zone reading that are calibrated cold and read from -340 inches to +60 inches.</p> <p>REQUIRED REFERENCES: None.</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

41

ID: 27947

Points: 1.00

Given the following conditions for Unit 2:

- A TIP trace is being run in the automatic mode.

The following events occur:

- The TIP detectors are between Core Top & Bottom Limits
- Power is lost Bus 29
- Drywell pressure reaches 2.5 psig

Based on these conditions, the TIPs will _____ .

- A. remain in their present positions and the Shear Valves cannot be fired.
- B. remain in their present positions and the Shear Valves will have to be fired.
- C. shift to the manual reverse mode and withdraw to their in-shield position. Then the Ball Valves will close.
- D. shift to the manual reverse mode and withdraw to their in-shield position. The associated Ball Valve will **NOT** close, and the Shear Valve will have to be fired.

Answer: B

Answer Explanation

DW pressure greater than 2 psig is an automatic withdraw signal for the TIP that is fully inserted into the core (Group II). However, the drive motors are powered from MCC 29-1. With a loss of Bus 29 (and therefore, MCC 29-1), the drive mechanisms will not operate and the ball valves will lose power - trying to close with the tip detector running through it. The squib shear valves are powered from 2A-1 and are fired manually from the control room (there is no automatic firing associated).

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

Question 41 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:	27947
User-Defined ID:	27947
Cross Reference Number:	
Topic:	41 - 215001.A4.03
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	<p>Objective: DRE215LN001.06 Reference: DOP 0700-06, 12E-2463B K/A: 215001.A4.03 3.0/3.1 K/A: Traversing In-Core Probe: Ability to manually operate and/or monitor in the control room: Isolation valves. CFR: 41.7 Safety Function: 7 Level: High Pedigree: Bank History: 2015 Cert</p> <p>Explanation:</p> <p>A. Incorrect - The squib shear valves are powered from 2A-1 and are fired manually from the control room. Plausible because part 1 is correct and must determine the power supply to the sheer valves is still available with a loss of Bus 29.</p> <p>B. Correct - DW pressure greater than 2 psig is an automatic withdraw signal for the TIP that is fully inserted into the core (Group II). However, the drive motors are powered from MCC 29-1. With a loss of Bus 29 (and therefore, MCC 29-1), the drive mechanisms will not operate and the ball valves will lose power - trying to close with the tip detector running through it. The squib shear valves are powered from 2A-1 and are fired manually from the control room (there is no automatic firing associated).</p> <p>C. Incorrect - The TIP drive motors do not have power with a loss of Bus 29. Plausible because this is the correct answer without the loss of power.</p> <p>D. Incorrect - The TIP drive motors do not have power with a loss of Bus 29. Plausible because this correct except that the Ball Valves will attempt to close with the detector running through it.</p> <p>REQUIRED REFERENCES: None.</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

42

ID: 27939

Points: 1.00

Unit 2 is operating at rated power for the past 6 months when the Main Turbine spuriously trips.

How does the plant respond and why?

- A. **ONLY** Main Turbine Bypass Valves open to reduce RPV pressure rise and maintain margin to the Minimum Critical Power Ratio (MCPR) limit.
- B. **ONLY** ERVs open to reduce RPV pressure rise and maintain margin to the Minimum Critical Power Ratio (MCPR) limit.
- C. Main Turbine Bypass Valves **AND** ERVs open to reduce RPV pressure rise and maintain margin to the Minimum Critical Power Ratio (MCPR) limit.
- D. Main Turbine Bypass Valves **AND** ERVs open to reduce RPV pressure rise and maintain margin to the Power minus Precondition State (P-PCS) limit.

Answer: A

Answer Explanation

During a scram from rated power, no ERV actuation occurs without additional component failures. The thermal limit of concern is MCPR.

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

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:	4.00
System ID:	27939
User-Defined ID:	27939
Cross Reference Number:	
Topic:	42 - 295005.K2.07
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	<p>Objective: DRE245LN001.03 Reference: UFSAR 15.2.2.1, Tech Spec Bases 3.7.7 K/A: 295005.K2.07 3.6 / 3.7 K/A: Knowledge of the reasons for the following responses as they apply to MAIN TURBINE GENERATOR TRIP: Bypass valve operation CFR: 41.5 Safety Function: 3 Level: High Pedigree: New History: N/A</p> <p>Explanation:</p> <p>A. Correct - During a scram from rated power, no ERV actuation occurs without additional component failures. The thermal limit of concern is MCPR.</p> <p>B. Incorrect - P-PCS violations are not of concern given the conditions in the stem. Plausible because only the Bypass valves will open to limit RPV pressure rise.</p> <p>C. Incorrect - ERV actuation does not occur during a scram from full power without additional failures. Plausible because MCPR is the thermal limit of concern.</p> <p>D. Incorrect - ERV actuation does not occur during a scram from full power without additional failures. Plausible because at that high a power, if a group 1 isolation or bypass valve failure, the ERV's would open.</p> <p>REQUIRED REFERENCES: None.</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

43

ID: 27940

Points: 1.00

<QQ 986923(1410)><<Given the following conditions:

- Unit 2 startup is in progress.
- Rx Pressure is 950 psig.
- Core flow is 30%.
- Both recirc loops are in operation.

A feedwater transient occurs on Unit 2.

Which one of the following violates a reactor core Safety Limit under these conditions?>>

- A. MCPR at 1.07
- B. Rx power rises to 30%
- C. RPV level drops to -130 inches
- D. Steam dome pressure rises to 1300 psig

Answer: A

Answer Explanation

Per TS 2.1.1.2, MCPR is required to be ≥ 1.08 for 2 loop operation.

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

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:	2.00
System ID:	27940
User-Defined ID:	27940
Cross Reference Number:	
Topic:	43 - 295014.A2.05
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	<p><QQ 986923(1411)Objective: 299LN001.03 Reference: TS 2.0 K/A: 295014.A2.05 4.2 / 4.6 K/A: Ability to determine and/or interpret the following as they apply to INADVERTENT REACTIVITY ADDITION : Violation of safety limits CFR: 41.10 / 43.5 / 45.13 Safety Function: 1 Level: Memory Pedigree: New History: N/A</p> <p>Explanations: A. Correct - Per TS 2.1.1.2, MCPR is required to be ≥ 1.08 for 2 loop operation. B. Incorrect - Power must be $< 25\%$ if pressure is < 685 psig and core flow is $\geq 10\%$ rated core flow. Plausible because a feedwater transient can affect Rx power. C. Incorrect - The safety limit for level is TAF (-170"). Plausible because a feedwater transient can affect level. D. Incorrect - Pressure safety limit is 1345 psig. Plausible because a feedwater transient can affect pressure.</p> <p>REQUIRED REFERENCES: None</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

44

ID: 27944

Points: 1.00

Unit 2 is at full power when the following alarms come in:

- 902-4 D-16, PUMPBACK SYS PRESS LO
- 902-4 A-16, DRYWELL PNEU SPLY TANK PRESS LO
- 902-4 B-16, DRYWELL PNEU SPLY TROUBLE

Two (2) minutes later the following occurred:

- The inboard MSIV's go closed immediately followed by a scram.

The Assist NSO is directed to control pressure with ADS valves and Iso Condenser.

Which statement describes the availability of the Target Rock relief valve to control pressure?

The Target Rock is _____.

- A. available for limited use due to its accumulator.
- B. not available for use due to loss of Drywell Pneumatics.
- C. available for unlimited cycles due to cross-tie to instrument air.
- D. available for limited use due to the Drywell Pneumatic receiver.

Answer: A

Answer Explanation

The Target Rock relief valve is normally operated by Drywell pneumatics but has a backup accumulator for limited operation in case Drywell pneumatics is lost. The alarm and Group 1 isolation are definite indications of a loss of Drywell pneumatics to the Drywell.

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

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:	27944
User-Defined ID:	27944
Cross Reference Number:	LI
Topic:	44 - 218000.A1.03
Num Field 1:	0.00
Num Field 2:	0.00
Text Field:	
Comments:	<p>Objective: 239LN001.03 Reference: M-37, DEOP Technical Bases K/A: 218000.A1.03 3.2 / 3.4 K/A: Ability to predict and/or monitor changes in parameters associated with operation the AUTOMATIC DEPRESSURIZATION SYSTEM controls including: ADS valve air supply pressure. Safety Function: 3 PRA: No CFR: 41.5 Level: High Pedigree: New History: N/A</p> <p>Explanations:</p> <p>A. Correct - The Target Rock relief valve is normally operated by Drywell pneumatics but has a backup accumulator for limited operation in case Drywell pneumatics is lost. The alarms and Group 1 isolation are definite indications of a loss of Drywell pneumatics to the Drywell.</p> <p>B. Incorrect - an accumulator allows for limited use of the Target Rock on a loss of DW pneumatics. Plausible because DW pneumatics are lost and must be aware of the accumulator.</p> <p>C. There is no cross connect from Instrument air to the Drywell pneumatic loads in the Drywell. Plausible because instrument air is the normal supply to the outboard MSIVs</p> <p>D. Incorrect - The Drywell pneumatic Receiver is upstream of the isolation point of the Drywell pneumatics to the Drywell and cannot supply the loads in this situation. Plausible because DW pneumatics is the normal supply to the Target Rock but is not available as it is upstream of the isolation point to the DW.</p> <p>Required References: None</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

45

ID: 27941

Points: 1.00

The Unit Supervisor has directed you to initiate the Isolation Condenser and establish a 75°F/hr cooldown rate on Unit 2.

Over the next 30 minutes, how will the cooldown rate be reflected on TR 2-1340-1, U2 ISOL CONDR/DW ATMOS TEMPS?

ISOLATION CONDENSER		
VIOLET	1 ISOL COND TUBE SIDE INLET 2A	150 °F
RED	2 ISOL COND TUBE SIDE INLET 2B	150 °F
BLACK	3 ISOL COND SHELL SIDE	150 °F
GREEN	4 SPARE	
BLUE	5 2A RECIRC PP MTR AREA	180 °F
BROWN	6 2B RECIRC PP MTR AREA	160 °F
7	AIR INLET TO 2D IN CLR	160 °F
8	AIR INLET TO 2A IN CLR	180 °F
9	MSL RLF VLV AREA SOUTH	210 °F
10	MSL RLF VLV AREA NORTH	210 °F
11	SPARE	
12	SPARE	

- A. Points 1,2, and 3 will lower in value **ONLY**.
- B. Points 1,2, and 3 will initially rise, then points 1 and 2 will lower in value.
- C. Points 5 and 6 will rise in value.
- D. Points 9 and 10 will rise in value.

Answer: B

Answer Explanation

Upon initiation of the Isolation Condenser, IC tube side temperatures will rise and shell side temperature will rise. These temperatures are indicated on the 902-3 panel. Points 1 and 2 on TR 2-1340-1 reflect IC tube side temperature, while point 3 reflects IC shell side (heat sink) temperature.

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

Question 45 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:	27941
User-Defined ID:	27941
Cross Reference Number:	
Topic:	45 - 207000.K5.07
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	<p>Objective: 207LN001.11 Reference: M-28 K/A: 207000.K5.07 2.7 / 2.8 K/A: Knowledge of the operational implications of the following concepts as they apply to ISOLATION (EMERGENCY CONDENSER: Temperature sensing. CFR: 41.5 Safety Function: 4 Pedigree: Bank History: 2017 NRC Level: Memory</p> <p>Explanation:</p> <p>A. Incorrect - This is plausible because the cooldown rate would cause temperatures to lower in the RPV. The candidate must understand the IC temperatures will rise when placed in service.</p> <p>B. Correct - Upon initiation of the Isolation Condenser, IC tube side temperatures will rise and shell side temperature will rise. These temperatures are indicated on the 902-3 panel. Points 1 and 2 on TR 2-1340-1 reflect IC tube side temperature, while point 3 reflects IC shell side (heat sink) temperature.</p> <p>C. Incorrect - These points are used to determine Drywell temperature. Plausible because they are part of the Iso Condenser recorder and temperatures in some areas of the DW are affected by the use of the IC.</p> <p>D. Incorrect - These points reflect MSL RLF VLV areas. These points are used to determine if RPV level indication is valid. Plausible because they are part of the Iso Condenser recorder and temperatures in some areas of the DW are affected by the use of the IC.</p> <p>REQUIRED REFERENCES: None.</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

46

ID: 27981

Points: 1.00

The following plant conditions exist on Unit 2:

- RPV level is -240" and trending up at a rate of 5"/hour
- RPV pressure is 560 psig and trending down at a rate of 50 psig/hour
- Drywell temperature is 270 degrees F and trending up at a rate of 10 degrees/hour
- The 316 Containment Spray Permissive Keylock Switch is in MANUAL

The Unit Supervisor then orders initiation of Drywell Sprays.

Which of the following conditions would allow Drywell Spray valves to be opened?

Note: 317 is the 2/3 Core Coverage Override Keylock Switch
318 is the CCSW Pump Start Permissive Keylock Switch

- A. 317 in MANUAL OVERRIDE
- B. 318 in MANUAL OVERRIDE
- C. 318 in AUTO AND 30 second timer timed out
- D. 317 in AUTO AND Torus pressure greater than 1 psig

Answer: A

Answer Explanation

With RPV level below 2/3 core height, the 317 keylock switch must be taken to MANUAL OVERRIDE to allow the Drywell Sprays to open.

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

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:	4.00
System ID:	27981
User-Defined ID:	27981
Cross Reference Number:	
Topic:	46 - 295028 A1.01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	<p>Objective: 203LN001.6 References: DOP 1500-03 K/A: 295028 A1.01 3.8/3.9 K/A: Ability to operate and/or monitor the following as they apply to HIGH DRYWELL TEMPERATURE: Drywell spray. Safety Function: 5 CFR: 41.7 PRA: No Level: High Pedigree: New History: None</p> <p>Explanations:</p> <p>A. Correct - With RPV level below 2/3 core height, the 317 keylock switch must be taken to MANUAL OVERRIDE to allow the Drywell Sprays to open.</p> <p>B. Incorrect - The 318 keylock is required to be taken to MANUAL OVERRIDE with >2 psig in the Drywell in order to start the CCSW pumps. Plausible because the CCSW pumps are required to be started when placing Torus Cooling online.</p> <p>C. Incorrect - The 318 keylock is required to be taken to MANUAL OVERRIDE with >2 psig in order to start the CCSW pumps. Plausible because the CCSW pumps are required to be started when placing Torus Cooling online and the 2-1501-11 valves are interlocked open for 30 seconds on an initiation signal.</p> <p>D. Incorrect - The 317 keylock switch is require to be in MANUAL OVERRIDE with RPV level <-191 in order for the Drywell Spray valves to opened. The logic does not monitor Torus pressure. Plausible because if RPV levels is <-191 then the 317 keylock switch is left in auto to operate the Drywell Spray valves and the logic monitors Drywell pressure and it must be >1 psig for the Drywell Spray valves to be opened.</p> <p>REQUIRED REFERENCE: None</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

47

ID: 27923

Points: 1.00

Unit 2 was operating at near rated power when a loss of air supply and a failure of the Bailey System occurred and the FWRVs cannot be opened.

The Operating Team was directed to open the valves manually.

The **FIRST** action to perform per DOA 0600-01, TRANSIENT LEVEL CONTROL, is to

- A. isolate air to the FWRV that will be operated.
- B. fully inserting the fork into the center stem groove.
- C. install the handwheel and rotate the handwheel clockwise.
- D. install the handwheel and rotate the handwheel counter clockwise.

Answer: A

Answer Explanation

Upon a loss of air supply and failure of the Bailey System, the FWRVs cannot be operated. To operate the valves manually, air must first be isolated to the FWRV before bleeding air from the actuator, unscrewing the coupling fork, installing the handwheel and rotating it clockwise, then inserting the fork before rotating the handwheel counter clockwise to open the valve.

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

Question 47 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:	27923
User-Defined ID:	27923
Cross Reference Number:	
Topic:	47 - 259002.G.2.4.35
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	<p>Objective: DRE259LN002.08 Reference: DOA 0600-01 K/A: 259002.G.2.4.35 3.8 / 4.0 K/A: Reactor Water Level Control System: Knowledge of local auxiliary operator tasks during an emergency and the resultant operational effects. CFR: 41.10 / 43.5 / 45.13 Safety Function: 2 Level: Memory Pedigree: Bank History: None</p> <p>Explanation:</p> <p>A. Correct - Upon a loss of air supply and failure of the Bailey System, the FWRVs cannot be operated. To operate the valves manually, air must first be isolated to the FWRV before bleeding air from the actuator, unscrewing the coupling fork, installing the handwheel and rotating it clockwise, then inserting the fork before rotating the handwheel counter clockwise to open the valve.</p> <p>B. Incorrect - Fully inserting the fork is not performed until after isolating air to the valve and installing the handwheel. Plausible because fully inserting the fork is a correct step, just not first.</p> <p>C. Incorrect - The handwheel is not installed and rotated clockwise until after air is bled from the valve. Plausible because fully inserting the fork is a correct step, just not first.</p> <p>D. Incorrect - The handwheel is not installed and rotated counter-clockwise until after air is bled from the valve and the fork is installed into the center of the stem groove. Plausible because fully inserting the fork is a correct step, just not first.</p> <p>REQUIRED REFERENCES: None</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

48

ID: 12903

Points: 1.00

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

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

The EO was dispatched to the AEER and reported the following on the 902-63B panel:

- Normal A/C power has FAILED to ESS UPS.
- The LOW DC VOLTAGE light is illuminated.
- The DC VOLT meter indicates 175 volts and lowering.

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

- A. MCC 28-2
- B. Bus 25
- C. Bus 29
- D. Unit 2 250 VDC system

Answer: B

Answer Explanation

The order of power supplies to the Unit 2 ESS is: Bus 29, 250 VDC MCC 2, Bus 25 (through Voltage Regulator), MCC 28-2. Upon a loss of 24-1, Bus 29 is lost. With a loss of 250Vdc MCC 2, the next possible power supply is Bus 25 (through Voltage Regulator).

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

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:	3.00
System ID:	12903
User-Defined ID:	12903
Cross Reference Number:	
Topic:	48 - 262002.K6.02
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	<p>Objective: 262LN005.02, 03, & 06 Reference: DOP 6800-01 & DAN 902-8 F-8, 12E-2325 sh 1 & 2, DOP 0500-03 K/A: 262002.K6.02 2.8 / 3.1 K/A: Uninterruptable Power Supply (A.C./D.C.): Knowledge of the effect that a loss or malfunction of the following will have on the UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.): D.C. electrical power. CFR: 41.7 / 45.7) Level: High Safety Function: 6 Pedigree: Bank History: 2001 Cert, 2008 NRC</p> <p>Explanation:</p> <p>A. Incorrect - MCC 28-2 is a power supply to the ESS Bus but in the order in the last one after Bus 25. Plausible because it is a power supply to ESS.</p> <p>B. Correct - The order of power supplies to the Unit 2 ESS is: Bus 29, 250 VDC MCC 2, Bus 25 (through Voltage Regulator), MCC 28-2. Upon a loss of 24-1, Bus 29 is lost. With a loss of 250Vdc MCC 2, the next possible power supply is Bus 25 (through Voltage Regulator).</p> <p>C. Incorrect - Bus 29 is the normal A/C power to ESS and per the stem it has failed. Plausible because it is a power supply to ESS.</p> <p>D. Incorrect - U2 250 VDC is a power supply to the ESS Bus but with the DC Volt meter reading 175 volts and lowering it will not be selected to power the ESS. Plausible because it is a power supply to the ESS Bus.</p> <p>REQUIRED REFERENCES: None.</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

49

ID: 27927

Points: 1.00

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

The Unit 2 Emergency Diesel Generator auto started and powered Bus 24-1.

Several minutes later the U2 D/G TO BUS 24-1 ACB tripped.

With **NO** Operator action, which of the following components will no longer have power available?

1. 2B SDC Pump
2. 2C SDC Pump
3. 2A and 2B LPCI Pumps
4. 2C and 2D LPCI Pumps

- A. 1 **ONLY**
- B. 2 **ONLY**
- C. 1 **AND** 4
- D. 2, **AND** 4

Answer: C

Answer Explanation

Upon a loss of feed to Bus 24-1, based on U2 D/G TO BUS 24-1 ACB tripping, all power will be lost to the division 2 powered 2B SDC Pump and the 2C and 2D LPCI Pumps.

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

Question 49 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:	27927
User-Defined ID:	27927
Cross Reference Number:	
Topic:	49 - 264000.K3.03
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	<p>Objective: 262LN001.12 Reference: DOS 6700-04, DOA 6500-01 K/A: 264000.K3.03 4.1/4.2 K/A: Knowledge of the effect that a loss or malfunction of the EMERGENCY GENERATORS (DIESEL/JET) will have on following: Major loads powered from electrical buses fed by the emergency generator(s). CFR: 41.7 Safety Function: 6 PRA: Yes Level: High Pedigree: Bank History: None</p> <p>Explanations:</p> <p>A. Incorrect - Plausible because upon a loss of feed to division 2 Bus 24-1, power will be lost to the 2B SDC Pump, but will also be lost to the 2C and 2D LPCI Pumps.</p> <p>B. Incorrect - The 2C SDC Pump is powered from division 1 Bus 23-1, which is still energized from the 2/3 EDG. Plausible because the orientation of 4 Kv loads does not always follow conventional divisional separation.</p> <p>C. Correct - Upon a loss of feed to Bus 24-1, based on U2 D/G TO BUS 24-1 ACB tripping, all power will be lost to the division 2 powered 2B SDC Pump and the 2C and 2D LPCI Pumps.</p> <p>D. Incorrect - The 2C SDC Pump AND 2A and 2B LPCI Pumps are powered from division 1 Bus 23-1, which is still energized from the 2/3 EDG. Plausible because must understand that on a LOCA both EDGs start and load to dash busses.</p> <p>REQUIRED REFERENCES: None</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

50

ID: 27919

Points: 1.00

Unit 2 was operating at rated power with the 2A, 2B, 2C and 2D CCSW Pumps running to support Torus Cooling, when the 2A CRD Pump lost power due to BUS OVERCURRENT.

Which of the CCSW Pumps would still be running?

- A. 2C ONLY
- B. 2A and 2B **ONLY**
- C. 2C and 2D **ONLY**
- D. 2A, 2B, 2C, and 2D

Answer: C

Answer Explanation

The 2A CRD pump is powered by Bus 23, which experienced an overcurrent condition, causing it to become de-energized. The 2A and 2B CCSW pumps are also powered by Bus 23, and also would have lost power and would not be running. The 2C and 2D CCSW pumps are powered by Bus 24, which is still energized, so they would continue running.

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

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:	27919
User-Defined ID:	27919
Cross Reference Number:	
Topic:	50 - 400000.K2.01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	<p><QQ 329053(1411)><<Objective: DRE208LN001.12 Reference: DAN 902-8 G-3, DOS 6700-04, DOP 0300-01, DOP 1500-03 K/A: 400000.K2.01 2.9 / 3.0 K/A: Knowledge of electrical power supplies to the following: CCW pumps Safety Function: 8 CFR: 41.7 Level: High Pedigree: Bank History: 15-1 NRC, 16-1 NRC</p> <p>Explanation:</p> <p>A. Incorrect - The 2C and 2D CCSW pumps are powered by Bus 24, which has not experienced a loss of power, so they would continue running. Plausible because other systems are not powered from Bus 23.</p> <p>B. Incorrect - The 2A and 2B CCSW are powered by Bus 23, which has lost power, causing the pumps to become de-energized and not running. Plausible if Bus 24 had experienced the overcurrent.</p> <p>C. Correct - The 2A CRD pump is powered by Bus 23, which experienced an overcurrent condition, causing it to become de-energized. The 2A and 2B CCSW pumps are also powered by Bus 23, and also would have lost power and would not be running. The 2C and 2D CCSW pumps are powered by Bus 24, which is still energized, so they would continue running.</p> <p>D. Incorrect - The 2A and 2B CCSW are powered by Bus 23, which has lost power, causing the pumps to become de-energized and not running, while the 2C and 2D would still be energized by Bus 24 and running. Plausible if all CCSW pumps were powered from Bus 23.</p> <p>REQUIRED REFERENCES: None.</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

51

ID: 27942

Points: 1.00

Unit 2 is at 100% power when a leak develops in the drywell.

Drywell pressure is trending up at the following rate.

0700 - 1.2 psig
0715 - 1.35 psig
0730 - 1.5 psig

At the current trend, what is the **EARLIEST** an automatic RPS actuation will have **ACTUALLY** occurred?

- A. 0745
- B. 0815
- C. 0830
- D. 0845

Answer: B

Answer Explanation

Per DAN 902(3)-5 D-11 a RPS Scram must occur by ≤ 1.94 psig Drywell Pressure

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

Question 51 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	2
Difficulty:	4.00
System ID:	27942
User-Defined ID:	27466
Cross Reference Number:	
Topic:	51 - 212000.K1.13
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	<p>Objective: DRE212LN001.06.h Reference: DAN 902(3)-5 D-11 K/A: 212000.K1.13 3.5 / 3.6 K/A: Knowledge of the physical connections and/or cause-effect relationships between REACTOR PROTECTION SYSTEM and the following: containment pressure CFR: 41.2 to 41.9 Safety Function: 7 Level: High Pedigree: Bank History: 18-1 NRC</p> <p>Explanation:</p> <p>A. Incorrect - This time corresponds to 1.65 psig, the actuation will occur at 1.94 psig. Plausible due to the fact a math error could correlate to the right answer.</p> <p>B. Correct - This time would correlate to 1.95 psig, the actuation would have occurred at 1.94 psig.</p> <p>C. Incorrect - This time would correlate to 2.10 psig, the actuation will occur at 1.94 psig. Plausible due to greater than 2 psig.</p> <p>D. Incorrect - This time would correlate to 2.25 psig., the actuation will occur at 1.94 psig. Plausible due to the fact a math error could correlate to the right answer.</p> <p>REQUIRED REFERENCES: None.</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

52

ID: 27943

Points: 1.00

Unit 2 was operating at 100%

IMD reports the following LPRMs are out of tolerance (each APRM level has at least 2 good LPRM inputs):

- 11 LPRM inputs into APRM 1
- 8 LPRM inputs into APRM 3
- 8 LPRM inputs into APRM 5
- 11 LPRM inputs into APRM 6

What actions are required?

- A. Bypass APRM 1 **ONLY**
- B. Bypass APRMs 1 **AND** 6
- C. Insert a full scram
- D. Insert a 1/2 scram on RPS channel B

Answer: B

Answer Explanation

More than 50% of the LPRM inputs to APRM 6 and APRM 1 are unavailable. Therefore the APRM must be placed in bypass.

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

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:	4.00
System ID:	27943
User-Defined ID:	27472
Cross Reference Number:	
Topic:	52 - 215005.K4.08
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	<p>Objective: DRE215LN005.12.b Reference: DIS 0700-21, DOP 0700-04, DOP 0700-08 K/A: 215005.K4.08 2.7 / 3.1 K/A: Knowledge of APRM / LPRM design feature(s) and/or interlocks which provide for the following: Sampling of overall core power in each APRM (accomplished through LPRM assignments and symmetrical rod patterns) CFR: 41.7 Safety Function: 7 Level: High Pedigree: Bank History: 18-1 NRC, 19-1 CERT</p> <p>Explanation:</p> <p>A. Incorrect - APRM 1 and APRM 6 are both inoperable with 11 LPRM inputs inoperable. Plausible because APRM 1 is inoperable.</p> <p>B. Correct - More than 50% of the LPRM inputs to APRM 6 and APRM 1 are unavailable. Therefore the APRMs must be placed in bypass.</p> <p>C. Incorrect - This would be correct if insufficient LPRM inputs were available to 2 APRMs in RPS channel B. This is plausible because if both APRMs fed into the same RPS channel a 1/2 scram would be required.</p> <p>D. Incorrect - Bypassing APRM 1 and 6 will address this issue, since they are both inoperable. A SCRAM is not required. Plausible because both trains of RPS have 1 APRM channel inop.</p> <p>REQUIRED REFERENCES: None.</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

53

ID: 27980

Points: 1.00

Unit 2 was at 100% power when the following events occurred:

- RPV pressure rises to 1060 psig then returns to 1002 psig.
- Annunciator 902-5 H-5, RPV PRESSURE HI, was received and is not reset.
- Annunciator 902-5 C-13, CHANNEL A/B RPV PRESSURE HI-HI, was received and is now reset.
- All eight RPS Scram Solenoid Group Lights at 902-5 Panel remain lit.

Reactor Power is ____ (1) ____, a(an) ____ (2) ____ has occurred.

- A. (1) 0%
(2) Full Scram
- B. (1) 50%
(2) Half scram
- C. (1) 100%
(2) Electric ATWS
- D. (1) 100%
(2) Hydraulic ATWS

Answer: C

Answer Explanation

Reactor pressure rose above the high pressure scram setpoint, and returned to normal at power value. A RPS full scram should have occurred. Scram/HI pressure alarms were received, but the Scram Group Solenoid lights remained lit, indicating an electrical ATWS.

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

Question 53 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:	27980
User-Defined ID:	27980
Cross Reference Number:	
Topic:	53 - 212000 A1.06
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	<p>Objective: DRE212LN001.10 References: DAN 902(3)-5 H-5, DAN 902(3)-5 C-13 K/A: 212000 A1.06 4.2/-- K/A: Ability to predict and/or monitor changes in parameters associated with operating the REACTOR PROTECTION SYSTEM controls including: Reactor power. Safety Function: 7 CFR: 41.5 PRA: No Level: High Pedigree: New History: None</p> <p>Explanations:</p> <p>A. Incorrect - Power would not change due to electric ATWS. Plausible because RPS received a valid full scram signal and this would be correct without the electric ATWS.</p> <p>B. Incorrect - No rod movement would occur with a full electric ATWS. Plausible because the alarms stated could also provide a 1/2 scram signal if only 1 channel is activated. Must understand a 1/2 scram would not cause rod movement.</p> <p>C. Correct - Reactor pressure rose above the high pressure scram setpoint, and returned to normal at power value. A RPS full scram should have occurred. Scram/HI pressure alarms were received, but the Scram Group Solenoid lights remained lit, indicating an electrical ATWS.</p> <p>D. Incorrect - Power would not change, but the scram lights being lit are an indication of an electric ATWS. Plausible because part 1 is correct and part 2 would be correct if the scram lights were extinguished.</p> <p>REQUIRED REFERENCE: None</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

54

ID: 27977

Points: 1.00

Unit 3 was operating at near rated power, with DOS 1400-05 CORE SPRAY SYSTEM PUMP OPERABILITY AND QUARTERLY IST TEST WITH TORUS AVAILABLE, in progress on the 3A Core Spray subsystem.

MO 3-1402-4A, FLOW TEST VLV, had been throttled open for 15 seconds when the 3A Core Spray pump breaker was inadvertently tripped locally by a janitorial contractor.

What operator action is required?

- A. Declare A Core Spray inoperable and notify the Unit Supervisor.
- B. Declare A and B Core Spray inoperable and notify the Unit Supervisor.
- C. Immediately close MO 3-1402-4A and vent **ONLY** the 3A CS subsystem within 4 hours.
- D. Immediately close MO 3-1402-4A and vent **BOTH** Core Spray subsystems within 4 hours.

Answer: C

Answer Explanation

Per DOS 1400-05, Should the 2(3)A Core Spray Pump trip with MO 2(3)-1402-4A, FLOW TEST VLV, open, MO 2(3)-1402-4A must be IMMEDIATELY closed AND when the Core Spray System is required to be operable the 2(3)A Core Spray subsystem shall be vented per DOP 1400-03 within four (4) hours. Core Spray would not be INOP based on a accidental breaker trip.

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

Question 54 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:	27977
User-Defined ID:	27977
Cross Reference Number:	
Topic:	54 - 209001.K5.05
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	<p><QQ 14524(1411)><Objective: DRE209LN001.12 Reference: DOS 1400-05, I.T.S. 3.5.1 K/A: 209001.K5.05 2.5/-- K/A: Knowledge of the operational implications of the following concepts as they apply to LOW PRESSURE CORE SPRAY SYSTEM: System venting. Safety Function: 4 CFR: 41.5 PRA: Yes Level: High Pedigree: New History: None</p> <p>Explanations:</p> <p>A. Incorrect - A Core Spray is not inoperable because the vlv was closed after 15 seconds. Plausible because if the valve was not closed in 37 seconds this would be correct.</p> <p>B. Incorrect - Core Spray is not inoperable because the vlv was closed after 15 seconds. Plausible because if the valve was not closed in 37 seconds this would be correct.</p> <p>C. Correct - Per DOS 1400-05, Should the 2(3)A Core Spray Pump trip with MO 2(3)-1402-4A, FLOW TEST VLV, open, MO 2(3)-1402-4A must be <u>IMMEDIATELY</u> closed <u>AND</u> when the Core Spray System is required to be operable the 2(3)A Core Spray subsystem shall be vented per DOP 1400-03 within four (4) hours.</p> <p>D. Incorrect - the 3A CS subsystem ONLY (NOT both) is required to be vented with the CS System required to be operable. Plausible because 3A does need to be vented. 3B would need to be vented if declared inop.</p> <p>REQUIRED REFERENCES: None</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

55

ID: 27955

Points: 1.00

Unit 3 was operating at near rated power, when the 3A Recirc pump tripped. The Shift Manager has decided to remain in single loop operation while trouble shooting the cause of the pump trip.

Which of the following Thermal Limits are required to have correction factors implemented within 24 hours and why?

- A. LHGR, MCPR, and MAPLHGR, to insert conservative limits due to the effects of flow through the idle loop.
- B. LHGR, MAPLHGR, and MFLPD, to insert conservative limits due to the effects of flow through the idle loop.
- C. LHGR, MCPR, and MAPLHGR, assure plastic strain limit on the cladding nor the fuel centerline melt temperature is exceeded during operation.
- D. LHGR, MAPLHGR, and MFLPD, assure plastic strain limit on the cladding nor the fuel centerline melt temperature is exceeded during operation.

Answer: A

Answer Explanation

DGP 03-03 require MAPLHGR, LHGR, and MCPR limits to be adjusted for single loop operation within 24 hours. This is done to compensate for flow through the idle loop.

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

Question 55 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:	27955
User-Defined ID:	27955
Cross Reference Number:	
Topic:	55 - 295001.K3.05
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	<p>Objective: 29501LK095 Reference: DGP 3-3, DOS 500-15, TS 3.4.1 K/A: 295001.K3.05 3.2/3.5 K/A: Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION: Reduced loop operating requirements: Plant-Specific Safety Function: CFR: 41.8 to 41.10 PRA: No Level: Memory Pedigree: New History: N/A</p> <p>Explanations:</p> <p>A. Correct - DGP 03-03 require MAPLHGR, LHGR, and MCPR limits to be adjusted for single loop operation within 24 hours. This is done to compensate for flow through the idle loop.</p> <p>B. Incorrect - DGP 03-03 require MAPLHGR, LHGR, and MCPR limits to be adjusted for single loop operation within 24 hours not MFLPD. Plausible because the items listed are valid thermal limits, and the student must know which three are required to meet the single loop requirements and part 2 is correct.</p> <p>C. Incorrect - DGP 03-03 require MAPLHGR, LHGR, and MCPR limits to be adjusted for single loop operation within 24 hours. Plausible because the items listed are valid thermal limits. Part 2 is correct for MFLPD but not required for a pump trip.</p> <p>D. Incorrect - DGP 03-03 require MAPLHGR, LHGR, and MCPR limits to be adjusted for single loop operation within 24 hours not MFLPD. Plausible because the items listed are valid thermal limits. Part 2 is correct for MFLPD but not required for a pump trip.</p> <p>REQUIRED REFERENCES: None.</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

56

ID: 27946

Points: 1.00

Unit 2 is operating at 48% power with RPV water level in the normal band when the RWCU system tripped.

While restoring the RWCU system to operation following maintenance, RPV water level drops to +22 inches, resulting in a low level alarm.

This condition...

- A. can be attributed to improper filling and venting of the system.
- B. is a normal response since there is no filling and venting capabilities installed on the RWCU system.
- C. is a normal response and is caused by FWLC sensing the additional RWCU flow as it enters the feed piping.
- D. can be attributed to the addition of cold water to the downcomer region, raising its density and lowering the indicated level.

Answer: A

Answer Explanation

Per DOP 1200-03, inadequate filling and venting of the RWCU System can result in a reactor water level drop of up to 10 inches. This will result in a feedwater flow spike and power level increase.

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

Question 56 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:	27946
User-Defined ID:	27946
Cross Reference Number:	LI
Topic:	56 - 204000.K3.02
Num Field 1:	0.00
Num Field 2:	0.00
Text Field:	
Comments:	<p>Objective: 20400LK006 ILTS004 Reference: DOP 1200-03 K/A: 204000K3.02 3.1/-- K/A: Knowledge of the effect that a loss or malfunction of the RWCU will have of following: Reactor water level. CFR: 41.7 Safety Function: 2 PRA: No Level: Memory Pedigree: Bank History: N/A</p> <p>Explanation:</p> <p>A. Correct - Per DOP 1200-03, Inadequate filling and venting of the RWCU System can result in a reactor water level drop of up to 10 inches. This will result in a feedwater flow spike and power level increase.</p> <p>B. Incorrect - Filling and venting is available and required per procedure. Plausible because of potential transient it can cause. Filling and venting of the RWCU System with MO 2(3)-1201-7, RX RETURN VLV, closed can result in a pressure transient that can cause a reactor scram.</p> <p>C. Incorrect - If piping properly filled and vented, level will not drop to low level alarm setpoint. Plausible because level indication and power could be affected by the change in temperature and flow.</p> <p>D. Incorrect - If piping properly filled and vented, level will not drop to low level alarm setpoint. Plausible because per DOP 1200-03, Restart of the RWCU System will result in an injection of a cold slug of water into the reactor.</p> <p>REQUIRED REFERENCE: None</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

57

ID: 27957

Points: 1.00

During the Unit 2 EOs round, he notices breaker C-4 on MCC 29-3 which feeds the Unit 2 Refueling Platform receptacle is damaged.

Unit 3 Refueling Platform is OOS.

Because of the failure described above ____ (1) ____ . Fuel moves may be performed by ____ (2) ____ .

- A. (1) only the frame mounted and monorail hoists lose power
(2) installing temporary power supply per DFP 0800-21 REFUELING PLATFORM AND FUEL HANDLING GRAPPLE OPERATION
- B. (1) only the frame mounted and monorail hoists lose power
(2) connecting to the Unit 3 refueling platform receptacle per DOS 0800-01 REFUELING INTERLOCK CHECKS
- C. (1) all power to the refueling platform and interlocks are lost
(2) connecting to the Unit 3 refueling platform receptacle per DOS 0800-01 REFUELING INTERLOCK CHECKS
- D. (1) all power to the refueling platform and interlocks are lost
(2) installing temporary power supply per DFP 0800-21 REFUELING PLATFORM AND FUEL HANDLING GRAPPLE OPERATION

Answer: D

Answer Explanation

Normal Power Supply for Unit 2 is MCC 29-3 cubical C-4. The Refuel Platform is unique in that the power supply feed is integral to the cable that provides the RPIS Interlocks. Therefore, when powering the Outage Refuel Platform from a temporary power source, a special "Interlock Jumper Box" is utilized to separate the RPIS Interlock cabling from the power cabling. Any other methodology for powering the Refuel Platform with temporary power will result in no RPIS Interlocks.

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

Question 57 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:	27957
User-Defined ID:	27957
Cross Reference Number:	
Topic:	57 - 234000 A2.01
Num Field 1:	0.00
Num Field 2:	0.00
Text Field:	
Comments:	<p>Objective: DRE23400LK014 Reference: DAN 902(3)-5 H-3DAN 902(3)-5 H-3DAN 902(3)-5 H-3DAN 902(3)-5 H-3 DFP 0800-21 K/A: 234000 A2.01 3.3 / -- K/A: Ability to (a) predict the impacts of the following on the Fuel Handling Equipment: and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Interlock failure. CFR: 41.5 Safety Function: 8 Level: High Pedigree: New History: N/A</p> <p>Explanations:</p> <p>A. Incorrect - power is lost to all of the refueling platform as well as interlocks. Plausible because must know the power supplies for the platform as well as the interlocks and part 2 is correct.</p> <p>B. Incorrect - power is lost to all of the refueling platform as well as interlocks. Plausible because must know the power supplies for the platform as well as the interlocks. The U3 is powered from 39-3 but is currently OOS. Interlock checks would have to be performed prior to restarting moves.</p> <p>C. Incorrect - U3 power supply is OOS and no procedural guidance to tie it to U2. Plausible because part 1 is correct and U3 has a separate power supply. Interlock checks would have to be performed prior to restarting moves.</p> <p>D. Correct - Normal Power Supply for Unit 2 is MCC 29-3 cubical C-4. The Refuel Platform is unique in that the power supply feed is integral to the cable that provides the RPIS Interlocks. Therefore, when powering the Outage Refuel Platform from a temporary power source, a special "Interlock Jumper Box" is utilized to separate the RPIS Interlock cabling from the power cabling. Any other methodology for powering the Refuel Platform with temporary power will result in no RPIS Interlocks.</p> <p>Required Reference: None</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

58

ID: 27945

Points: 1.00

During a Reactor Startup on Unit 2:

- Rod H-7 has a target position of 24.
- Rod H-7 is currently being single notched from position 12 to position 14.
- The RMCS sequence timer locks up during the "drive out" portion of the cycle and the rod continues to move.

This condition would be interrupted by...

- A. an RWM rod block.
- B. a timer malfunction select block.
- C. nothing. The rod would continue to drift out.
- D. a rod drift condition which would generate a rod block.

Answer: B

Answer Explanation

The timer malfunction timer timing out generates a timer malfunction select block.

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

Question 58 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:	27945
User-Defined ID:	331372
Cross Reference Number:	LI
Topic:	58 - 201002.A3.04
Num Field 1:	0.00
Num Field 2:	0.00
Text Field:	
Comments:	<p>Objective: 201L-S2-10 Reference: DAN 902(3)-5 D-3, DOP 0400-01 K/A: 201002.A3.04 2.8 / -- K/A: Ability to monitor automatic operations of the REACTOR MANUAL CONTROL SYSTEM including: Rod movement sequence timer malfunction alarm: Plant-Specific CFR: 41.7 Level: Memory Safety Function: 7 Pedigree: New History: N/A</p> <p>Explanations:</p> <p>A. Incorrect - a RWM block would not occur until the rod was outside the target position. Plausible because if the timer malfunction block did not occur the rod would continue to move.</p> <p>B. Correct - The timer malfunction timer timing out generates a timer malfunction select block.</p> <p>C. Incorrect - motion would be stopped by the timer malfunction. Plausible because the drift would continue until greater than position 24 if not for the timer malfunction block.</p> <p>D. Incorrect - a rod drift condition would not directly cause a rod block. Plausible because when the rod drifted outside of boundary the rod will be blocked by the RWM.</p> <p>REQUIRED REFERENCE: None</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

59

ID: 27958

Points: 1.00

An ATWS has occurred on Unit 2 and the following conditions exist:

- A Group 1 Isolation is in.
- Reactor power is 30%.
- Reactor Pressure is 1200 psig.
- Reactor Water Level is -110 inches.

___(1)___ safety valve(s) have opened.

Tailpipe temperatures can be observed on ___(2)___ panel.

- A. (1) 1
(2) 902-3 Panel
- B. (1) 1
(2) 902-21 Panel
- C. (1) 3
(2) 902-3 Panel
- D. (1) 3
(2) 902-21 Panel

Answer: B

Answer Explanation

The target rock is considered a safety valve and per T.S. 3.4.3 is required to open at 1135 psig plus or minus 34.1 pounds. The next two safety valves open at 1240 psig plus or minus 37.2 pounds, therefore would not be open yet. Although light indication is available on the 902-3 panel, Tailpipe temperatures are located on the 902-21 panel.

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

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:	27958
User-Defined ID:	27958
Cross Reference Number:	
Topic:	59 - 239002.A3.03
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	<p>Obj: 239LN001.06 Reference: TS 3.4.3, DAN 902(3)-4 H-19 K/A: 239002.A3.03 3.6 / -- K/A: Ability to monitor automatic operations of the RELIEF/SAFETY VALVES including: Tail pipe temperatures CFR: 41.7 Safety Function: 3 Level: High Pedigree: Bank History: None</p> <p>Explanation:</p> <p>A. Incorrect - Tailpipe temps are located on the 902-21 panel. Plausible because the first part is correct and other temperatures are on the 902-3 panel.</p> <p>B. Correct - The target rock is considered a safety valve and per T.S. 3.4.3 is required to open at 1135 psig plus or minus 34.1 pounds. The next two safety valves open at 1240 psig plus or minus 37.2 pounds, therefore would not be open yet. Although light indication is available on the 902-3 panel.</p> <p>C. Incorrect - Three safety valves will be open at 1240 psig. Plausible because two safety valves open at 1240 psig plus the target rock and the second part is correct.</p> <p>D. Incorrect - Three safety valves will be open at 1240 psig and tailpipe temps are located on the 902-21 panel. Plausible because two safety valves open at 1240 psig plus the target rock and other temperatures are on the 902-3 panel.</p> <p>REQUIRED REFERENCE: None</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

60

ID: 27959

Points: 1.00

A malfunction occurred that caused a partial blockage of the 2-3399-132A, 2A OFFGAS CDSR COND INLET VLV, causing condensate flow to lower. Downstream Off Gas temperatures will ____ (1) ____ and Off Gas radiation levels will ____ (2) ____.

- A. (1) rise
(2) rise
- B. (1) rise
(2) lower
- C. (1) lower
(2) rise
- D. (1) lower
(2) lower

Answer: A

Answer Explanation

Lowering of the condensate flow through the Off Gas Condenser would result in increased carryover of moisture in the form of steam, thus placing a greater heat load/moisture load on the downstream components. The resultant effect would be an increase in temperature and in the release rate of radio-nuclides discharged to the atmosphere.

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

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:	4.00
System ID:	27959
User-Defined ID:	27959
Cross Reference Number:	
Topic:	60 - 271000.K6.10
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	<p>Obj: 271LN001.12b Reference: M-43 Sheet 2 K/A: 271000.K6.10 2.7 / -- K/A: Knowledge of the effect that a loss or malfunction of the following will have on the OFF GAS SYSTEM: Condensate system flow CFR: 41.7 / 45.7 Level: High Safety Function: 9 Pedigree: New History: N/A</p> <p>Explanation:</p> <p>A. Correct - Lowering of the condensate flow through the Off Gas Condenser would result in increased carryover of moisture in the form of steam, thus placing a greater heat load/moisture load on the downstream components. The resultant effect would be an increase in temperature and in the release rate of radio-nucleides discharged to the atmosphere.</p> <p>B. Incorrect - The first part is correct. With lowering of the ability to condense the steam in the Off Gas System then steam with radioactive material will carryover causing radiation levels to rise. Plausible because in other systems a loss of flow could cause rad levels to decrease.</p> <p>C. Incorrect - With lowering of the condensate flow through the Off Gas Condenser less cooling will take place therefore steam will carry over causing downstream temperatures to rise. The second part is correct. Plausible because in other systems a loss of flow could cause temperatures to rise.</p> <p>D. Incorrect - With lowering of the condensate flow through the Off Gas Condenser less cooling will take place therefore steam will carry over causing downstream temperatures to rise. With lowering of the ability to condense the steam in the Off Gas System then steam with radioactive material will carryover causing radiation levels to rise. Plausible because in other systems a loss of flow could cause temperatures to decrease and loss of flow could cause rad levels to decrease.</p> <p>REQUIRED REFERENCE: None</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

61

ID: 27962

Points: 1.00

Following a scram on Unit 2:

- Drywell pressure has exceeded 2 psig.
- Drywell temperature is 160 degrees F.

The US has ordered you to bypass DW cooler isolations per DEOP 500-2, BYPASSING INTERLOCKS AND ISOLATIONS.

The RO would ____ (1) ____ in the Aux Electric Room to ____ (2) ____ DW coolers.

- A. (1) lift leads
(2) AUTO start
- B. (1) pull fuses
(2) AUTO start
- C. (1) lift leads
(2) allow MANUAL restart
- D. (1) pull fuses
(2) allow MANUAL restart

Answer: C

Answer Explanation

The conditions in the stem require entry into DEOP 200-1. In order to restart the DW coolers, DEOP 500-2 is performed which requires the RO to lift leads in the AUX Electric room. This would allow a Manual restart of coolers.

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

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:	27962
User-Defined ID:	27962
Cross Reference Number:	
Topic:	61 - Generic 2.4.34
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	<p>Objective: 29502LK060(b) References: DEOP 200-1, DEOP 500-2, 12E-2393 K/A: Generic 2.4.34 4.2/4.1 K/A: Knowledge of RO tasks performed outside the main control room during an emergency and the resultant operational effects. Safety Function: 5 CFR: 41.10 PRA: No Level: High Pedigree: New History: N/A</p> <p>Explanations:</p> <p>A. Incorrect - Lifting leads is required to remove the trip logic but does not cause and auto start of the coolers. Plausible because part 1 is correct and removing the trip signal could cause an auto start in some cases.</p> <p>B. Incorrect - Pulling fuses is not required for DW coolers. Lifting leads is required to remove the trip logic but does not cause an auto start of the coolers. Plausible because pulling fuses is required to bypass RB vent isolations. removing the trip signal could cause an auto start in some cases.</p> <p>C. Correct - The conditions in the stem require entry into DEOP 200-1. In order to restart the DW coolers, DEOP 500-2 is performed which requires the RO to lift leads in the AUX Electric room. This would allow a Manual restart of coolers.</p> <p>D. Incorrect - removing the trip signal could cause an auto start in some cases. Plausible because pulling fuses is required to bypass RB vent isolations. Part 2 is correct</p> <p>REQUIRED REFERENCE: None</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

62

ID: 27975

Points: 1.00

Torus Temperature is 95°F and rising SLOWLY and no testing is in progress that would add heat to the Torus. You have been directed to place Torus Cooling in service per DOP 1500-02, TORUS WATER COOLING MODE OF LOW PRESSURE COOLANT INJECTION SYSTEM (Hardcard).

What is the reason for placing Torus Cooling in service under these conditions?

- A. To cool non-condensable gases and condense steam following a spurious ERV opening.
- B. To maintain Tech Spec operability of the Torus Average Bulk Temperature.
- C. To prevent violating the Torus Heat Capacity Limit following a Blowdown.
- D. To ensure there is sufficient NPSH for ECCS pump operation.

Answer: B

Answer Explanation

Per LCO 3.6.2.1: Suppression pool average temperature shall be: 95°F with THERMAL POWER > 1% RTP and no testing that adds heat to the suppression pool is being performed

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

Question 62 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:	27975
User-Defined ID:	27975
Cross Reference Number:	
Topic:	62 - 295026.K3.02
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	<p>Objective: 29502LK011 Reference: UFSAR 6.2.1, EOP-DEOP Technical Bases K/A: 295026.K3.02 3.9 / 4.0 K/A: Knowledge of the reasons for the following responses as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE: Suppression pool cooling CFR: 41.5 / 45.6 Safety Function: 5 Level: High Pedigree: New History: N/A</p> <p>Explanations:</p> <p>A. Incorrect - Torus cooling does not cool non-condensable gases. Plausible because it would be true for torus sprays.</p> <p>B. Correct - Per LCO 3.6.2.1: Suppression pool average temperature shall be: 95 ° F with THERMAL POWER > 1% RTP and no testing that adds heat to the suppression pool is being performed</p> <p>C. Incorrect - This is the basis for why an emergency depressurization is performed when torus temperature cannot be maintained below HCTL. Plausible because it would be true if temperature was at or above HCTL. This would be correct if the temperature was exceeded prior to blowdown.</p> <p>D. Incorrect - ECCS flow for Torus Cooling is not high enough at this torus temp to violate the LPCI / Core Spray NPSH Limit. Plausible because at a higher torus temperature or higher ECCS Flow then NPSH is an issue.</p> <p>REQUIRED REFERENCES: None</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

63

ID: 22583

Points: 1.00

Both Units were operating at near rated power, when annunciator 923-5 A-1, U2 RX BLDG VENT/EXH FAN TRIP, alarmed.

The NSO discovered amber TRIP lights illuminated for:

- 2A RX BLDG EXHAUST FAN
- 2C RX BLDG VENT FAN

Which 480V Bus is the NSO required to dispatch an EO to check the breaker indications for the above fans?

- A. Bus 25
- B. Bus 26
- C. Bus 28
- D. Bus 29

Answer: C

Answer Explanation

2A Rx Bldg Vent Fan and the 2C Rx Bldg Exhaust Fans are both powered from Bus 28. This is a common misconception, as the supply and vent fans are reversed, as stated below:

Bus 28: Vent "A",	Vent "C",	Exh, "A"
Bus 29: Vent "B",	Exh "B",	Exh "C"

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

Question 63 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:	22583
User-Defined ID:	22583
Cross Reference Number:	
Topic:	63 - 288000.K1.01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	<p>Objective: DRE288LN001.02 Reference: Electrical Print 12E-2306, DOP 6700-18 K/A: 288000.K1.01 2.6 / 2.6 K/A: Knowledge of the physical connections and/or cause effect relationships between PLANT VENTILATION SYSTEMS and the following: A.C. electrical. CFR: 41.2 to 41.9 / 45.7 to 45.8 Safety Function: 9 Level: Memory Pedigree: Bank History: 2009 NRC</p> <p>Explanation:</p> <p>A. Incorrect - Bus 25 is a non Safety Related Bus. Plausible because N TB Supply Fan and RFP Vent Fans are powered by this bus.</p> <p>B. Incorrect - Bus 26 is a non Safety Related Bus. Plausible because N TB Supply Fan and RFP Vent Fans are powered by this bus.</p> <p>C. Correct - 2A Rx Bldg Vent Fan and the 2C Rx Bldg Exhaust Fans are both powered from Bus 28. This is a common misconception, as the supply and vent fans are reversed, as stated below: Bus 28: Vent "A", Vent "C", Exh, "A" Bus 29: Vent "B", Exh "B", Exh "C"</p> <p>D. Incorrect - Bus 29 does not power 2A RB Vent Fan and 2C RB Exh Fan. Plausible because other RB Vent and Exh Fans are powered by Bus 29.</p> <p>REQUIRED REFERENCE: None.</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

64

ID: 27965

Points: 1.00

During performance of DFPS 4195-01, HALON SYSTEMS OPERABILITY, the AEER Halon system has one initial discharge cylinder and one extended discharge cylinder less than 80% of full charge.

What Tech Spec/TRM action is required?

- A. A dedicated roving fire watch.
- B. Stage backup fire suppression equipment for unprotected areas **ONLY**.
- C. A dedicated continuous fire watch with backup fire suppression equipment for unprotected areas.
- D. An hourly fire watch patrol with backup fire suppression equipment in the unprotected area.

Answer: C

Answer Explanation

Per TRM 3.7.k, 8 of 9 initial discharge cylinders and ALL 5 extended discharge cylinders must meet 95% full. With 1 extended discharge cylinder low, Halon to the AEER is INOP and requires a continuous fire watch with backup fire suppression equipment for unprotected areas.

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

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:	3.00
System ID:	27965
User-Defined ID:	27965
Cross Reference Number:	
Topic:	64 - 600000.G.2.2.39
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	<p>Objective: DRE286LN002.07 Reference: DFPS 4195-01, TRM 3.7.k K/A: 600000 G2.2.39 3.9/-- K/A: Ability to apply Technical Specification for a system: Fire on Site Safety Function: 8 CFR: 41.10 PRA: No Level: High Pedigree: New History: None</p> <p>Explanations:</p> <p>A. Incorrect - Continuous fire watch required. Plausible because this would be correct for Main Computer room halon inop.</p> <p>B. Incorrect - the conditions also require a continuous fire watch. Plausible because this is a requirement but not complete.</p> <p>C. Correct - Per TRM 3.7.k, 8 of 9 initial discharge cylinders and ALL 5 extended discharge cylinders must meet 95% full. With 1 extended discharge cylinder low, Halon to the AEER is INOP and requires a continuous fire watch with backup fire suppression equipment for unprotected areas.</p> <p>D. Incorrect - Continuous fire watch required. Plausible because this would be correct for CO2 subsystem inop.</p> <p>REQUIRED REFERENCE: None</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

65

ID: 27968

Points: 1.00

Unit 2 was operating at near rated power when a transient occurred, resulting in the following sequence of events at 0900:

- Annunciator 902-7 G-3 TURB BYP VLV OPEN, alarmed.
- RPV water level is +20 inches and trending down at 3 inches a minute.
- Main Steam line pressure is 915 psig and trending down at 30 psig a minute.

What is the earliest time an automatic scram would have occurred?

- A. 0902
- B. 0904
- C. 0906
- D. 0908

Answer: B

Answer Explanation

With the current pressure trend, after 3 minutes steam line pressure would be 825 psig with the mode switch in run resulting in a group I isolation and a scram.

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

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:	27968
User-Defined ID:	27968
Cross Reference Number:	
Topic:	65 - 295006.A2.04
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	<p>Objective: DRE239LN001.06</p> <p>References: DAN 902(3)-5 B-13, DAN 902(3)-5 A-9</p> <p>K/A: 295006 A2.04 4.1/--</p> <p>K/A: Ability to determine and/or interpret the following as they apply to SCRAM: Reactor pressure.</p> <p>Safety Function: 1</p> <p>CFR: 41.10</p> <p>PRA: No</p> <p>Level: High</p> <p>Pedigree: New</p> <p>History: None</p> <p>Explanations:</p> <p>A. Incorrect - pressure and level would still above SCRAM setpoints. Plausible because RPV LVL LO alarm is in on the 902-5 panel, CHANNEL A/B RPV LVL LO setpoint is where the scram would occur.</p> <p>B. Correct - With the current pressure trend, after 3 minutes steam line pressure would be 825 psig with the mode switch in run resulting in a group I isolation and a scram.</p> <p>C. Incorrect - Scram would already have occurred on pressure. Plausible because reactor water level would be below 8 inches which is the nominal setpoint for a reactor scram.</p> <p>D. Incorrect - Scram would already have occurred on pressure. Plausible because reactor water level would be below 8 inches which is the nominal setpoint for a reactor scram.</p> <p>REQUIRED REFERENCE: None</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

66

ID: 27970

Points: 1.00

Unit 2 is operating at full power when a transient occurs causing 125 VDC Bus 2B-1 to de-energize.

What is the effect of the loss of 125 VDC Bus 2B-1?

- A. RPS Channel A half scram is in
- B. Bus 23-1 has lost control power
- C. Isolation Condenser has automatically initiated
- D. HPCI Initiation logic is powered from reserve power

Answer: D

Answer Explanation

The HPCI Isolation logic is powered from HPCI 125 VDC which gets its normal power from 125 VDC Bus 2B-1 and reserve power from 125 VDC Bus 2A-1. With the loss of the normal power supply HPCI Initiation logic will swap over and get its power from its reserve power supply.

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

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:	27970
User-Defined ID:	27970
Cross Reference Number:	
Topic:	66 - 206000.K2.03
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	<p>Objective: 206LN001.12b Reference: DOA 6900-T1, DOA 6900-02, DOP 6800-05 K/A: 206000.K2.03 2.8 / 2.9 K/A: HPCI: Knowledge of electrical power supplies to the following: Initiation logic: BWR-2,3,4 CFR: 41.7 PRA: Yes Level: Memory Safety Function: 2 & 4 Pedigree: New History: N/A</p> <p>Explanation:</p> <p>A. Incorrect - Loss of 125 VDC 2B-1 will cause a RPS Channel B half scram. Plausible because if 125 VDC 2A-1 was lost then the half scram would come in on RPS Channel A.</p> <p>B. Incorrect - Bus 23-1 control power is powered from 125 VDC 2A-1. Plausible because 125 VDC 2B-1 is the reserve control power for Bus 23-1.</p> <p>C. Incorrect - Both the initiation and isolation logic for the Isolation Condenser are powered from 125 VDC and they are de-energize to actuate. Plausible because while both would try and work, the isolation logic takes precedence and isolates the Isolation Condenser</p> <p>D. Correct - The HPCI Isolation logic is powered from HPCI 125 VDC which gets its normal power from 125 VDC Bus 2B-1 and reserve power from 125 VDC Bus 2A-1. With the loss of the normal power supply HPCI Initiation logic will swap over and get its power from its reserve power supply.</p> <p>REQUIRED REFERENCE: None</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

67

ID: 27971

Points: 1.00

Unit 3 is in STARTUP with a Control Rod selected for movement.

- APRM 3 failed downscale.
- APRM 5 failed downscale.

RBM ____ (1) ____ is downscale

In order to withdraw rods, the MINIMUM action the operator is required to bypass APRM(s) ____ (2) ____.

- A. (1) 7
(2) 3 **ONLY**
- B. (1) 7
(2) 3 **AND** 5
- C. (1) 8
(2) 3 **ONLY**
- D. (1) 8
(2) 3 **AND**>> 5

Answer: A

Answer Explanation

APRM 3 must be bypassed to clear the RBM INOP condition, APRM 5 does not need to be bypassed to move rods. APRM downscale does not cause a rod block with Mode switch not in RUN.

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

Question 67 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:	27971
User-Defined ID:	27736
Cross Reference Number:	
Topic:	67 - 215002.A1.01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	<p>Objective: 215LN002.06 References: DAN 902(3)-5 C-6, DOP 0700-11, DOA 0700-03, DAN 902(3)-5 C-3 K/A: 215002.A1.01 2.7/2.8 K/A: Ability to predict and/or changes in parameters associated with operating the ROD BLOCK MONITOR SYSTEM controls including: Trip reference. CFR: 41.5 Safety Function: 7 Level: High Pedigree: Bank History: 15-1 NRC</p> <p>Explanation:</p> <p>A. Correct - APRM 3 must be bypassed to clear the RBM INOP condition, APRM 5 does not need to be bypassed to move rods. APRM downscale does not cause a rod block with Mode switch not in RUN.</p> <p>B. Incorrect - RBM 7 reference APRM is downscale. Bypassing APRM 3 will force APRM 2 into the reference position for RBM 7. APRM 5 downscale will not initiate an additional rod block due to Mode switch in Startup. Plausible because APRM 3 needs to be bypassed and APRM 5 would need to be bypassed if mode switch was in RUN..</p> <p>C. Incorrect - Plausible because RBM 8 alternate reference APRM has been lost. No issues with proper operation of RBM 8 will result from conditions in the stem. This requires the candidate to display knowledge of APRM reference channel relationships with RBMs. The second part is correct.</p> <p>D. Incorrect - Plausible because RBM 8 has lost the alternate reference APRM given stem conditions. Normal reference APRM (4) is not affected. APRMs 3 and 5 would need to be bypass if the Mode switch was in run.</p> <p>REQUIRED REFERENCES: None</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

68

ID: 27972

Points: 1.00

Unit 2 was operating in MODE 3, with the following set of conditions:

- SDC in operation, with the 2B pump and heat exchanger in service.
- RBCCW in operation, with the 2B AND 2/3 pumps and heat exchangers in service.

The following annunciators are then received:

- Time 05:15:00; 902-4 A-23, SDC HX/FUEL POOL WTR TEMP HI.
- Time 05:45:00; 902-4 B-23, SDC PP TRIP.

Which of the following actions could have caused the **SDC Pump** to trip?

- A. Loss of U2 250 VDC.
- B. Bus 23-1 undervoltage.
- C. Trip of the 2B RBCCW pump.
- D. The 2B SDC pump discharge AOV drifting closed.

Answer: C

Answer Explanation

This occurs because after the RBCCW pump trips, the SDC temperatures start to rise (as indicated by both annunciators). When the SDC temperature reaches 339°F at the suction of the pumps, this causes the SDC pump to trip.

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

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:	4.00
System ID:	27972
User-Defined ID:	24149
Cross Reference Number:	
Topic:	68 - 295021.G.2.4.31
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	<p>Objective: DRE205LN001.06 Reference: DAN 902-4 B-23 K/A: 295021.G.2.4.31 4.2/-- K/A: Knowledge of annunciator alarms, or response procedures. Loss of Shutdown cooling. Safety Function: 4 PRA: No Level: High Pedigree: Bank History: 2008 NRC, 14-1 NRC</p> <p>Explanation:</p> <p>A. Incorrect - Loss of U2 250 VDC does not cause parameters to change within the SDC system. Plausible because U2 250 VDC is the power supply to a number of SDC valves.</p> <p>B. Incorrect - Bus 23-1 undervoltage is incorrect as the pumps listed as operating in the initial conditions are powered from Bus 24-1. Plausible because common misconception of power supplies since both the SDC and RBCCW systems have 3 pumps - with opposite power supplies from each other.</p> <p>C. Correct - This occurs because after the RBCCW pump trips, the SDC temperatures start to rise (as indicated by both annunciators). When the SDC temperature reaches 339°F at the suction of the pumps, this causes the SDC pump to trip.</p> <p>D. Incorrect - Closing of the 2B SDC pump discharge valve is incorrect as it would cause SDC pump suction pressure to increase. Plausible because this is the opposite of the SDC pump trip on LOW suction pressure.</p> <p>Justification of HIGH order: The candidate must determine system response to transients and compare to knowledge of SDC pump trip setpoints.</p> <p>REQUIRED REFERENCES: None</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

69

ID: 27973

Points: 1.00

Unit 3 is operating at 80% power. DOS 2300-10, HIGH PRESSURE COOLANT INJECTION SYSTEM IST COMPREHENSIVE/PRESERVICE PUMP TEST, is in progress.

- SBTG is in operation
- Torus Cooling is running IAW DOP 1500-02, TORUS WATER COOLING MODE OF LOW PRESSURE COOLANT INJECTION SYSTEM, with 3A & 3C LPCI pumps and 3A & 3D CCSW pump operating

After approximately 10 minutes of running Torus Bulk Temperature is 93°F and rising at 1°F per minute

In 13 minutes what action(s) is(are) required?

- A. Secure HPCI
- B. Enter DEOP 200-1 and continue the surveillance
- C. Increase Torus Cooling flow and continue the surveillance
- D. Slow the HPCI turbine to minimum speed until Torus Temperature is <95°F

Answer: A

Answer Explanation

IAW TS 3.6.2.1 when Suppression pool average temperature >105°F AND thermal power is >1% RTP AND performing testing that adds heat to the suppression pool then the required action is to suspend all testing that adds heat to the suppression pool IMMEDIATELY, therefore securing HPCI is the proper action.

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

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:	3.00
System ID:	27973
User-Defined ID:	27973
Cross Reference Number:	
Topic:	69 - 295013.K3.02
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	<p>Objective: 223LN001.8 Reference: DOS 2300-10, DEOP 200-1, TS 3.6.2.1 K/A: 295013.K3.02 3.6 / 3.8 K/A: Knowledge of the reasons for the following responses as they apply to HIGH SUPPRESSION POOL TEMPERATURE: Limiting heat additions CFR: 41.5 / 45.6 PRA: Yes Level: High Safety Function: 5 Pedigree: New History: N/A</p> <p>Explanation:</p> <p>A. Correct - IAW TS 3.6.2.1 when Suppression pool average temperature >105°F AND thermal power is >1% RTP AND performing testing that adds heat to the suppression pool then the required action is to suspend all testing that adds heat to the suppression pool IMMEDIATELY, therefore securing HPCI is the proper action.</p> <p>B. Incorrect - DEOP 200-1 will be entered when Torus Bulk Temperature >95°F but the surveillance will be stopped. Plausible because there have been multiple times when actually performing the surveillance and having to enter DEOP 200-1 while continuing with the surveillance.</p> <p>C. Incorrect - Max Torus Cooling will be placed in service when Torus Bulk Temperature >95°F as directed by DEOP 200-1 and DOP 1500-02 but the surveillance will be stopped. Plausible because if the math is done incorrectly and it is determined to be <105°F then Max Torus Cooling would be established and the surveillance would continue.</p> <p>D. Incorrect - HPCI is required to be secured if Torus Bulk Temperature >105°F. Plausible because there are multiple HPCI surveillances that require HPCI to run at minimum speed for extended periods of time.</p> <p>REQUIRED REFERENCES: None</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

70

ID: 27974

Points: 1.00

U3 is operating at rated power.

- Annunciator 923-5 G-6, U3 LPCI/CS PP AREA TEMP HI, alarms
- No ECCS pumps are running
- Local temperatures have been taken at all 3 levels

What is the **LOWEST** temperature that would require Engineering to evaluate the elevated temperature impact on instrumentation?

- A. Average temperature is above 100°F
- B. Any one area temperature above 104°F
- C. Average temperature is above 104°F
- D. Average temperature is above 150°F

Answer: C

Answer Explanation

Per DAN 923-5 G-6, IF the average temperature is above 104°F, with ALL Core Spray AND LPCI pumps secured, THEN notify Engineering to evaluate the elevated temperature condition.

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

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:	4.00
System ID:	27974
User-Defined ID:	27974
Cross Reference Number:	
Topic:	70 - 295032.K2.05
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	<p>Objective: 203LN001.08 References: DAN 923-5 G-6 K/A: 295032 K2.05 3.2/-- K/A: Knowledge of the interrelations between HIGH SECONDARY CONTAINMENT AREA TEMPERATURE and the following: Temperature sensitive instrumentation. Safety Function: 7 CFR: 41.7 PRA: Yes Level: High Pedigree: New History: N/A</p> <p>Explanations:</p> <p>A. Incorrect - Temperature above 100°F does not require an Engineering eval. Plausible because 100°F is the alarm setpoint for corner room temp Hi.</p> <p>B. Incorrect - Engineering evaluation is not required until the average of all three levels exceeds 104°F. Plausible because 104°F is the temperature at which the eval would be required if averaged over all 3 areas.</p> <p>C. Correct - Per DAN 923-5 G-6, <u>IF</u> the average temperature is above 104°F, with <u>ALL</u> Core Spray <u>AND</u> LPCI pumps secured, <u>THEN</u> notify Engineering to evaluate the elevated temperature condition.</p> <p>D. Incorrect - At 150°F an evaluation would be required but not the lowest temperature. Plausible because 150°F equates to Max Normal corner room temperature and requires entry into DEOP 300-1.</p> <p>REQUIRED REFERENCE: None</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

71

ID: 27982

Points: 1.00

Unit 2 was operating at near rated power, when the feed breaker to Bus 29 de-energized on overcurrent.

Which PCIS **alarm(s)** would occur due to this condition?

- A. GROUP 3 **ONLY**
- B. GROUP 1 AND GROUP 2 **ONLY**
- C. GROUP 2 AND GROUP 3 **ONLY**
- D. GROUP 1, GROUP 2 **AND** GROUP 3

Answer: D

Answer Explanation

A loss of Bus 29 will cause MCC 29-1 to de-energize. When MCC 29-1 de-energizes, this will cause a loss of power to ATS panel 2202-73B. A loss of power to the ATS panel will cause a loss of RPV water level instruments, which will cause a half isolation and drive up annunciators for the Group 1, 2, and 3 PCIS isolations.

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

Question 71 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:	27982
User-Defined ID:	13609
Cross Reference Number:	
Topic:	71 - 223002.K1.20
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	<p>Objective: DRE223LN005.12 Reference: DANs 902-4 H-20, 902-5 B-13 , and 902-5 B-15; DOP 6800-05 K/A: 223002.K1.20 2.8 / 3.0 K/A Knowledge of the physical connections and/or cause-effect relationships between PCIS/Nuclear Steam Supply Shutoff and the following: A.C. distribution. CFR: 41.2 to 41.9 Safety Function: 5 PRA: No Level: High Pedigree: Bank History: 2007 NRC</p> <p>Explanations:</p> <p>A. Incorrect - Plausible because Group 3 alarm would be illuminated, but so would Gr 1 and 2. B. Incorrect - Plausible because Group 1 and Group 2 would be illuminated, but so would Gr 3. C. Incorrect - Plausible because Group 2 and Group 3 would be illuminated, but so would Gr 1. D. Correct - A loss of Bus 29 will cause MCC 29-1 to de-energize. When MCC 29-1 de-energizes, this will cause a loss of power to ATS panel 2202-73B. A loss of power to the ATS panel will cause a loss of RPV water level instruments, which will cause a half isolation and drive up annunciators for the Group 1, 2, and 3 PCIS isolations.</p> <p>REQUIRED REFERENCES: None</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

72

ID: 27983

Points: 1.00

U2 is operating at rated power with the following conditions.

- Alarm 902-56 A-5, DRYWELL/TORUS H₂ CONC HI, is actuated
- Drywell pressure is 2.1 psig and slowly rising.

To obtain hydrogen and oxygen readings from the CAMs, they have been/must be ____ (1) ____ initiated and the **EARLIEST** readings can be taken is ____ (2) ____ minutes, to allow the system to stabilize.

- A. (1) manually
(2) 10 - 15
- B. (1) manually
(2) 20 - 30
- C. (1) automatically
(2) 10 - 15
- D. (1) automatically
(2) 20 - 30

Answer: C

Answer Explanation

902-56 A-5 alarm setpoint is an entry condition into DEOP 200-1 primary containment control. This alarm does NOT initiate the CAMS however Per DEOP 200-1 the CAMS will start automatically on Core Spray Logic with 2 psig signal. WHEN readings on CH A POST-LOCA H2 RECORDER, 2(3)-2406A, AND/OR CH B POST LOCA H2 RECORDER, 2(3)-2406B, stabilize (after approximately 10 to 15 minutes of operation), THEN hydrogen and oxygen readings can be taken.

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

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:	4.00
System ID:	27983
User-Defined ID:	27983
Topic:	72 - 500000.G.2.1.32
Comments:	<p>Objective: DRE223LN006-11 Reference: DEOP 200-1, DOP 2400-01, DAN 902(3)-56 A-5. K/A: 500000 G2.1.32 3.8/-- K/A: Ability to explain and apply system limits and precautions. High CTMT Hydrogen Conc. Safety Function: 5 CFR: 41.10 PRA: No Level: High Pedigree: New History: None</p> <p>Explanation:</p> <p>A. Incorrect - Drywell pressure of greater than 2 psig will auto start the drywell CAM system. <u>WHEN</u> readings on CH A POST-LOCA H2 RECORDER, 2(3)-2406A, <u>AND/OR</u> CH B POST LOCA H2 RECORDER, 2(3)-2406B, stabilize (after approximately 10 to 15 minutes of operation), <u>THEN</u> hydrogen and oxygen readings can be taken. Plausible because the 902-56 A-5 DRYWELL/TORUS H2 CONC HI alarm does not auto start the system. Part 2 is correct.</p> <p>B. Incorrect - Drywell pressure of greater than 2 psig will auto start the drywell CAM system. <u>WHEN</u> readings on CH A POST-LOCA H2 RECORDER, 2(3)-2406A, <u>AND/OR</u> CH B POST LOCA H2 RECORDER, 2(3)-2406B, stabilize (after approximately 10 to 15 minutes of operation), <u>THEN</u> hydrogen and oxygen readings can be taken. Plausible because the 902-56 A-5 DRYWELL/TORUS H2 CONC HI alarm does not auto start the system. Part 2 is plausible because of 20 to 30 minute actions. Readings would be available but it would not be the earliest.</p> <p>C. Correct - 902-56 A-5 alarm setpoint is an entry condition into DEOP 200-1 primary containment control. This alarm does NOT initiate the CAMS however Per DEOP 200-1 the CAMS will start automatically on Core Spray Logic with 2 psig signal. <u>WHEN</u> readings on CH A POST-LOCA H2 RECORDER, 2(3)-2406A, <u>AND/OR</u> CH B POST LOCA H2 RECORDER, 2(3)-2406B, stabilize (after approximately 10 to 15 minutes of operation), <u>THEN</u> hydrogen and oxygen readings can be taken.</p> <p>D. Incorrect - The 902-56 A-5 setpoint does NOT initiate the CAMS automatically. Plausible because part 1 is correct. Part 2 is plausible because of 20 to 30 minute actions. Readings would be available but it would not be the earliest.</p> <p>REQUIRED REFERENCES: None</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

73

ID: 27984

Points: 1.00

Unit 2 is in Startup.

- Shorting links are INSTALLED
- IRM 18 High Voltage power supply fails low

A ____ (1) ____ will occur, required operator action is to take actions of ____ (2) ____ to bypass IRM 18.

- A. (1) Rod Block ONLY
(2) DAN 902-5 C-5, IRM DOWNSCALE,
- B. (1) Rod Block AND a 'B' Half Scram
(2) DAN 902-5 C-5, IRM DOWNSCALE,
- C. (1) Rod Block ONLY
(2) DAN 902-5 C-15, Channel B IRM Hi HI/INOP,
- D. (1) Rod Block AND a 'B' Half Scram
(2) DAN 902-5 C-15, Channel B IRM Hi HI/INOP,

Answer: D

Answer Explanation

With the High Voltage < 88 to 90 VDC (per DAN 902(3)-5 C-15) and the shorting links installed then a Rod Block and 'B' Half Scram will occur. The actions necessary to be taken are in DAN 902(3)-5 C-15.

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

Question 73 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:	27984
User-Defined ID:	27984
Cross Reference Number:	
Topic:	73 - 215003.A2.01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	<p>Objective: 215LN003.12b Reference: DAN 902(3)-5 C-5 & C-15 K/A: 215003.A2.01 2.8 / 3.2 K/A: Ability to (a) predict the impacts of the following on the INTERMEDIATE RANGE MONITOR (IRM) SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Power supply degraded CFR: 41.5 / 45.6 Level: Memory Safety Function: 7 Pedigree: New History: N/A</p> <p>Explanation:</p> <p>A. Incorrect - A Rod Block will occur but will not the only plant response to the conditions, a 'B' Half Scram will also occur. 902-5 C-15 will come in and this will be the correct procedure to reference. Plausible because a Rod Block will happen but so will a 'B' half scram and 902-5 C-5 will come in as well but will not direct resetting the half scram.</p> <p>B. Incorrect - First part of the answer is correct. 902-5 C-15 will come in and this will be the correct procedure to reference. Plausible because 902-5 C-5 will come in as well but will not direct resetting the half scram.</p> <p>C. Incorrect - A Rod Block will occur but will not the only plant response to the conditions, a 'B' Half Scram will also occur. Plausible because a Rod Block will happen but so will a 'B' half scram. Second part is correct</p> <p>D. Correct - With the High Voltage < 88 to 90 VDC (per DAN 902(3)-5 C-15) and the shorting links installed then a Rod Block and 'B' Half Scram will occur. The actions necessary to be taken are in DAN 902(3)-5 C-15.</p> <p>REQUIRED REFERENCES: None</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

74

ID: 27988

Points: 1.00

Unit 2 is operating at 100% Reactor Power when grid disturbances occur causing the following:

- Trip of Unit 2 Main Power Transformer
- Trip of Unit 2 Main Generator

____(1)____ causes the Unit 2 Reactor to trip to prevent fuel damage and ____ (2)____.

- A. (1) Reactor Overpressure
(2) actuation of ERVs
- B. (1) Reactor Overpressure
(2) actuation of reactor safety valves
- C. (1) Turbine Control Valve Fast Closure
(2) actuation of ERVs
- D. (1) Turbine Control Valve Fast Closure
(2) actuation of reactor safety valves

Answer: D

Answer Explanation

Loss of generator load would cause the turbine generator to speed up. The turbine speed governor would react by closing turbine control valves. The reduction in steam flow would cause RPV pressure to rise. To prevent fuel damage and lifting of reactor safety valves, the reactor will scram by the fast closure of the turbine control valves.

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

Question 74 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:	27988
User-Defined ID:	27988
Cross Reference Number:	
Topic:	74 - 700000.K3.01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	<p>Objective: DRE245LN002.12 Reference: DOA 6000-01, DAN 902(3)-8 A-12, UFSAR 15.2.2 K/A: 700000.K3.01 3.9 / 4.2 K/A: Knowledge of the reasons for the following responses as they apply to GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES: Reactor and turbine trip criteria CFR: 41.4, 41.5, 41.7, 41.10 / 45.8 Safety Function: 6 PRA: Yes Level: Memory Pedigree: New History: N/A</p> <p>Explanations:</p> <p>A. Incorrect - Pressure switches on the control valve solenoids initiate the scram before the reactor overpressure trip is reached. ERVs opening is not a concern since they discharge to the Torus. Plausible because reactor overpressure would cause the scram if the fast closure of turbine control valves did not. Also ERVs will actuate before the safety valves do.</p> <p>B. Incorrect - Pressure switches on the control valve solenoids initiate the scram before the reactor overpressure trip is reached. Plausible because reactor overpressure would cause the scram if the fast closure of turbine control valves did not. Also the second part is correct.</p> <p>C. Incorrect - ERVs opening is not a concern since they discharge to the Torus. Plausible because ERVs will open before the safety valves do. Also the first part is correct.</p> <p>D. Correct - Loss of generator load would cause the turbine generator to speed up. The turbine speed governor would react by closing turbine control valves. The reduction in steam flow would cause RPV pressure to rise. To prevent fuel damage and lifting of reactor safety valves, the reactor will scram by the fast closure of the turbine control valves.</p> <p>REQUIRED REFERENCE: None</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

75

ID: 28000

Points: 1.00

Reactor power ascension is in progress on Unit 3. The unit is holding load at 475 MWe for APRM gain adjustment.

A failure causes the DEHC Pressure Controller to sense LOWERING reactor pressure and slowly CLOSE the Turbine Control Valves.

With NO operator actions, Reactor Power will _____.

- A. rise until pressure exceeds 1005 psig requiring an emergency load drop
- B. rise until pressure exceeds 1005 psig requiring a Manual Scram
- C. remain steady due to control valves readjusting to maintain pressure steady.
- D. lower requiring the Turbine Control valves to be jacked open.

Answer: A

Answer Explanation

Reactor pressure will rise causing reactor power to rise. When pressure exceeds 1005 psig an emergency load drop is required per DOA 5650-05

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

Question 75 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:	28000
User-Defined ID:	28000
Cross Reference Number:	
Topic:	75 - 295007.K1.03
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	<p>Objective: 241LN001A.12b(5) Reference: DOA 5650-05 K/A: 295007k1.03 3.8/3.9 K/A: Knowledge of the operational implications of the following concepts as they apply to HIGH REACTOR PRESSURE: Pressure effects on reactor power. CFR: 41.8 to 41.10 Level: High Safety Function: 3 Pedigree: New History: None</p> <p>Explanations: A. Correct - Reactor pressure will rise causing reactor power to rise. When pressure exceeds 1005 psig an emergency load drop is required per DOA 5650-05. B. Incorrect - Plausible because pressure will continue to rise. A scram is not required at 1005 psig. C. Incorrect - As the control valve closes down reactor pressure will rise and power will also increase. Plausible because the other control valves will adjust to maintain pressure steady if the signal failed Bad Quality.. D. Incorrect - As the control valve closes down pressure and power will rise. Plausible because must understand that control valves closing will increase reactor pressure and power. Bypass valves can be jacked open, not control valves.</p> <p>Required References: None</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

76

ID: 24186

Points: 1.00

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

- A. Chemistry Manager
- B. Shift Operations Supervisor
- C. Radiation Protection Manager
- D. Senior Manager Ops Support and Services

Answer: A

Answer Explanation

With the monitor not in use DOP 2000-111 Attachment A requires signatures from both the Shift Manager and the Chemistry Manager prior to discharge.

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

Question 76 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:	24186
User-Defined ID:	24186
Cross Reference Number:	
Topic:	76 - Generic 2.3.06
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	<p><QQ 9796(1411)>Objective: DRE268LN001.14 Reference: DOP 2000-111 K/A: Generic.2.3.06 -- / 3.8 K/A: Ability to approve release permits. CFR: 43.4 Safety Function: N/A Level: Memory Pedigree: Bank History: 14-1 NRC exam, 18-1 NRC exam</p> <p>SRO Criteria: 10CFR55.43(b)(4) - Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions.</p> <p>Explanations:</p> <p>A. Correct - With the monitor not in use DOP 2000-111 Attachment A requires signatures from both the Shift Manager and the Chemistry Manager prior to discharge.</p> <p>B. Incorrect - With the monitor not in use the Shift Operations supervisor can verify calculations, but Shift Manager and Chemistry Manager are required signatures.</p> <p>C. Incorrect - Radiation Protection Manager signature is not required in spite of the radiation monitor not being used. This plausible because RP Manager signature is required during off normal events leading to plant releases. RP Manager must also be notified in DOP 2000-110 with monitor operable</p> <p>D. Incorrect - This is plausible due to the SMOS is a Licensee senior SRO</p> <p>REQUIRED REFERENCES: None</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

77

ID: 27505

Points: 1.00

Unit 2 was operating at 100% power when a transient occurred causing reactor water level dropped to 0 inches.

- DGP 02-03 Hard Card actions are being taken.
- ARI has been actuated at the 902-5 panel.
- Recirc Pumps are running at minimum speed.
- 2 APRM downscale lights are illuminated.
- Torus Temperature is 109 F.

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

- A. (1) will
(2) this will provide a prompt reduction in power.
- B. (1) will
(2) they did not trip on ARI initiation.
- C. (1) will not
(2) this would reduce boron mixing efficiency.
- D. (1) will not
(2) reactor power is less than 6%.

Answer: A

Answer Explanation

The purpose of tripping the RR pumps is to provide a prompt reduction in power. This is only required if reactor power is above 6% (APRM downscale). Power is above 6% in this example, because 3 APRMs are NOT downscale.

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

Question 77 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:	27505
User-Defined ID:	27505
Cross Reference Number:	
Topic:	77 - Generic 2.4.09
Comments:	<p>Objective: 29502LK046.C Reference: EPG B-6-52-54, DGP 02-03, DEOP 0400-05, DEOP Technical Bases K/A: Generic 2.4.09 -- / 4.2 K/A: Knowledge of low power/shutdown implications in accident (e.g., loss of coolant accident or loss of residual heat removal) mitigation strategies. CFR: 43.5 Safety Function: 1 Pedigree: Bank Level: High History: 18-1 NRC</p> <p>SRO Criteria: 10CFR55.43(b)(5) - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.</p> <p>Explanation:</p> <p>A. Correct - The purpose of tripping the RR pumps is to provide a prompt reduction in power. This is only required if reactor power is above 6% (APRM downscale), which is indicated by having only 3 APRMs downscale.</p> <p>B. Incorrect - Although ARI will trip the RR pumps, this is only if ARI is automatically initiated via logic. If manually initiated from the control room, the RR pumps do not trip. Plausibility: The event is an ATWS, and the DGP 02-03 actions include actuating ARI. Plausible because the candidate must remember that use of manual ARI does not trip recirc.</p> <p>C. Incorrect - While tripping the RR pumps will reduce the boron mixing efficacy (if boron is injected), the reduction in power is of precedence. Plausibility: The reduction in boron mixing efficacy is a valid concern, but it is overridden by the need to reduce power rapidly in an ATWS above 6% power.</p> <p>D. Incorrect - Reactor power is not below 6% with only 3 APRM downscale lit. (since all APRMs are not downscale), and with recirc pumps at minimum speed, the recirc pumps are required to be tripped per DGP 02-03. Plausibility: The candidate may mistakenly believe that power is below 6% power, since three APRMs are downscale.</p> <p>REQUIRED REFERENCES: None</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

78

ID: 27502

Points: 1.00

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

- Torus Bottom Pressure is 20 psig and going up at 0.5 psig/min.
- Torus water level is 15 feet and steady.
- Reactor water level is -140 inches and lowering at 2 in/min.
- Drywell temperature is 200 degrees F and going up 10 degrees/min.
- Reactor Pressure is 800 psig and lowering 30 psig/min

Which safety function is the first priority and why?

- A. Containment Integrity
Blowdown due to Drywell Temperature
- B. Containment Integrity
Blowdown due to Pressure Suppression Limit (PSP)
- C. Reactor Water Inventory Control
Entering DEOP 400-3 for Steam Cooling
- D. Reactor Water Inventory Control
Blowdown due to Rx water level below TAF

Answer: D

Answer Explanation

With water level dropping at 2 in/min, in addition to reactor pressure dropping at 30 psig/min, will cause the level correction factor to not be in affect after 10 minutes due to reactor pressure below 500 psig. This will require a Reactor Blowdown due to Rx water level below TAF.

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

Question 78 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:	27502
User-Defined ID:	27502
Cross Reference Number:	
Topic:	78 - Generic 2.4.22
Comments:	<p>Objective: 29800LP041 Reference: DEOP 100, DEOP 200-1, DEOP 400-3 K/A: Generic 2.4.22 -- / 4.4 K/A: Knowledge of the bases for prioritizing safety functions during abnormal/emergency operations. CFR: 43.5 Safety Function: 2 and 5 Level: High Pedigree: Bank History: 18-1 NRC</p> <p>SRO Criteria: 10CFR55.43(b)(5) - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.</p> <p>Explanation:</p> <p>A. Incorrect - The limit for Drywell temperature is 338 degrees. This would be met in 14 minutes at the current rate. This is plausible because if it was not recognized that the correction factor was no longer in effect, then reactor water level would not have reached TAF for 15 minutes. In addition in the previous revision of the DEOPs the limit was 281 degrees which would have been met first.</p> <p>B. Incorrect - The limit for PSP is 26 psig for Torus level at 15 feet. This would be met in 12 minutes at the current rate. This is plausible because if it was not recognized that the correction factor was no longer in effect, then reactor water level would not have reached TAF for 15 minutes.</p> <p>C. Incorrect - The limit for reactor water level entry into STEAM COOLING with no injection source is 162 inches. This would be at the 11 minute mark. In addition there is no mention of which injection sources are available. This is plausible because if it was not recognized that the correction factor was no longer in effect, then reactor water level would not have reached TAF for 15 minutes.</p> <p>D. Correct - With water level dropping at 2 in/min, in addition to reactor pressure dropping at 30 psig/min, will cause the level correction factor to not be in affect after 10 minutes due to reactor pressure below 500 psig. This will require a Reactor Blowdown due to Rx water level below TAF.</p> <p>REQUIRED REFERENCES: None</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

79

ID: 24233

Points: 1.00

Given the following:

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

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

- A. 30 days
- B. 60 days
- C. 90 days
- D. 120 days

Answer: C

Answer Explanation

IAW LS-AA-104-1000 4.2.2, a temporary alternation is exempt from performing a 50.59 review as long as it is in direct support of maintenance. Temporary alternations that support maintenance including jumpering terminals, lifting leads, etc. An approved procedure must exist (CC-AA-112) for tracking and controlling the 90 day period. Otherwise at a minimum a 50.59 screening must be performed. CC-AA-112 step 2.5 declares: although the "90 days from the current date" is conservative, the addition of this date will result in a "flag" in the schedule that reflects the earliest possible date that the temporary configuration change has to be removed, UNLESS the maintenance activity that it supports is completed sooner. Note, that MR90s are required to be removed when the maintenance activity is completed or within 90 days of the temporary change installation, whichever comes first. 50.59 uses terms like "90 days at power" and not just 90 days. While "90 days at power" are the absolute limits, the ability to control the limits of temporary change installation is based on just 90 days. The reason for using 90 days and not taking credit for the "at power" is that the NEI guidance has defined "at power" as beginning when the reactor goes critical. The difficulty in controlling duration limits based on a "reactor criticality date" that may come earlier or later than scheduled introduces a variable that is difficult to use in establishing an easily identified removal date. Risk considerations associated with the temporary change are addressed as part of the maintenance activity being performed. Procedure WC-AA-101 governs the risk evaluation for the on-line maintenance activity. MR90 work orders should have the term 'MR90' in their title to further identify that it is a Maintenance Rule (a)(4) temporary change.

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

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:	24233
User-Defined ID:	24233
Cross Reference Number:	
Topic:	79 - Generic 2.2.05
Num Field 1:	
Num Field 2:	
Text Field:	

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

<p>Comments:</p>	<p>Objective: 29900LK148 Reference: LS-AA-104-1000 K/A: Generic 2.2.05 2.2 / 3.2 K/A: Knowledge of the process for making design or operating changes to the facility. CFR: 43.3 PRA: No Level: Memory Pedigree: Bank History: 14-1 NRC, 18-1 NRC, 19-1 CERT</p> <p>SRO CRITERIA: 10CFR55.43(b)(3) - Facility licensee procedures required to obtain authority for design and operating changes in the facility.</p> <p>Comments:</p> <p>A. Incorrect - Less than the 90 day requirement Plausible because this is time allowed for a 50.59 screening.</p> <p>B. Incorrect - Less than the 90 day requirement This is plausible if the candidate believes 90 days will require a 50.59 screening, and subtracts 30 days from the allowed time in order to do the 50.59 screening.</p> <p>C. Correct - IAW LS-AA-104-1000 4.2.2, a temporary alternation is exempt from performing a 50.59 review as long as it is in direct support of maintenance. Temporary alternations that support maintenance including jumpering terminals, lifting leads, etc. An approved procedure must exist (CC-AA-112) for tracking and controlling the 90 day period. Otherwise at a minimum a 50.59 screening must be performed. CC-AA-112 step 2.5 declares: although the "90 days from the current date" is conservative, the addition of this date will result in a "flag" in the schedule that reflects the earliest possible date that the temporary configuration change has to be removed, UNLESS the maintenance activity that it supports is completed sooner. Note, that MR90s are required to be removed when the maintenance activity is completed or within 90 days of the temporary change installation, whichever comes first. 50.59 uses terms like "90 days at power" and not just 90 days. While "90 days at power" are the absolute limits, the ability to control the limits of temporary change installation is based on just 90 days. The reason for using 90 days and not taking credit for the "at power" is that the NEI guidance has defined "at power" as beginning when the reactor goes critical. The difficulty in controlling duration limits based on a "reactor criticality date" that may come earlier or later than scheduled introduces a variable that is difficult to use in establishing an easily identified removal date. Risk considerations associated with the temporary change are addressed as part of the maintenance activity being performed. Procedure WC-AA-101 governs the risk evaluation for the for the on-line maintenance activity. MR90 work orders should have the term 'MR90' in their title to further identify that it is a Maintenance Rule (a)(4) temporary change.</p> <p>D. Incorrect - Greater than the 90 day requirement. This is plausible if the candidate incorrectly applies SR 3.0.2 which allows which allows for a 25% extension of completion times for Technical Specification surveillance Requirements.</p>
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EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

	Required References: None.
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80

ID: 27897

Points: 1.00

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

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

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

- A. (1) Refuel Floor **ONLY**;
(2) Unusual Event
- B. (1) Refuel Floor **ONLY**;
(2) Alert
- C. (1) Drywell **AND** Refuel Floor;
(2) Unusual Event
- D. (1) Drywell **AND** Refuel Floor;
(2) Alert

Answer: D

Answer Explanation

Local ARM reading above the alert value, and confirmation of high value (SBGT autostarts at 100 mr/hr on the refuel floor) Alert (RA2) criteria has been reached. With a dropped fuel bundle and local rad levels confirmed to be above alarm setpoint, the SRO is required to direct evacuation of BOTH the Drywell and refuel floors.

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

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:	4.00
System ID:	27897
User-Defined ID:	27897
Cross Reference Number:	
Topic:	80 - 295023.G.2.4.41
Comments:	<p>Objective: 23400LK001 Reference: DFP 0850-03, DOA 0010-08, DAN 923-5 A-1, EP-AA-1004 Addendum 3 K/A: 295023.G2.4.41 --/4.6 K/A: Knowledge of the emergency action level thresholds and classifications: Refueling Accidents CFR: 43.5 Safety Function: 8 Level: High Pedigree: Bank History: NRC 18-1</p> <p>SRO Criteria: 10CFR55.43(b)(5) - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. EP classification is a SRO only task at Dresden Station</p> <p>Explanation:</p> <p>A. Incorrect - A dropped fuel bundle is a symptom of drywell evacuation DOA 0010-08. The first part is plausible because the accident and indications are located on the refuel floor. The second part is plausible because unplanned ARM readings rising is one condition that factors into an Unusual Event classification.</p> <p>B. Incorrect - A dropped fuel bundle is a symptom of drywell evacuation DOA 0010-08. The first part is plausible because the accident and indications are located on the refuel floor. The second part is correct.</p> <p>C. Incorrect - This is plausible due to the first part being correct. The second part is plausible because unplanned ARM readings rising is one condition that factors into an Unusual Event classification but an Unusual event requires loss of water level AND unplanned ARM readings rising.</p> <p>D. Correct - Local ARM reading above the alert value, and confirmation of high value (SBGT autostarts at 100 mr/hr on the refuel floor) Alert (RA2) criteria has been reached. With a dropped fuel bundle and local rad levels confirmed to be above alarm setpoint, the SRO is required to direct evacuation of BOTH the Drywell and refuel floors.</p> <p>REQUIRED REFERENCES: EP-AA-1004 Addendum 3</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

81

ID: 27903

Points: 1.00

Unit 3 is at full power.

When the following occurs:

- 903-4 C-19, LPCI/CS EAST SUMP LVL HI is received in the control room.
- An EO reports a large unisolable leak on the Core Spray suction piping and there is 10 inches of water on the floor in the East Corner Room.

20 minutes later:

- 903-4 D-19, LPCI/CS WEST SUMP LVL HI is received in the control room.
- An EO reports there is 9 inches of water on the floor in the West Corner Room.

Based on these conditions, at a minimum, the SRO is required to direct ____ (1) ____ and the basis for this is ____ (2) ____.

- A. (1) a reactor scram per DGP 02-03
(2) a direct threat exists relative to secondary containment integrity, to equipment located in the secondary containment and to continued safe operation of the plant.
- B. (1) a reactor shutdown per DGP 02-01
(2) a direct threat exists relative to secondary containment integrity, to equipment located in the secondary containment and to continued safe operation of the plant.
- C. (1) a reactor scram per DGP 02-03
(2) to reduce to decay heat levels to the energy that the RPV may be discharging to the secondary containment.
- D. (1) a reactor shutdown per DGP 02-01
(2) to reduce to decay heat levels to the energy that the RPV may be discharging to the secondary containment.

Answer: B

Answer Explanation

Per BWROG EPG/SAGs, Appendix B, When an area water level exceeds its maximum safe operating water level (Table SC-1) in more than one area, shut down the reactor. When the accumulation of water can no longer be confined to one secondary containment area, a direct threat exists relative to secondary containment integrity, to equipment located in the secondary containment, and to continued safe operation of the plant. Irrespective of the source of water, it is prudent to commence an orderly reactor shutdown.

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

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:	4.00
System ID:	27903
User-Defined ID:	27903
Topic:	81 - 295036.A2.03
Comments:	<p>Objective: 295L049 Reference: BWROG EPG/SAGs, EOP-DEOP Technical Bases, DEOP 0300-01 K/A: 295036.A2.03 3.4/3.8 K/A: Ability to determine and/or interpret the following as they apply to Secondary Containment High Sump / Area Water Level: Cause of the high water level Safety Function: 5 CFR: 41.10, 43.5, 45.13 PRA: Yes Level: High Pedigree: Bank History: ILT 11-1 NRC, ILT 14-1 NRC</p> <p>SRO Criteria: 10CFR55.43(b)(5) - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.</p> <p>Explanation:</p> <p>A. Incorrect - Plausible because the reactor would be scrammed if a primary system was discharging and unisolable.</p> <p>B. Correct - Per BWROG EPG/SAGs, Appendix B, When an area water level exceeds its maximum safe operating water level (Table SC-1) in more than one area, shut down the reactor. When the accumulation of water can no longer be confined to one secondary containment area, a direct threat exists relative to secondary containment integrity, to equipment located in the secondary containment, and to continued safe operation of the plant. Irrespective of the source of water, it is prudent to commence an orderly reactor shutdown.</p> <p>C. Incorrect - Plausible because the reactor would be scrammed if a primary system was discharging and unisolable</p> <p>D. Incorrect - Plausible because this is the basis for scramming the reactor with an unisolable leak of a primary system.</p> <p>SRO Justification: Examinee must identify that a leak from core spray suction is not a leak from a primary system and a reduction in RPV pressure will not affect the leak rate. Per DEOP 300-1 a decision must be made by the SRO whether a Scram or Unit Shutdown must be performed. This is the responsibility of the Unit Supervisor.</p> <p>REQUIRED REFERENCES: None</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

82

ID: 13808

Points: 1.00

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

- A. obtain special permission from the NRC Regional office for reactivation.
- B. at a minimum, have received a passing grade on a special reactivation exam.
- C. complete a minimum of 60 hours of shift functions under the direction of an operator or senior operator and in the position to which the individual will be assigned.
- D. participate in a complete plant tour as part of a minimum of 40 hours of shift functions under the direction of an operator or senior operator and in the position to which the individual will be assigned.

Answer: D

Answer Explanation

Before resumption of functions authorized by a license issued under this part, an authorized representative of the facility licensee shall certify the following:
That the licensee has completed a minimum of 40 hours of shift functions under the direction of an operator or senior operator as appropriate and in the position to which the individual will be assigned. The 40 hours must have included a complete tour of the plant.

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

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:	2.00
System ID:	13808
User-Defined ID:	13808
Cross Reference Number:	
Topic:	82 - Generic.2.1.04
Num Field 1:	0.00
Num Field 2:	0.00
Text Field:	
Comments:	<p>Objective: DRE299LK187 Reference: OP-AA-105-102, 10 CFR 55.53 K/A: Generic.2.1.04 --- / 3.8 K/A: Knowledge of individual licensed operator responsibilities related to shift staffing, such as medical requirements, "no-solo" operation, maintenance of active license status, 10CFR55, etc. CFR: 43.2 Safety Function: N/A Level: Memory Pedigree: Bank History: 2007 NRC, 2013 Cert, 18-1 NRC</p> <p>SRO Criteria: 10CFR55.43(b)(2) - Facility operating limitations in the technical specifications and their bases. This is an SRO function as the Shift Operations Superintendent (or designee) will document and approve each individual apply for reactivation status.</p> <p>Explanation:</p> <p>A. Incorrect - As long as the individual has no NRC restrictions special permission is not required. This is plausible as this would be correct if the reason for deactivating the license was NRC restrictions, i.e. INPO assignment, college, military, or foreign exchange program.</p> <p>B. Incorrect - There is no special reactivation exam as long as the individual is current in the requalification program. This is plausible due to the requirement of current in a training program (not more than 1 cycle behind).</p> <p>C. Incorrect - The number of hours under direction or observation is 40 hours not 60. This plausible because 60 hours are required per quarter if the individual is on 12 hour shifts.</p> <p>D. Correct - Before resumption of functions authorized by a license issued under this part, an authorized representative of the facility licensee shall certify the following: That the licensee has completed a minimum of 40 hours of shift functions under the direction of an operator or senior operator as appropriate and in the position to which the individual will be assigned. The 40 hours must have included a complete tour of the plant.</p> <p>REQUIRED REFERENCES: None</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

83

ID: 27904

Points: 1.00

Unit 3 was operating at 100% power when an earthquake occurred.

20 MINUTES LATER:

- Torus Level is 15 feet and steady.
- LPCI Pumps are **NOT** available.
- Torus bottom pressure is 20 psig and rising 0.5 psig per minute.
- RPV level is -139 inches and lowering 1 inch per minute.
- RPV pressure is 850 psig and steady.
- Torus temperature is 150°F and rising at 1°F per minute.
- Drywell temperature is 331°F and rising 1°F per minute.

If **NO** other operator actions are taken, when will a blowdown **FIRST** be required?

- A. 4 minutes
- B. 7 minutes
- C. 10 minutes
- D. 12 minutes

Answer: B

Answer Explanation

RPV blowdown required due to being above the drywell temperature limit of 338°F, with no ability to restore and hold.

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

Question 83 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:	27904
User-Defined ID:	27904
Cross Reference Number:	
Topic:	83 - 295028.G.2.4.47
Num Field 1:	
Num Field 2:	
Text Field:	

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

Comments:	<p><QQ 284525(1411)Objective: 29502LK015 Reference: DEOP 200-1, DEOP 0100 K/A: 295028.G2.4.47 -- / 4.2 K/A: Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material. CFR: 43.5 Safety Function:5 PRA: No Level: High Pedigree: New History: N/A</p> <p>SRO Only Criteria: 10CFR55.43(b)(5) - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. Additionally, the curves for PSP and for HCTL are SRO ONLY learning objectives.</p> <p>Explanation: A. Incorrect - The correct time is 7 minutes, since this is when the ADS design temperature will be exceeded. Plausible because this would be true if pressure were less than 500 psig. B. Correct - In seven minutes, we will reach 338°F, which will require a blowdown, since LPCI is not available and therefore DW sprays cannot be used. C. Incorrect - The first time that a blowdown would be required is in seven minutes. Plausible because a blowdown would be required at this value, due to exceeding HCTL. Also, if DW sprays were available, and could lower DW temperature, the blowdown at 338°F would not be required. D. Incorrect - The first time that a blowdown would be required is in seven minutes. Plausible because this is when PSP will be exceeded.</p> <p>REQUIRED REFERENCES: DEOP 0100 and DEOP 200-1 with entry conditions blanked out.</p>
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EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

84

ID: 27027

Points: 1.00

Given the following conditions:

- Unit 2 is operating at rated power.
- Unit 3 is in Day 3 of a refueling outage.
- U3 SAC is OOS for overhaul.
- 2B IAC is OOS for desiccant replacement.
- 2A and 3C IACs are running supplying Unit 2.
- 3A and 3B IACs are running supplying Unit 3.

A transient occurs resulting in the following:

- 10:05 Unit 2 IA header pressure begins lowering
- 10:20 902-6 H-10, FW REG VLVS BACKUP AIR ACTIVE alarm is received

What actions will the SRO direct? (Assume Shift Manager concurrence has been obtained, if necessary)

- A. Start all available **Service Air** compressors, per DOA 4600-01 Service Air System Failure
- B. Crosstie Unit 2 and Unit 1 **Service Air** systems, per DOA 4600-01 Service Air System Failure
- C. Crosstie Unit 2 and Unit 3 **Instrument Air** systems, per DOA 4700-01 Instrument Air System Failure
- D. Close the 2/3-4701-501A, U2 SERV AIR TO INST AIR X-TIE MANUAL ISOL VLV, per DOA 4700-01 Instrument Air System Failure

Answer: C

Answer Explanation

Given the time from the beginning of the leak to the alarm, the candidate must identify IA header pressure is dropping at approximately 1 psig per minute. Direction to cross tie U2 and U3 IA headers per DOP 4700-03 is appropriate

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

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:	27027
User-Defined ID:	27027
Cross Reference Number:	
Topic:	84 - 295019.A2.01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	<p>Objective: DRE278LN001.08 Reference: DOA 4700-01, DOP 4700-03, DAN 902(3)-6 H-10, DAN 923-1 F-4 K/A: 295019.A2.01 3.5 / 3.6 K/A: Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR: Instrument air system pressure CFR: 43.5 Safety Function: 8 Level: High Pedigree: Bank History: 15-1 NRC, 20-1 Cert</p> <p>SRO Only Criteria: 10CFR55.43(b)(5) - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.</p> <p>Explanation:</p> <p>A. Incorrect - There are multiple IA compressors (5 total). Only 2 SA compressors (1 on each unit). Given in the stem Unit 3 SAC is OOS (Service Air is no longer unit specific when in a normal lineup. Normal lineup is 2/3 SA crosstie open, 1 SAC running and the other in PTL), U2 SAC is running and the SA-IA crosstie is already open based on alarms given. Unit 2 SAC is unable to keep up with SA loads and IA loads/leakage.</p> <p>B. Incorrect - Unit 1 and Unit 2 Service Air systems can be cross-connected, however Unit 2 supplies Unit 1 SA. Unit 2 IA is capable of supplying U1 IA. This would result in an additional load on the U2 and U3 SA systems.</p> <p>C. Correct - Given the time from the beginning of the leak to the alarm, the candidate must identify IA header pressure is dropping at approximately 1 psig per minute. Direction to cross tie U2 and U3 IA headers per DOP 4700-03 is appropriate</p> <p>D. Incorrect - This action would be appropriate if Unit 2 had experienced a loss of offsite power.</p> <p>REQUIRED REFERENCES: None</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

85

ID: 27992

Points: 1.00

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

Aux Power fast transfer failed to occur.

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

What is the SRO required to direct **NEXT**?

- A. Control RPV water level between +8" and +48" using condensate and feed system
- B. Start 1 SBLC pump and 2A CRD pump to control RPV level between +8" and + 48"
- C. Start both SBLC pumps and 2A CRD pump to control RPV level between -170" and +48"
- D. Exit DEOP 100, RPV CONTROL and enter DEOP 400-2 EMERGENCY DEPRESSURIZATION

Answer: C

Answer Explanation

Due to RPV level above TAF, attempts to restore RPV level shall be made. The only high pressure injection sources available are SBLC and 1 CRD pump. If the decision to use SBLC is made, the band for RPV level is above TAF

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

Question 85 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:	27992
User-Defined ID:	27992
Cross Reference Number:	
Topic:	85 - 295031.A2.01
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	<p>Objective: 29501LK024 Reference: DEOP 100, DOP 1200-02 K/A: 295031.A2.01 4.6/4.6 K/A: Ability to determine and/or interpret the following as they apply to REACTOR LOW WATER LEVEL: Reactor Low Water Level CFR: 43.5 Safety Function: 5 Pedigree: Bank Level: High History: N/A</p> <p>SRO Only Criteria: 10CFR55.43(b)(5) - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.</p> <p>Explanation:</p> <p>A. Incorrect - This is prohibited due to the initial loss of all condensate pumps. Restart of condensate system is not allowed. This is plausible because it would be the correct answer if aux power had transferred as designed.</p> <p>B. Incorrect - This would be the correct number of SBLC pumps to start for an ATWS, however when using SBLC for level, both pumps are required. This is plausible because the level band would be correct if SBLC was not used. (i.e. only CRD)</p> <p>C. Correct - Due to RPV level above TAF, attempts to restore RPV level shall be made. The only high pressure injection sources available are SBLC and 1 CRD pump. If the decision to use SBLC is made, the band for RPV level is above TAF</p> <p>D. Incorrect - This is plausible because this would be correct if RPV pressure was less than 500 psig.</p> <p>REQUIRED REFERENCES: None</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

86

ID: 27964

Points: 1.00

Unit 3 is operating at near rated power, with HPCI out of service, when a Group I occurred:

- All Reactor Feed pumps have tripped
- All appropriate DEOPs have been entered
- RPV pressure is 1020 psig and steady being controlled by the Isolation Condenser
- RPV water level is +6 inches and steady

Then the following occurs:

- These 902-3 annunciators are in alarm:
 - ☐ B-3, ISOL CONDR VENT RAD HI
 - ☐ C-4, ISOL CONDR TEMP HI
- Isolation Condenser shell side water level is rising

The SRO will direct what actions to be taken?

- A. Maintain operation of Isolation Condenser to control pressure IAW DEOP 100
- B. Manually restore RPV water level to +8 to +48 IAW DOA 0600-01, TRANSIENT LEVEL CONTROL, IAW DEOP 100.
- C. Drain the Isolation Condenser to normal band IAW DOP 1300-01, STANDBY OPERATION OF ISOLATION CONDENSER
- D. Isolate the Isolation Condenser IAW DOA 1300-01, ISOLATION CONDENSER TUBE LEAK, and control pressure IAW DEOP 100

Answer: D

Answer Explanation

With the tube leak in the Isolation Condenser the proper action is to isolate the Isolation Condenser and then utilize ERVs to maintain RPV pressure.

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

Question 86 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:	27964
User-Defined ID:	27964
Cross Reference Number:	
Topic:	86 - 207000.G.2.4.8
Num Field 2:	
Text Field:	
Comments:	<p>Objective: 207LN001.8 Reference: DOA 1300-01, DEOP 100 K/A: 207000.G.2.4.8 3.8 / 4.5 K/A: ISOLATION (EMERGENCY) CONDENSER: Knowledge of how abnormal operating procedures are used in conjunction with EOPs. CFR: 41.10 / 43.5 / 45.13 Level: High PRA: Yes Safety Function: 4 Pedigree: New History: N/A</p> <p>SRO Only Criteria: 10CFR55.43(b)(5) - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.</p> <p>Explanation:</p> <p>A. Incorrect - With the Isolation Condenser having a tube leak per the conditions in the stem the proper action would be to isolate the Isolation Condenser and utilize ERVs for RPV pressure control. Plausible because there are cases where pieces of equipment would be continued to run even with a leak to protect the plant or personnel.</p> <p>B. Incorrect - With no RFPs available per the stem there are no actions in DOA 0600-01 to be able to restore RPV water level. Plausible because with the tube leak in the Isolation Condenser RPV water level will be lowering and will need to be restored to 8 to 48 with Alternate Water Injection systems.</p> <p>C. Incorrect - The Isolation Condenser will need to be isolated to prevent RPV level from going down and radiation levels from going up in the plant. Plausible because the Isolation Condenser has a normal range and per DOP 1300-01 the level should be maintained within the normal band.</p> <p>D. Correct - With the tube leak in the Isolation Condenser the proper action is to isolate the Isolation Condenser and then utilize ERVs to maintain RPV pressure.</p> <p>REQUIRED REFERENCES: None</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

87

ID: 27909

Points: 1.00

<QQ 284525(1410)><<<QQ 27907(1410)><<<QQ 284525(1410)><<Unit 2 was operating at 100% power with the Isolation Condenser OOS when a transient occurred.

10 MINUTES LATER:

- All Rods are in.
- Drywell Pressure is 1.4 psig and steady.
- RPV Pressure is 460 psig and steady.
- RPV level is -145 inches and lowering at 2 inches per minute.
- The 2A and 2B Core Spray Pumps are the **ONLY** available injection sources.
- The NSO reports that **ALL** relief valve indicating lights on the 902-3 panel are extinguished.

The Unit Supervisor must direct the NSO to ____ (1) ____

- A. enter DEOP 0400-02, EMERGENCY DEPRESSURIZATION **ONLY**.
- B. enter DEOP 0500-03, ALTERNATE WATER INJECTION SYSTEMS, and **START** lining up alternate injection systems.
- C. enter DEOP 0500-03, ALTERNATE WATER INJECTION SYSTEMS, and **MAXIMIZE** injection with alternate injection systems.
- D. enter DEOP 0400-02, EMERGENCY DEPRESSURIZATION, **AND** then enter DEOP 0500-07, ALTERNATE EMERGENCY DEPRESSURIZATION SYSTEMS.

Answer: D

Answer Explanation

Based on the stem of the question, all relief valves have lost control power and cannot be opened from the main control room. Additionally, RPV level is at TAF, with two Detail F subsystems available to inject. Therefore, entry into DEOP 0400-02 is required, and the SRO must also enter DEOP 500-7 because less than 5 ADS valves are available and the difference between drywell and RPV pressure is greater than 67 psid.

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

Question 87 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:	27909
User-Defined ID:	27909
Cross Reference Number:	
Topic:	87 - 295004.A2.02

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

<p>Comments:</p>	<p>Objective: 26302LK002 Reference: DOA 6900-02, DEOP 100 K/A: 295004.A2.02 3.5/3.9 K/A: Ability to determine and/or interpret the following as they apply to Partial or Total Loss of DC power: Extent of partial or complete loss of D.C. power CFR: 43.5 PRA: Yes Level: High Pedigree: New History: N/A</p> <p>SRO Only Criteria: 10CFR55.43(b)(5) - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. Requires specific procedure content knowledge</p> <p>Explanation:</p> <p>A. Incorrect - Given the conditions in the stem, entry into DEOP 0400-02 is required. However, with the loss of all relief valve indicating lights on the 902-3 panel, this means that all relief valves have no control power and will not work in the relief mode. Therefore, entry into DEOP 0500-07 is also required. Plausible because the answer is partially correct. The candidates must recognize that the loss of indicating lights means a loss of relief valve capability, and also recognize that the conditions given in the stem will require alternate emergency depressurization systems to be used. If conditions were different (for example, if the difference between RPV pressure and containment pressure were less than 66 psid, then this answer would be correct.</p> <p>B. Incorrect - This would be required if less than two subsystems in DEOP 100 Detail F were available, and if RPV level was above TAF. Plausible because RPV level would still be above TAF if RPV pressure were above 500 psig, and because this would be correct if only one Core Spray pump were available.</p> <p>C. Incorrect - This would be required if no subsystems in DEOP 100 Detail F were available. Plausible because if other injection sources, for example CRD or SBLC, were the available sources then this would be required.</p> <p>D. Correct - Based on the stem of the question, all relief valves have lost control power and cannot be opened from the main control room. Additionally, RPV level is at TAF, with two Detail F subsystems available to inject. Therefore, entry into DEOP 0400-02 is required, and the SRO must also enter DEOP 500-7 because less than 5 ADS valves are available and the difference between drywell and RPV pressure is greater than 67 psid.</p> <p>REQUIRED REFERENCES: DEOP 0100, 0400-02 with entry conditions blanked out.</p>
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EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

88

ID: 27979

Points: 1.00

Unit 3 was at rated conditions when a LOCA occurred. Torus water level rose to 20 feet immediately after the LOCA occurred then over 3 to 5 seconds it returned to a level of 15 feet.

The Unit Supervisor will direct _____.

- A. performing an RPV Blowdown due to torus water level above 18.5 feet
- B. dispatching an EO to locally determine torus water level due to erratic indication
- C. restoring torus water level between 14 feet 6.5 inches to 14 feet 10.5 inches within 2 hours
- D. holding torus water level below 18.5 ft due to "pool swell" occurring as described in the design bases

Answer: D

Answer Explanation

"Pool Swell" occurs when a LOCA causes the 'jet' of air/water into the Torus causing torus water level to rise and then lower over 3-5 seconds. The UFSAR explains this phenomenon and the design basis accounts for its occurrence. Due to torus water level returning below 18.5 feet, the direction would be to maintain torus water level below 18.5 feet per DEOP 0200-01.

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

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:	27979
User-Defined ID:	27979
Cross Reference Number:	
Topic:	88 - 295029.A2.03
Num Field 1:	0.00
Num Field 2:	0.00
Text Field:	
Comments:	<p>Objective: 223L-S1-07 Reference: UFSAR section 6.2.1.3.4.1.1, DEOP 0200-01, T.S. 3.6.2.2 K/A: 295029.A2.03 3.4 / 3.5 K/A: Ability to determine and/or interpret the following as they apply to HIGH SUPPRESSION POOL WATER LEVEL: Drywell/containment water level CFR: 41.10 / 43.5 / 45.13 Safety Function: Level: High Pedigree: New History: N/A</p> <p>SRO Only Criteria: 10CFR55.43(b)(5) - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.</p> <p>Explanations: A - Incorrect: Blowdown is not required since torus water level returned to less than 18.5 feet. Plausible because if torus water level remained above 18.5 feet, a blowdown would be required. B - Incorrect: Locally verifying torus water level is not required in this scenario. Plausible because locally verifying torus water level is an action that is performed in other scenarios. C - Incorrect: Tech Spec torus water level not required in this scenario. Plausible because if a LOCA had not occurred, it would be required. D - Correct: "Pool Swell" occurs when a LOCA causes the 'jet' of air/water into the Torus causing torus water level to rise and then lower over 3-5 seconds. The UFSAR explains this phenomenon and the design basis accounts for its occurrence. Due to torus water level returning below 18.5 feet, the direction would be to maintain torus water level below 18.5 feet per DEOP 0200-01.</p> <p>Required Reference: DEOP 0200-01 with entry criteria blanked out, T.S. 3.6.2.2</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

89

ID: 27935

Points: 1.00

With a unit at near rated power, the following conditions exist:

- At 1430 on 11/3/21 it was discovered that a Technical Specification surveillance with a 24 hour frequency was last performed satisfactorily at 1230 hours on 11/1/21.

The Limiting Condition for Operation (LCO) required actions state:

Condition A:

Restore the equipment to OPERABLE status within 4 hours

Condition B:

Be in MODE 3 in 12 hours

AND

Be in MODE 4 in 36 hours

If the surveillance was then performed with unsatisfactory results at 1230 on 11/4/21, which of the following will be the LATEST time that meets the requirement to be in MODE 4?

- A. By 1430 hours on 11/5/21
- B. By 1830 hours on 11/5/21
- C. By 0430 hours on 11/6/21
- D. By 1830 hours on 11/6/21

Answer: C

Answer Explanation

Given the last time that the surveillance was performed (1230 on 11/1/21) and not performing the surveillance again, the time that the unit would need to be in MODE 4 by is 0430 on 11/6/21 (24 hours from discovery of not performing the surveillance plus 4 hours to perform the surveillance, plus 36 hours). The latest time, out of the distractors, that would meet this time is 0430 11/6/21.

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

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:	27935
User-Defined ID:	27935
Cross Reference Number:	
Topic:	89 - Generic 2.2.23
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	<p>Objective: DRE299LN001.04 Reference: Tech Specs, TS SR 3.0.3 K/A: Generic.2.2.23 3.1 / 4.6 K/A: Ability to track Technical Specification limiting conditions for operations. CFR: 43.2/45.13 Safety Function: N/A Level: High Pedigree: New History: N/A</p> <p>SRO Only Criteria: 10CFR55.43(b)(1) - Conditions and limitations in the facility license.</p> <p>Explanations:</p> <p>A. Incorrect - Plausible because 11/5/21 @ 1430 would be assumed if the 24 hours from discovery was NOT counted.</p> <p>B. Incorrect - Plausible because 11/5/21 @ 1830 would be assumed if the 24 hours from discovery was not counted but the 4 hours for completion of the surveillance was added.</p> <p>C. Correct - Given the last time that the surveillance was performed (1230 on 11/1/21) and not performing the surveillance again, the time that the unit would need to be in MODE 4 by is 0430 on 11/6/21 (24 hours from discovery of not performing the surveillance plus 4 hours to perform the surveillance, plus 36 hours). The latest time, out of the distractors, that would meet this time is 0430 11/6/21.</p> <p>D. Incorrect - Plausible because 11/6/21 @ 1830 would be assumed if the 12 and 36 hours were added.</p> <p>REQUIRED REFERENCES: None</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

90

ID: 27951

Points: 1.00

DEOP 100, RPV CONTROL
DEOP 400-2, EMERGENCY DEPRESSURIZATION
DEOP 400-3, STEAM COOLING
DEOP 500-3, ALTERNATE WATER INJECTION SYSTEMS

Unit 3 was operating at near rated power when a scram occurred. The following reports were received:

- Time 00:00:00 - RPV pressure is 450 psig.
- Time 00:00:10 - Drywell pressure is 2.3 psig.
- Time 00:00:30 - RPV water level is -150 inches.
- Time 00:01:00 - The only source of injection is U2 CRD cross-tied to Unit 3.
- Time 00:02:00 - RPV water level is -160 inches.

What action(s) is/are the SRO required to take?

- A. Exit ALL DEOPs and enter the SAMGs.
- B. Exit DEOP 100 ONLY then enter DEOP 400-3.
- C. Exit DEOP 100 AND DEOP 500-3 then enter DEOP 400-3.
- D. Enter DEOP 400-2 while continuing in DEOP 100 and DEOP 500-3

Answer: D

Answer Explanation

Given the set of conditions, with only single detail F injection source and RPV level approaching -164 inches, the DEOPs direct the SRO to enter DEOP 400-2, while continuing (not exiting) on in DEOP 100.

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

Question 90 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:	27951
User-Defined ID:	27951
Cross Reference Number:	
Topic:	90 - 295009.G.2.4.21
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	<p>Objective: 29501LK029 Reference: DEOP 100, DEOP 400-2 K/A: 295009.G.2.4.21 --/4.6 K/A: Knowledge of the parameters and logic used to assess the status of safety functions, such as reactivity control, core cooling and heat removal, reactor coolant system integrity, containment conditions, radioactivity release control, etc. CFR: 43.5 Safety Function: N/A PRA: No Level: High Pedigree: Bank History: 2009 NRC</p> <p>SRO Only Criteria: 10CFR55.43(b)(5) - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.</p> <p>Explanations:</p> <p>A. Incorrect - SAMGs are not entered unless there is an injection source and a blowdown has been performed AND level cannot still be restored. Plausible because level is still dropping.</p> <p>B. Incorrect - An injection source is available and criteria for Steam Cooling not met. Plausible because this would be correct for no injection source available and level below TAF.</p> <p>C. Incorrect - criteria for Steam Cooling not met. Plausible because this would be correct for less than 2 injection sources available and level below TAF. Blowdown must be performed first to lower pressure for LP injection systems.</p> <p>D. Correct - Given the set of conditions, with only single detail F injection source and RPV level approaching -164 inches, the DEOPs direct the SRO to enter DEOP 400-2, while continuing (not exiting) on in DEOP 100.</p> <p>REQUIRED REFERENCES: DEOP Charts, with the entry conditions blanked out</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

91

ID: 27952

Points: 1.00

Unit 2 was shutdown, with the following set of conditions:

- NO Recirc pumps are running.
- RPV water temperature is 320°F.
- 2A SDC pump is running and aligned to the RPV.
- 2B SDC pump is running and aligned to the RPV.
- 2C SDC pump is running and aligned to the FPC system.

If Recirc Loop water temperature trends up to 360°F, what action(s) is/are the Unit Supervisor required to direct?

- A. Control RPV water temperature/pressure per DOA 1000-1, RESIDUAL HEAT REMOVAL ALTERNATIVES **ONLY**.
- B. Align 2C SDC pump to the RPV to control RPV water temperature/pressure per DOP 1000-3, SHUTDOWN COOLING MODE OF OPERATION **ONLY**.
- C. Align 2C SDC pump to the RPV to control RPV water temperature/pressure per DOP 1000-3, SHUTDOWN COOLING MODE OF OPERATION **AND** monitor fuel storage pool water level once per hour per DOA 1900-1, LOSS OF FUEL POOL COOLING.
- D. Monitor fuel storage pool water level once per hour per DOA 1900-1, LOSS OF FUEL POOL COOLING **AND** control RPV water temperature/pressure per DOA 1000-1, RESIDUAL HEAT REMOVAL ALTERNATIVES.

Answer: A

Answer Explanation

The SDC Pump Suction temperature (Recirc Loop Temp) exceeds the setpoint of 339°F for SDC pump trip, for the pumps lined up to the RPV, which causes 'A' and 'B' to trip. The 'C' pump will NOT trip, as it still has suction from the Fuel Pool Cooling system. Entry into DOA 1000-01 is the only required entry.

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

Question 91 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:	27952
User-Defined ID:	27952
Cross Reference Number:	
Topic:	91 - 205000.G.2.1.23
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	<p><QQ 13066(1411)>Objective: DRE205LN001.06 Reference: DOA 1000-01, DOA 1900-01, DAN 902-4 B-23, DAN 902-4 H-4 K/A: 205000.G2.1.23 4.3/4.4 K/A: Ability to perform specific system and integrated plant procedures during all modes of plant operation. Shutdown Cooling Mode. Safety Function: 4 PRA: No Level: High Pedigree: Bank History: 2012 NRC</p> <p>SRO Only Criteria: 10CFR55.43(b)(5) - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.</p> <p>Explanations:</p> <p>A. Correct - The SDC Pump Suction temperature (Recirc Loop Temp) exceeds the setpoint of 339°F for SDC pump trip, for the pumps lined up to the RPV, which causes 'A' and 'B' to trip. The 'C' pump will NOT trip, as it still has suction from the Fuel Pool Cooling system. Entry into DOA 1000-01 is the only required entry.</p> <p>B. Incorrect - Aligning 2C SDC pump to the RPV would not start due to the temperature that caused 2A and 2B to trip. Plausible because 2C did not trip originally when temperature spiked. Must understand that the temperature would prohibit operation to the RPV.</p> <p>C. Incorrect - Aligning 2C SDC pump to the RPV would not start due to the temperature that caused 2A and 2B to trip. Level in fuel pool storage should not be affected by the current configuration of SDC pumps. Plausible because 2C did not trip originally when temperature spiked. Must understand that the temperature would prohibit operation to the RPV. Part 2 would be plausible if part 1 was performed.</p> <p>D. Incorrect - Level in fuel pool storage should not be affected by the current configuration of SDC pumps. Plausible because part 2 is correct and part 1 would be a concern if all pumps tripped.</p> <p>REQUIRED REFERENCES: None</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

92

ID: 28001

Points: 1.00

An NSO is performing DOS 0300-01, CONTROL ROD EXERCISE. Rod C-5 is being moved from position 46 to 48.

____(1)____ on the full core display would be an indication of overtravel.

The SRO would enter Tech Spec 3.1.3, Control Rod Operability, and direct the NSO to ____ (2) ____ per DOA 0300-05, INOPERABLE OR FAILED CONTROL ROD DRIVES.

- A. (1) Blank indication
(2) attempt to re-couple the rod up to 4 times
- B. (1) Red double dashes
(2) attempt to re-couple the rod up to 4 times
- C. (1) Blank indication
(2) immediately drive the rod to full in position
- D. (1) Red double dashes
(2) immediately drive the rod to full in position

Answer: A

Answer Explanation

The full core display for the rod would go blank to indicate an uncoupled rod. With power greater than 10% T.S. allows attempts to recouple the rod (up to 4 times).

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

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:	28001
User-Defined ID:	28001
Cross Reference Number:	
Topic:	92 - 214000.A2.03
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	<p>Objective: DRE201LN002.08 References: DAN 902(3) -5 E-3, DOA 0300-05, T.S. 3.1.3 K/A: 214000.A2.03 3.6/3.9 K/A: Ability to predict the impacts of the following on the ROD POSITION INFORMATION SYSTEM: and based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Overtravel/in-out. CFR: 41.5/45.6 Safety Function: 7 Level: High Pedigree: New History: None</p> <p>Explanations: A. Correct - The full core display for the rod would go blank to indicate an uncoupled rod. With power greater than 10% T.S. allows attempts to recouple the rod (up to 4 times). B. Incorrect - Plausible because the indication is correct and part 2 would be correct for power less than 10%. C. Incorrect - Plausible because during rod exercising 48 would go away and go to Red double dashes before settling back to 48. Part 2 is correct. D. Incorrect - Plausible because during rod exercising 48 would go away and go to Red double dashes before settling back to 48. Part 2 would be correct for power less than 10%.</p> <p>Required References: None</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

93

ID: 27961

Points: 1.00

Unit 2 is in Refuel Mode. Fuel moves are being performed.

Which of the following are SRO responsibilities?

1. From the Fuel Bridge, supervise movement of fuel assembly and blade guides per the approved Move Sheet.
2. Verify the correct fuel bundle will be moved by observing the Grapple over the correct fuel bundle in the cavity per the Move Sheet.
3. Verify the fuel bundle is placed in the correct Spent Fuel Pool location by observing rack coordinates in the Spent Fuel Pool.
4. Initial the "FROM" location and "TO" location boxes of the Move Sheet.

- A. 1, 2 ONLY
- B. 2, 3, 4 ONLY
- C. 1, 3, 4 ONLY
- D. 1, 2, 3, 4

Answer: D

Answer Explanation

Per DFP 0800-01, MASTER REFUELING PROCEDURE, all 4 of the duties listed in the stem are responsibilities of the SRO.

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

Question 93 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:	27961
User-Defined ID:	27961
Cross Reference Number:	
Topic:	93 - Generic 2.1.35
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	<p>Objective: 234LN001.12 References: DFP 0800-01 K/A: Generic 2.1.35 --/3.9 K/A: Knowledge of the Fuel Handling responsibilities of SROs. Safety Function: 8 CFR: 43.7 Level: Memory Pedigree: New History: N/A</p> <p>SRO Only Criteria: 10CFR55.43(b)(7) - Fuel handling facilities and procedures.</p> <p>Explanations:</p> <p>A. Incorrect - 1 and 2 are correct but so are 3 and 4. Plausible because both 1 and 2 are responsibilities of the fuel handling SRO.</p> <p>B. Incorrect - 1 is also a correct answer. Plausible because all 3 are responsibilities of the fuel handling SRO.</p> <p>C. Incorrect - 2 is also a correct answer. Plausible because all 3 are responsibilities of the fuel handling SRO.</p> <p>D. Correct - Per DFP 0800-01, MASTER REFUELING PROCEDURE, all 4 of the duties listed in the stem are responsibilities of the SRO.</p> <p>REQUIRED REFERENCES: None</p>

None

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

94

ID: 27966

Points: 1.00

A LOCA has occurred on Unit 2, concurrently with a LOOP, with the following conditions:

- 2A Core Spray pump failed to start
- Reactor Pressure is 400 psig and LOWERING
- HPCI and SBLC are the ONLY high pressure systems available AND injecting into the Reactor
- RPV water level is -162 inches and LOWERING
- BOTH loops of Torus Sprays are in operation
- BOTH loops of Torus Cooling are in operation
- BOTH loops of Drywell sprays cannot be opened remotely, operators are en route for local operation
- Drywell Pressure is 18 psig and RISING slowly
- Torus Bottom Pressure is 23 psig and RISING slowly
- Torus Level is 14 feet and STABLE

The SRO is required to direct the NSO to ____ (1) ____ Torus Cooling AND Torus Sprays; blowdown is required based upon ____ (2) ____ .

- A. (1) CONTINUE to operate
(2) Torus Bottom Pressure ONLY
- B. (1) CONTINUE to operate
(2) Reactor Water Level AND Torus Bottom Pressure
- C. (1) STOP
(2) Reactor Water Level ONLY
- D. (1) STOP
(2) Reactor Water Level AND Torus Bottom Pressure

Answer: C

Answer Explanation

Since LPCI is needed for core cooling, Torus cooling and sprays need to be secured. With Reactor Pressure at 300 psig, Top of Active Fuel is -143 inches so a blowdown is required on reactor water level.

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

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:	2.00
System ID:	27966
User-Defined ID:	27966
Cross Reference Number:	
Topic:	94 - 230000.G.2.2.44
Comments:	<p>Objective: DRE203LN001.06 Reference: DEOPs 100 and 200-1 K/A: 230000.G.2.2.44 4.2 / 4.4 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. LPCI: Torus/Suppression Spray Mode CFR: 41.5 / 43.5 / 45/12 Safety Function: 5 Level: High Pedigree: Bank History: Quad Cities 2005 NRC exam</p> <p>SRO Only Criteria: 10CFR55.43(b)(5) - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.</p> <p>A - Incorrect: Torus cooling and sprays should be secured since LPCI will be needed for core cooling. At 14 ft in the Torus, you would need to wait for 25 psig Torus Bottom Pressure to Blowdown. Plausible because at 19 psig in the Drywell, you would continue to run Torus cooling and sprays if LPCI was not needed for core cooling. If Reactor Pressure was > 500 psig then Top of Active Fuel would be -170 inches and would not warrant a blowdown.</p> <p>B - Incorrect: Torus cooling and sprays should be secured since LPCI will be needed for core cooling. At 14 ft in the Torus, you would need to wait for 25 psig Torus Bottom Pressure to Blowdown. Plausible because at 19 psig in the Drywell, you would continue to run Torus cooling and sprays if LPCI was not needed for core cooling.</p> <p>C - Correct: Since LPCI is needed for core cooling, Torus cooling and sprays need to be secured. With Reactor Pressure at 400 psig, Top of Active Fuel is -143 inches so a blowdown is required on reactor water level.</p> <p>D - Incorrect: At 14 ft in the Torus, Torus bottom pressure does not exceed the figure "L" on DEOP 200-1. Plausible because at a lower Torus water level, 24 psig Torus Bottom Pressure would warrant a blowdown.</p> <p>REQUIRED REFERENCES: DEOP 0100 and 0200-01 with the entry conditions blanked out</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

95

ID: 27986

Points: 1.00

Unit 2 is operating at 80% power.

- EMD is working on the breaker for 2-1501-22A, U2 LPCI LOOP I COOLANT INJECTION INBD ISOL VLV at MCC 28-7/29-7
- The appropriate LCO is entered at 0600 today

During the work on the breaker the racking linkage broke and the breaker is unable to be racked in. There is no replacement breaker on site and one **CANNOT** be obtained for seven (7) days.

What is the required action that needs to be taken?

- A. Be in MODE 3 within 12 hours **ONLY**
- B. Be in MODE 3 within 13 **AND** be in MODE 4 within 37 hours
- C. Restore one LPCI subsystem to OPERABLE status within 72 hours
- D. Restore low pressure ECCS injection/spray subsystem to OPERABLE status within 7 days

Answer: C

Answer Explanation

If two (2) LPCI subsystems are inoperable (3.5.1 Condition E) then the Required Action is to restore one LPCI subsystem to OPERABLE status within 72 hours. Because it is determined that two LPCI subsystems are inoperable with the conditions then this would be the proper LCO to enter.

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

Question 95 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:	27986
User-Defined ID:	27986
Cross Reference Number:	
Topic:	95 - 203000.G.2.2.36
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	<p>Objective: 203LN001.7b Reference: TS 3.5.1 K/A: 203000 G.2.2.36 K/A: RHR/LPCI: Injection Mode (Plant Specific): Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operations. CFR: 41.10 / 43.2 / 45.13 PRA: Yes Level: High Safety Function: 2 & 4 Pedigree: New History: N/A</p> <p>SRO Only Criteria: 10CFR55.43(b)(2) - Facility operating limitations in the technical specifications and their bases.</p> <p>Explanation:</p> <p>A. Incorrect - The timeclock for the original LCO (3.5.1 Condition B) will run out in 72 hours after it was entered. If the Required Action and associated Completion Time of condition A, B, or C not met then the required actions will be to be in MODE 3 within 12 hours per 3.5.1 Condition D.</p> <p>B. Incorrect - Conditions for entry into T.S. 3.0.3 has not been met. Plausible because two LPCI subsystems are inoperable, if support systems with the conditions then this would be the proper LCO to enter.</p> <p>C. Correct - If two (2) LPCI subsystems are inoperable (3.5.1 Condition E) then the Required Action is to restore one LPCI subsystem to OPERABLE status within 72 hours. Because it is determined that two LPCI subsystems are inoperable with the conditions then this would be the proper LCO to enter.</p> <p>D. Incorrect - The original LCO that would have been entered with the conditions is 3.5.1 Condition B and the Required Action is to restore low pressure ECCS injection/spray subsystem to OPERABLE status within 7 days. Plausible because this is the original LCO timeclock and if the timeclock didn't expire then the LCO would still be valid.</p> <p>REQUIRED REFERENCES: TS 3.5.1 with 1 hour or less blanked out</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

96

ID: 27987

Points: 1.00

Both units were operating at 100% power.
The cribhouse is on in-direct open cycle with heavy rainfall.

Annunciator 902(3)-7 B-15, SCREEN WASH CONTROL PANEL TROUBLE, illuminates.
Moments later annunciator 923-1 C-3 U2 OR U3 SERV WATER PP TRIP, illuminates.
Followed by annunciator 923-1 E-2, U2 or U3 TBCCW TEMP HI illuminates.

1. X-area temperature is 202°F and rising.
2. Alternator air temperature is 46°C and steady.
3. Service Water pressure is 84 psig and dropping.
4. Stator Cooling Water outlet temperature is 75°C and rising.

The Unit Supervisor is required to direct

- A. tripping both Recirc Pumps and entering DGP 02-03, REACTOR SCRAM.
- B. verifying PCIS Group I isolation has occurred and enter DEOP 100.
- C. cross-connecting the fire water and service water systems per DOA 3900-01, LOSS OF COOLING WATER SYSTEM.
- D. verifying runback of Main Generator load set and entering DOA 7400-01, FAILURE OF THE STATOR COOLANT SYSTEM.

Answer: B

Answer Explanation

X-area temperature has exceeded the setpoint for PCIS group 1 actuation and a subsequent scram on MSIV closure has occurred. Entry into DEOP 100 as a result of setpoint setdown will result.

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

Question 96 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:	27987
User-Defined ID:	27987
Cross Reference Number:	
Topic:	96 - 400000.A2.03
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	<p>Objective: DRE276LN001.08 Reference: DOA 3900-01, DAN 902(3)-7 B-15, DAN 923-1 C-3, DAN 923-1 E-2, DAN 902(3)-5 D-4 K/A: 400000 A2.03 --/3.0 K/A: Ability to predict the impacts of the following on the CCWS and based on those predictions, use procedures to correct, control, or mitigate the consequences of the abnormal operation: High/low CCW temperature. CFR: 45.6 Safety Function: 8 Level: High Pedigree: New History: None</p> <p>SRO Only Criteria: 10CFR55.43(b)(5) - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.</p> <p>Explanations:</p> <p>A. Incorrect - Tripping Recirc Pumps would only be correct if all Service Water flow has been lost resulting in a loss of RBCCW. Plausible because pressure is lowering but nothing indicates that a total loss of Service Water occurs.</p> <p>B. Correct - X-area temperature has exceeded the setpoint for PCIS group 1 actuation and a subsequent scram on MSIV closure has occurred. Entry into DEOP 100 as a result of setpoint setdown will result.</p> <p>C. Incorrect - Cross-connecting fire water and service water is not a REQUIRED procedural action. Plausible because it is still a viable option.</p> <p>D. Incorrect - Stator Cooling Water temperature is below the setpoint to verify runback of Main Generator load set. Plausible because if temperature were to get above 80°F then this would be a correct answer.</p> <p>REQUIRED REFERENCES: None.</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

97

ID: 27989

Points: 1.00

Unit 3 was operating at full power when a transient occurred causing the following:

- 3A Recirc Pump tripped
- OPRMs sensed oscillations such that an automatic scram has occurred
- Reactor power is 4% as read on the APRMs
- RPV water level is 0" and lowering slowly
- Appropriate DEOPs are being executed

What action will the Unit Supervisor direct **NEXT**?

- A. Initiate one train of SBLC
- B. Trip running Recirc pump
- C. Take running Recirc to minimum
- D. Terminate and prevent injection to at least -35"

Answer: C

Answer Explanation

With Reactor power <6% taking the running Recirc pump to minimum speed is the proper action.

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

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:	3.00
System ID:	27989
User-Defined ID:	27989
Cross Reference Number:	
Topic:	97 - 295001.G.2.4.6
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	<p>Objective: 29502LK046b Reference: DEOP 400-5 K/A: 295001 G2.4.6 -- / 4.7 K/A: Partial or Complete Loss of Forced Core Flow Circulation: Knowledge of EOP mitigation strategies. CFR: 41.10 / 43.5 / 45.13 PRA: Yes Level: High Safety Function: 1 & 4 Pedigree: New History: N/A</p> <p>Explanation: A. Incorrect - Plausible because SBLC is injected if Reactor power is >6% during an ATWS. B. Incorrect - Plausible because tripping the running Recirc pump would be required if Reactor power is >6% C. Correct - With Reactor power <6% taking the running Recirc pump to minimum speed is the proper action. D. Incorrect - Plausible because if Reactor power is >6% then Terminating and Preventing to at least -35" would be a proper action.</p> <p>SRO Only Criteria: 10CFR55.43(b)(5) - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.</p> <p>REQUIRED REFERENCE: None</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

98

ID: 27993

Points: 1.00

Unit 2 was operating at full power when an event occurred and the following conditions currently exist:

- The reactor scrammed and all rods fully inserted.
- Reactor water level is at -194" and slowly dropping.
- 'A' Main steam line failed to isolate at -59" reactor water level.
- The 2/3 Chimney SPING indicates 2.7E+09 uCi/sec. for the last 20 minutes.
- Plant Parameter Display System indicates a wind direction of 120 degrees and a wind speed of 25 mph.

The correct EP recommendations are to declare a ...

- A. Site Area Emergency and evacuate 2 mile radius and 5 miles downwind for subareas 1, 3, 4.
- B. Site Area Emergency and evacuate 2 mile radius and 10 miles downwind for subareas 1, 2, 3, 4, 5, 6, 8.
- C. General Emergency and evacuate 2 mile radius and 5 miles downwind for subareas 1, 3, 4.
- D. General Emergency and evacuate 2 mile radius and 10 miles downwind for subareas 1, 2, 3, 4, 5, 6, 8.

Answer: D

Answer Explanation

General Emergency is correct based on the current release rate. Max PARs and the listed subareas are correct based on the off-site release and the wind direction of 120 degrees. This would require a state and NRC notification.

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

Question 98 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:	27993
User-Defined ID:	27993
Cross Reference Number:	
Topic:	98 - 295038.G.2.4.30
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	<p>Objective: 29501LP032 References: EP-AA-1004, EP-AA-111-F4 K/A: 295038 G2.4.30 --/4.1 K/A: High Off-Site Release Rate: Knowledge of events related to system operation/status that must be reported to internal organization or external agencies, such as the State, the NRC, or transmission operator. Safety Function: N/A CFR: 43.5 Level: High Pedigree: New History: None</p> <p>Explanations:</p> <p>A. Incorrect - The 2/3 SPING reading greater than 2.05 E+9 for 15 minutes drives us to a RG1. Plausible because the other system parameters do not drive us to a General at this time, but do require outside notifications.</p> <p>B. Incorrect - Plausible because this would be correct for a General, the areas listed would be correct for Shelter with a Hostile Action event in Progress.</p> <p>C. Incorrect - Plausible because this would be correct for a General that did not meet the requirement of a Rapidly Progressing Severe Accident.</p> <p>D. Correct - General Emergency is correct based on the current release rate. Max PARs and the listed subareas are correct based on the off-site release and the wind direction of 120 degrees. This would require a state and NRC notification.</p> <p>SRO Only Criteria: 10CFR55.43(b)(5) - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.</p> <p>Required References: EP-AA 1004 Addendum 3, EP-AA-111-F4</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

99

ID: 27995

Points: 1.00

Unit 2 was operating at near rated power when the Main Turbine tripped and an ATWS occurred.

RPV pressure peaked at 1115 psig then lowered to 1005, then rose to 1115 psig again.

___(1)___ relief valves are cycling open AND the SRO is required to direct initiating the Isolation Condenser ___(2)___ to control RPV pressure.

- A. (1) 2
(2) ONLY
- B. (1) 2
(2) and taking manual control of the ERVs
- C. (1) 5
(2) ONLY
- D. (1) 5
(2) and taking manual control of the ERVs

Answer: B

Answer Explanation

The ERVs open at the following RPV pressure: B & C @ 1097 psig, A (target rock), D, and E open at 1120 psig. All the relief valves re-close at ~70 psig below their individual open setpoint. With an RPV pressure peaking at 1115 psig, this indicates that only 2 (B and C) of the ERVs would be open. With pressure decreasing to 1005, these ERVs would close, thus allowing RPV pressure to re-increase (this cycling makes the distractors plausible). The correct action, per the DEOP, is to control RPV pressure with the Iso Cond and ERVs.

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

Question 99 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:	27995
User-Defined ID:	27995
Topic:	99 - 239002.A2.06
Comments:	<p>Objective: DRE239LN001.06 Reference: DANs 902-3 D-9, C-13, D-13, E-13, E-12; DEOP 100 and 400-5 K/A: 239002.A2.06 4.1 / 4.3 K/A: Ability to (a) predict the impacts of the following on the RELIEF/SAFETY VALVES; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Reactor high pressure.</p> <p>Level: High Safety Function: 3 Pedigree: Bank History: None</p> <p>Explanation:</p> <p>A. Incorrect - The first part is correct. With ERVs cycling the proper action to direct is to utilize the Iso Condenser and the ERVs to control pressure. Plausible because there will be 2 ERVs open and could deduce that allowing them to open on their own to help control pressure.</p> <p>B. Correct - The ERVs open at the following RPV pressure: B & C @ 1097 psig, A (target rock), D, and E open at 1120 psig. All the relief valves re-close at ~70 psig below their individual open setpoint. with an RPV pressure peaking at 1115 psig, this indicates that only 2 (B and C) of the ERVs would be open. With pressure decreasing to 1005, these ERVs would close, thus allowing RPV pressure to re-increase (this cycling makes the distractors plausible). The correct action, per the DEOP, is to control RPV pressure with the Iso Cond and ERVs.</p> <p>C. Incorrect - Only 2 ERVs, B & C, would be open with RPV pressure <1120 psig. With ERVs cycling the proper action to direct is to utilize the Iso Condenser and the ERVs to control pressure. Plausible because if RPV pressure went >1120 psig then all 5 ERVs would be open and there will be 2 ERVs open and could deduce that allowing them to open on their own to help control pressure.</p> <p>D. Incorrect - Only 2 ERVs, B & C, would be open with RPV pressure <1120 psig. Second part is correct. Plausible because if RPV pressure went >1120 psig then all 5 ERVs would be open.</p> <p>SRO Only criteria: 10 CFR 55.43(b).5 - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.</p> <p>REQUIRED REFERENCES: None.</p>

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

100

ID: 27994

Points: 1.00

During a Reactor Startup on Unit 2 with Reactor Power at 7%, CRD H-11 is withdrawn one notch past its target position.

CRD H-11 will display a ____ (1) ____ background on the Rod Worth Minimizer.

The Unit Supervisor will enter Tech Spec ____ (2) ____.

- A. (1) RED
(2) Tech Spec 3.1.6, Rod Pattern Control
- B. (1) CYAN
(2) Tech Spec 3.1.6, Rod Pattern Control
- C. (1) RED
(2) Tech Spec 3.3.2.1, Control Rod Block Instrumentation
- D. (1) CYAN
(2) Tech Spec 3.3.2.1, Control Rod Block Instrumentation

Answer: A

Answer Explanation

Per T.S. 3.1.6 Rod Pattern Control all OPERABLE control rods shall comply with the requirements of the analyzed rod position sequence. This applicable in MODE 2. Per DOP 0400-02 ROD WORTH MINIMIZER, the indication would be Red.

EXAMINATION ANSWER KEY

20-2 (2021-302) NRC - SRO

Question 100 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:	27994
User-Defined ID:	27994
Cross Reference Number:	
Topic:	100 - 201006.A2.05
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	<p>Objective: 201LN006.08 Reference: DOA 0300-12, TS 3.1.3, TS 3.1.6, DOP 0400-02 K/A: 201006.A2.05 3.1/3.5 K/A: Ability to (a) predict the impacts of the following on the ROD WORTH MINIMIZER SYSTEM (RWM); and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Out of sequence rod movement: P-Spec (Not-BWR-6)</p> <p>CFR: 41.5 Safety Function: 7 Pedigree: New History: N/A Level: High</p> <p>Explanation: A. Correct - Per T.S. 3.1.6 Rod Pattern Control all OPERABLE control rods shall comply with the requirements of the analyzed rod position sequence. This applicable in MODE 2. Per DOP 0400-02 ROD WORTH MINIMIZER, the indication would be Red. B. Incorrect - Cyan represents a rod OOS not mispositioned. Plausible because T.S. 3.1.6 is correct and Cyan would be correct for an OOS rod. C. Incorrect - Plausible because RED is correct. T.S. would be correct for control rod block instrumentation. D. Incorrect - Plausible because Cyan is a correct color for a mispositioned rod. Part 2 would be correct if the cause of the misposition was a control rod failure.</p> <p>SRO Only criteria: 10 CFR 55.43(b).5 - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.</p> <p>Required References: None</p>