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1	K/A Importance: 2.9/3.0			**Points: 1.00**
R01	Difficulty: 2.00	Level of Knowledge: Higher cognitive level	Source: NEW	87852

Which of the responses below best completes the following statement to describe indications of a failed inlet elbow hold down bolt causing the inlet ram's head to separate from a pair of Reactor Recirculation Jet Pumps?

Jet Pump Total Flow will __(1)__ and the affected loop's Recirculation Loop Flow will __(2)__.

A. (1) decrease due to a reduction in core flow
(2) increase due to a reduction in system head loss

B. (1) increase due to an increase in core flow
(2) increase due to a reduction in system head loss

C. (1) increase due to an increase in core flow
(2) decrease due to an increase in system head loss

D. (1) decrease due to a reduction in core flow
(2) decrease due to an increase in system head loss

Answer: A

Answer Explanation:
The conditions given in the stem indicate a partial loss of core flow due to a reduction in flow to a pair of jet pumps caused by separation of the inlet ram's head.

Per 20.138.02, Jet Pump Failure AOP and BASES:

The average power range monitor simulated thermal power upscale trip function trip level is varied as a function of recirculation drive flow. The drive flow signal is used to give a total core flow signal, which is meant to provide a representative upscale trip signal. Because a failed jet pump could cause a lower total core flow than is depicted by the drive flow, the average power range monitor simulated thermal power upscale trips could be non-conservative in value. This reason also applies to the oscillation power range monitor, which is enabled at 60% drive flow decreasing. The jet pump failure may result in higher drive flow, which would result in the actual enabling of the oscillation power range monitor at a less conservative core flow. Because this affects all the APRMs and OPRMs, the operating staff needs to be aware of the requirements specified in the technical specifications for these instruments, which could have non-conservative setpoints due to a jet pump failure. Example: If drive flow has increased and power has decreased, the setpoints will be non-conservative.

The candidate must recall that the broken jet pump ram's head will cause a reduction in flow through the affected jet pumps, which will cause a reduction in total jet pump flow. The candidate must then recall that the broken jet pump will result in less back pressure (head loss) in the affected recirculation loop which will cause loop flow to increase.

Distractor Explanation:
Distractors are incorrect and plausible because:

B. (1) The candidate may incorrectly apply Caution 2 in 20.138.02, which states "Indicated core flow may be higher than actual. Caution should be exercised when reducing core flow to prevent thermal-hydraulic instability." The flow referred to in this Caution is Loop Flow and not Jet Pump Flow as explained in the AOP Bases and described above. (2) is correct.
C. (1) The candidate may incorrectly apply Caution 2 in 20.138.02, which states "Indicated core flow may be higher than actual. Caution should be exercised when reducing core flow to prevent thermal-hydraulic instability." The flow referred to in this Caution is Loop Flow and not Jet Pump Flow as explained in the AOP Bases and described above. (2) The candidate could determine that the broken jet pump inlet elbow (ram's head) represents a more tortuous flow path for water to take from the jet pump riser, through the inlet and finally into the jet pump pair supplied by the ram's head. This could lead the candidate to conclude that Loop Flow will decrease because of the increased head loss, which is incorrect because head loss will actually decrease due to water exiting the ram's head and returning to the downcomer region rather than travelling through the jet pumps to the core inlet.
D. (1) is correct. (2) The candidate could determine that the broken jet pump inlet elbow (ram's head) represents a more tortuous flow path for water to take from the jet pump riser, through the inlet and finally into the jet pump pair supplied by the ram's head. This could lead the candidate to conclude that Loop Flow will decrease because of the increased head loss, which is incorrect because head loss will actually decrease due to water exiting the ram's head and returning to the downcomer region rather than travelling through the jet pumps to the core inlet.

Reference Information:

20.138.02, Jet Pump Failure AOP and BASES

Question Use
Closed Reference
RO

NUREG 1123 KA Catalog Rev. 2
295001 Partial or Complete Loss of Forced Core Flow Circulation
295001 AK3. Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION :
295001 AK3.06 Core flow indication

10CFR55 RO/SRO Written Exam Content
10 CFR 55.41(b) (5) Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

NRC Exam Usage
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Higher cognitive level
New
RO

Associated objective(s):
Reactor Recirculation System
Cognitive Enabler
Given Reactor Recirculation system performance data, detect abnormalities, and determine possible causes for performance problems.

Reactor Recirculation System
Cognitive Terminal
Given the system operating conditions/parameters, in accordance with approved plant procedures: Discuss potential modes of Reactor Recirculation system component failures.

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2	K/A Importance: 4.2/4.3			**Points: 1.00**
R02	Difficulty: 3.00	Level of Knowledge: Higher cognitive level	Source: BANK	87588

Following a loss of 120 kV, the following indications exist on panel H11-P809:

<u>Breaker Positions</u>

- BUS 64B POS B8, 4160V X-Tie to Bus 11EA .... CLOSED
- BUS 64B POS B6, 4160V Normal Feed to Bus 64B .... TRIPPED
- BUS 11EA POS EA3, EDG 11 4160V Output Breaker .... CLOSED
- BUS 12EB POS EB3, EDG 12 4160V Output Breaker .... OPEN
- BUS 64C POS C8, 4160V X-Tie to Bus 12EB .... CLOSED
- BUS 64C POS C6, 4160V Normal Feed to Bus 64C .... TRIPPED

Based on these indications what event has occurred?

A. Lockout of BUS 64B

B. Lockout of BUS 64C

C. EDG 11 failure

D. EDG 12 failure

Answer: D

Current View

Answer Explanation:
20.300.120kV Page 8, Notes 3 and 4 state that "If Bus 64B (64C) Pos B6 and B8 (C6 and C8) are open, this is an indication of a bus fault. The EDG is designed not to start on a bus fault."
The candidate must evaluate the indications given and determine that a bus lockout does NOT exist.

20.300.120kV directs 20.307.01 and 20.300.72C if the EDG 12 output breaker fails to close (Condition K). Additionally, no bus lockout is indicated so 20.307.01 would be effective if the cause of the EDG failure is correctable.

Distracter Explanation:

A. Is incorrect and plausible because there is no indication of a 64B BUS LOCKOUT (indicated by BUS isolation).

B. Is incorrect and plausible because there there is no indication of a 64C BUS LOCKOUT (indicated by BUS isolation).

C. Is incorrect and plausible because the BUS 11EA POS EA3 is CLOSED so EDG 11 has started and loaded.

Reference Information:

20.300.120kv, page 9, Condition K.

Plant Procedures
20.300.120kv

Question Use
Closed Reference
RO

NUREG 1123 KA Catalog Rev. 2
295003 Partial or Complete Loss of A.C. Power
295003 AA1. Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER :
295003 AA1.02 Emergency generators

10CFR55 RO/SRO Written Exam Content
10 CFR 55.41(b)(10) Administrative, normal, abnormal, and emergency operating procedures for the facility.

NRC Exam Usage
ILO 2015 Exam
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Bank
Higher cognitive level
RO

Associated objective(s):

Emergency Diesel Generator (R3000)

Cognitive Enabler

Identify abnormal and emergency operating procedures associated with the Emergency Diesel Generator System.

Integrated Electrical Events

Performance Enabler

Recognize, respond to, and correct Emergency Diesel Generator Failure

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3	K/A Importance: 3.3/3.4			**Points: 1.00**
R03	Difficulty: 3.00	Level of Knowledge: Higher cognitive level	Source: BANK	70212

The plant is operating at full power when the following annunciators alarm:

- 9D17 DIV I ESS 130 V BATTERY 2PA TROUBLE
- 9D21 DIV I EDG SEQUENCER TROUBLE
- 1D6 DIV I CSS LOGIC POWER FAILURE
- 1D8 RHR LOGIC A 125 VDC BUS POWER FAILURE
- 1D62 STM LK DET HPCI LOGIC POWER FAILURE

Select the correct diagnosis and effect, if any, on Division I EDGs.

A. ONLY the Division I Batteries have been lost. Division I EDGs will AUTO START upon receipt of a valid start signal.

B. ONLY the Division I Batteries have been lost. Division I EDGs will NOT START upon receipt of a valid start signal.

C. BOTH Division I Battery Chargers AND Division I Batteries have been lost. Division I EDGs will AUTO START upon receipt of a valid start signal.

D. BOTH Division I Battery Chargers AND Division I Batteries have been lost. Division I EDGs will NOT START upon receipt of a valid start signal.

Answer: D

Answer Explanation:
K/A Match justification:

Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER :
Cause of partial or complete loss of D.C. power.

Alarms provided in the STEM indicate a DC POWER LOSS. The candidate INTERPRETs the DC POWER LOSS based on the listed alarms and then DETERMINEs if the cause of the LOSS OF DC POWER was due to either:
BOTH Division I Battery Chargers AND Division I Batteries
– OR –

ONLY the Division I Batteries.

Answer explanation:
Examinee must diagnose PARTIAL or COMPLETE Loss of Division I DC based on listed annunciators.
Diversity of DC power loss symptoms indicates that Division I Chargers AND Batteries are lost.
Per 20.300.260VESF the impact of a Complete Loss of DC on EDGs there is no AUTO START function.

Distractors Explanation:
Distractors are incorrect and plausible because:

A. Batteries AND Chargers must be lost to have a Complete Loss of Div I DC Power, therefore ONLY Division 1 Batteries is incorrect. AUTO START of DIV I EDGs are disabled by Loss of DIV I DC as indicated by the listed alarms therefore Division I EDGs will AUTO START is incorrect. This answer is plausible because the candidate may diagnose the indication as a partial loss of DC.
B. Batteries AND Chargers must be lost to have a Complete Loss of Div I DC Power, therefore ONLY Division 1 Batteries is incorrect. This answer is plausible because the candidate may diagnose the indication as a partial loss of DC.
C. AUTO START of DIV I EDGs are disabled by Loss of DIV I DC as indicated by the listed alarms therefore Division I EDGs will AUTO START is incorrect. This answer is plausible because the candidate may diagnose the indication as a partial loss of DC.

Reference Information:
20.300.260VESF

Plant Procedures
20.300.260VESF
23.309

Question Use
Closed Reference
RO

NUREG 1123 KA Catalog Rev. 2

295004	Partial or Complete Loss of D.C. Power
295004 AA2.	Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER :
295004 AA2.01	Cause of partial or complete loss of D.C. power

10CFR55 RO/SRO Written Exam Content

10 CFR 55.41(b) (7)	Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

NRC Exam Usage
LOR 2009 Exam
LOR 2015 Exam
ILO 2012 Exam
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Bank
Higher cognitive level
RO

Associated objective(s):
Cycle 16-4 Objectives
Cognitive Terminal
Describe the impact associated with a fault on the EF2 electrical distribution system

DC Electrical Distribution (R3200 & S3102)
Cognitive Enabler
Given DC Electrical Distribution System performance data, detect abnormalities, and determine possible causes for performance problems.

DC Electrical Distribution
Cognitive Terminal
In accordance with approved plant procedures, given various controls and indications for system operations: Describe the impact on plant operations of a loss of the following DC buses, and describe the actions necessary to correct, control or mitigate the loss of DC power: 260/130VDC ESF bus, 260/130VDC BOP bus, 48/24VDC Bus

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4	K/A Importance: 4.6 / 4.6			**Points: 1.00**
R04	Difficulty: 3.00	Level of Knowledge: Higher cognitive level	Source: BANK	87667

Reactor power is 50% during a plant startup with the following in alarm:

- 4D9, Main Turbine Vibration High-High.
- 4D13, Main Turbine Vibration High.

The CRLNO reports the following main turbine vibration levels:

Bearing	Vibration level
1	1 mil
2	1 mil
3	1 mil
4	2 mils
5	3 mils
6	4 mils
7	4 mils
8	7 mils
9	11 mils
10	7 mils

Which action(s) will the crew perform?

A. Scram the reactor, and trip the main turbine.

B. Lower turbine load by 50 MWe and reevaluate.

C. Fully unload the turbine, and shutdown to hot standby.

D. Maintain turbine load at the present load and contact system engineering.

Answer: A

Answer Explanation:
Per 4D9, Turbine vibration Hi-HI trip setpoint has been exceeded on turbine bearing #9.
Per 4D13, Turbine vibration Hi setpoint has been exceeded on bearings 8 and 10. (7.5 mils for #8, and 6.0 mils for #10.)

Per both 4D9 and 4D13, Hi-Hi on bearing 9 with a Hi on bearing 8 or 10 will cause an automatic turbine trip. At the current power level, the proper course of action is to place the Reactor Mode Switch in Shutdown, then trip the turbine to prevent the automatic RPS actuation.

Distractor Explanation:
B. Plausible because this would be a conservative course of action if a hi vibration issue existed, but without exceeding the setpoint.Incorrect because the turbine trip setpoint has been exceeded.
C. Plausible because this would be a conservative course of action if a hi vibration issue existed, but without exceeding the setpoint.Incorrect because the turbine trip setpoint has been exceeded.
D. Plausible because the call to system engineering is a recommendation within the procedure when turbine vibration hi (4D13) or hi-hi (4D9) is in alarm. Incorrect because the turbine has exceeded a turbine trip setpoint, and the operators must trip the turbine.

Plant Procedures
20.109.01
23.109
04D009

Question Use
Closed Reference
RO

NUREG 1123 KA Catalog Rev. 2
G2.1.20 Ability to interpret and execute procedure steps
295005 Main Turbine Trip

10CFR55 RO/SRO Written Exam Content
10 CFR 55.41(b)(10) Administrative, normal, abnormal, and emergency operating procedures for the facility.

NRC Exam Usage
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Bank
Higher cognitive level
RO

Associated objective(s):
Turbine Steam
Cognitive Terminal
Given the Turbine Steam/parameters, evaluate those conditions/parameters for the appropriate operator response in accordance with approved plant procedures.

Turbine Supervisory Equipment and Protection
Cognitive Terminal
Given the system operating conditions/parameters, in accordance with approved plant procedures: Identify alarm response procedures associated with the Turbine Supervisory Equipment and Protection System.

5	K/A Importance: 3.7 / 4.0			**Points: 1.00**
R05V2	Difficulty: 3.00	Level of Knowledge: Fundamental	Source: BANK 2014 RIVER BEND	88808

To ensure sufficient time is allowed to fully insert the control rods following a scram, the scram signal from the __(1)__ is present __(2)__.

A. (1) RPS manual pushbuttons
(2) for 10 seconds

B. (1) RPS manual pushbuttons
(2) until bypassed with a keylock switch

C. (1) Reactor Mode Switch
(2) for 10 seconds

D. (1) Reactor Mode Switch
(2) until bypassed with a keylock switch

Answer: C

Answer Explanation:
Per 3D73, the mode switch position when taken to shutdown provides a scram signal that is bypassed after 10 seconds.

On logic diagram I-2155-08, it can be seen the relay K16A and K17A work together to provide the 10 second time delay around the mode switch. K16A is energized when in the Shutdown position (D4), which de-energizes K17A (D3). K17A has a 10 second delay on drop-out. At F5, an "a" contact from K16A, together with a "b" contact from K17A provide the bypass around the mode switch after the 10 second drop out time is expired.

TS Bases page 3.3.1.1-2 states "This 10 second delay on reset ensures that the scram function will be completed."

Distractor Explanation:
Distractors are incorrect and plausible because:

A. Plausible because the RPS pushbuttons do provide a scram signal. Incorrect because as can be seen on I-2155-08, the pushbuttons only interrupt the current path while physically pressed, with no automatic (or manual bypass) around the pushbutton.
B. Plausible because the RPS pushbuttons do provide a scram signal. Incorrect because as can be seen on I-2155-08, the pushbuttons only interrupt the current path while physically pressed, with no automatic (or manual bypass) around the pushbutton.
D. Plausible because the Reactor Mode Switch does provide a scram signal. Incorrect because the automatic bypass around the mode switch after 10 seconds to allow a scram to be reset is accomplished automatically by relays K16A and K17A, as shown on I-2155-08, and not by the means of a keylock switch.

Reference Information:

3D73
I-2155-08
TS Bases 3.3.1.1

Question Use
Closed Reference
RO

NUREG 1123 KA Catalog Rev. 2
295006 SCRAM
295006 AK1. Knowledge of the operational implications of the following concepts as they apply to SCRAM :
295006 AK1.03 Reactivity control

10CFR55 RO/SRO Written Exam Content
10 CFR 55.41(b) (6) Design, components, and function of reactivity control mechanisms and instrumentation.

NRC Exam Usage
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Bank
Fundamental
RO

Associated objective(s):
Reactor Protection System (C7100)

Cognitive Enabler

Describe general Reactor Protection System operation, including component operating sequence, normal operating parameters, and expected system response.

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6	K/A Importance: 4.4/4.5			**Points: 1.00**
R06	Difficulty: 2.00	Level of Knowledge: Fundamental	Source: BANK: 2017 FERMI NRC EXAM	88007

Which one of the following correctly identifies the RCIC controls or indications available BOTH in the Main Control Room AND at the Remote Shutdown Panel?

A. RCIC Pump Flow indication.

B. RCIC Pump Discharge Pressure indication.

C. E5150-F010, RCIC Pump CST Suction Isolation Valve, control pushbuttons.

D. RCIC Pump Flow controller with Manual and Automatic Setpoint Adjustments.

Answer: A

Answer Explanation:
RCIC Flow Indication exists at both locations.

Distractor Explanation:
B. Is incorrect and plausible because it exists only in the MCR.
C. Is incorrect and plausible because it exists only in the MCR.
D. Is incorrect and plausible because it exists only in the MCR.

Reference Information:
20.000.19, page 6

Plant Procedures
20.000.19

Question Use
Closed Reference
RO

NUREG 1123 KA Catalog Rev. 2
295016 Control Room Abandonment
295016 AK2. Knowledge of the interrelations between CONTROL ROOM ABANDONMENT and the following:
295016 AK2.01 Remote shutdown panel: Plant-Specific

10CFR55 RO/SRO Written Exam Content
10 CFR 55.41(b) (8) Components, capacity, and functions of emergency systems.

NRC Exam Usage
ILO 2012 Exam
ILO 2017 Exam
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Bank
Fundamental
RO

Associated objective(s):
Reactor Core Isolation Cooling
Cognitive Enabler
Identify associated remote and local instrumentation, indications, alarms, and controls for the RCIC System.

Remote Shutdown System
Cognitive Terminal
In accordance with approved plant procedures/references, under all conditions of the Remote Shutdown System. Identify remote and local instrumentation, indications, alarms, and controls for the Remote Shutdown System.

7	K/A Importance: 3.3 / 3.4			**Points: 1.00**
R07	Difficulty: 2.00	Level of Knowledge: Higher cognitive level	Source: NEW	87568

The plant is operating at 100% power.

The following then occurs:

- 2D112, RBCCW Makeup Tank Level High/Low is received.
- 2D116, RBCCW Makeup Tank Flow High is received
- G1101-C001A, RW DW Floor Drain Sump West Sump Pump starts.
- P42-F410, RBCCW Head Tank Demin Water Makeup LCV is 100% open
- P42-R400, RBCCW Head Tank Level Ind is -18" and lowering.
- Both running RBCCW pumps trip.

Which of the following describes actions the crew will take and the reason for those actions?

A. Commence GOP shutdown to allow for leak isolation.

B. Commence GOP shutdown to comply with Technical Specifications.

C. Place the Reactor Mode Switch in Shutdown because of the total loss of RBCCW.

D. Trip Recirc Pump A, Isolate Div 1 EECW to and from the Drywell and then monitor reactor power to perform leak isolation.

Answer: C

Answer Explanation:
Per 20.127.01, "A total loss of RBCCW is defined as, no pump running with no reasonable likelihood of restarting any RBCCW pump."

With the given stem conditions, both running RBCCW pumps have tripped, and there is no reasonable likelihood of restarting any RBCCW pump because the trip occurred on low makeup tank level, and the level is dropping with the makeup valve fully open.

Distractor Explanation:
Distractors are incorrect and plausible because:

A. Plausible because a GOP shutdown would be preferred if the leak rate was between 5 and 25 gpm. The examinee may disregard the trip of RBCCW because the automatic actuation of EECW will provide cooling to the safety related loads. Also plausible because the 2nd half is correct if the leak rate was between 5 and 25 gpm. Incorrect because the leak rate is not manageable, and a reactor scram is directed by procedure.
B. Plausible because a GOP shutdown would be preferred if the leak rate was between 5 and 25 gpm. The examinee may disregard the trip of RBCCW because the automatic actuation of EECW will provide cooling to the safety related loads. Also plausible because there are TS implications of EECW leakage. If two EECW systems needed to be declared inoperable, then the plant would be required to be in mode 3 within 12 hours. Incorrect because the leak rate is not manageable, and a reactor scram is directed by procedure.
D. Plausible because this action is direction for a leak inside the DW from Div 1 EECW. However this is incorrect because this is done with the mode switch in SHUTDOWN.

Reference Information:

2D112 RBCCW Makeup Tank Level High/Low
2D116 RBCCW Makeup Tank Flow High
20.127.01 Bases
20.127.01 Loss of Reactor Building Closed Cooling Water System

Question Use
Closed Reference
RO

NUREG 1123 KA Catalog Rev. 2

295018	Partial or Complete Loss of Component Cooling Water
295018 AK3.02	Reactor power reduction
295018 AK3	Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER:

10CFR55 RO/SRO Written Exam Content

10 CFR 55.41(b) (5)	Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

NRC Exam Usage
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Higher cognitive level
New
RO

Associated objective(s):
Reactor Building Closed Cooling Water Emergency Equipment Cooling Water (P4200/P4400)

Cognitive Enabler

Given Reactor Building Closed Cooling Water/Emergency Equipment Cooling Water System performance data, detect abnormalities, and determine possible causes for performance problems.

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8	K/A Importance: 3.5.3.3			**Points: 1.00**
R08V2	Difficulty: 3.00	Level of Knowledge: Higher cognitive level	Source: BANK	88787

Due to a leak in the pipe supplying NIAS from IAS, the following conditions currently exist in the plant's air systems (NOTE: The air pressures listed are the LOWEST sensed during the transient):

- Division 1 NIAS to Dryer Pressure .... 84 psig (lowering)
- Division 2 NIAS to Dryer Pressure .... 84 psig (lowering)
- Interruptible Air to NIAS Header Pressure .... 79 psig (lowering)

Based on these conditions, the current status of plant air system components is:

(1) P5002-D001(D002) Division 1(2) Control Air Compressors are ________.
(2) P5000-F440(F441) Division 1(2) Control Air Isolation Valves are ________.

A. (1) running
(2) closed

B. (1) not running
(2) closed

C. (1) running
(2) open

D. (1) not running
(2) open

Answer: C

Answer Explanation:
20.129.01, Loss of Station and Control Air AOP, Condition B and C contain the information necessary to answer this question.

The candidate must evaluate system conditions given in the stem of the question and determine that at 85 psig both Control Air Compressors would have started. The candidate must also recall that, since system pressure has not yet lowered below 75 psig, the P5000-F440 and F441 would still be open.

Distractor Explanation:
Distractors are incorrect and plausible because:

A. This would be correct if the lowest pressures sensed were < 75 psig since the Control Air Compressors receive an auto start at 75 psig. However, this distractor is incorrect with the air pressures listed.
B. This would be correct if the setpoints for the valves closing and the compressors starting were reversed, which is a common misconception for initial license candidates since it makes sense to some that the system might try to isolate NIAS from the damaged part of the system prior to starting the Control Air Compressors. This is incorrect, however, because the CACs start at 85 psig while the valves isolate at 75 psig.
D. This is is the normal system lineup and would be correct if the lowest pressures sensed were > 85 psig. However, since pressure is <85 psig, the CACs would be running.

Reference Information:
20.129.01, Loss of Station and/or Control Air
20.129.01 BASES, Loss of Station and/or Control Air Bases

Plant Procedures
20.129.01
23.129

Question Use
Closed Reference
RO

NUREG 1123 KA Catalog Rev. 2

295019	Partial or Complete Loss of Instrument Air
295019 AA1.	Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR:
295019 AA1.01	Backup air supply

10CFR55 RO/SRO Written Exam Content

10 CFR 55.41(b) (7)	Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

NRC Exam Usage
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Bank
Higher cognitive level
RO

Associated objective(s):
Compressed Air Systems (P5001 & P5002)

Cognitive Terminal

Given the Compressed Air Systems operating conditions/parameters, evaluate those conditions/parameters for the appropriate operator response in accordance with approved plant procedures.

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9	K/A Importance: 3.4/3.4			**Points: 1.00**
R09	Difficulty: 3.00	Level of Knowledge: Higher cognitive level	Source: BANK	87630

The plant is in HOT SHUTDOWN, with the following conditions:

- Reactor Pressure is 25 psig.
- RPV Water Level is 250 inches.
- RHR Pump B is operating in Shutdown Cooling Mode.
- 2D26, Div II RHR System Low Flow Bypass Initiated, alarms due to a faulty flow switch.
- E1150-F007B, Div 2 RHR Pmps Min Flow Vlv, OPENS

Which one of the following describes the effect of the above on ACTUAL flow through the B RHR Pump and continued RHR Loop B Shutdown Cooling operation?

A. Flow will increase; Reactor Water Level will lower until Shutdown Cooling isolates.

B. Flow will remain the same**;** RHR Loop B will remain operating in Shutdown Cooling with this condition.

C. Flow will decrease; Reactor Water Level will lower until Shutdown Cooling isolates.

D. Flow will increase; RHR Heat Exchanger Bypass Valve has opened; Reactor Coolant System flow will decrease until Shutdown Cooling isolates.

Answer: A

Answer Explanation:
The alarm 2D26 indicates the RHR pumps min flow valve has opened resulting in increased pump flow (two paths). The min flow valve will reject Rx water to the Torus causing level to lower until a low Rx water level isolation is reached.

B. is incorrect because the alarm indicates the min flow valve has opened which will affect flow and level.
C. is incorrect because RHR pump flow will increase. This could be confused with RHR Loop flow which would slightly decrease as a result of the min flow valve opening.
D. is incorrect because it could identify a misconception about the word BYPASS in the alarm title. If the RHR HX were fully BYPASSED, coolant system flow would increase with no isolation.

References: 2D26, page 1; 3D79, page 1

Plant Procedures
03D079
02D026

Question Use
Closed Reference
RO

NUREG 1123 KA Catalog Rev. 2
295021 Loss of Shutdown Cooling
295021 AA2. Ability to determine and/or interpret the following as they apply to LOSS OF SHUTDOWN COOLING:
295021 AA2.02 RHR/shutdown cooling system flow

10CFR55 RO/SRO Written Exam Content
10 CFR 55.41(b) (7) Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

NRC Exam Usage
ILO 2012 Exam
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Bank
Higher cognitive level
RO

Associated objective(s):
Residual Heat Removal (E1100)
Cognitive Enabler
Given RHR System performance data, detect abnormalities, and determine possible causes for performance problems.

10	K/A Importance: 4.4/4.7			**Points: 1.00**
R10	Difficulty: 3.00	Level of Knowledge: Higher cognitive level	Source: MODIFIED: 2018 FERMI NRC EXAM	87631

A refueling accident has caused the following:

- 16D1, RB Refueling Area Fifth Floor High Radiation is received.
- D21-K717, RB5 Refuel Floor Lo Range ARM Ind trip Unit, indicates 10 mR/hr and slowly lowering.
- 3D35, Div I/II FP Vent Exh Radn Monitor Upscale Trip is received.
- All Fuel Pool Vent Exhaust Radiation Monitors peaked at 3.5 mR/hr and are now slowly lowering.

(1) What automatic actions, if any, will occur AND (2) which of the following procedure(s) entry conditions is (are) met?

A. (1) RBHVAC isolates.

(2) 20.000.02, Abnormal Release of Radioactive Material, ONLY

B. (1) RBHVAC isolates.

(2) 20.710.01, Refueling Floor High Radiation, AND 20.000.02, Abnormal Release of Radioactive Material

C. (1) No automatic actions occur.

(2) 20.710.01, Refueling Floor High Radiation, AND 20.000.02, Abnormal Release of Radioactive Material

D. (1) No automatic actions occur.

(2) 20.710.01, Refueling Floor High Radiation, AND 29.100.01, Sheet 5, Secondary Containment Control

Answer: B

Answer Explanation:
Per 16D1, if Rad Monitor Channel 15, 17 or 18 are verified greater than alarm setpoint, immediately evacuate the area, enter 20.710.01, Refueling Floor High Radiation, and enter 20.000.02, Abnormal Release Of Radioactive Material.
Per 3D35 if the setpoint of 3mr/hr is reached, RBHVAC will isolate, CCHVAC will shift to the Recirculation Mode, and SGTS will automatically start. Also, if FP Vent Exh Rad Monitors are verified greater than 3 mr/hr, perform 20.000.02, Abnormal Release Of Radioactive Material and 20.710.01, Refueling Floor High Radiation. Finally, if Fuel Pool Vent Exh Duct Radiation is verified greater than 5 mr/hr, perform 29.100.01 SH 5, "Secondary Containment and Rad Release,".

Therefore, the candidate must recall the above and determine that RBHVAC will isolate and 20.000.02 and 20.710.01 must be entered.

Distractor Explanation:
Distractors are incorrect and plausible because:

A. Part (1) is correct in that RBHVAC isolates. Part (2) is plausible if the candidate recognizes the need to enter Abnormal Release of Radioactive Material due to the RB5 radiation levels but failed to recognize that entry conditions exist for 20.710.01, Refueling Floor High Radiation. This distractor is incorrect because 20.710.01 is also required to be entered.
C. Note: This was the previously correct answer on the original (bank) version. This distractor is plausible because the examinee could have memorized the bank version but failed to recognize that, in this version, FP Vent Exh Rad Monitors have exceeded their trip setpoints which will cause RBHVAC to isolate.
D. Part (1) was previously correct and is plausible if the candidate failed to recognize that the FP Vent Exh Rad Monitors have exceeded their trip setpoints which will cause RBHVAC to isolate. Part 2 is incorrect because 29.100.01 is only required to be entered if FP Vent Exh Rad levels exceed 5 mr/hr.

Reference Information:
20.710.01, Refueling Floor High Radiation.
16D1, RB Refueling Area Fifth Floor High Radiation
3D35, Div I/II FP Vent Exh Radn Monitor Upscale Trip.

Plant Procedures
16D01

Question Use
Closed Reference
RO

NUREG 1123 KA Catalog Rev. 2

G2.1.7	Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation
295023	Refueling Accidents

10CFR55 RO/SRO Written Exam Content

10 CFR 55.41(b) (7)	Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.
10 CFR 55.41(b)(13)	Procedures and equipment available for handling and disposal of radioactive materials and effluents.

NRC Exam Usage
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Higher cognitive level
Modified
RO

Associated objective(s):
Refueling
(F1500)

Cognitive Terminal

Given Refueling operating conditions/parameters, evaluate those conditions/parameters for the appropriate operator response in accordance with approved plant procedures.

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11	K/A Importance: 4.1/4.2			Points: 1.00
R11	Difficulty: 3.00	Level of Knowledge: Fundamental	Source: BANK	87632

Which of the responses below completes the following regarding the operational implications of High Drywell Pressure?
In MODES 1, 2 and 3 the maximum Drywell Pressure allowed by Technical Specification LCO 3.6.1.4, Primary Containment Pressure, is __(1)__. This limit is based on __(2)__.

A. (1) 1.5 psig

(2) maintaining the resultant differential pressure below the maximum primary containment design differential pressure in the event of inadvertent drywell spray actuation.

B. (1) 2.0 psig

(2) maintaining the resultant differential pressure below the maximum primary containment design differential pressure in the event of inadvertent drywell spray actuation.

C. (1) 1.5 psig

(2) maintaining the resultant peak primary containment accident pressure below the primary containment design pressure in the event of a Design Basis Accident (DBA).

D. (1) 2.0 psig

(2) maintaining the resultant peak primary containment accident pressure below the primary containment design pressure in the event of a Design Basis Accident (DBA).

Answer: D

Answer Explanation:
The examinee should correctly conclude that, in accordance with LCO 3.6.1.4, the maximum Drywell Pressure in the specified MODEs is 2.0 psig. The examinee should also conclude that the bases for this limit is to preserve the initial conditions assumed in the accident analysis for the Design Basis Accident (DBA) thus ensuring that the peak primary containment pressure does not exceed the design pressure of the primary containment pressure.

Distracter Explanation:

A. Is incorrect because the examinee could fail to recall the LCO 3.6.1.4 maximum allowable Drywell Pressure of 2.0 psig and incorrectly recall that the limit is 1.5 psig, which is plausible because this is the setpoint for 3D81, Primary Containment Pressure High Alarm on the P603 panel. Also, the examinee could fail to recognize that the basis given in the second part of the distractor is for the minimum allowable Drywell Pressure specified by LCO 3.6.1.4 in MODES 1, 2 and 3, which protects containment integrity by keeping external to internal drywell differential pressure below the maximum allowable design D/P.
B. Is incorrect and plausible because the first part of the distractor does provide the correct maximum Drywell Pressure allowed by LCO 3.6.1.4 in the MODEs specified. However, the examinee could fail to recognize that the basis given in the second part of the distractor is for the minimum allowable Drywell Pressure specified by LCO 3.6.1.4 in MODES 1, 2 and 3, which protects containment integrity by keeping external to internal drywell differential pressure below the maximum allowable design D/P.
C. Is incorrect because the examinee could fail to recall the LCO 3.6.1.4 maximum allowable Drywell Pressure of 2.0 psig and incorrectly recall that the limit is 1.5 psig, which is plausible because this is the setpoint for 3D81, Primary Containment Pressure High Alarm on the P603 panel.

Reference Information:

For this question the T.S.B. is used as a reference for system information that is taught in the systems course under lesson plan LP-315-0127 per training objective C013:

C013. Describe the Reactor Protection system technical specification limiting conditions for operation, their bases, the associated surveillance requirement(s), and their relationship to operability.

T.S. 3.6.1.4 (pg 3.6-18) LCO statement
T.S. 3.6.1.4 BASES (pg B3.3.1.4-1 to 2) Bases for limit.

Plant Procedures
03D081

Question Use
Closed Reference
RO

NUREG 1123 KA Catalog Rev. 2

295024	High Drywell Pressure.
295024 EK1.	Knowledge of the operational implications of the following concepts as they apply to HIGH DRYWELL PRESSURE:
295024 EK1.01	Drywell integrity: Plant-Specific

Technical Specifications
3.6.1.4 Primary Containment Pressure

10CFR55 RO/SRO Written Exam Content
10 CFR 55.41(b) (8) Components, capacity, and functions of emergency systems.

NRC Exam Usage
ILO 2015 Exam
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Bank
Fundamental
RO

Associated objective(s):
Containment Systems (T2200 & T2300)
Cognitive Enabler
Describe the Containment Systems technical specification limiting conditions for operation, their bases, the associated surveillance requirement(s), and their relationship to operability.

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12	K/A Importance: 3.7/3.7			**Points: 1.00**
R12	Difficulty: 3.00	Level of Knowledge: Higher cognitive level	Source: BANK	87633

The plant is operating at 90% power. The #1 pressure regulator is in service, and you are asked to raise the setpoint 1 PSI. When you take your finger off the raise button, the button stays depressed, and you are unable to release it from this position.

Which of the following describes the response of the plant?

A. Reactor pressure will lower and the reactor will scram on MSIV closure.

B. Reactor pressure will raise and the reactor will scram on high reactor pressure.

C. The #2 pressure regulator will take over and control reactor pressure approximately 3.5 psi below the setpoint of the #1 regulator.

D. The #2 pressure regulator will take over and control reactor pressure approximately 3.5 psi above the setpoint of the #1 regulator.

Answer: B

Answer Explanation:
Raising the setpoint of the pressure regulator will result in reactor pressure increasing until the High Reactor Pressure Scram setpoint in reached without the backup regulator taking over. This is because, as the pushbutton is held, the raising signal is applied to both the in-service and the backup regulator, preventing the backup regulator from taking control, as it normally would, to prevent a high reactor pressure condition.

Distractor Explanation:
Distractors are incorrect and plausible because:

A. This is plausible because this is how the system would respond to, as the AOP states, "Pressure Regulator signal fails high". The candidate could confuse the term "signal" with "setpoint" and determine that the setpoint going high is the same as the "signal failing high" but is incorrect because the push button sticking in the raise position causes the Pressure Regulator signal to go low and thus reactor pressure will rise.
C. The candidate could incorrectly recall the system diagram and determine that this is how the system would work if the setpoint to only the in-service regulator were failing and the backup pressure regulator was able to take control, which is plausible because it is logical to assume that the system would be designed to prevent a high RPV pressure condition due to the impact the reactor pressure on core reactivity. This is incorrect because the sticking pushbutton would impact both pressure regulators so that backup would not be able to take control.
D. The candidate could determine that only the input to the in-service regulator were failing high because eventually the backup regulator would take over and control pressure 3.5 psi higher. This is incorrect because the sticking pushbutton would impact both pressure regulators so that backup would not be able to take control.

Reference Information:

20.109.02, page 3 and Enclosure A.

Plant Procedures
20.109.02

Question Use
Closed Reference
RO

NUREG 1123 KA Catalog Rev. 2
295025 EK2.08 3.7/3.7 Reactor/turbine pressure regulating system: Plant-Specific
295025 High Reactor Pressure
295025 EK2. Knowledge of the interrelations between HIGH REACTOR PRESSURE and the following:

10CFR55 RO/SRO Written Exam Content
10 CFR 55.41(b) (7) Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

NRC Exam Usage
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Bank
Higher cognitive level
RO

Associated objective(s):
Governor/Pressure Control (N3012)

Cognitive Enabler

Describe Governor/Pressure Control System response to failure of instrument channels selected for control.

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13	K/A Importance: 3.9 / 4.0			Points: 1.00
R13	Difficulty: 3.00	Level of Knowledge: Higher cognitive level	Source: BANK	87687

The Reactor Core Isolation Cooling (RCIC) system is the only high pressure injection source available.

It is operating and supplying the RPV from the Torus with the following conditions:

- RCIC Turbine supply pressure........900 psig
- RCIC Turbine exhaust pressure........45 psig
- RCIC area ambient temperature........145°F
- RCIC Discharge flow........500 gpm
- Torus Temperature........175°F
- RPV level........10" and stable

(1) What operator action(s) will be taken for these conditions, and (2) what indications will the operator verify / monitor?

A. (1) Arm & Depress the RCIC Manual Isolation Pushbutton.

(2) Verify E5150-F008, RCIC Stm Line Otbd Iso Vlv, closes.

B. (1) Arm & Depress the RCIC Manual Trip Pushbutton.

(2) Verify E5150-F059, RCIC Turb Trip Throttle Vlv, closes.

C. (1) Place RHR in Torus Cooling.

(2) Monitor the RCIC Turbine for signs of damage or unusual operation.

D. (1) Place the RCIC Discharge Flow Controller in Manual, and lower RCIC Turbine speed.

(2) Monitor the RCIC Turbine for signs of damage or unusual operation.

Answer: C

Answer Explanation:
Per BWROG EPGs/SAGs Appendix B Vol 1 Cautions:

The lube oil and control oil for both HPCI and RCIC are cooled by the water being pumped. The maximum allowable cooling water temperature will vary with the particular brand and grade of lube oil being used in the machine. The BWROG performed a generic assessment of RCIC turbine performance under degraded conditions.(73) Threshold temperatures for degraded performance and loss of function are listed in the Caution table.

At Fermi, From EOP Sheet 6, Caution 2, operation of RCIC turbine with suction temperature >140°F causes increased equipment wear, and >170°F may result in equipment damage.

RCIC system trips and isolations are given in 1D94 RCIC Turbine Tripped, and 1D51 RCIC Isolation Trip Signal A. A RCIC trip occurs on any RCIC isolation, or if RCIC Turbine Steam Exhaust pressure exceeds 50 psig. A RCIC isolation occurs if the RCIC room temperature exceeds 154°F based on steam leak detection.

K/A Match Justification:
KA is knowledge of the reason to provide cooling to the Torus water during a high torus water temperature. 29.100.01 Sh 2 is the Fermi EOP that addresses the Torus Temp entry condition. TWT-3 directs the operators to operate all TWT cooling, using only pumps not required for RPV injection. Following this direction, caution 2 is included on the chart on the high torus water temperature leg, and is one of the reasons that torus cooling is provided.

Distractor Explanation:
Distractors are incorrect and plausible because:

A. Plausible because the RCIC area ambient temperature is approaching the RCIC isolation setpoint of 154°F. Incorrect because RCIC operation is required, and RCIC area ambient temperature is still below the isolation setpoint.
B. Plausible because the RCIC exhaust pressure is approaching the RCIC Turbine trip setpoint of 50 psig. Incorrect because RCIC operation is required, and RCIC turbine exhaust pressure is still below the setpoint of 50 psig.
D. Plausible because the actions given would reduce the heat load on the RCIC oil coolers, and thus allow a slightly lower oil temperature, which could extend RCIC equipment life. Incorrect because RCIC operation is required, and the action listed would result in lower RCIC system flow. The second half of the answer choice is correct.

Reference Information:

1D94
1D51
BWROG EPGs/SAGs Appendix B Vol 1 Caution #5

Plant Procedures
29.100.01 SH 6
29.100.01 SH 2

Question Use
Closed Reference
RO

NUREG 1123 KA Catalog Rev. 2
295026 EK3.02 3.9/4 Suppression pool cooling
295026 Suppression Pool High Water Temperature
295026 EK3. Knowledge of the reasons for the following responses as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE:

10CFR55 RO/SRO Written Exam Content
10 CFR 55.41(b)(10) Administrative, normal, abnormal, and emergency operating procedures for the facility.

NRC Exam Usage
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Bank
Higher cognitive level
RO

Associated objective(s):
Containment Systems (T2200 & T2300)
Cognitive Terminal
Given the Containment Systems operating conditions/parameters, evaluate those conditions/parameters for the appropriate operator response in accordance with approved plant procedures.

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14	K/A Importance: 3.2 / 3.4			**Points: 1.00**
R14	Difficulty: 3.00	Level of Knowledge: Higher cognitive level	Source: NEW	87708

A LOCA and ATWS have occurred, with the following plant conditions:

- Drywell pressure is 27 psig
- Drywell temperature is 260°F
- Torus pressure is 20 psig
- Torus temperature is 125°F
- Torus level is 10"
- HPCI is maintaining RPV level at -15"
- RPV pressure is 750 psig

You have been directed to place division 1 of RHR in the Torus Cooling, Torus Spray and Drywell Spray Mode to lower Drywell temperature.

To place Division 1 of RHR into Torus Cooling, Torus Spray and Drywell Spray, the Containment Spray 2/3 Core Height Override keylock switch will ___(1)___.

The maximum flow allowed from Division 1 of RHR is ___(2)___.

A. (1) remain in OFF
(2) 28,000 gpm

B. (1) remain in OFF
(2) 14,000 gpm

C. (1) be placed in MANUAL OVERRIDE
(2) 28,000 gpm

D. (1) be placed in MANUAL OVERRIDE
(2) 14,000 gpm

Answer: C

Answer Explanation:
Per 23.205, Enclosure A RHR Containment Cooling Modes Operation, when the RPV level is < 0", the Containment Spray 2/3 Core Height Override keylock switch is placed in MANUAL OVERRIDE.

Per 23.205, Enclosure A, when using RHR in the Drywell Spray mode, the maximum flow is 14,000 gpm per pump.

The given Torus conditions do not restrict RHR flow to a lower value due to either the Vortex limit or the NPSH limit, which are both 31012 gpm for two pumps.

Distractor Explanation:
Distractors are incorrect and plausible because:

A. Plausible because the 2/3 Core Height interlock prevents an actuation that occurs at -42", and that therefore an examinee could conclude that the override is not yet needed. Incorrect because the procedure directs placing the switch in MANUAL OVERRIDE if placing Drywell Spray into operation when core level is below 0". The second half of the response is correct.
B. Plausible because the 2/3 Core Height interlock prevents an actuation that occurs at -42", and that therefore an examinee could conclude that the override is not yet needed. Incorrect because the procedure directs placing the switch in MANUAL OVERRIDE if placing Drywell Spray into operation when core level is below 0". Also plausible because the limit for one RHR pump operation is 14,000 gpm. Incorrect because there are two RHR pumps available.
D. Plausible because the limit for one RHR pump operation is 14,000 gpm. Incorrect because there are two RHR pumps available.

Reference Information:

23.205
Rx L0 Trip Setpoint from CECO
Sheet 6

ILT Retake Reference Provided:

29.100.01 sheet 6, EOP CURVES, CAUTIONS & TABLES

Question Use
Open Reference
RO

NUREG 1123 KA Catalog Rev. 2
295028 EA1.01 3.8/3.9 Drywell spray: Mark-I&II
295028 High Drywell Temperature
295028 EA1. Ability to operate and/or monitor the following as they apply to HIGH DRYWELL TEMPERATURE :

10CFR55 RO/SRO Written Exam Content
10 CFR 55.41(b)(10) Administrative, normal, abnormal, and emergency operating procedures for the facility.

NRC Exam Usage
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Higher cognitive level
New
RO

Associated objective(s):

Residual Heat Removal (E1100)

Cognitive Terminal

Given the RHR system operating conditions/parameters, evaluate those conditions/parameters for the appropriate operator response in accordance with approved plant procedures.

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15	K/A Importance: 3.7/3.9			**Points: 1.00**
R15	Difficulty: 4.00	Level of Knowledge: Higher cognitive level	Source: BANK: 2018 FERMI NRC EXAM	88567

With a Torus Water Level < -112 inches (1) which of the following Reactor Pressures require EOP action and (2) what action must be taken?

A. (1) 50 PSIG.

(2) Depressurize the RPV by opening 5 SRVs (ADS preferred)

B. (1) 50 PSIG.

(2) Rapidly depressurize the RPV ignoring cooldown rates using the Main Condenser and other steam Loads.

C. (1) 500 PSIG.

(2) Depressurize the RPV by opening 5 SRVs (ADS preferred)

D. (1) 500 PSIG.

(2) Rapidly depressurize the RPV ignoring cooldown rates using the Main Condenser and other steam Loads.

Answer: D

Answer Explanation:
Torus level of -115 inches would be below the level of the SRV T-Quenchers and could possibly pressurize the Torus there per ED-7 SRV is not used and rapidly depressurize the RPV ignoring cooldown rates using another system is required. The need to depressurize is required per the EOPs TWL-5

Distractor Explanation:
50 PSIG is incorrect and plausible because the branch at ED-6 shows that no action is needed if already at 50 PSIG or below
50 psig is a significant number for the candidates to know because it defines the term "Decay Heat Removal Pressure (DHRP)" for Fermi 2. Per the BWROG EPGs, Appendix B, the definition of DHRP is "The lowest differential pressure between the RPV and the suppression chamber at which steam flow through the Minimum Number of SRVs Required for Emergency Depressurization (MNSRED) is sufficient to remove all decay heat from the core." The EPGs then go on to say that "the DHRP is utilized to define the depressurized state of the RPV."
The candidate should recognize that, if RPV pressure is above 50 psig then actions need to be taken to depressurize below the DHRP. If pressure is at or below the DHRP, then additional actions to depressurize are not necessary since, by definition, the plant is depressurized.
5 SRVs is plausible because this is the normal method of emergency depressurization. However per answer explanation this is incorrect.

Plant Procedures
29.100.01 SH 3

Question Use
Closed Reference
RO

NUREG 1123 KA Catalog Rev. 2
295030 EA2.03 Reactor pressure
295030 EA2. Ability to determine and/or interpret the following as they apply to LOW SUPPRESSION POOL WATER LEVEL :

10CFR55 RO/SRO Written Exam Content
10 CFR 55.41(b) (8) Components, capacity, and functions of emergency systems.

NRC Exam Usage
ILO 2018 Exam
ILO 2019 Retake Exam

NRC Question Use (ILO 2018)
BANK
High
RO

NRC Question Use (ILO 2019 Retake)
Bank
Higher cognitive level
RO

Associated objective(s):
Containment Systems (T2200 & T2300)
Cognitive Terminal
Given the Containment Systems operating conditions/parameters, evaluate those conditions/parameters for the appropriate operator response in accordance with approved plant procedures.

16	K/A Importance: 4.6/4.4			Points: 1.00
R16V2	Difficulty: 2.00	Level of Knowledge: Higher cognitive level	Source: NEW	87787

The plant was operating at 100% power when ONE Reactor Feed Pump tripped.

- RPV Water Level is 190”, trending down SLOWLY
- Reactor power is stable at 68%
- Recirc flow is stable at 55% of rated

Which of the following IMMEDIATE ACTIONS must the crew take?

A. Restore RPV level using HPCI.

B. Restore RPV level using SBFW and one RFP.

C. Place the reactor mode switch to SHUTDOWN.

D. Depress the RECIRC MANUAL RUNBACK pushbutton.

Answer: B

Answer Explanation:
Per 20.107.01, LOSS OF FEEDWATER OR FEEDWATER CONTROL, Immediate action IB.1 directs operators to restore RPV WL with SBFW and/or RFPs if adequate pumping capacity is determined to be available. In this case, RPV level is trending down slowly, therefore it can be concluded that "adequate pumping capacity exists" and an attempt should be made to restore level with available feed sources (SBFW and the remaining RFP) in accordance with immediate action IB.1.
Distractor Explanation:
A is incorrect because use of HPCI as an immediate action is not directed by AOP 20.107.01. Applicants may choose this if they incorrectly determine that HPCI should be used to restore level.
C would be correct if it is determined that adequate pumping capacity does not exist. However, in this case, since level is dropping slowly and one RFP and SBFW are available, an immediate attempt to inject with SBFW, and raise injection from the remaining RFP is appropriate. Applicants may choose this if they incorrectly conclude that insufficient injection sources exist.
D is incorrect because a runback of recirc flow has already taken place as indicated by reactor power and recirc flow values. Additionally, conditions that require this action do not exist. Applicants may choose this since manually initiating a runback is required immediately by 20.107.01 if heater drain pumps have failed, and they misapply or incorrectly recall this immediate action.
Reference Information:
20.107.01, LOSS OF FEEDWATER OR FEEDWATER CONTROL

K/A Match justification:
This question matches the selected K/A since RO applicants are required to recall, without reference to procedures, immediate actions resulting from the reactor feed pump trip and resulting low water level.

Question Use
Closed Reference
RO

NUREG 1123 KA Catalog Rev. 2

G2.4.49	Ability to perform without reference to procedures those actions that require immediate operation of system components and controls.
295031	Reactor Low Water Level.

10CFR55 RO/SRO Written Exam Content

10 CFR 55.41(b)(10)	Administrative, normal, abnormal, and emergency operating procedures for the facility.

NRC Exam Usage
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Higher cognitive level
New
RO

Associated objective(s):
Reactor Operator
Performance Enabler
Without reference to procedures, perform AOP immediate actions.

17	K/A Importance: 4.0/4.2			Points: 1.00
R17	Difficulty: 3.00	Level of Knowledge: Higher cognitive level	Source: NEW	89608

You are the Control Room LNO (CRLNO).

The plant has experienced a FAILURE TO SCRAM. Conditions are as follows:

- RPV pressure is stable at 900 psig.
- Standby Liquid Control is injecting.
- SLC tank level is 70” and lowering.

(1) At what SLC tank level will a cooldown be allowed?

While cooling down at 50°F/hr you observe the following:

- Power on IRMs and SRMs begins to rise.
- Period is positive and getting shorter.

(2) How will you proceed?

A. (1) 15”.
(2) Stabilize RPV Pressure at the current value.

B. (1) 15".
(2) Lower cooldown rate to 25°F/hr.

C. (1) 44”.
(2) Stabilize RPV Pressure at the current value.

D. (1) 44”.
(2) Lower cooldown rate to 25°F/hr.

Answer: A

Answer Explanation:
Per 29.100.01, Sheet 1A – RPV Control ATWS step FSP-4, cooldown (depressurization) can commence when the Cold S/D boron weight has been injected into the RPV. Table 16 defines the CSBW as SLC tank level <16". Therefore, the candidate must recognize that the CRS can direct cooldown to commence at SLC tank level of (1) 15".

Per BWROG EPGs/SAGs, Appendix B, RPV Control (page B-6-39) if while executing a cooldown the reactor is not shutdown, the cooldown must be terminated and RPV pressure stabilized. The candidate must recognize that the indications show the cooldown has caused the reactor to NOT be shutdown and determine that (2) RPV pressure must be stabilized to prevent further positive reactivity addition.

Distractor Explanation:
Distractors are incorrect and plausible because:

B. Part (1) is correct. Part (2) is plausible because the candidate could determine that lowering the cooldown rate will have the desired effect by lowering the rate at which positive reactivity is being added to the core due to the cooldown. However, although the reactivity addition rate will be slowed by lowering the cooldown, the EOP charts require suspending the cooldown by stabilizing RPV pressure, thus making this distractor incorrect.
C. Part (1) is plausible because Hot Shutdown Boron Weight has been injected when SLC tank level drops <45". However, Cooldown is commenced when the CSBW is injected and NOT the HSBW. Part (2) is correct.
D. Part (1) is plausible because Hot Shutdown Boron Weight has been injected when SLC tank level drops <45". However, Cooldown is commenced when the CSBW is injected and NOT the HSBW. Part (2) is plausible because the candidate could determine that lowering the cooldown rate will have the desired effect by lowering the rate at which positive reactivity is being added to the core due to the cooldown. However, although the reactivity addition rate will be slowed by lowering the cooldown, the EOP charts require suspending the cooldown by stabilizing RPV pressure, thus making this distractor incorrect.

Reference Information:

29.100.01, Sheet 1A – RPV Control ATWS.
BWROG EPGs/SAGs, Appendix B.

Question Use
Closed Reference
RO

NUREG 1123 KA Catalog Rev. 2
295037 EK1.06 4.0*/4.2* Cooldown effects on reactor power

295037	SCRAM Condition Present and Reactor Power Above APRM Downscale or unknown.
295037 EK1.	Knowledge of the operational implications of the following concepts as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN:

10CFR55 RO/SRO Written Exam Content

10 CFR 55.41(b)(10)	Administrative, normal, abnormal, and emergency operating procedures for the facility.

NRC Exam Usage
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Higher cognitive level
New
RO

Associated objective(s):

18	K/A Importance: 3.6/3.8			**Points: 1.00**
R18	Difficulty: 2.00	Level of Knowledge: Fundamental	Source: BANK	87768

Following an unisolable steam leak in the Turbine Building, the following conditions exist:

- An Offsite Radiation Release is in progress.
- 29.100.01 Sheet 5, Radioactivity Release has been entered.
- Turbine Building Ventilation has TRIPPED and ISOLATED.

Which of the following actions, if any, are required and why?

A. Maintain Turbine Building Ventilation ISOLATED to lower the radioactivity released.

B. Maintain Turbine Building Ventilation ISOLATED to prevent an unmonitored release.

C. Defeat the isolation and RESTART Turbine Building Ventilation to ensure the release is monitored.

D. Defeat the isolation and RESTART Turbine Building Ventilation to lower the radioactivity release rate.

Answer: C

Answer Explanation:
Per 29.100.01, Sheet 5, TBHVAC is restarted per RR-OR1 by defeating interlocks as necessary per 29.ESP.25, which defeats the high radiation trip/isolation. The candidate must recall that the Turbine Building is NOT an airtight structure, and radioactivity release inside the building would lead to an unmonitored ground level release. Operating TBHVAC is allowed because it discharges radioactivity through an elevated, monitored release point.

Distractor Explanation:
Distractors are incorrect and plausible because:
A. This distractor is true for Reactor Building HVAC, which automatically isolates when High Radiation is detected, and the candidate could assume that maintaining TBHVAC shut down and isolated is required.
B. This distractor is true for Reactor Building HVAC, which is exhausted through Standby Gas Treatment and discharged to a monitored stack when High Radiation is detected.
D. The candidate could assume that dilution of the Turbine Building would lower the specific activity released, however, the reason TBHVAC is restarted is to discharge from an elevated and monitored release path and not to lower activity release.

Reference Information:

29.100.01, Sheet 5, Secondary Containment Control and Radiation Release.

Question Use
Closed Reference
RO

NUREG 1123 KA Catalog Rev. 2
295038 EK2.03 3.6/3.8 Plant ventilation systems
295038 High Off-Site Release Rate
295038 EK2. Knowledge of the interrelations between HIGH OFF-SITE RELEASE RATE and the following:

10CFR55 RO/SRO Written Exam Content
10 CFR 55.41(b)(10) Administrative, normal, abnormal, and emergency operating procedures for the facility.

NRC Exam Usage
ILO 2009 Exam
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Bank
Fundamental
RO

Associated objective(s):
Reactor Operator
    Performance Enabler
    Perform other EOP / AOP actions per site procedures as directed.

19	K/A Importance: 2.8/3.4			**Points: 1.00**
R19V3	Difficulty: 2.00	Level of Knowledge: Fundamental	Source: NEW	89127

The plant is operating at 100% power when the following alarms/indications are received:

- 16D27, Fire Alarm.
- White FIRE light is lit for Zone 5, RCIC Quad on H11-P816.
- 7D6, Diesel Fire Pump Auto Start.
- 7D10, Electric Fire Pump Auto Start.

Which of the following actions needs to be performed FIRST and WHY?

A. An operator needs to be dispatched to investigate because indications of a confirmed fire have NOT been received.

B. 20.000.18, Control of the Plant from the Dedicated Shutdown Panel, must be performed because equipment malfunctions, due to hot shorts, may jeopardize safe operation of the plant.

C. 2PA2-6 Ckt 10, HPCI Valve Logic, must be opened because HPCI is the preferred HP feed source and isolation logic is de-energized to prevent spurious isolations.

D. The Fire Alarm must be sounded and the fire announced because plant personnel, including the fire brigade, need to be alerted.

Answer: D

Answer Explanation:
The candidate must evaluate the alarms and attempt to determine if they indicate a confirmed fire or not. They must recall that the BASES for AOP 20.000.22, Plant Fires, states that "a confirmed fire is a fire that is either reported in person, or secondary indications received such as fire pump auto start." Therefore, the candidate must conclude that a confirmed fire exists from the indications provided in the stem of the question.

Next, the candidate must recall the bases for the Immediate Actions taken in the Plant Fires AOP. Among other things, the candidate must recall that It is necessary to sound the fire alarm and announce the event over the Hi-Com to alert plant personnel, which includes fire brigade members, to the potential hazards in the area. This information is located in 20.000.22, Plant Fires BASES document as the basis for Actions IA.1 - IA.4.

Distractor Explanation:
Distractors are incorrect and plausible because:

A. The bases given in this distractor are the bases for Subsequent Action A.1 of 20.000.02 to Dispatch operator to investigate. This basis is incorrect because this action is only necessary for a fire alarm received with NO confirmation. Since the indications in the stem of the question are consistent with indications of a confirmed fire, as spelled out in the BASES document for 20.000.02, this action is not necessary so this distractor is incorrect.
B. The bases given in this distractor are the bases for Subsequent Action C.1 of 20.000.02 to Perform 20.000.18, Control of the Plant from the Dedicated Shutdown Panel. This basis is incorrect because this action is only necessary for a confirmed fire in one of the 3L Zones (8, 9A, 11, 12, 12A, 14 or 16). Since the stem of the question indicates a fire in Zone 5, RCIC Quad, this action is not necessary so this distractor is incorrect.
C. The bases given in this distractor are the bases for Subsequent Action D.1 of 20.000.02 to (in summary) take actions to isolate RCIC and protect HPCI as the preferred high pressure injection source. This distractor is incorrect because, although the fire is in the RCIC Quad and HPCI will therefore be the preferred injection source, the action that must be taken next are IA.1 through IA.4 before actions to protect HPCI are taken.

Reference Information:
20.000.22, Plant Fires AOP
20.000.22 Bases - Plant Fires AOP BASES

Question Use
Closed Reference
RO

NUREG 1123 KA Catalog Rev. 2

600000 AK3.04	Actions contained in the abnormal procedure for plant fire on site
600000	Plant Fire On Site
600000 AK3.	Knowledge of the reasons for the following responses as they apply to PLANT FIRE ON SITE:

10CFR55 RO/SRO Written Exam Content

10 CFR 55.41(b)(10)	Administrative, normal, abnormal, and emergency operating procedures for the facility.

NRC Exam Usage
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Fundamental
New
NRC Early Review
RO

Associated objective(s):

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20	K/A Importance: 4.1 / 4.1			**Points: 1.00**
R20	Difficulty: 2.00	Level of Knowledge: Higher cognitive level	Source: BANK	87748

The plant is operating at 80% power during a rod pattern adjustment. The Systems Operations Center notifies the main control room that there is potential for degraded grid conditions due to erratic loading in Monroe County.

The following conditions exist for the Fermi 2 Main Generator:

- S13-R800, MTG XFMR Output Voltage -- 119.5 Meter Volts
- S13-R809 MTG Frequency Recorder -- 57.70 Hertz

Which of the following operator actions is required and why?

A. Reduce generator load to raise frequency.

B. Raise generator voltage to stabilize the grid.

C. Start CTG 11-1 to minimize the impact of a potential loss of the 120Kv Mat.

D. Place the Mode Switch in SHUTDOWN to protect the plant from conditions that might damage the Main Generator.

Answer: D

Answer Explanation:
With generator frequency less than 57.8Hz and power greater than 30% (conditions which could damage the generator), 20.300.GRID, Override 1, directs placing the mode switch in Shutdown to protect the plant.

Distractor Explanation:

A. Plausible because low frequency indications on a generator can generally be solved by reducing the system load to allow the frequency to recover to the setpoint. Incorrect because Fermi operations may not take independent action to stabilize the grid. SOC/ITC must coordinate those actions.
B. Plausible because an examinee may interpret that a rise in voltage will help stabilize the grid and help with the frequency issue. Incorrect because raising voltage will not automatically correct a frequency issue, and the action to place the Reactor Mode Switch in Shutdown would take priority.
C. Plausible because these actions are directed for low voltage conditions within 20.300.GRID. Incorrect because the reason for entry into 20.300.GRID is low frequency, not low voltage, and the action to place the Reactor Mode Switch in Shutdown would take priority.

RO Question justification:
This is not an SRO question because it can be answered solely by knowing entry conditions for AOPs or plant parameters that require direct entry into major EOPs. Asking questions that would test the detail of a step within 20.300.GRID that does not require a Reactor Scram may prevent this KA from being tested at the RO level.

Reference Information:
20.300.GRID

Plant Procedures
20.300.GRID
20.300.GRID BASES

Question Use
Closed Reference
RO

NUREG 1123 KA Catalog Rev. 2
700000 AA1.04 4.1/4.1 Reactor Controls
700000 Generator Voltage and Electric Grid Disturbances
700000 AA1 Ability to operate and/or monitor the following as they apply to GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES:

10CFR55 RO/SRO Written Exam Content
10 CFR 55.41(b)(10) Administrative, normal, abnormal, and emergency operating procedures for the facility.

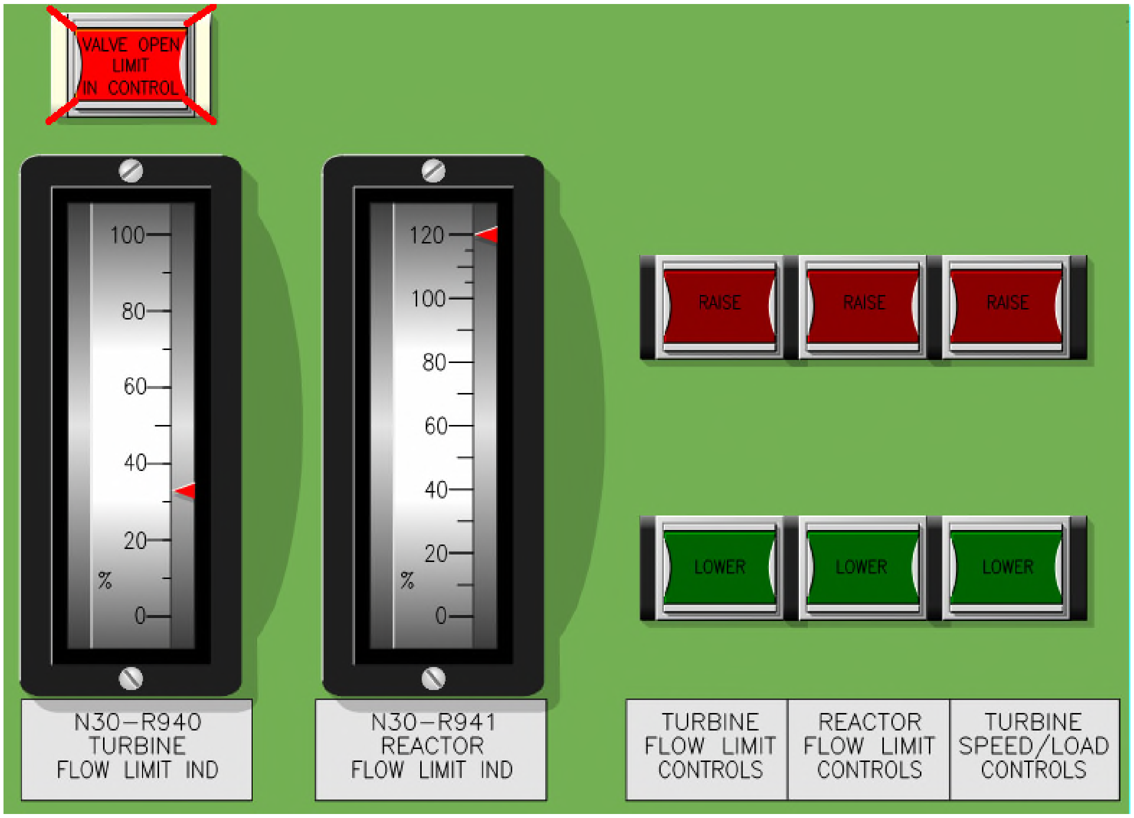
NRC Exam Usage
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Bank
Higher cognitive level
RO

Associated objective(s):
Reactor Operator
Performance Enabler
Perform other EOP / AOP actions per site procedures as directed.

21	K/A Importance: 2.7 / 2.8			**Points: 1.00**
R21	Difficulty: 3.00	Level of Knowledge: Higher cognitive level	Source: NEW	87767

The plant was initially at 60% power. A malfunction occured, as shown below:



Additionally:

- Generator output lowers to 350 MWe
- Turbine bypass valves (TBVs) open to 100%
- RPV pressure rises and stabilizes at 1055 psig
- Reactor power is stable at 62%

The Crew attempts to raise the TURBINE FLOW LIMIT. There is no response.

What action will the crew take?

A. trip the main turbine

B. lower Reactor Recirc flow

C. throttle closed on the TBVs

D. manually open A or G SRV

Answer: B

Answer Explanation:

The failure indicated is a failure of the Turbine Governor System, the Turbine Flow limit setpoint has failed to 1/3 scale.

From the conditions given in the stem, there is no indicated failure of the Reactor Pressure Controller. The turbine valves have closed consistent with the "VALVE OPEN LIMIT IN CONTROL" and the TBVs have opened fully as expected for the given power and conditions.

The given pressure is above the TS value of 1045 psig, and also above the alarm setpoint for 3D168.

The applicable Initial Response actions taken per 3D168 are as follows:
1. Verify Reactor Pressure greater than 1045 psig on the following:
•C32-R609, Reactor Pressure Recorder.
•C32-R605A, Div 1 RPV Pressure Ind.
•C32-R605B, Div 2 RPV Pressure Ind.
•C32-K816, FW & RR Flat Panel Display.
2. IF Reactor Pressure has reached 1093 psig, ENTER 20.000.21, "Reactor Scram."
3. Verify Pressure Regulator setpoint is between 944 and 949 psig.
4. Verify Reactor Flow Limiter is set correctly for current plant conditions.
5. IF Pressure Regulator fault or failure has occurred, PERFORM 20.109.02, "Reactor Pressure Controller Failure," concurrently with this procedure.

Subsequent actions are as follows:
1. Within 15 minutes, reduce Reactor Pressure to less than 1045 psig.
2. **IF Reactor Pressure cannot be restored to less than 1045 psig with minor flow adjustments within 15 minutes, PERFORM rapid power reduction per 23.623, Enclosure I (hard card).**
3. Comply with Technical Specifications, Section 3.4.11, Reactor Steam Dome Pressure.

- Pressure is > 1045.
- Pressure has not reached 1093 psig, therefore no scram is required.
- No failure is indicated for PR setpoint, and the given indications are not consistent with a PR setpoint failure.
- The reactor flow limiter is set correctly for current plant conditions.
- No pressure regulator fault or failure is indicated.
- Direction is therefore to reduce Reactor Pressure to less than 1045 within 15 minutes.
- The method directed to reduce Reactor Pressure to less than 1045 is to use Reactor Recirc flow adjustments.

<u>Distractor Explanation:</u>
A. Plausible because the turbine governor control system has a fault, and tripping the turbine would stabilize the system. Incorrect because tripping the turbine at this point would result in an automatic reactor scram, and it is contrary to operations practice to cause an automatic protective action to occur.
C. Plausible because the TBVs are not normally open at a reactor power of 60% and it may be interpreted that a TBV control failure has occurred. Incorrect because the TBVs are operating properly and should remain in automatic control.
D. Plausible because opening an SRV would cause a reduction in reactor pressure. Incorrect because it does not solve the problem, and simply adds heat to the torus, which will require additional actions to remove.

<u>10 CFR 55.41(b)(10) RO Justification:</u>
This question does NOT meets ES-401 Attachment 2 requirements to be SRO-Only because the question can be answered solely by knowing "systems knowledge". With the given indications, reactor power is too high for the available steam flow paths, but can be reduced with RR flow to be within the capability of the Main Turbine and TBVs.

Reference Information:

3D168

Question Use
Closed Reference
RO

NUREG 1123 KA Catalog Rev. 2
295007 AK1.04 2.7/2.8 Turbine load
295007 AK1. Knowledge of the operational implications of the following concepts as they apply to HIGH REACTOR PRESSURE:

10CFR55 RO/SRO Written Exam Content
10 CFR 55.41(b)(10) Administrative, normal, abnormal, and emergency operating procedures for the facility.

NRC Exam Usage
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Higher cognitive level
New
RO

Associated objective(s):
Governor/Pressure Control (N3012)
Cognitive Enabler
Identify Governor/Pressure Control System related technical specifications, with emphasis on action statements requiring prompt actions (for example, one hour or less).

22	K/A Importance: 3.2 / 3.3			**Points: 1.00**
R22-ALT	Difficulty: 3.00	Level of Knowledge: Fundamental	Source: NEW	87788

The plant has scrammed due to a LOCA. You have been directed to Spray the Drywell.

(1) What is the MINIMUM Torus Water Level above which initiation of Drywell Spray is prohibited?
(2) Why is Drywell Spray prohibited above this level?

A. (1) 569 feet as read on T50-R810, Div 2 Primary Containment Water Level Recorder.

(2) The Suppression Chamber to Drywell Vacuum Breakers are being submerged.

B. (1) 569 feet as read on T50-R810, Div 2 Primary Containment Water Level Recorder.

(2) Water has reached the elevation of the Torus Vent.

C. (1) 45 inches as read on T50-R804A(B) Div 1(2) Torus Level Recorder.

(2) The Suppression Chamber to Drywell Vacuum Breakers are being submerged.

D. (1) 45 inches as read on T50-R804A(B) Div 1(2) Torus Level Recorder.

(2) Water has reached the elevation of the Torus Vent.

Answer: C

Answer Explanation:
Per the BWROG EPGs / SAGs, appendix B, Vol 1, SP/L-3.2:

Maintain suppression pool water level below [17 ft. 2 in. (elevation of bottom of suppression chamber to drywell vacuum breaker openings [less vacuum breaker opening pressure in feet of water])]. If suppression pool water level cannot be maintained below [17 ft. 2 in. (elevation of bottom of suppression chamber to drywell vacuum breaker openings [less vacuum breaker opening pressure in feet of water])]:
• Terminate drywell sprays.
• If adequate core cooling is assured, terminate injection into the RPV from sources external to the primary containment except from systems required to shut down the reactor.

Step SP/L-3.2 applies only to plants with suppression chamber-to-drywell vacuum breaker penetrations significantly below the top of the suppression chamber (i.e., those in which a significant volume of noncondensibles could be trapped if the suppression chamber is flooded). If the penetrations are submerged, the vacuum breakers cannot function as designed to relieve noncondensibles into the drywell and equalize drywell and suppression chamber pressures. Suppression pool water level must therefore be maintained below the bottom of the vacuum breaker openings to permit initiation and operation of drywell sprays.

Per Sheet 2, DWT-7, PCP-8, and TWL-12, at Fermi this level corresponds to +45 inches in the Torus.

Distractor Explanation:
Distractors are incorrect and plausible because:
A. (1) 569' is plausible because it is the highest level at which Torus Spray, which is closely related to Drywell Spray, may be initiated as per Sheet 2. Drywell spray will already have been terminated by this point, so this is not the LOWEST level at which it would be required to terminate Drywell Spray. The candidate who fails to correctly recall the elevation of the Suppression Chamber to Drywell Vacuum Breakers may choose this distractor. (2) Is the correct response for +45" This distractor is incorrect because Drywell Spray may continue until +45".
B. (1) 569' is plausible because it is the highest level at which Torus Spray, which is closely related to Drywell Spray, may be initiated as per Sheet 2. Drywell spray will already have been terminated by this point, so this is not the LOWEST level at which it would be required to terminate Drywell Spray. The candidate who fails to correctly recall the elevation of the Suppression Chamber to Drywell Vacuum Breakers may choose this distractor. (2) Is the correct response for 569' because this is the elevation of the highest Torus Vent and therefore the reason why Torus Spray is prohibited. However, this distractor is incorrect because Drywell Spray would have been terminated at 45" and it is Torus Spray that is terminated at this elevation for this reason.
D. (1) +45" is the highest torus water level that allows the use of Drywell Spray. (2) Is the reason for terminating Torus Spray (not Drywell Spray) at 569' because this is the elevation of the highest Torus Vent and therefore the reason why Torus Spray is prohibited. However, this distractor is incorrect because Drywell Spray is terminated because the vacuum breakers are becoming submerged and not the Torus Vent.

10 CFR 55.41(b)(10) RO Justification:
This question DOES NOT meet ES-401 Attachment 2 requirements to be SRO-Only because it can be answered with systems knowledge (how the system works) by knowing the elevation of the Torus to Drywell vacuum breakers and that covering these with water could result in failure.

Reference Information:

BWROG EPGs / SAGs
Sheet 2
Sheet 6

Question Use
Closed Reference
RO

NUREG 1123 KA Catalog Rev. 2
295010 AK2.01 3.2/3.3 Suppression pool level
295010 High Drywell Pressure
295010 AK2 Knowledge of the interrelations between HIGH DRYWELL PRESSURE and the following:

10CFR55 RO/SRO Written Exam Content
10 CFR 55.41(b)(10) Administrative, normal, abnormal, and emergency operating procedures for the facility.

NRC Exam Usage
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Fundamental
New
RO

Associated objective(s):
Containment Systems (T2200 & T2300)
Cognitive Enabler
Describe the characteristics and locations of major Containment Systems components.

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23	K/A Importance: 3.7 / 3.7			**Points: 1.00**
R23	Difficulty: 3.00	Level of Knowledge: Fundamental	Source: NEW	88428

Power ascension is in progress.

Reactor power is 80%.

The EARLIEST the Rod Block Monitor system will prevent control rod withdrawal is when RBM A __(1)__ RBM B reach the trip setpoint.

The basis for this rod block is __(2)__.

A. (1) OR

(2) to terminate an inadvertant rod withdrawal

B. (1) OR

(2) to mitigate the consequences of a dropped rod event

C. (1) AND

(2) to terminate an inadvertant rod withdrawal

D. (1) AND

(2) to mitigate the consequences of a dropped rod event

Answer: A

Answer Explanation:
Per TS Bases, During high power operation, the rod block monitor (RBM) provides protection for control rod withdrawal error events. During low power operations, control rod blocks from the rod worth minimizer (RWM) enforce specific control rod sequences designed to mitigate the consequences of the control rod drop accident (CRDA).

The RBM has two channels, either of which can initiate a control rod block when the channel output exceeds the control rod block setpoint.

Distractor Explanation:
B. The first half is correct. The second half is plausible because this is the reason for the Rod Worth Minimizer's rod block, which is often confused with the function of the Rod Block Monitor.
C. This is plausible because the RMCS is not part of the safety system, and a examinee may incorrectly determine that the RPS will provide the protection in this situation, and therefore a 2 out of 2 scheme is allowed. The second half is correct.
D. This is plausible because the RMCS is not part of the safety system, and a examinee may incorrectly determine that the RPS will provide the protection in this situation, and therefore a 2 out of 2 scheme is allowed. The second half is plausible because this is the reason for the Rod Worth Minimizer's rod block, which is often confused with the function of the Rod Block Monitor.

Reference Information:

TS 3.3.2.1
TS 3.3.2.1 Bases

Question Use
Closed Reference
RO

NUREG 1123 KA Catalog Rev. 2
295014 AK3.02 3.7/3.7 Control rod blocks
295014 Inadvertent Reactivity Addition
295014 AK3 Knowledge of the reasons for the following responses as they apply to INADVERTENT REACTIVITY ADDITION:

10CFR55 RO/SRO Written Exam Content
10 CFR 55.41(b) (7) Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

NRC Exam Usage
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Fundamental
New
RO

Associated objective(s):
Power Range Neutron Monitoring (C5112, C5113 & C5114)
Cognitive Enabler
State the purpose of the Power Range Neutron Monitoring System.

24	K/A Importance: 3.6/3.8			**Points: 1.00**
R24	Difficulty: 2.00	Level of Knowledge: Fundamental	Source: NEW	87789

The plant was operating at 100% power when an ATWS occurred.

- The CRS has directed 29.ESP.03, ALTERNATE CONTROL ROD INSERTION METHODS
- The P603 operator is preparing to manually insert control rods per section 3.0

Which one of the following methods must be used by the P603 operator to manually insert control rods initially?

Insert rods...

A. in a spiral pattern using the “Rod Movement Control” switch.

B. in a spiral pattern using the “Rod Out Notch Override” switch.

C. from the cram array using the “Rod Movement Control” switch.

D. from the cram array using the “Rod Out Notch Override” switch.

Answer: D

Answer Explanation:
29.ESP.03 section 3.0 directs operators to insert rods using the RONOR switch in “emergency in” following the sequence specified in the cram array.

Distractor Explanation:

A is incorrect since the normal rod movement control switch should not be used and the rods should be inserted first using the cram array, not a spiral pattern. A spiral pattern is not directed until after the cram array has been fully inserted. Applicants may choose this if they incorrectly believe the normal rod movement switch should be used and rods initially inserted in a spiral pattern.

B is incorrect since rods should be inserted first using the cram array, not a spiral pattern. A spiral pattern is not directed until after the cram array has been fully inserted. Applicants may choose this if they incorrectly believe rods should initially be inserted in a spiral pattern

C is incorrect since the normal rod movement control switch should not be used. Applicants may choose this if they incorrectly believe the normal rod movement switch should be used.

Reference Information:

29.ESP.03, ALTERNATE CONTROL ROD INSERTION METHODS

K/A Justification:

This question matches the specified K/A since RO applicants are asked to determine which RMCS switch should be used to insert control rods during an incomplete scram situation without reference to procedures.

Question Use
Closed Reference
RO

NUREG 1123 KA Catalog Rev. 2
295015 AA1.03 3.6/3.8 RMCS: Plant-Specific
295015 Incomplete SCRAM
295015 AA1 Ability to operate and/or monitor the following as they apply to INCOMPLETE SCRAM:

10CFR55 RO/SRO Written Exam Content
10 CFR 55.41(b)(10) Administrative, normal, abnormal, and emergency operating procedures for the facility.

NRC Exam Usage
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Fundamental
New
RO

Associated objective(s):
EOP task training
Performance Enabler
Insert rods IAW 29.ESP.03

25	K/A Importance: 3.6 / 3.6			**Points: 1.00**
R25	Difficulty: 3.00	Level of Knowledge: Higher cognitive level	Source: NEW	88172

The plant is at 100% power.

A rupture of the instrument line on the VARIABLE leg associated with the below level instruments occurs.

- B21-N080A, Div 1 Reactor Level Narrow Range Transmitter
- B21-N080B, Div 1 Reactor Level Narrow Range Transmitter
- B21-N095A, Div 1 Reactor Level Narrow Range Transmitter
- B21-N095C, Div 1 Reactor Level Narrow Range Transmitter

The associated excess flow check valve closed and the instrument line depressurized.

Of the following automatic actions, which should occur based on this event?

- Trip of Main Turbine and Reactor Feed Pumps
- Group 13: Drywell Sump Pump Isolation
- Trip of the North and South Reactor Recirc Pumps

A. Group 13: Drywell Sump Pump Isolation ONLY

B. Trip of Main Turbine and Reactor Feed Pumps ONLY

C. Trip of the North and South Reactor Recirc Pumps ONLY

D. Group 13: Drywell Sump Pump Isolation AND Trip of the North and South Reactor Recirc Pumps

Answer: A

Answer Explanation:
Per GFES, when the variable leg is depressurized, the instrument fails to a low level. For a NR instrument, it fails to the bottom of scale, which will trip the L3 function.

Per 23.601, page 11, a trip of B21-N080A and B will cause a Group 13 isolation.

Per 23.601, page 22, a high level trip of B21-N095A and C will cause a Main Turbine trip as well as a North and South RFP trip.

Per 23.601, page 17 a low (L2) trip of B21-N091A and C will cause a trip of the RRMG sets.

In the given scenario, only NR instruments are affected, so the L2 trip of the RRMGs will not occur. Additionally, since the failure is a downscale failure, the L8 trip of the Main Turbine, N and S RFPs will also not occur.

Distractor Explanation:
Distractors are incorrect and plausible because:

B. Plausible if the examinee incorrectly determines that indicated level rises after a failure of the variable leg. Incorrect because indicated level lowers after a failure of the variable leg.
C. Plausible if the examinee incorrectly determines that the trip of the RRMGs is at L3, and also incorrectly determines that the Drywell Sump Pump isolation is at L2. Incorrect because the trip of the RRMGs is at L2, and the trip of the Drywell Sump Pumps is at L3.
D. Plausible if the examinee incorrectly determines that the NR instrument includes the L2 functions. Incorrect because the NR instrument includes the L3 and L8 trips only.

Reference Information:

23.601

Question Use
Closed Reference
RO

NUREG 1123 KA Catalog Rev. 2
295020 AA2.05 3.6/3.6 Reactor water level
295020 Inadvertent Containment Isolation
295020 AA2 Ability to determine and/or interpret the following as they apply to INADVERTENT CONTAINMENT ISOLATION:

10CFR55 RO/SRO Written Exam Content
10 CFR 55.41(b) (7) Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

NRC Exam Usage
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Higher cognitive level
New
RO

Associated objective(s):
Primary Containment Isolation System

Cognitive Enabler

Given various system operating parameters, relate system/equipment operation to fundamental concepts to determine proper operation/response as described in the BWR Fundamentals Catalog.

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26	K/A Importance: 3.8/4.3			**Points: 1.00**
R26	Difficulty: 2.00	Level of Knowledge: Fundamental	Source: NEW	87810

The plant has experienced an ATWS with significant fuel damage.
Which one of the following activities contains a caution that, when performed under these conditions, could release radioactive steam into secondary containment and significantly increase secondary containment radiation levels?

A. Venting the scram air header

B. Injecting into the RPV using RCIC

C. Venting the control rod over piston volume

D. Bypassing isolations and re-opening MSIVs

Answer: C

Answer Explanation:
29.100.01 sheet 1A, RPV Control - ATWS specifies that caution #7 should be observed while venting CRD over piston volumes per step FSQ-11. Caution #7 on sheet 6 states: “Executing the following may release pressurized radioactive steam or water from the RPV”. Since this activity is conducted within secondary containment, higher than normal rad levels may be expected under ATWS conditions with fuel damage.

Distractor Explanation:

A is incorrect since the scram air header will not contain radioactive steam. Applicants may choose this since this procedure does caution operators that higher than normal rad levels may occur and they incorrectly conclude that a release of radioactive steam is associated with this caution.

B is incorrect. Although this activity has a caution associated, it reminds operators that injecting cold water into a critical reactor can raise reactor power and has nothing to do with the potential for steam release and SC rad levels. Applicants may choose this if they incorrectly associate the wrong caution with this action.

D is incorrect since re-opening MSIVs would not be permitted if indications of a steam line rupture exist. Therefore, no formal procedural caution exists for this procedure. Applicants may choose this if they believe that re-opening MSIVs would be permitted if the potential for a steam leak exists.

Reference Information:

29.100.01 sheet 1A, RPV Control – ATWS
29.100.01 sheet 6, Curves, Cautions & Tables

K/A Justification:

This question matches the selected K/A since RO applicants must recall EOP cautions associated with high secondary containment rad levels.

Question Use
Closed Reference
RO

NUREG 1123 KA Catalog Rev. 2

G2.4.20	Knowledge of operational implications of EOP warnings, cautions, and notes.
295033	High Secondary Containment Area Radiation Levels

10CFR55 RO/SRO Written Exam Content

10 CFR 55.41(b)(10)	Administrative, normal, abnormal, and emergency operating procedures for the facility.

NRC Exam Usage
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Fundamental
New
RO

Associated objective(s):
Control Rod Drive Hydraulics (C1150)
Cognitive Enabler
Describe any industrial safety precautions associated with the Control Rod Drive Hydraulic system.

27	K/A Importance: 3.9 / 4.2			**Points: 1.00**
R27	Difficulty: 3.00	Level of Knowledge: Fundamental	Source: BANK	87807

If 8D46 DIV 1 REACTOR BLDG PRESSURE HIGH/LOW alarms and pressure is +1" wc, why is 29.100.01 SH 5 "Secondary Containment and Rad Release" entered?

A. A primary system is discharging into the Secondary Containment.

B. An indication that radioactivity will not be treated or monitored prior to release exists.

C. The continued operability of equipment needed to stabilize the plant may be compromised.

D. Radioactivity is being released to the environment when the system should have automatically isolated.

Answer: B

Answer Explanation:
Per BWROG EPGs / SAGs, Appendix B, vol 1, SC entry conditions:

A high secondary containment differential is indicative of a potential loss of reactor building structural integrity and could result in uncontrolled release of radioactivity to the environment.

Distractor Explanation:

A. Plausible because this is the basis for High Temperature above the Max Normal Operating Level, and because large steam releases often cause high secondary containment pressure. Incorrect because other conditions exist which cause high RB pressure to exist, and there is separate entry into sheet 5 due to steam leak into secondary containment.

C. Plausible because this is the basis for High Water Levels above the Max Normal Operating Limit. Incorrect because d/p alone does not jeopardize any equipment require to stabilize the plant.

D. Plausible because this is the basis for the High HVAC exhaust radiation entry into 29.100.01 Sh 5, and because the high RB pressure could cause an unmonitored radiation release. Incorrect because RBHVAC does not trip until +2.5" WC, and the automatic isolation are on the HVAC system, which is a monitored release, not on the unmonitored leakage pathway.

Reference:
BWROG EPGs / SAGs

Plant Procedures
BWROG EPGs Volume 2

Question Use
Closed Reference
RO

NUREG 1123 KA Catalog Rev. 2
295035 EK1.01 3.9/4.2* Secondary containment integrity
295035 Secondary Containment High Differential Pressure
295035 EK1 Knowledge of the operational implications of the following concepts as they apply to SECONDARY CONTAINMENT HIGH DIFFERENTIAL PRESSURE :

10CFR55 RO/SRO Written Exam Content
10 CFR 55.41(b)(10) Administrative, normal, abnormal, and emergency operating procedures for the facility.

NRC Exam Usage
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Bank
Fundamental
RO

Associated objective(s):
Secondary Containment Control and Radioactive Release
Performance Terminal
List the conditions requiring entry or re-entry into 29.100.01 Sh 5, Secondary Containment Control and Radioactive Release.

28	K/A Importance: 3.1 / 3.3			**Points: 1.00**
R28	Difficulty: 4.00	Level of Knowledge: Higher cognitive level	Source: BANK: 2017 FERMI NRC EXAM	87808

The plant is operating at 100% power with RHR Pump "A" operating in Torus Cooling. A transient occurs, resulting in the following conditions:

- Control Rods .... All fully inserted
- MSIVs .... All closed
- 1D6, DIV I CSS LOGIC POWER FAILURE .... Alarming
- 1D8, RHR LOGIC A 125V DC BUS POWER FAILURE .... Alarming
- 9D17, DIV I ESS 130V BATTERY TROUBLE .... Alarming
- ALL Div. 1 ECCS Pumps .... CMC indication lost (no lights)

Immediately after the above conditions are evident, but before operator action, a LOCA occurs.

Based on these conditions, RHR Pump "A" is __(1)__ and ____(2)____.

A. (1) RUNNING
(2) CAN be tripped from the MCR

B. (1) RUNNING
(2) CANNOT be tripped from the MCR

C. (1) TRIPPED
(2) WILL auto start on receipt of a valid signal

D. (1) TRIPPED
(2) WILL NOT auto start on receipt of a valid signal

Answer: B

Answer Explanation:
From 1D8, RHR LOGIC A 125V DC BUS POWER FAILURE:

CAUTION

Div 1 RHR equipment may be manually operated, but all automatic start, stop, valve position, isolation, pressure, pump safety interlocks, and inputs to other systems will not be active.

RHR Pump "A" has lost control power, indicated by the DC Annunciators; therefore RHR Pump "A" will continue to run and cannot be electrically tripped. Control power is required for both CMC indication and trip function.

Distractor Explanation:

A. Plausible because an examinee could incorrectly conclude that the trip coil is powered by an AC power source, or misinterpret the given indications and fail to conclude that DC power is lost. THe first half is correct. Incorrect because DC power is lost to the breaker cubicle, and this power is required to trip the pump.
C. Plausible if the candidate believes that there is an undervoltage coil in the breaker that can cause a breaker trip without control power. The second half would be plausible if the candidate misinterprets the given conditions and believes that control power is still available. Incorrect because DC power is required to start **OR** stop an RHR pump.
D. Plausible if the candidate believes that there is an undervoltage coil in the breaker that can cause a breaker trip without control power. The second half is correct. Incorrect because DC power is required to start **OR** stop an RHR pump.

Reference Information:
ARP 1D8 (pg 1)

Plant Procedures
01D08

Question Use
Closed Reference
RO

NUREG 1123 KA Catalog Rev. 2
203000 K1.07 3.1/3.3 D.C. electrical power
203000 RHR/LPCI: Injection Mode
203000 K1. Knowledge of the physical connections and/or causeeffect relationships between RHR/LPCI: INJECTION MODE (PLANT SPECIFIC) and the following:

10CFR55 RO/SRO Written Exam Content
10 CFR 55.41(b) (7) Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

NRC Exam Usage
LOR 2009 Exam
LOR 2015 Exam
ILO 2017 Exam
LOR 2019 Exam
ILO 2019 Retake Exam

I-Drawings
I-2201-01

NRC Question Use (ILO 2017):
Bank
RO
High

NRC Question Use (ILO 2019 Retake)
Bank
Higher cognitive level
RO

Associated objective(s):
DC Electrical Distribution (R3200 & S3102)
Cognitive Enabler
Discuss the DC Electrical Distribution System interrelationships with other systems.

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29	K/A Importance: 3.9 / 4.6			**Points: 1.00**
R29 - ALT	Difficulty: 3.00	Level of Knowledge: Fundamental	Source: NEW	89087

Initial conditions:

- The plant is at 100% power.
- LCO 3.5.1 Condition A was entered due to RHR pump A out of service for a planned outage.

Event:

- RHR Pump B has just been declared inoperable.
- LCO 3.5.1 Condition B was entered as a result.

Which of the following additional inoperabilities would require entry into LCO 3.0.3?

A. SRV H

B. CS Pump A

C. RCIC System

D. SBFW System

Answer: B

Answer Explanation:
Per TS 3.5.1, condition K, two or more low pressure ECCS injection / spray subsystems inoperable for reasons other than Condition B or C,

Required action: Enter LCO 3.0.3.

Completion time: Immediately.

The stem conditions place the unit in to TS 3.5.1.A and 3.5.1.B due to the inoperability of one LPCI pump in each LPCI subsystem.

T.S. Entries into Conditions A and B are listed because these conditions are 7 day TS actions statements, and are not required to be memorized by RO candidates. Removing these from the stem would make the question SRO only.

Distractor Explanation:
Distractors are incorrect and plausible because:

A. Plausible because an ADS valve inoperability would require entry into TS 3.5.1.G. If TS action 3.5.1.C had also been entered, then this combination would require entry into TS 3.5.1.K, resulting in entry into 3.0.3.
C. Plausible because verifying RCIC operability is an immediate TS action if HPCI is inoperable per TS action 3.5.1.E. If this check were not able to be completed, due to simultaneous inoperability of HPCI and RCIC, then TS action 3.5.1.I would apply, and the required action is to be in mode 3 within 12 hours, which is similar to the action of entering TS action 3.0.3. Incorrect because HPCI is not inoperable.
D. Plausible Standby Feedwater is a TRM required system, and serves a purpose similar to RCIC and HPCI. Incorrect because this system is not included in TS 3.5.1.

Reference Information:

TS 3.5.1

Question Use
Closed Reference
RO

NUREG 1123 KA Catalog Rev. 2
G2.2.42 Ability to recognize system parameters that are entry-level conditions for Technical Specifications
203000 RHR/LPCI: Injection Mode

10CFR55 RO/SRO Written Exam Content
10 CFR 55.41(b)(10) Administrative, normal, abnormal, and emergency operating procedures for the facility.

NRC Exam Usage
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Fundamental
New
NRC Early Review
RO

Associated objective(s):

30	K/A Importance: 3.1 / 3.1			Points: 1.00
R30	Difficulty: 3.00	Level of Knowledge: Higher cognitive level	Source: NEW	87847

The plant is cooling down in mode 3, with Division 1 of Shutdown Cooling (SDC) in service

- SDC Flow is 10,000 gpm
- E1150-F003A, Div 1 RHR Hx Outlet Vlv, is throttled
- E1150-F048A, Div 1 RHR Hx Bypass Vlv, is full open

You have been directed to raise the cooldown rate. What is the proper method to raise cooldown rate in this condition?

A. Throttle open E1150-F003A ONLY

B. Throttle closed E1150-F048A ONLY

C. Throttle closed E1150-F017A, then throttle open E1150-F003A.

D. Throttle closed E1150-F048A, then throttle open E1150-F017A

Answer: A

Answer Explanation:
Per 23.205, a caution is repeated frequently:

CAUTION
A flow rate of 10,000 to 10,700 gpm should be maintained through the SDC loop. Less than 10,000 gpm may cause temperature stratification in the Reactor Vessel unless the RPV Head is removed and the cavity flooded. More than 10,700 gpm through the RHR Heat Exchanger could cause damage to the Heat Exchanger.

A limit and precaution for operation in SDC mode states the following:

Either E1150-F048A (B), Div 1 (2) RHR Hx Bypass Vlv, or E1150-F003A (B), Div 1 (2) RHR Hx Outlet Vlv, must be full open to prevent stratification in Rx Vessel.

With the conditions in the stem, to raise cooldown rate while complying with each of these limits and precautions, it is required to throttle open on E1150-F003A to raise the flow through the RHR HX.

To answer this question, the applicant must know that the normal SDC flowpath is through E1150-F611A. An alternate but incorrect flowpath is through E1150-F017A, which is the normal flowpath for LPCI flow in emergency operation.

The SDC flowpath is orificed and E1150-F611A is fully open. Neither E1150-F611A nor 1150-F017A are mentioned in the stem. Giving either valve position in the stem would cue the applicant regarding the flowpath.

Knowledge of the normal flowpath for SDC is a necessary component to being able to properly control SDC flow.

Distractor Explanation:
Distractors are incorrect and plausible because:
B. Plausible because this would raise the cooldown rate by forcing more flow through the RHR HX, and would be a correct response if E1150-F003A was full open and flow started at 10,700 gpm. Incorrect because procedures require at least one of F048A and F003A to be fully open, and throttling flow to less than 10,000 gpm also violates the procedure.
C. Plausible because this would raise the cooldown rate by allowing more flow through the RHR HX if SDC flow were controlled by E1150-F017A. Incorrect because E1150-F017A is not used during the SDC mode.
D. Plausible because this would raise the cooldown rate by allowing more flow through the RHR HX if SDC flow were controlled by E1150-F017A. Incorrect because E1150-F017A is not used during the SDC mode.

10 CFR 55.41(b)(10) RO Justification:
This question meet does not meet ES-401 Attachment 2 requirements to be SRO-Only because although it cannot be answered solely on the basis of "systems knowledge", nor solely by knowing immediate actions, nor by knowing entry conditions for AOPs or plant parameters that require direct entry into major EOPs, nor by knowing the purpose, overall sequence or overall mitigative strategy of a procedure, it also does not require assessment of plant conditions and then the selection of a procedure or section, nor when to implement attachments and appendices, nor knowledge of diagnostic steps and decision points in EOPs nor knowledge of administrative procedures regarding coordination between normal, abnormal and emergency procedures.

The license level is indeterminate per ES-401, attachment 2, figure 2-2, but directly involves a task that is routinely performed by an RO during outages, and is therefore judged to be an RO level question.

Reference Information:

23.205

Question Use
Closed Reference
RO

NUREG 1123 KA Catalog Rev. 2
205000 A4.09 3.1/3.1 System flow
205000 A4. Ability to manually operate and/or monitor in the control room:

10CFR55 RO/SRO Written Exam Content
10 CFR 55.41(b)(10) Administrative, normal, abnormal, and emergency operating procedures for the facility.

NRC Exam Usage
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Higher cognitive level
New
RO

Associated objective(s):
Plant Shutdown from 85% to Cold Shutdown
Performance Enabler
Shutdown and cooldown the reactor following the soft shutdown and recriticality prevention guidelines.

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31	K/A Importance: 3.9/3.8			Points: 1.00
R31	Difficulty: 3.00	Level of Knowledge: Higher cognitive level	Source: BANK	87848

A transient results in the following:

- Drywell pressure is 4 psig and rising.
- HPCI automatically starts and injects.
- HPCI then trips on High RPV Water Level.
- RPV Level is 200" and lowering slowly.

How will the HPCI system valve lineup change if the Reactor High Water Level Signal RESET Pushbutton is depressed?

A. E4150-F067, HPCI Turbine Steam Stop Valve will immediately re-open.

B. E4150-F001, HPCI Turbine Steam Supply Isolation Valve will immediately re-open.

C. E4150-F067, HPCI Turbine Steam Stop Valve will re-open if RPV level reaches 110".

D. E4150-F001, HPCI Turbine Steam Supply Isolation Valve will re-open if RPV level reaches 110".

Answer: A

Answer Explanation:
Per 23.202, HPCI System SOP:

Section 7.1 Recovery from a Trip.
The HPCI system will trip on Level 8 (214") which will cause the E4100-F067, HPCI Turbine Steam Stop Valve (among others) to close. The E4150-F001 does not close on this signal.

Note on Page 24:

If the Reactor Water Level reaches Level 8, the HPCI Turbine will trip and 2D28, HPCI SYSTEM ACTUATED, will clear. If water level falls again to Level 2 the HPCI Turbine will re-start. The high water level trip can be reset when level is less than Level 8 by depressing the High Level Reset Switch. If an auto initiation signal is present, HPCI will restart when Level 8 is reset, or HPCI Turbine can be restarted manually, E4150-F006, HPCI Pump Disch Inbd Iso Vlv, will reopen automatically when level 2 is reached or can be opened manually. Preferred method of Level indication is RPV wide range level indication.

Therefore, the candidate must determine that, because HPCI tripped on Level 8 the F067 will close, and because the Reactor High Water Level Signal RESET Pushbutton is depressed AND because the High Drywell Pressure signal is present, HPCI will automatically restart meaning the F067 will re-open.

Distractor Explanation:
Distractors are incorrect and plausible because:

B. Closure of the F001 will cause the HPCI turbine to shut down and this valve does go closed for some trips. However, it will not close on a Level 8 trip, therefore it would not re-open when the Reactor High Water Level Signal RESET Pushbutton is depressed.
C. The candidate could incorrectly determine that the HPCI trip cleared the automatic initiation signal, which is true for the Low Level (Level 2) initiation signal as can be seen in the note above. However, the High Drywell pressure signal will not clear so HPCI will automatically restart when Reactor High Water Level Signal RESET Pushbutton is depressed and not wait until RPV level dropped below Level 2 (110.8").
D. Closure of the F001 will cause the HPCI turbine to shut down and this valve does go closed for some trips. However, it will not close on a Level 8 trip, therefore it would not re-open. Even if the F001 did close, with the high Drywell Pressure signal present, it would not wait until RPV level dropped to Level 2 (110.8") to re-open.

Reference Information:

23.202 HPCI System SOP.

Question Use
Closed Reference
RO

NUREG 1123 KA Catalog Rev. 2
206000 A3.03 3.9/3.8 System lineup: BWR-2,3,4
206000 HPCI System.
206000 A3. Ability to monitor automatic operations of the HIGH PRESSURE COOLANT INJECTION SYSTEM including::

10CFR55 RO/SRO Written Exam Content
10 CFR 55.41(b) (7) Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

NRC Exam Usage
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Bank
Higher cognitive level
RO

Associated objective(s):
High Pressure Coolant Injection
Cognitive Enabler
Describe general HPCI System operation, including component operating sequence, normal operating parameters, and expected system response.

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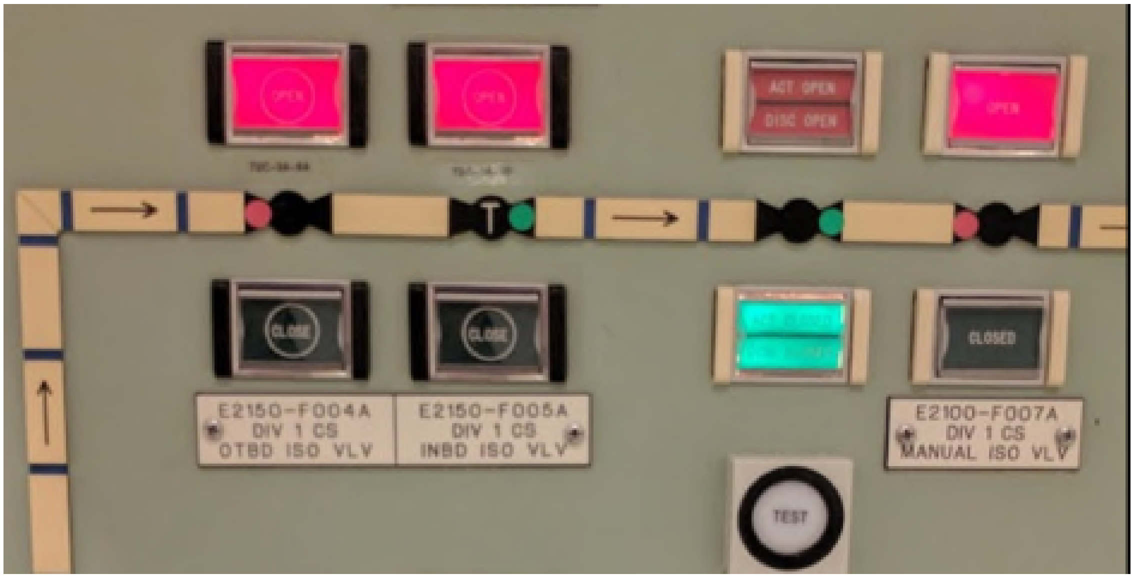
32	K/A Importance: 3.1 / 3.1			**Points: 1.00**
R32	Difficulty: 4.00	Level of Knowledge: Higher cognitive level	Source: NEW	87849

The plant is at 100% power

An electrician was investigating the MCC for E2150-F005A and made a mistake, causing E2150-F005A to open, as shown below:



A fault exists on testable check valve E2150-F006A, allowing 6 gpm of reverse flow at full d/p:

(1) What Alarm is consistent with these conditions?

To correct the condition, the crew will close E2150-F005A and then __(2)__.

A. (1) 1D14 - DIV I CSS INJ VAL LEAK PRESS HIGH

(2) Close F004A. Cycle F005A open then shut. Open F004A. Relieve pressure through E2150-F015A, Div 1 CS Test Line Iso Vlv. Close E2150-F015A

B. (1) 1D14 - DIV I CSS INJ VAL LEAK PRESS HIGH

(2) Close F007A. Cycle F005A open then shut. Relieve pressure through E2150-F015A, Div 1 CS Test Line Iso Vlv, close E2150-F015A. Open F007A

C. (1) 1D10 - CORE PLATE TO SP HDR A DIFF PRESS HIGH

(2) Close F007A. Cycle F005A open then shut. Open F007A.

D. (1) 1D10 - CORE PLATE TO SP HDR A DIFF PRESS HIGH

(2) Close F004A. Cycle F005A open then shut. Open F004A.

Answer: A

Answer Explanation:
Per 1D14, DIV 1 CSS INJ VAL LEAK PRESS HIGH,

NOTE: This annunciator is an indication that there is leakage through E2100-F006A, Div 1 CS Testable Check Valve, and E2150-F005A, Div 1 CS Inbd Iso Vlv.

After the actions already listed in the stem to close E2150-F005A, the remaining actions are specified:

a. Close E2150-F004A, Div 1 CS Otbd Iso Vlv.
b. Open, then close E2150-F005A, Div 1 CS Inbd Iso Vlv.
c. Open E2150-F004A, Div 1 CS Otbd Iso Vlv.
d. To reduce pressure, slowly throttle open E2150-F015A, Div 1 CS Test Line Iso Vlv.
e. When desired pressure has been obtained, close E2150-F015A, Div 1 CS Test Line Iso Vlv.

The listed failure is similar to an event that occurred at Fermi.

Operating Experience
CARD 02-16969, Simultaneous Opening of E2150F005B while the E2150F004B was Open During MPM Testing
During the core spray system division 2 system outage the E2150F005B (core spray loop B inboard isolation valve) was stroked open with the E2150F004B (core spray loop B outboard isolation valve) already open. During normal operation with reactor pressure greater than 461 psig, an interlock requires one of these system isolation valves be closed. The purpose is to prevent an overpressure condition from occurring in the core spray piping upstream of the E2100F004B which could result in an intersystem LOCA.

Distractor Explanation:
Distractors are incorrect and plausible because:
B. Plausible because the first half is correct, and actions in the second half would relieve the trapped pressure. Incorrect because these actions are not specified in ARP 1D14, which would be the controlling procedure for restoration.
C. Plausible on the first half because 1D10 is an indication of a fault on the CS system piping inside the drywell, and an examinee could incorrectly determine that the CS system piping pressure would lower as a result of the stem conditions. The second half would then be a method to reset F005A to a closed position while d/p was zero to get the best seal. Incorrect because the piping pressure would not be lowered, but instead rises as a result of stem conditions.
D. Plausible on the first half because 1D10 is an indication of a fault on the CS system piping inside the drywell, and an examinee could incorrectly determine that the CS system piping pressure would lower as a result of the stem conditions. The second half would then be a method to reset F005A to a closed position while d/p was zero to get the best seal. Incorrect because the piping pressure would not be lowered, but instead rises as a result of stem conditions.

Reference Information:

1D14
M-2034

Question Use
Closed Reference
RO

NUREG 1123 KA Catalog Rev. 2
209001 A2.08 3.1/3.1 Valve openings

209001	Low Pressure Core Spray System.
209001 A2.	Ability to (a) predict the impacts of the following on the LOW PRESSURE CORE SPRAY SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:

10CFR55 RO/SRO Written Exam Content

10 CFR 55.41(b) (7)	Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

NRC Exam Usage
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Higher cognitive level
New
RO

Associated objective(s):
Core Spray System

Cognitive Enabler

Discuss potential modes of Core Spray System component failures and any industry operating experience related to the failure.

Standby Liquid Control

Cognitive Enabler

Discuss failure modes of Standby Liquid Control System controls and vital instruments, including design features that could result in erroneous operation or indication.

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33	K/A Importance: 4.0*/4.1			**Points: 1.00**
R33	Difficulty: 4.00	Level of Knowledge: Higher cognitive level	Source: MODIFIED: 2018 FERMI NRC EXAM	87647

The plant is operating at 100% power. The control power fuse for Motor Control Center (MCC) 72B-4C Pos 2A-R, which provides power to ONE of the SLC Pump control circuits, has blown.

Which of the responses below accurately completes the following statement regarding the status of the SLC system?

Currently, only the __(1)__ squib circuit continuity light is lit and __(2)__ squib valve(s) would open upon system initiation.

A. A; both

B. B; both

C. A; only the A

D. B; only the B

Answer: D

Answer Explanation:
The A and B SLC Pumps are powered from MCCs 72B-4C, Pos 2A-R and 72E-5B, Pos 2B respectively.
Note: This question tests a common misconception among licensed operator candidates in that, regardless of what SLC pump is started, both explosive (squib) valves will always open. What actually happens is that, when either SLC pump is started, contacts in the control circuits for both squibs close that would enable both squibs to fire. However, control power is still needed to fire the squib and, since control power comes from the A and B SLC pump control circuits, both squibs will only fire if both pumps have power via their respective MCCs with control power available.
Since the stem of the question states that control power to MCC 72B-4C, Pos 2A-R has been lost, the operator should recognize that power to the A SLC Pump, and A explosive (squib) control circuits, has been lost. This will cause a loss of continuity through the A squibs and loss of continuity indication (amber light). Upon system initiation, the A SLC pump will be unavailable. Upon starting the B SLC pump, even though a contact will close in the control circuit for the A squib valve (C4104F004A), without control power available, this squib will not fire and the valve will not open. Only the B squib will fire for the B squib valve (C4104F004B) and only the B valve will open.

Distractor Explanation:
Distractors are incorrect and plausible because:

A. The operator could incorrectly determine that 72B-4C is the power supply to the B squib circuit and, therefore, determine that only the A squib continuity light will be lit, which is incorrect because 72E-5B is the power supply to the B squib circuit. Furthermore, this distractor is plausible because of the common misconception that both squibs will always fire, regardless of which SLC pump is started. This is incorrect because each squib valve still requires control power from the MCC that powers the valve's respective SLC pump. The operator could apply this misconception and assume that both valves will still open upon system initiation, which is incorrect.
B. The operator could recognize that 72B-4C is the power supply to the A squib circuit and, therefore, recognize that only the B squib continuity light will be lit. The operator could then conclude that, regardless of this failure, both squib circuits will fire and both squib valves will open, which is plausible because of the common misconception that both squibs will always fire, regardless of which SLC pump is started. This is incorrect because each squib valve still requires control power from the MCC that powers the valve's respective SLC pump. The operator could apply this misconception and assume that both valves will still open upon system initiation, which is incorrect.
C. This distractor is plausible because the operator could incorrectly determine that the MCC given in the stem of the question (72B-4C Pos 2A-R) powers the B SLC Pump and B squib control circuit. This is not true since the A SLC Pump and squib valve control circuits are powered from 72B-4C, Pos 2A-R.

Reference Information:
I-2131-01, SLC Pumps Schematic Diagram
Student Text ST-OP-315-0014, SLC System

Question Use
Closed Reference
RO

NUREG 1123 KA Catalog Rev. 2

211000 A1.09	SBLC system lineup
211000	SLC System
211000 A1.	Ability to predict and/or monitor changes in parameters associated with operating the STANDBY LIQUID CONTROL SYSTEM controls including:

10CFR55 RO/SRO Written Exam Content

10 CFR 55.41(b) (7)	Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

NRC Exam Usage
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Higher cognitive level
Modified
RO

Associated objective(s):

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34	K/A Importance: 3.5/3.7			**Points: 1.00**
R34	Difficulty: 3.00	Level of Knowledge: Fundamental	Source: MODIFIED: 2018 FERMI NRC EXAM	87877

The plant is operating at 96% Power.

A transient results in indicated RPV pressure rising to 1100 psig.

During the transient, the following instruments respond as indicated:

- B21-N678A, Reactor Vessel Steam Dome Pressure - High Trip Unit, remains UNTRIPPED.
- B21-N678B, Reactor Vessel Steam Dome Pressure - High Trip Unit, TRIPPED.
- B21-N678C, Reactor Vessel Steam Dome Pressure - High Trip Unit, remains UNTRIPPED.
- B21-N678D, Reactor Vessel Steam Dome Pressure - High Trip Unit, TRIPPED.

How will the Reactor Protection System (RPS) be affected by the above conditions?

A. BOTH RPS Trip Systems will TRIP.

B. ONLY RPS Trip System A will TRIP.

C. ONLY RPS Trip System B will TRIP.

D. NEITHER RPS Trip System will TRIP.

Answer: C

Answer Explanation:
This question was modified by changing conditions in the stem to make previously correct answer (A) incorrect and previously incorrect distractor (C) correct.

The candidate must recognize that the transient has caused Nuclear Boiler pressure to rise above the trip setpoint of 1093 psig.
The candidate must then determine that (1) MALFUNCTION of 2 nuclear boiler instruments has resulted in the two instruments associated with RPS Trip System A NOT tripping.
The candidate must then determine that (2) the two instruments associated with RPS Trip System B DID trip.
Finally, the candidate must conclude that the transient, coupled with failure of the Nuclear Boiler instruments in the stem, will result in the receipt of ONLY a half scram of RPS Trip System B per the logic below:

NOTE: Refer to 23.601, Page 10 for this discussion. Trip Logic for the Reactor Vessel Steam Dome Pressure - High Trip Function for RPS is as follows:
Trip System A = Trip Channels A (A1), C (A2)
Trip System B = Trip Channels B (B1), D (B2)

Trip Channel A1 = B21-N678A; A2 = B21-N678C
Trip Channel B1=B21-N678B; B2 = B21-N678D

The Trip Setpoint for this Function is </= 1093 psig.

Trip Logic for this Function is (A1 or A2) and (B1 or B2) = Full Rx Scram

With the instrument malfunctions given in the stem of the question, ONLY the B and D instruments will trip, resulting in Trip Channels B1 and B2 tripping, which will result in only RPS Trip System B tripping, which will result in a HALF reactor scram.

Distractor Explanation:
Distractors are incorrect and plausible because:

A. The candidate could incorrectly recall RPS Trip Logic and assume that the failures listed will not prevent either RPS Trip System from tripping, which is incorrect as can be seen from the Trip Logic described above.
B. The candidate could incorrectly recall RPS Trip Logic and assume that the failures listed will prevent only the A RPS Trip System from tripping, which is incorrect as can be seen from the Trip Logic described above.
D. The candidate could incorrectly recall RPS Trip Logic and assume that the failures listed will prevent BOTH RPS Trip Systems from tripping, which is incorrect as can be seen from the Trip Logic described above

Reference Information:
23.601 , Instrument Trip Sheets.

Question Use
Closed Reference
RO

NUREG 1123 KA Catalog Rev. 2
212000 K6.03 3.5/3.7 Nuclear boiler instrumentation
212000 RPS
212000 K6. Knowledge of the effect that a loss or malfunction of the following will have on the REACTOR PROTECTION SYSTEM:

10CFR55 RO/SRO Written Exam Content
10 CFR 55.41(b) (7) Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

NRC Exam Usage
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Fundamental
Modified
RO

Associated objective(s):
Reactor Protection System (C7100)
Cognitive Enabler
Describe Reactor Protection System response to failure of instrument channels selected for control.

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35	K/A Importance: 4.4/4.7			**Points: 1.00**
R35	Difficulty: 4.00	Level of Knowledge: Fundamental	Source: BANK	87649

The reactor is critical during a reactor startup. IRM Ranges and indications are as follows:

IRM A	30 (Range 1)	IRM B	32 (Range 1)
IRM C	40 (Range 2)	IRM D	BYPASSED
IRM E	28 (Range 1)	IRM F	25 (Range 1)
IRM G	45 (Range 2)	IRM H	30 (Range 2)

Which of the following describes the status of the IRM Retract Permit lights and the ability to withdraw IRM detectors?

A. NONE of the IRMs have a Retract Permit light lit but ALL would retract.

B. NONE of the IRMs have a Retract Permit light lit and NONE would retract.

C. Only IRM D has a Retract Permit light lit and ONLY it could be retracted without receiving a rod block.

D. IRMs C, D, G and H have a Retract Permit light lit and ALL 4 could be retracted without receiving a rod block.

Answer: C

Answer Explanation:
23.603 INTERMEDIATE RANGE MONITORING SYSTEM Section 7.0 BYPASS OPERATION OF AN IRM DETECTOR provides guidance for the expected response of the white Bypass light as follows:

NOTE: When an IRM Channel is bypassed, the following IRM functions are defeated:
- IRM Upscale Trip to RPS
- Associated IRM trips to Rod Withdrawal Block circuits of Reactor Manual Control System
- IRM outputs to annunciator and sequence recorder
- IRM outputs to indicating lights at COP H11-P603 (not Panel H11-P606, Start-up Range Neutron Monitoring), with the exception of the Retract Permit light which will remain ON as long as the IRM detector is not full out.

7.2.2 Verify white Bypass light comes ON for IRM Channel bypassed (if Reactor Mode switch is in RUN, the white bypass light will stay OFF at the H11-P603).

In summary:

Retract permit light indication will be illuminated when either the Reactor Mode Switch is in Run or the IRM is bypassed and the IRM detector is not full out.

IRM related ROD BLOCKS are listed in ARP 3D113 and will occur a follows:
Upscale
Downscale
Detec not full in
Inop

Therefore:

With IRM D Bypassed the IRM Upscale is bypassed, therefore the ROD BLOCK will not occur and because the MODE SW is not in RUN and IRM D is bypassed ONLY IRM D will have a retract permit light. Section 1.1 of the SOP (23.603) describes the detector insert and retract mechanism which is always functional as long as it has power, so the IRM D would retract and not cause a ROD BLOCK.

Distractor Explanation:
A. While IRMs could be retracted however this is incorrect because IRM D being bypassed means that a retract permit light is illuminated. This is plausible if the examinee does not know the NOTE listed in the answer.
B. This is incorrect because IRM D is bypassed so the retract permit light would be ON for IRM D and IRMs can always be retracted. This is plausible if the examinee recalls step 6.2.4 of 23.603 which states "If withdrawing detectors, verify Retract Permit light for desired IRM is ON." but does not know that the retract permit light does not effect the ability to withdraw IRM detectors and will only cause a rod block as listed in ARP 3D113.
D. This is incorrect because Only IRMs that are bypassed would have a retract light illuminated unless the Mode switch is in Run. IRM C, G and H are the highest listed and the examinee may believe that a slight withdraw will not result in a ROD BLOCK due to UPSCALE, however a ROD BLOCK due to DETEC NOT FULL IN will occur.

Reference Information:
23.603 Section 7.0 - Bypass operation of an IRM Detector - Pg 12
ARP 3D113 ROD BLOCK SUMMARY

Plant Procedures
23.603

Question Use
Closed Reference
RO

NUREG 1123 KA Catalog Rev. 2

215003 K5.01	Detector operation
215003	IRM System
215003 K5.	Knowledge of the operational implications of the following concepts as they apply to INTERMEDIATE RANGE MONITOR (IRM) SYSTEM :

10CFR55 RO/SRO Written Exam Content

10 CFR 55.41(b) (5)	Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

NRC Exam Usage
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Bank
Fundamental
RO

Associated objective(s):
Intermediate Range Monitoring (C5111)
Cognitive Enabler
List the interlocks associated with Intermediate Range Monitoring System components.

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36	K/A Importance: 3.4/3.4			**Points: 1.00**
R36	Difficulty: 3.00	Level of Knowledge: Higher cognitive level	Source: NEW	88207

A startup is in progress:

- SRM C Bypassed and fully withdrawn.
- All other IRMs and SRMs fully inserted.
- IRMs A, C, E and G on Range 2.
- IRMs B, D, F and H on range 3.
- All IRMs and SRMs rising.

Predict the impact on RPS if the "A" SRM high voltage power supply was to drift to 250 VDC and the only operator action taken was to range the IRMs during the power ascension.

A half scram on RPS A will …

A. not occur.

B. immediately occur.

C. occur later than expected.

D. occur earlier than expected.

Answer: A

Answer Explanation:
The malfunction given in the stem of the question indicates a low voltage condition. The candidate should recognize that normal SRM voltage is ~350VDC and voltage below that value will cause SRM output to lower. The candidate should recall that voltage below 300VDC will cause alarm 3D55, and an accompanying control rod withdrawal block, due to SRM INOP. The candidate should then recall that RPS will remain unaffected because the SRM shorting links are installed.

Distractor Explanation:
Distractors are incorrect and plausible because:
B. This distractor would be correct if detector voltage failed for an IRM since this would cause 3D59, IRM Upscale/INOP, which generates a trip signal to RPS. This is incorrect because loss of SRM detector voltage causes SRM INOP, which does not input into RPS.
C. This distractor would be correct if the SRM shorting links were removed because, with low detector voltage, SRM A output would read low and the resulting upscale trip from SRM A would occur later than expected.
D. This distractor would be correct if the source range high volts was the input into the rest of the SRM drawer and not just the input to the detector. The candidate could recognize the voltage as being lower than the nominal 350VDC but fail to recall what the high voltage power supply supplies. If the candidate mis-remembers the SRM block diagram and assumes that the failed power supply caused a reduction in voltage to the SRM trip circuits, then it is plausible that the candidate could assume that this would lower the setpoint of the SRM upscale trip function causing the SRM output to reach this value earlier than expected. However, source range high voltage supplies only the SRM detector so its going low will only impact detector output and not the upscale trip circuitry, which is powered separately.

Reference Information:

3D55 SRM Upscale/Inop.
3D59 IRM A/E/C/G Upscale Trip/Inop.

Question Use
Closed Reference
RO

NUREG 1123 KA Catalog Rev. 2
215004 K3.01 3.4/3.4 RPS
215004 SRM System
215004 K3. Knowledge of the effect that a loss or malfunction of the SOURCE RANGE MONITOR (SRM) SYSTEM will have on the following:

10CFR55 RO/SRO Written Exam Content
10 CFR 55.41(b) (7) Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

NRC Exam Usage
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Higher cognitive level
New
RO

Associated objective(s):
Source Range Monitoring (C5110)
Cognitive Enabler
Discuss the Source Range Monitoring System interrelationships with other systems.

37	K/A Importance: 3.4/3.5			Points: 1.00
R37	Difficulty: 2.00	Level of Knowledge: Higher cognitive level	Source: NEW	88107

The plant is in MODE 2 with a reactor startup in progress.
Control rod withdrawal results in the following SRM channel indications:

- SRM A = 3×10^5 counts per second
- SRM B = 1×10^5 counts per second
- SRM C = 5×10^4 counts per second
- SRM D = 6×10^4 counts per second

This condition will result in…

A. ONLY a FULL control rod withdrawal block

B. a FULL scram AND a FULL control rod withdrawal block

C. a HALF scram on RPS channel A AND a FULL control rod withdrawal block

D. a HALF scram on RPS channel A AND a HALF control rod withdrawal block

Answer: A

Answer Explanation:
Only SRM A is above the upscale trip setpoint of $2x10^5$. In mode 2, the shorting links will be installed bypassing all SRM inputs to RPS. As a result, only a full control rod block will be received since rod block logic is 1 out of 4 and is NOT affected by shorting links.

Distractor Explanation:

Distractors are incorrect and plausible because:
B is incorrect since the shorting links prevent any input to RPS from SRMs. Applicants may select this if they do not recall that the shorting links will be installed in this condition or do not recall the effect of shorting links on SRM trip logic.
C is incorrect since the shorting links prevent any input to RPS from SRMs. Applicants may select this if they do not recall that the shorting links will be installed in this condition or the effect of shorting links on SRM trip logic, and further incorrectly believe that SRM scram signals have coincident logic.
D is incorrect since the control rod block logic for SRM channels is 1 out of 4, therefore, a full control rod block will occur, shorting links prevent any SRM input to RPS, and SRM scram logic is 1 out of 4. Applicants may select this if they do not recall the correct rod block logic arrangement for SRM channels, the effect of shorting links on SRM RPS logic, and further incorrectly believe that SRM scram signals have coincident logic .

Reference Information:

LP-OP-315-0022, Source Range Monitors.

K/A Match Justification:

This question matches the selected k/a since RO applicants must recall SRM design features associated with scram signals such as trip setpoints, logic configuration, and bypasses.

Question Use
Closed Reference
RO

NUREG 1123 KA Catalog Rev. 2
215004 K4.02 3.4/3.5 Reactor SCRAM signals
215004 SRM System
215004 K4. Knowledge of SOURCE RANGE MONITOR (SRM) SYSTEM design feature(s) and/or interlocks which provide for the following:

10CFR55 RO/SRO Written Exam Content
10 CFR 55.41(b) (7) Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

NRC Exam Usage
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Higher cognitive level
New
RO

Associated objective(s):
Source Range Monitoring (C5110)
Cognitive Enabler
Discuss the Source Range Monitoring System interrelationships with other systems.

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38	K/A Importance: 3.2/3.3			**Points: 1.00**
R38	Difficulty: 3.00	Level of Knowledge: Higher cognitive level	Source: BANK	87829

A plant startup is in progress. Reactor power is 30% by the APRMs and being increased by control rod withdrawal. APRM 2 malfunctions and is indicating 0% power.
Which one of the following describes the effect on the Rod Block Monitors because of the APRM failure?

A. RBM A and B are OPERABLE. Control rod withdrawal can continue.

B. RBM A and B are INOPERABLE. Control rod withdrawal is blocked.

C. RBM A is INOPERABLE. Bypassing APRM 2 will allow control rod withdrawal.

D. RBM B is INOPERABLE. Bypassing APRM 2 will allow control rod withdrawal.

Answer: D

<u>Answer Explanation:</u>
The examinee should correctly recognize that APRM #2 is the primary APRM for RBM B and determine that, under the current condition of 30% power, the RBM is bypassed due to APRM 2 power input of 0%. The RBM is bypassed until power exceeds the setpoint of 27% power from the primary APRM. The secondary ARPM (APRM 4 in this case) will be automatically selected after the ARPM 2 bypass joystick is operated.

This question is made more challenging by the fact that Tech Specs requires the RBM to be OPERABLE when Power is ≥ 30%, while the RBM actually starts enforcing when Power is >27% by the reference APRM.

<u>Distracter Explanation:</u>

A. Is incorrect and plausible because the examinee may incorrectly recall that technical specification allowable setpoint for RBM bypassing is ≥30% and determine that the RBM are not yet affected. Additionally RBM A is associated with APRMs #1 and 3, and since it is may seem more logical to associate APRMs 1&2 with RBM A, rather than APRMs 2&4 being associated with RBM B. Also with the APRM downscale with the mode switch in run, outward rod motion could not occur due to a APRM downscale control rod withdrawal block.

B. Is incorrect but plausible because the examinee could incorrectly determine that the ARPM downscale effects both RBMs. (Recirc flow transmitter failures could affect both RBMs flow comparator). Outward control rod motion is blocked by the APRM downscale failure.

C. Is incorrect but plausible because the examinee may incorrectly recall that RBM A is associated with APRM #2 since it is may seem more logical to associate APRMs 1&2 with RBM A, rather than APRMs 2&4 being associated with RBM B.

<u>Reference Information:</u>
23.607 (pg 3) 1.1 System Description

Plant Procedures
23.605
03D099
03D103
23.607

Question Use
Closed Reference
RO

NUREG 1123 KA Catalog Rev. 2
215005 APRM/LPRM

215005 K3.	Knowledge of the effect that a loss or malfunction of the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM will have on following:
215005 K3.07	Rod block monitor: Plant-Specific

Technical Specifications
3.3.2.1 Control Rod Block Instrumentation

10CFR55 RO/SRO Written Exam Content
10 CFR 55.41(b) (7) Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

NRC Exam Usage
ILO 2015 Exam
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Bank
Higher cognitive level
RO

Associated objective(s):
Power Range Neutron Monitoring (C5112, C5113 & C5114)
Cognitive Enabler
Discuss the Power Range Neutron Monitoring System interrelationships with other systems.

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39	K/A Importance: 2.7/2.8			**Points: 1.00**
R39V3	Difficulty: 3.00	Level of Knowledge: Fundamental	Source: NEW	87853

A loss of power to RCIC Inverter, E51-K603, will result in a loss of...

A. ability to automatically control RCIC flow.

B. ability to automatically trip the RCIC turbine if running.

C. isolation capability for E5150-F008, RCIC Stm Line Otbd Iso Valve ONLY.

D. isolation capability for E5150-F007, RCIC Stm Line Inbd Iso Valve AND E5150-F008, RCIC Stm Line Otbd Iso Valve.

Answer: A

Answer Explanation:

Power Supplies

Power to RCIC logic circuits is supplied by 130 VDC Buses A and B.
The RCIC inverter is powered from H11-P613 via 2PA2-5, pos 9.

Per 1D60, RCIC Inverter Failure, 1D56, RCIC Logic Bus Power Supply Failure, the following are the results of loss of logic bus A, B and RCIC inverter power:

Failure of **logic A** power will result in:

- **Loss of isolation capability for F008,** F062, and F031.
- RCIC will not isolate when RCIC Logic A Manual Isolation Pushbutton.
- RCIC suction will not shift on Low CST level.
- E51-F045, F095, and F013 will close if open.

Failure of **logic B** will result in the following:

- **F007** and F084 **will not isolate**.
- **RCIC will not trip on auto trip signal**.

Failure of the RCIC Inverter, E51-K603, will result in loss of the following:

E51-K600, RCIC Power Supply
RCIC Pump suction and discharge pressure.
RCIC Turbine Steam Inlet/Exhaust pressure.
E51-K615, RCIC Moore Controller/Auto RCIC Controller would be disabled

Reference:

1D60, RCIC Logic Bus Power Failure
1D56 RCIC Logic Bus Power Failure.
I-2235-01, RCIC power supply schematic

Distractor explanation:

B is incorrect since this results from a loss of logic power B, and will be unaffected by loss of the inverter. Applicants may choose this if they incorrectly recall RCIC power supplies.
C is incorrect since this results from a loss of logic power A, and will be unaffected by loss of the inverter. Applicants may choose this if they incorrectly recall RCIC power supplies.
D is incorrect since only F007 isolation logic is powered by logic power B. F008 isolation logic is powered by logic power A. Applicants may choose this if they incorrectly recall RCIC power supplies.

K/A Match Justification:

This question matches the selected k/a since RO applicants must recall the power supply to the RCIC flow controller without reference to procedures.

Question Use
Closed Reference
RO

NUREG 1123 KA Catalog Rev. 2
217000 K2.03 2.7*/2.8* RCIC flow controller
217000 RCIC System
217000 K2 Knowledge of electrical power supplies to the following:

10CFR55 RO/SRO Written Exam Content
10 CFR 55.41(b) (7) Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

NRC Exam Usage
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Fundamental
New
RO

Associated objective(s):
Reactor Core Isolation Cooling
Cognitive Enabler
Describe the normal and alternate power supplies to RCIC System components.

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40	K/A Importance: 3.9/3.9			Points: 1.00
R40	Difficulty: 2.00	Level of Knowledge: Higher cognitive level	Source: NEW	87908

The plant is experiencing a LOCA and loss of all High Pressure Feedwater sources.

- Drywell pressure is 0.6 psig, stable
- RPV Water level is 77”, lowering

If operators do NOT bypass ADS, how will ADS SRVs be affected when RPV level reaches the level 1 setpoint?

A. Open after a 105 second time delay ONLY.

B. Open after a 7 minute plus 105 sec time delay.

C. Remain closed until drywell pressure rises above 1.68 psig.

D. Remain closed until a low pressure ECCS pump can be manually started.

Answer: B

Answer Explanation:
Per LP-OP-315-0042: With no operator action ADS will automatically initiate with RPV water level < level 1 for 7 minutes (plus 105 sec), provided drywell pressure remains below 1.68# and low pressure ECCS pumps auto start as expected at L1. In this case, drywell pressure should remain stable, so the ADS 105 second timer will start after a 7 minute time delay, resulting in ADS SRVs opening after 7 min + 105 sec.

Per ARP 1D44, With 1D40, ADS RELAY ENERGIZED, in alarm, ADS actuation will occur in 105 seconds ("0" seconds indicated on timer). It is initiated by ADS Drywell Pressure Bypass Timer Timed Out, when Rx water level is ≤ 31.8" for 7 minutes.

Per ARP 1D31, "he ADS Drywell Pressure Bypass Timers are actuated from a Reactor Vessel Low Water Level (L1) when a High Drywell Pressure signal (> 1.68 psig) is not present. When timed out (7 minutes), a Hi Drywell Pressure Signal is not required to satisfy the ADS initiation logic."

Distractor Explanation:

A is incorrect since the ADS logic adds a 7 minute time delay when RPV level is below L1, but DW pressure is <1.68#. This is plausible if applicants do not correctly recall ADS logic with low level only.
C is incorrect since the ADS logic does not require high DW pressure to actuate. If RPV W/L is <L1, the logic adds a 7 minute time delay. This is plausible if applicants do not correctly recall ADS logic with low level only
D is incorrect since ECCS logic will auto-start all LP ECCS pumps when RPV W/L reaches L1. Manual start of pumps is not required. This is plausible if applicants do not recall the auto start conditions for LP ECCS pumps or if they believe high DW pressure is necessary for ECCS Auto Start.

Reference Information:

LP-OP-315-0042, ADS

K/A Justification:

This question matches the selected k/a since RO applicants must recall the physical and cause/effect relationship between ADS and SRVs without reference to procedures.

Question Use
Closed Reference
RO

NUREG 1123 KA Catalog Rev. 2
218000 K1.06 3.9*/3.9* Safety/relief valves
218000 ADS
218000 K1 Knowledge of the physical connections and/or cause-effect relationships between AUTOMATICDEPRESSURIZATION SYSTEM and the following:

10CFR55 RO/SRO Written Exam Content
10 CFR 55.41(b) (7) Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

NRC Exam Usage
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Higher cognitive level
New
RO

Associated objective(s):
Automatic Depressurization System (B2104)

Cognitive Enabler

Describe general ADS operation, including component operating sequence, normal operating parameters, and expected system response.

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41	K/A Importance: 4.3 / 4.4			Points: 1.00
R41	Difficulty: 3.00	Level of Knowledge: Higher cognitive level	Source: NEW	88109

The plant was operating at 100% power when the following occurred:

- Main Turbine trip.
- ATWS.
- Reactor Power is 55%.
- RPV level is 150" and lowering.

You have been directed to "Bypass and Restore Drywell Pneumatics."

Which of the following actions, or sets of actions, is/are the minimum necessary to carry out the order as directed?

A. Place Division 1 and 2 Drywell Pneumatic Supply Isolation Bypass Keylock switches to ON ONLY

B. Place Division 1 and 2 Drywell Pneumatic Supply Isolation Bypass Keylock switches to ON

AND

Open T4901-F465(468), Div 1(2) DW Pneumatics Supply Outboard Iso Valves

AND

Open T4901-F601(602), Div 1(2) DW Pneumatics Supply Inboard Iso Valves

C. Depress A7100-M120(M146), Inboard (Outboard) MSIV Iso Reset Switches

AND

Open T4901-F465(468), Div 1(2) DW Pneumatics Supply Outboard Iso Valves

AND

Open T4901-F601(602), Div 1(2) DW Pneumatics Supply Inboard Iso Valves

D. Place Division 1 and 2 Drywell Pneumatic Supply Isolation Bypass Keylock switches to ON

AND

Depress A7100-M120(M146), Inboard (Outboard) MSIV Iso Reset Switches

AND

Open T4901-F465(468), Div 1(2) DW Pneumatics Supply Outboard Iso Valves

AND

Open T4901-F601(602), Div 1(2) DW Pneumatics Supply Inboard Iso Valves

Answer: A

Answer Explanation:
The conditions given in the stem of the question have RPV level lowering towards Level 2.

23.601 indicates that the Group 18, Primary Containment Pneumatic Supply System, isolation valves will close at Reactor Low Water Level - Level 2 (RPV level 110.8"). This can be seen on on Page 20 of Enclosure B in 23.601.

The direction to "Bypass and Restore Drywell Pneumatics" is a generic order given regardless of the status of the Group 18 isolation valves. The candidate must assess plant conditions and determine the actions necessary to carry out the order.

In the listed conditions, all that is required is to take the keylock switches to ON. This is done per the hardcard from 23.406.

Distractor Explanation:
Distractors are incorrect and plausible because:

B. This would be true if RPV Level had dropped below Level 2. During an ATWS, the direction to bypass and restore drywell pneumatics is often given PRIOR TO lowering RPV water level (for Terminate and Prevent purposes for example). In this case, all that is necessary to carry out the order is turn the two keylock switches. However, this distractor is incorrect because, with RPV level > L2, the valves have not yet closed.
C. This would be a valid method if RPV level dropped below Level 2 but was subsequently restored >Level 2, thus clearing the isolation condition. In this case, the isolation valves would be reset, IAW 23.427 Enclosure 2B page 4, by depressing the two Main Steam Isolation Reset Push Buttons on the H11-P601 and P602. This is incorrect because RPV Level is still <Level 2.
D. This would be true if it was necessary to reset MSIV (PCIS or NSSSS) isolation logic after turning the keylock switches, which is the case for the Group 13 PCIS isolation valve G1154-F018, DW Equip Drain Sump Inbd Containment Iso Valve (see 23.427 Enclosure 1B, Page 1). This is incorrect because Group 18 isolation valves do not require this additional step.

Reference Information:

23.427, Primary Containment Isolation System.
23.601, Instrument Trip Sheets.
23.406, Primary Containment Nitrogen Inerting and Purge System

Question Use
Closed Reference
RO

NUREG 1123 KA Catalog Rev. 2

G2.1.23	Ability to perform specific system and integrated plant procedures during different modes of plant operation
223002	PCIS/NSSS

10CFR55 RO/SRO Written Exam Content

10 CFR 55.41(b) (7)	Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

NRC Exam Usage
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Higher cognitive level
New
RO

Associated objective(s):

Primary Containment Isolation System

Cognitive Enabler

Describe general Primary Containment Isolation System operation, including component operating sequence, normal operating parameters, and expected system response.

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42	K/A Importance: 3.9/4.1			**Points: 1.00**
R42	Difficulty: 3.00	Level of Knowledge: Fundamental	Source: NEW	87827

Which of the responses below completes the following statement from the Precautions and Limitations section of 23.201, Safety Relief Valves and Automatic Depressurization System?

Reactor Water Level should be as near to normal level as possible during manual SRV operation to prevent inadvertent Reactor __(1)__ Level trips due to __(2)__.

A. (1) Low
(2) shrink

B. (1) High
(2) swell

C. (1) Low
(2) inventory loss

D. (1) High
(2) increased injection

Answer: B

Answer Explanation:
Note: During validation it was noted that this question relates directly to recent plant OE whereby a licensed operator received numerous High RPV Level 8 trips of the RCIC system while attempting to control level with RCIC while SRVs were automatically maintaining pressure in the Low-Low Set mode of operation.

Per 23.201, Safety Relief Valves and Automatic Depressurization System SOP, Precaution and Limitation 3.5, Reactor Water Level should be as near to normal level as possible during manual SRV operation to prevent inadvertent Reactor Hi Level trips due to swell.

To answer this question, and satisfy the K/A, the candidate must first recall that SRV actuation causes the phenomenon known as "swell" to occur. Then, the candidate must recall that "swell" causes RPV level to rise and potentially a Reactor Hi Level trip.

Distractor Explanation:
Distractors are incorrect and plausible because:

A. The candidate could conclude that RPV level will lower, causing a Low RPV water level trip, because the candidate could determine that SRV operation causes shrink to occur in the core (fuel) region due to the pressure reduction, which would draw water from the downcomer region, where RPV water level is sensed, which would cause RPV level to lower and thus contribute to a Low Level Trip. This is incorrect because the pressure reduction is felt across the entire RPV, not just the core (fuel) region, and it causes some water in the RPV (at saturated conditions) to expand and form steam bubbles, thus causing indicated level to "swell" in the downcomer.
C. The candidate could determine that RPV level will lower, causing a Low RPV water level trip, when SRVs are actuated, which is plausible because the SRVs take water from the RPV and discharge it to the Suppression Pool which means inventory is lost and RPV level will lower. This is incorrect because the P&L is concerned with the INITIAL SRV operation and not the long-term SRV operation so initially RPV level will rise and might cause an inadvertent RPV High Level trip
D. The candidate could conclude that RPV level will rise, as pressure is lowered with SRVs, causing increased injection and that a Reactor High Level Trip could occur. While increased injection is a possibility, it is within the control of the operator to prevent excessive injection and not the basis for the precaution referenced in the stem of the question.

Reference Information:

23.201, Safety Relief Valves and Automatic Depressurization System SOP.

Question Use
Closed Reference
RO

NUREG 1123 KA Catalog Rev. 2
239002 A4.06 3.9/4.1 Reactor water level
239002 SRVs
239002 A4. Ability to manually operate and/or monitor in the control room:

10CFR55 RO/SRO Written Exam Content
10 CFR 55.41(b) (5) Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

NRC Exam Usage
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Fundamental
New
RO

Associated objective(s):
Reactor Pressure Vessel Instrumentation (B2100)

Cognitive Enabler

Discuss design considerations, capabilities, and limitations related to Reactor Pressure Vessel Instrumentation component operation.

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43	K/A Importance: 2.7/2.7			**Points: 1.00**
R43	Difficulty: 2.50	Level of Knowledge: Higher cognitive level	Source: BANK	88788

The plant is operating at 100% power with the following:

- C32-R616A North Reactor Feedwater Pump (RFP) Controller in AUTO.
- C32-R616B South Reactor Feedwater Pump (RFP) Controller in AUTO.
- RPV Startup LCV Mode Switch in RUN.
- Level Control Mode Switch in 3-Element.

A reactor scram subsequently occurs.

Two minutes after the scram, (1) what is the status of the RFP Controllers and (2) what are the approximate RFP Turbine speeds?

A. (1) Auto
(2) 1600 rpm

B. (1) Auto
(2) 2650 rpm

C. (1) Manual
(2) 1600 rpm

D. (1) Manual
(2) 2650 rpm

Answer: D

Answer Explanation:
Per 3D157, After a Scram, Post Scram Feedwater Logic (PSFWL) actuates to perform the following:

6 seconds after Reactor scram, DCS logic causes C32-R618, Master Feedwater Level Controller, setpoint (SP) setdown to 150 inches (below minimum scale of 160 inches of SP display). This action occurs regardless of RPV Startup LCV Mode Switch position.
30 seconds after Reactor scram, DCS logic causes the following actions if RPV Startup LCV Mode Switch is in RUN:

- For 2 seconds, C32-R620, N21-F403 Startup LCV Controller, is pulsed to AUTO with a setpoint (SP) of 197 inches.
- N2100-F607, N RFP Disch Line Iso Valve, and N2100-F608, S RFP Disch Line Iso Valve, receive close signals (approximately 90 seconds to close).
- N2100-F045A, N RFP Disch HYD Stop Valve, and N2100-F045B, S RFP Disch HYD Stop Valve, receive close signals (approximately 8-10 seconds to close).

60 seconds after Reactor scram, DCS logic causes the following actions if RPV Startup LCV Mode Switch was in RUN at 30 seconds after Reactor scram:

- For 2 seconds, C32-K616A, North Reactor Feed Pump Controller, and C32-K616B, South Reactor Feed Pump Controller, are pulsed to MANUAL with outputs at approximately 29.23% corresponding to approximately 2650 rpm on the North and South Reactor Feed Pump Turbine. This action occurs in AUTO or MANUAL.
- N2100-F607, N RFP Disch Line Iso Valve, and N2100-F608, S RFP Disch Line Iso Valve, close signals are clear.
- N2100-F045A, N RFP Disch HYD Stop Valve, and N2100-F045B, S RFP Disch HYD Stop Valve, close signals are clear.

The candidate must first evaluate the conditions in the stem and determine that, with the RPV startup LCV Mode Switch in RUN, PSFWL will actuate. The candidate must then determine that 2 minutes after the scram PSFWL will have set the RFPT speed controllers to (1) Manual and the RFPT speeds to approximately (2) 2650 rpm.

Distractor Explanation:
Distractors are incorrect and plausible because:

A. (1) Auto is the normal mode for the RFPT controllers and also the mode they would be in if the RPV Startup LCV Mode Switch was in the START position. However, with the RPV Startup LCV Mode Switch was in the RUN position, the RFP speed controllers will be in MANUAL 2 minutes after the scram. (2) is the speed that the RFPTs would be turning at if the RPV Startup LCV Mode Switch was in the START position. This is true because, 6 seconds after Reactor scram, DCS logic causes C32-R618, Master Feedwater Level Controller, setpoint (SP) setdown to 150 inches (below minimum scale of 160 inches of SP display), regardless of RPV Startup LCV Mode Switch position. This causes the RFPTs to lower to the lower demand setpoint of the controllers (approximately 1600 rpm) and, if the RPV Startup LCV Mode Switch was in START after 60 seconds, the RFPTs would not get a signal to increase speed so they would remain at 1600 rpm. However, with the RPV Startup LCV Mode Switch in the RUN position, the RFPT speeds will get set to approximately 2650 rpm by PSFWL.
B. (1) Auto is the normal mode for the RFPT controllers and the candidate could fail to recall that PSFWL sets the RFP controllers to MANUAL 60 seconds after a scram. However, with the RPV Startup LCV Mode Switch was in the RUN position, the RFP speed controllers will be in MANUAL 2 minutes after the scram. (2) is correct and is plausible with part (1) if the candidate incorrectly determines that the final condition is 2650 rpm with the controllers in AUTO.
C. (1) Manual is correct and is the final mode of the RFP controllers after PSFWL has finished its sequence. (2) is plausible because it represents the minimum speed setpoint of the RFPs, which is the lowest speed attainable with the RFP controllers and the point when speed control shifts to the MOP (Motor Operated Potentiometer). The candidate could recall seeing the RFPTs at 1600 rpm following a scram and incorrectly determine that this is their final speed. This is incorrect because the RFP controllers will be set to an output corresponding to approximately 2650 rpm 60 seconds after the scram.

Reference Information:
3D157, Post Scram Feedwater Logic Actuated

Plant Procedures
20.000.21

Question Use
Closed Reference
RO

NUREG 1123 KA Catalog Rev. 2
259002 K5.07 2.7/2.7 Turbine speed control mechanisms: TDRFP
259002 Reactor Water Level Control System
259002 K5 Knowledge of the operational implications of the following concepts as they apply to REACTOR WATER LEVEL CONTROL SYSTEM:

10CFR55 RO/SRO Written Exam Content
10 CFR 55.41(b) (7) Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

NRC Exam Usage
ILO 2009 Exam
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Bank
Higher cognitive level
RO

Associated objective(s):
Reactor Feedwater (N2100)
Cognitive Enabler
Describe general Reactor Feedwater system operation, including component operating sequence, normal operating parameters, and expected system response.

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44	K/A Importance: 3.2/3.2			Points: 1.00
R44	Difficulty: 3.00	Level of Knowledge: Fundamental	Source: NEW	87569

The plant is operating in MODE 1 when an SRV fails open.
After the plant stabilizes with the SRV open, the feedwater control system (FWCS) will control RPV water level …

A. HIGHER than normal because the additional steam flow results in a rise in feedwater flow.

B. at the normal setpoint because feedwater flow will rise to compensate for the added steam flow.

C. LOWER than normal because the steam flow through the open SRV is NOT sensed by the FWCS.

D. at the normal setpoint because the SRV failure will cause the FWLC system to shift to single element.

Answer: C

Answer Explanation:
Per 6M721-5701-1, the steam line flow sensors are downstream of the SRVs. With an SRV open, the steam flow through the valve is not sensed by the FWCS, resulting in a permanent FF/SF mismatch (flow error) with indicated FF higher than indicated SF once RPV level is stabilized. The system must compensate for the flow error by creating an opposing level error resulting in indicated/actual level lower than the level setpoint.

Distractor Explanation:
Distractors are incorrect and plausible because:
A is incorrect because RPV level will be controlled lower. Although feed flow will initially rise, level will stabilize lower due to the permanent FF/SF mismatch. Applicants may choose this if they incorrectly recall that level will be controlled higher, as is true with an inadvertent RCIC injection at power.

B is incorrect because the FWCS WILL be affected by the un-sensed steam flow and control lower than normal. Applicants may choose this if they incorrectly believe that the SRV flow will have no effect on the FWCS.

D is incorrect because the FWCS will not shift to single element upon SRV failure. Applicants may choose this because there are several component failures that DO cause a shift to single element and they may incorrectly believe that the SRV failure will.

Reference Information:

6M721-5701-1, Nuclear Boiler System Functional Operating Sketch.

K/A Match:

This question matches the K/A since applicants must predict the response of the FWCS to changes in steam flow in order to properly monitor automatic system operation.

Question Use
Closed Reference
RO

NUREG 1123 KA Catalog Rev. 2
259002 A3.03 3.2/3.2 Changes in main steam flow
259002 Reactor Water Level Control System
259002 A3. Ability to monitor automatic operations of the REACTOR WATER LEVEL CONTROL SYSTEM including:

10CFR55 RO/SRO Written Exam Content
10 CFR 55.41(b) (7) Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

NRC Exam Usage
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Fundamental
New
RO

Associated objective(s):
Feedwater Control (C3200)

Cognitive Enabler

Describe general Feedwater Control System operation, including component operating sequence, normal operating parameters, and expected system response.

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45	K/A Importance: 3.0/3.3			Points: 1.00
R45	Difficulty: 2.00	Level of Knowledge: Higher cognitive level	Source: NEW	87928

The plant is operating at 100% power.

- Standby Gas Treatment System (SGTS) Div 1 is out of service for maintenance
- A spurious Secondary Containment isolation signal is received

Five minutes later the CRLNO reports that Reactor Building HVAC is tripped, and SGTS Div 2 is running with system flow at 1000 SCFM and stable.

SGTS Div 2 flow is…

A. normal and secondary containment ΔP will remain within limits.

B. insufficient and secondary containment ΔP may become less negative than allowable.

C. slightly lower than normal but secondary containment ΔP will remain stable and within limits.

D. higher than normal and secondary containment ΔP may become more negative than allowable.

Answer: B

Answer Explanation:
Per 23.404, Standby Gas Treatment System, system flow following an auto start of SGTS must be between 3879 and 4180 scfm. This flow is required to maintain SC pressure at least -0.25"H2O. The flow indicated here is significantly below the required value and because SGTS div 1 is not available, it would likely result in the system not maintaining SC pressure sufficiently negative. The allowable SC pressure is between -0.125 and -0.750 "H2O.

1000 SCFM is listed in the UFSAR as the flow capacity of one of four parallel HEPA filters.

Distractor Explanation:
Distractors are incorrect and plausible because:
A is incorrect because SGTS flow is below the required value and SC pressure will likely not be maintained. Applicants may choose this if they do not recall the correct flow value to maintain SC pressure.
C is incorrect since 1000 scfm is insufficient to ensure SC DP remains within limits. Applicants may choose this if they believe that SC DP will remain within limits with flow very low. 1000 SCFM is plausible because this value is listed in the UFSAR as the flow through one of four parallel HEPA filters.
D is incorrect since the system flow is below the required value. Applicants may choose this if they do not recall the correct flow values for an operable SGTS system.

Reference Information:

23.404, Standby Gas Treatment System
LP-OP-315-0020

K/A Match justification:

This question matches the selected k/a because RO applicants must predict the impact of SGTS flow on SC ΔP.

Question Use
Closed Reference
RO

NUREG 1123 KA Catalog Rev. 2
261000 A1.04 3/3.3 Secondary containment differential pressure
261000 Standby Gas Treatment System
261000 A1 Ability to predict and/or monitor changes inparameters associated with operating the STANDBY GASTREATMENT SYSTEM controls including:

10CFR55 RO/SRO Written Exam Content
10 CFR 55.41(b) (7) Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

NRC Exam Usage
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Higher cognitive level
New
NRC Early Review
RO

Associated objective(s):
Standby Gas Treatment System
Cognitive Enabler
Describe general Standby Gas Treatment System operation, including component operating sequence, normal operating parameters, and expected system response.

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46	K/A Importance: 3.0 / 3.1			**Points: 1.00**
R46	Difficulty: 3.00	Level of Knowledge: Higher cognitive level	Source: MODIFIED: 2010 NMP NRC EXAM	88188

Initial Conditions:

- Drywell pressure is 12 psig.
- Both trains of SGTS auto-started.
- Division 1 of SGTS was stopped per 23.404 Enclosure A, Shutdown a Division of SGTS.
- RB differential pressure is stable at -0.6" wc.

Event:

- The exhaust fan for Division 2 SGTS trips on overload.

RB differential pressure will begin to trend __(1)__ negative.

Division 1 of SGTS __(2)__ in response to the event.

A. (1) less
(2) will be restarted per the hard card

B. (1) less
(2) will automatically restart

C. (1) more
(2) will be restarted per the hard card

D. (1) more
(2) will automatically restart

Answer: A

Answer Explanation:
Fermi does not have an auto-start feature on SGTS based on trip of the running division nor does SGTS auto start on Reactor Building D/P.

Per I-2642-01, lines 1A-1D and 2, the starting logic is seen not to depend upon the other division status or air flow.

23.404 Enclosure A leaves the Division 1 SGTS Exhaust Fan in Off/Reset, which prohibits any automatic starts.

The purpose of the SGTS is to

1. provide controlled filtered venting of the Primary Containment (Drywell and/or Suppression Chamber) or Secondary Containment (Reactor Building) following an accident or abnormal occurrence which might cause abnormally high airborne contamination in these areas.
2. During periods of containment isolation, maintain a negative pressure of at least negative 0.25 inches water column (wc) in the Reactor Building with respect to atmospheric pressure.

When the SGTS exhaust fan trips, the RB differential pressure will begin to trend towards 0", or become less negative.

This question was modified from NMP 2010 Q#15. It's listed as "MODIFIED" due to the fact that the difference in system design between the two plants makes one of NMP's distractors the correct answer at FERMI.

Distractor Explanation:
Distractors are incorrect and plausible because:
B. Plausible if there was an interlock based on division 2 flow, and the procedure referenced returned Div 1 of SGTS to automatic. Incorrect because the auto starts of Division 1 of SGTS do not check for SGTS flow or Div 2 exhaust fan status.
C. Plausible if an examinee understands that SGTS maintains DP and that upon losing SGTS flow the D/P lowers, and then interprets "lowers" as "becomes more negative."
D. Plausible if an examinee understands that SGTS maintains DP and that upon losing SGTS flow the D/P lowers, and then interprets "lowers" as "becomes more negative." Also plausible if there was an interlock based on division 2 flow, and the procedure referenced returned Div 1 of SGTS to automatic. Incorrect because the auto starts of Division 1 of SGTS do not check for SGTS flow or Div 2 exhaust fan status.

Reference Information:

I-2642-01
23.404
NMP Q#15 from 2010.

Question Use
Closed Reference
RO

NUREG 1123 KA Catalog Rev. 2
261000 A2.05 3/3.1 Fan trips
261000 Standby Gas Treatment System
261000 A2 Ability to (a) predict the impacts of the following on the STANDBY GAS TREATMENT SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:

10CFR55 RO/SRO Written Exam Content
10 CFR 55.41(b)(10) Administrative, normal, abnormal, and emergency operating procedures for the facility.

NRC Exam Usage
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Higher cognitive level
Modified
NRC Early Review
RO

Associated objective(s):
Standby Gas Treatment System
Cognitive Enabler
Identify normal operating procedures associated with the Standby Gas Treatment System.

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47	K/A Importance: 2.9 / 3.1			**Points: 1.00**
R47V2	Difficulty: 3.00	Level of Knowledge: Higher cognitive level	Source: BANK	89027

- Plant conditions: 100% power
- 24.307.14, Emergency Diesel Generator 11 Start and Load Test is in progress.
- Conditions have been established to parallel the EDG with the grid.

Which of the following describes the indications immediately after Bus 11EA Pos EA3, EDG 11 Generator Output Breaker is closed?

Bus 11EA voltage will __(1)__ voltage.

Diesel KVAR __(2)__ rise.

A. (1) rise to EDG output
(2) will

B. (1) rise to EDG output
(2) will NOT

C. (1) remain at previous
(2) will

D. (1) remain at previous
(2) will NOT

Answer: C

Answer Explanation:
Per 24.307.14, conditions are established that match the given information in the stem.

Per step 5.1.29, EDG voltage is established to be 50 volts higher than bus voltage, which will make the EDG a KVAR source when paralleled.

When the EDG is paralleled, the machine goes from a no-load condition to some positive value of VARs, and then immediately the operator adjusts frequency and voltage to achieve 750-1000 KW and 200-400 KVAR.

Distractor Explanation:
Distractors are incorrect and plausible because:

A. Plausible because the oncoming EDG voltage is higher than bus voltage. Some examinees may believe that the voltage on the bus will be the higher of the two (system or EDG) voltage as the EDG assumes some of the bus loading. Incorrect because the system voltage represents an infinite grid, and the local voltage will not rise appreciably based on the action of one small generator.
B. Plausible because the oncoming EDG voltage is higher than bus voltage. Some examinees may believe that the voltage on the bus will be the higher of the two (system or EDG) voltage as the EDG assumes some of the bus loading. Also plausible in that KVAR would not rise if the EDG voltage was set equal to bus loading prior to the parallel. Incorrect because the paralleling operation is being performed with an EDG at no-load across the feeder breaker to 11EA, with voltage set 50 v higher than running voltage.
D. plausible because if the paralleling across the normal feeder breaker to bus 64B during recovery from a LOOP, and with EDG and System voltages equal, these would be the indications observed. Incorrect because the paralleling operation is being performed with an EDG at no-load across the feeder breaker to 11EA.

Reference Information:

24.307.14

Question Use
Closed Reference
RO

NUREG 1123 KA Catalog Rev. 2
262001 A1.03 2.9/3.1 Bus voltage
262001 AC Electrical Distribution
262001 A1. Ability to predict and/or monitor changes in parameters associated with operating the A.C. ELECTRICAL DISTRIBUTION controls including:

10CFR55 RO/SRO Written Exam Content
10 CFR 55.41(b)(10) Administrative, normal, abnormal, and emergency operating procedures for the facility.

NRC Exam Usage
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Bank
Higher cognitive level
RO

Associated objective(s):

48	K/A Importance: 2.7/2.9			**Points: 1.00**
R48	Difficulty: 2.00	Level of Knowledge: Fundamental	Source: NEW	87868

A complete failure of UPS "A" static inverter occurs.
UPS "A" loads will now be supplied by the UPS …

A. "A" rectifier from the UPS battery.

B. "B" rectifier from Bus 72R.

C. "A" voltage regulator from Bus 72M.

D. "A" voltage regulator from Bus 72R.

Answer: D

Answer Explanation:
Per 23.308.01, each UPS unit contains automatic load transfer logic that will actuate a static transfer switch upon failure of the static inverter. In this case, the inverter has completely failed, resulting in auto transfer of loads to the alternate supply voltage regulator, which is supplied by bus 72R.

Distractor Explanation:

A is incorrect because the battery cannot supply loads to the UPS distribution cabinet unless the static inverter is functional. Additionally, the A rectifier is not fed by the battery and cannot provide power to loads unless the static inverter is functional. Applicants may choose this if they do not recall the basic functional layout of the UPS system.

B is incorrect because UPS B rectifier cannot supply the A distribution cabinet unless the A static inverter is functional. Applicants may choose this because the UPS B rectifier IS capable of supplying power to UPS A loads if the A rectifier or bus 72M fails. However, this is not possible with failure of the A inverter.

C is incorrect since the alternate supply for UPS A is 72R. Applicants may choose this if they confuse the alternate supply (72R) with the normal supply (72M).

Reference Information:

23.308.01, UNINTERRUPTIBLE POWER SUPPLY SYSTEM

K/A Match Justification:

This question matches the selected k/a because RO applicants must determine the effect on the UPS system following a failure of a static inverter.

Question Use
Closed Reference
RO

NUREG 1123 KA Catalog Rev. 2
262002 K6.03 2.7/2.9 Static inverter
262002 UPS (AC/DC)
262002 K6 Knowledge of the effect that a loss or malfunction ofthe following will have on the UNINTERRUPTABLE POWERSUPPLY (A.C./D.C.) :

10CFR55 RO/SRO Written Exam Content
10 CFR 55.41(b) (7) Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

NRC Exam Usage
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Fundamental
New
RO

Associated objective(s):
Uninterruptible Power Supply (R3100)
Cognitive Enabler
List the automatic features of Uninterruptible Power Supply System operations.

49	K/A Importance: 2.6 / 2.9			**Points: 1.00**
R49V2	Difficulty: 3.00	Level of Knowledge: Fundamental	Source: NEW	87927

The plant is operating at 100% power with the following:

- Division 1 130V ESF Batteries are undergoing an equalizing charge.
- T4100-B033, Battery Room AC Unit is in operation.
- Division 1 and 2 Battery Room Temperatures are at 72°F.

T4100-C007 and T4100-C008, Division 1 Battery Room East and West Exhaust Fans ...

A. are running to prevent undesirable buildup of explosive hydrogen gas.

B. will run If Battery Room AC Unit air flow stops to prevent unacceptable room temperatures.

C. will run if Battery Room ambient temperature exceeds 75°F to prevent unacceptable room temperatures.

D. will run if Battery Room AC Unit air flow stops to prevent undesirable buildup of explosive hydrogen gas.

Answer: D

Answer Explanation:
From 23.426, Reactor Building Heating Ventilation and Air Conditioning:

Air Conditioner (T4100-B033) fan units will operate continuously, but compressor will cycle on and off to supply cool air to Battery Room when Essential Battery Room ambient temperature exceeds 75°F. In AUTO, Exhaust Fans (T4100-C007 and T4100-C008) will auto start if air flow stops, preventing undesirable buildup of explosive hydrogen gas in the Battery Room(s).

Distractor Explanation:
Distractors are incorrect and plausible because:

A. This distractor would be true if the battery room exhaust fans auto started, or were manually started, when an equalizing charge was initiated. This distractor is incorrect because no direction exists to manually start the exhaust fans and the exhaust fans will only auto start upon receipt of a no air flow condition from the battery room AC unit.
B. This distractor would be true if the battery room exhaust fans started, to draw air across the cooling coils and cool the battery rooms, upon a low air flow condition on the Battery Room AC unit to prevent excessive room temperatures. However, the exhaust fans auto start, if air flow stops, to prevent an undesirable buildup of explosive hydrogen gas in the Battery Rooms
C. This distractor would be true if the battery room exhaust fans auto started when room temperature exceeds 75°F, which is how the battery room AC unit operates. If the candidate determined that the exhaust fans draw air across the cooling coils and are needed to cool the room, then this answer would be selected. This answer is incorrect because the AC fans operate continuously (for ventilation) and the compressor cycles (for cooling) based on room temperature, but the exhaust fans will only auto start upon receipt of a no air flow condition from the battery room AC unit.

Reference Information:

23.426, Reactor Building Heating Ventilation and Air Conditioning.

Question Use
Closed Reference
RO

NUREG 1123 KA Catalog Rev. 2
263000 K5.01 2.6/2.9 Hydrogen generation during battery charging
263000 DC Electrical Distribution
263000 K5 Knowledge of the operational implications of the following concepts as they apply to D.C. ELECTRICAL DISTRIBUTION :

10CFR55 RO/SRO Written Exam Content
10 CFR 55.41(b) (7) Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

NRC Exam Usage
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Fundamental
New
RO

Associated objective(s):
Reactor Building HVAC
Cognitive Enabler
State the design basis of the Reactor Building HVAC System.

50	K/A Importance: 2.6/2.7			**Points: 1.00**
R50	Difficulty: 3.00	Level of Knowledge: Higher cognitive level	Source: BANK	87870

Following a Loss of Offsite Power, the grid has been restored, conditions are as follows:

- EDG 11 is supplying ESF Bus 64B by way of EDG Bus 11EA and ESF-EDG Bus Tie Breaker B8.
- Synchroscope Switch for ESF Bus 64B Normal Feeder Breaker B6 is ON.
- ALL conditions are met for paralleling the EDG with Offsite Power.

When the B6 Breaker is CLOSED, the operating mode of EDG 11 will shift to the ___(1)___ mode.

Following breaker closure, the EDG 11 Governor must be immediately adjusted to prevent a(n) ___(2)___ condition.

A. (1) Speed Droop
(2) reverse power

B. (1) Speed Droop
(2) overload

C. (1) Isochronous
(2) overload

D. (1) Isochronous
(2) reverse power

Answer: A

Answer Explanation:
Per 23.321, ESF Aux Electrical Distribution System, section 6.3.2, Restoring Offsite Power to ESF Bus 64B and EDG Bus 11EA, the following caution exists:

CAUTION
When the Normal Feeder Breaker B6 is closed, the operating mode of the EDG will be Speed Droop. Operator action will be required to keep sufficient load on the EDG and thus prevent the EDG from tripping on Reverse Power.

Per ST-OP-315-0065:
EDG 11 governor mode is determined by the position of the Bus 64B Normal Feeder Breaker B6 and ESF-EDG Bus Tie Breaker B8. If the offsite feed breaker B6 is OPEN, control relays place the system in isochronous mode. Once the B6 is CLOSED, these relays place the control system in droop mode, allowing the EDG to share load with the offsite source. Once the EDG is paralleled with the grid, there is a risk of the grid assuming all the bus load thus decreasing the margin to a reverse power trip. For this reason, the EDG must be promptly loaded to prevent this from occurring.

Distractors:

B is incorrect because the EDG is promptly loaded to prevent reverse power. Overload would occur if the EDG attempted to assume more load than it is capable, which is not the case when paralleling in this way. Applicants may select this if they confuse the overload and reverse power concepts

C is incorrect because (1) the control shifts to droop mode when paralleled. (2) the EDG is promptly loaded to prevent reverse power. Overload would occur if the EDG attempted to assume more load than it is capable, which is not the case when paralleling in this way. Applicants may select this if they confuse the overload and reverse power concepts AND confuse the expected mode for parallel operation.

D is incorrect because the control shifts to droop mode when paralleled. Applicants may select this if they confuse the expected mode for parallel operation

Reference:
ST-OP-315-0065

K/A Match Justification:

This question matches the selected k/a because RO applicants are required to recall and apply knowledge of the EDG governor control system operation without reference to procedures.

Question Use
Closed Reference
RO

NUREG 1123 KA Catalog Rev. 2

264000 K4.06	Governor control
264000	Emergency Generators (Diesel/Jet)
264000 K4.	Knowledge of EMERGENCY GENERATORS (DIESEL/JET) design feature(s) and/or interlocks which provide for the following:

10CFR55 RO/SRO Written Exam Content

10 CFR 55.41(b) (7)	Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

NRC Exam Usage
ILO 2008 Exam
ILO 2017 Audit
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Bank
Higher cognitive level
RO

Associated objective(s):
Emergency Diesel Generator (R3000)
Cognitive Enabler
Summarize the process of paralleling an Emergency Diesel Generator with Offsite power.

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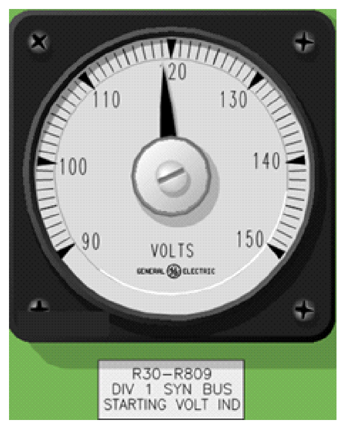
51	K/A Importance: 3.4/3.4			**Points: 1.00**
R51	Difficulty: 3.00	Level of Knowledge: Higher cognitive level	Source: MODIFIED: 2019 FERMI NRC EXAM	88707

The following conditions exist:

- ESF Bus 64B is being supplied from System Service Transformer #64.
- EDG Bus 11EA is being supplied from EDG 11.
- ESF-EDG Bus Tie Breaker B8 is open.
- No faults or trips exist.
- The CRS has directed you to synchronize the EDG with Bus 64B using 64B Pos B8, 4160V X-Tie to Bus 11EA.

When you place the Synchronize Switch for Bus 64B Pos B8, 4160V X-Tie to Bus 11EA in ON, you observe:

NOTE: The Synchroscope is rotating at 10 revolutions per minute (rpm) in the direction shown by the red arrow.



Which of the following adjustments are necessary to close ESF-EDG Bus Tie Breaker Pos. B8?

A. (1) The EDG Governor Control Switch would have to be operated in the LOWER direction.

(2) The EDG Voltage Control Switch would have to be operated in the LOWER direction.

B. (1) The EDG Governor Control Switch would have to be operated in the RAISE direction.

(2) The EDG Voltage Control Switch would have to be operated in the LOWER direction.

C. (1) The EDG Governor Control Switch would have to be operated in the LOWER direction.

(2) The EDG Voltage Control Switch would have to be operated in the RAISE direction.

D. (1) The EDG Governor Control Switch would have to be operated in the RAISE direction.

(2) The EDG Voltage Control Switch would have to be operated in the RAISE direction.

Answer: C

Answer Explanation:
Per 23.321 Section 6.3.3, the following conditions must be established prior to closing the breaker:

- With EDG 11 Governor Control Switch, adjust EDG speed until Synchroscope is rotating slowly in FAST direction.
- With EDG 11 Voltage Control Switch, adjust Starting Voltage until it is equal to or slightly higher than Running Bus Voltage indication.

Also from 23.321: **NOTE**: Synchroscope may be in phase and only show slight movement at first. Changing EDG frequency slightly will rotate scope

Therefore, the examinee must determine that the Synchroscope is not set correctly because it is rotating in the correct direction but at a fast rate. Therefore, the examinee must determine that EDG speed must be lowered and therefore the Governor Control Switch must be placed in the LOWER direction. The examinee must also determine that Starting Voltage represents the EDG's voltage and Running Voltage represents the bus' voltage. The examinee must recognize that EDG (starting) voltage is lower than the bus (running voltage) and therefore determine that the Voltage Control Switch must be placed in the RAISE position.

Distractor Explanation:
Distractors are incorrect and plausible because:

A. (1) This part is correct. (2) The examinee could incorrectly recall the requirement to have EDG voltage higher than bus voltage, and instead believe that EDG voltage should be lower. This would lead the examinee to conclude that EDG voltage is too high, and must be lowered. This is incorrect since EDG Voltage is required to be higher than Bus voltage, and therefore must be raised by placing the Voltage Control switch in the Raise position.
B. (1) The examinee could either misread the markings on the synchroscope or misunderstand the switch manipulation required to get the synchroscope to rotate slowly in the fast direction from its current condition of fast in the fast direction. (2) The examinee could incorrectly recall the requirement to have EDG voltage higher than bus voltage, and instead believe that EDG voltage should be lower. This would lead the examinee to conclude that EDG voltage is too high, and must be lowered. This is incorrect since EDG Voltage is required to be higher than Bus voltage, and therefore must be raised by placing the Voltage Control switch in the Raise position.
D. (1) The examinee could either misread the markings on the synchroscope or misunderstand the switch manipulation required to get the synchroscope to rotate slowly in the fast direction from its current condition of fast in the fast direction. (2) This part is correct.

Reference Information:

23.321, Engineered Safety Features Auxiliary Electrical Distribution System, Section 6.3.3 Manual Operation of ESF-EDG Bus Tie Breaker B8 to ESF Bus 64B when EDG Bus 11EA Power is Being Supplied by EDG 11.

Question Use
Closed Reference
LOR
Open Reference
RO

NUREG 1123 KA Catalog Rev. 2
264000 A4.02 3.4/3.4 Synchroscope
264000 Emergency Generators (Diesel/Jet)
264000 A4 Ability to manually operate and/or monitor in the control room:

10CFR55 RO/SRO Written Exam Content
10 CFR 55.41(b)(10) Administrative, normal, abnormal, and emergency operating procedures for the facility.

NRC Exam Usage
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Higher cognitive level
Modified
RO

Associated objective(s):
Emergency Diesel Generator (R3000)
Cognitive Enabler
Summarize the process of paralleling an Emergency Diesel Generator with Offsite power.

52	K/A Importance: 3.3 / 3.4			Points: 1.00
R52	Difficulty: 3.00	Level of Knowledge: Higher cognitive level	Source: NEW	87929

If the instrument air line to P42-F400, RBCCW Temperature Control Valve is severed, temperature in the RBCCW system will __(1)__.

If the instrument air line to P42-F403, RBCCW Recirculation Pressure Differential Control Valve is severed, differential pressure in the RBCCW system will __(2)__.

A. (1) increase
(2) increase

B. (1) increase
(2) decrease

C. (1) decrease
(2) increase

D. (1) decrease
(2) decrease

Answer: C

Answer Explanation:
Per 20.129.01, P42-F400 RBCCW Temp Control Vlv, is a fail open valve. This valve controls the GSW cooling water to RBCCW, so when it fails open, the temperature will decrease.

Per 20.129.01, P42-F403 RBCCW DP Control Vlv, is a fail closed valve. This valve controls the RBCCW bypass flow, maintaining proper DP as heat loads change. When it fails closed, no bypass flow is allowed, which raises DP.

Distractor Explanation:
Distractors are incorrect and plausible because:

A. Plausible because the RBCCW temperature control valve has a mechanical stop to prevent full closure. An examinee who believes this mechanical stop is to ensure a minimum flow may therefore conclude that the valve fails closed on loss of air and select this response. Incorrect because the valve is a fail open valve.
B. See plausibility for A and D.
D. Plausible if the examinee does not recall the method of D/P control. DP could be controlled by restricting the supply flow or return flow on the loop instead of opening or closing a bypass flow path around the loop. If this method of control were used, the examinee may then conclude that the valve should fail open to maintain flow, which would reduce DP. Incorrect because the flow is bypass flow around the loop.

Reference Information:

20.129.01
M-5727 RBCCW functional diagram

Question Use
Closed Reference
RO

NUREG 1123 KA Catalog Rev. 2
300000 K3.02 3.3/3.4 Systems having pneumatic valves and controls
300000 Instrument Air System
300000 K3 Knowledge of the effect that a loss or malfunction of the (INSTRUMENT AIR SYSTEM) will have on the following:

10CFR55 RO/SRO Written Exam Content
10 CFR 55.41(b) (7) Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

NRC Exam Usage
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Higher cognitive level
New
RO

Associated objective(s):
Reactor Building Closed Cooling Water Emergency Equipment Cooling Water (P4200/P4400)
Cognitive Enabler
Discuss the Reactor Building Closed Cooling Water/Emergency Equipment Cooling Water System interrelationships with other systems.

53	K/A Importance: 2.9/2.9			**Points: 1.00**
R53	Difficulty: 3.00	Level of Knowledge: Fundamental	Source: MODIFED: 2019 FERMI NRC EXAM	87892

If a loss of offsite power occurs, when will the P4400-F603B, Div 2 EECW Supply Iso Valve and the P4400-F601B, Div 2 EECW Return Iso Valve, CLOSE?

A. Only after offsite power is restored to Bus 72C.

B. Only after offsite power is restored to Bus 72F.

C. When the EDG 12 Output Breaker closes.

D. When the EDG 14 Output Breaker closes.

Answer: D

Answer Explanation:
Note: Information on Automatic Initiation of Div 2 EECW can be found in 23.127 Section 5.13. Information on Div 2 EECW Valve Power can be found in 23.137 Attachment 2B.

Per 23.127 Attachment 2B, the P4400-F603B and P4400-F601B are powered from Motor Control Center (MCC) 72F-4A which is fed from 480V ESF Bus 72F. Bus 72F is powered from EDG 14, via 4160V Bus 65F, following a loss of power event.

Per the second NOTE in 23.127 Section 5.13, under emergency electric power distribution conditions, Division 2 Isolation Valves will reposition immediately after the EDG Output Breaker closes.

Therefore, the examinee must recall that the 2 valves in the stem of the question are powered by Bus 72F, which will be de-energized on a loss of 345kV Offsite Power. The examinee must then recall that power to the valves will be restored, and the valves will close, when the EDG 14 output breaker closes to restore power to Bus 72F.

Distractor Explanation:
Distractors are incorrect and plausible because:

A. The examinee could determine that the valves listed are powered from an MCC that is fed off of bus 72C, which is plausible because P4400-F607B, Div 2 EECW DW Otbd Supply Vlv, is powered by MCC 72C-3A. The examinee could then determine that the MCC that powers the valves listed does not get sequenced back on by the EDG load sequencer, which is plausible because several breakers and MCCs off of bus 72F do not get sequenced back on with the EDG. This could lead the candidate to determine that power to these valves will only be restored upon re-energization of the MCC when offsite power is restored. This distractor is incorrect because the valves listed are powered off of an MCC powered by bus 72F, which will be restored when the EDG 14 Output Breaker closes.
B. The examinee could correctly determine that the valves listed are powered from an MCC that is fed off of bus 72F. The examinee could then determine that the MCC that powers the valves listed does not get sequenced back on by its EDG load sequencer, which is plausible because several breakers and MCCs off of bus 72F do not get sequenced back on with the EDG. These could lead the candidate to determine that power to these valves will only be restored upon re-energization of the MCC when offsite power is restored to bus 72F. This is incorrect because the MCC that powers these valves does get sequenced back on and will be re-energized when the EDG Output Breaker closes as described in 23.127 Section 5.13.
C. The examinee could determine that the valves listed are powered by Bus 72C, which will be restored when the EDG 12 Output Breaker closes. This is plausible because P4400-F607B, Div 2 EECW DW Otbd Supply Vlv, is powered by MCC 72C-3A, which will be restored when the EDG 12 Output Breaker closes. This distractor is incorrect because the valves listed are powered by Bus 72F via MCC 72F-4A, which will be restored when the EDG 14 Output Breaker closes.

Reference Information:

23.127, Reactor Building Closed Cooling Water / Emergency Equipment Closed Cooling Water System SOP.

Question Use
Closed Reference
RO

NUREG 1123 KA Catalog Rev. 2
400000 K2.02 CCW valves
400000 Component Cooling Water System
400000 K2. Knowledge of electrical power supplies to the following:

10CFR55 RO/SRO Written Exam Content
10 CFR 55.41(b) (4) Secondary coolant and auxiliary systems that affect the facility.

NRC Exam Usage
ILO 2019 Retake Exam

NRC Question Use (ILO 2019)
Low
NEW
RO

NRC Question Use (ILO 2019 Retake)
Fundamental
Modified
RO

Associated objective(s):
Reactor Building Closed Cooling Water Emergency Equipment Cooling Water (P4200/P4400)
Cognitive Enabler
Describe general Reactor Building Closed Cooling Water/Emergency Equipment Cooling Water System operation, including component operating sequence, normal operating parameters, and expected system response.

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54	K/A Importance: 3.3/3.3			**Points: 1.00**
R54	Difficulty: 3.00	Level of Knowledge: Fundamental	Source: NEW	88108

The plant is operating in MODE 1 with C1106-C001B, West CRD Pump in service.

Loss of AC power to 4160V Bus __(1)__ will cause the CRD Pump to trip.

One minute later, after the above bus is restored by its respective EDG, the CRD Pump will be __(2)__.

A. (1) 65E
(2) tripped

B. (1) 65E
(2) running

C. (1) 65F
(2) tripped

D. (1) 65F
(2) running

Answer: A

Answer Explanation:
Per 23.106, CRD Hydraulic System SOP, Attachment 2, the power supply to the West CRD Pump is Bus 65E.

Per 24.307.03, Enclosure B, it can be seen that the power supply to the West CRD Pump, namely 4160V breaker 65E Pos E11, will shed on loss of Bus 65E.
Per 24.307.03, Enclosure C, it can be seen that the power supply to the West CRD Pump, namely 4160V breaker 65E Pos E11, will NOT sequence by on when EDG 13 restores power to Bus 65E.

Therefore, the candidate must recall that the West CRD Pump is (1) powered by Bus 65E and (2) will not be running after EDG 13 restores power to 65E.

Distractor Explanation:
Distractors are incorrect and plausible because:

B. The candidate could (1) correctly identify the power supply to the West CRD Pump but (2) incorrectly recall that the West CRD Pump will be restored by the EDG 13 Load Sequencer, which is plausible because the CRD system is important for safe operation of the plant and CRD pump hydraulic pressure is needed to restore accumulator pressure and prevent a plant shutdown. Furthermore, other similarly sized loads, such as the B RHR Pump and the B Core Spray pump ARE sequenced on by the EDG load sequencer. However, the West CRD pump is not restored via the EDG 13 load sequencer, so this response is incorrect.
C. The candidate could (1) incorrectly identify the power supply to the West CRD Pump and select Bus 65F, which is plausible because 65F is the other Division II 4160V ESF bus. However, this is incorrect because 65E is the power supply to the West CRD Pump. Part (2) is correct as described above.
D. The candidate could (1) incorrectly identify the power supply to the West CRD Pump and select Bus 65F, which is plausible because 65F is the other Division II 4160V ESF bus. However, this is incorrect because 65E is the power supply to the West CRD Pump. For part (2) the candidate could incorrectly recall that the West CRD Pump will be restored by the EDG Load Sequencer, which is plausible because the CRD system is important for safe operation of the plant and CRD pump hydraulic pressure is needed to restore accumulator pressure and prevent a plant shutdown. Furthermore, other similarly sized loads, such as the B RHR Pump and the B Core Spray pump ARE sequenced on by the EDG load sequencer. However, the West CRD pump is not restored via the EDG load sequencer, so this response is incorrect.

Reference Information:

23.106, CRD Hydraulic System SOP.
24.307.03, EDG 13 - LOP and ECCS Start Test

Question Use
Closed Reference
RO

NUREG 1123 KA Catalog Rev. 2
201001 K6.05 3.3/3.3 A.C. power
201001 CRDH System
201001 K6. Knowledge of the effect that a loss or malfunction of the following will have on the CONTROL ROD DRIVE HYDRAULIC SYSTEM:

10CFR55 RO/SRO Written Exam Content
10 CFR 55.41(b) (7) Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

NRC Exam Usage
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Fundamental
New
RO

Associated objective(s):
Control Rod Drive Hydraulics (C1150)
Cognitive Enabler
Describe the normal and alternate power supplies to Control Rod Drive Hydraulic system components.

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55	K/A Importance: 3.4/3.5			**Points: 1.00**
R55	Difficulty: 3.00	Level of Knowledge: Fundamental	Source: BANK: 2011 VY NRC EXAM	87931

A power reduction is in progress. Plant conditions are as follows:

- feed flow is 32% and lowering
- steam flow is 28% and lowering

The EARLIEST the Rod Worth Minimizer will enforce control rod blocks is when _______.

A. EITHER feed flow OR steam flow is below 27%

B. BOTH feed flow AND steam flow are below 27%

C. EITHER feed flow OR steam flow is below 11%

D. BOTH feed flow AND steam flow are below 11%

Answer: C

Answer Explanation:
Per 23.608, The RWM enforces adherence to the selected (computer program) control rod sequence up to the Low Power Setpoint (LPSP). RWM Sequence Enforcement restricts movement of Control Rods that are not in compliance with the rod by rod order selected as listed in a programmed sequence below the LPSP. No rod motion that would result in an insert or withdraw error is permitted. The LPSP is determined by the digital outputs of the Feedwater Control System (FWCS), which are driven by flow transmitters used for steam and feedwater flow input to the FWCS. Rod Worth Minimizer: When decreasing power, LPSP enforcement is enabled when either total feedwater flow is < 11.275% or total steam flow is < 11.25%.
Distractor Explanation:

A is incorrect because the LPSP entry setpoint is not 27%. As power is reduced below 27% the RWM transition zone is entered. Rod blocks are not enforced in this zone. This is plausible since the RWM LPAP is set at 27%. Applicants may choose this if they confuse the LPSP and LPAP setpoints.

B is incorrect because the LPSP entry setpoint is not 27%. As power is reduced below 27% the RWM transition zone is entered. Rod blocks are not enforced in this zone. Additionally, only one signal is required to enable the enforcement zone. This is plausible since the RWM LPAP is set at 27%. Applicants may choose this if they confuse the LPSP and LPAP setpoints and incorrectly determine that both signals are required to cause enforcement.

D is incorrect because only one signal is required to enable rod blocks and enter the enforcement zone. This is plausible since both signals are required to exit the enforcement zone and disable rod blocks when raising power. Applicants may choose this if they confuse the requirements for entering and exiting the enforcement zone.

Reference Information:

Per 23.608, Rod Worth Minimizer
LP-OP-315-0013

K/A Match Justification:

This question matches the selected k/a because RO applicants must monitor and evaluate parameters used by the RWM and predict the status of rod block enforcement without reference to procedures.

Question Use
Closed Reference
RO

NUREG 1123 KA Catalog Rev. 2
201006 A1.02 3.4/3.5 Status of control rod movement blocks; P-Spec(Not-BWR6)
201006 RWM System
201006 A1 Ability to predict and/or monitor changes inparameters associated with operating the ROD WORTHMINIMIZER SYSTEM (RWM) (PLANT SPECIFIC) controls including:

10CFR55 RO/SRO Written Exam Content
10 CFR 55.41(b) (7) Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

NRC Exam Usage
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Bank
Fundamental
RO

Associated objective(s):
Rod Worth Minimizer (C1108)

Cognitive Enabler

Describe general Rod Worth Minimizer system operation, including component operating sequence, normal operating parameters, and expected system response.

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56	K/A Importance: 2.6 / 2.8			**Points: 1.00**
R56	Difficulty: 3.00	Level of Knowledge: Higher cognitive level	Source: NEW	88087

The plant is at 100% power

A Feedwater transient occurs, resulting in a RPV level reduction to 170".

(1) What is a possible impact of this condition?

(2) What procedure will be used to correct the condition?

A. (1) Lowering Reactor Recirc Pump NPSH due to carryunder
(2) 29.100.01 Sheet 1, RPV Control

B. (1) Lowering Reactor Recirc Pump NPSH due to carryunder
(2) 23.623 Enclosure I, Rapid Power Reduction

C. (1) Neutron Flux instability due to saturated conditions in the downcomer
(2) 29.100.01 Sheet 1, RPV Control

D. (1) Neutron Flux instability due to saturated conditions in the downcomer
(2) 23.623 Enclosure I, Rapid Power Reduction

Answer: A

Answer Explanation:
Per UFSAR, the feedwater control system maintains RPV water level:

Section 7.7.1.3.1.1:
... The range of water level is based on the requirements of the steam separators, including limiting carryover and carryunder, which affects turbine performance and recirculation pump operation....

Section 7.7.1.3.3.1:

... The separators limit water carryover in the steam going to the turbines and limit steam carryunder in water returning to the core....

From TS bases for the L3 Low RPV level trip:

The Reactor Vessel Water Level-Low, Level 3 Allowable Value is selected to ensure that the function will perform as predicted in the recirculation line break analysis. It also ensures that during normal operation the separator skirts are not uncovered (this protects available recirculation pump net positive suction head (NPSH) from significant carryunder)

With RPV water level less than the L3 setpoint, the correct action is to scram the reactor, and the procedure will be 29.100.01 Sheet 1, RPV control.

Distractor Explanation:
Distractors are incorrect and plausible because:

B. Plausible because performing a rapid power reduction could give the feed system a chance to respond if the nature of the feedwater transient was a lack of capability due to a pump trip. Additionally, the rapid power reduction is effected by lowering the speed of the reactor recirc pumps and would therefore improve the NPSH. Incorrect because with the RPV water level < 173.4", the correct action is to take the Reactor Mode switch to SHUTDOWN.
C. Plausible because higher temperatures in the downcomer will lead to increased boiling in the reactor. Some examinees will not realize that this decreases the chance for Neutron Flux Instability by allowing boiling earlier, and in fact this reduction in level (and reduction in Neutron Flux Instability) is the basis for the 114" level in the ATWS terminate and prevent leg of the EOPs. If Neutron Flux Instability were the issue, then a Reactor Scram would be a plausible corrective action.
D. Plausible because higher temperatures in the downcomer will lead to increased boiling in the reactor. Some examinees will not realize that this decreases the chance for Neutron Flux Instability by allowing boiling earlier, and in fact this reduction in level (and reduction in Neutron Flux Instability) is the basis for the 114" level in the ATWS terminate and prevent leg of the EOPs. If Neutron Flux Instability were the issue, then a rapid power reduction would be a plausible corrective action.

Reference Information:

UFSAR Section 7.7.1.3 selected pages
TS bases 3.3.1.1-13

Question Use
Closed Reference
RO

NUREG 1123 KA Catalog Rev. 2
202001 A2.13 2.6/2.8 Carryunder
202001 Recirculation System
202001 A2 Ability to (a) predict the impacts of the following on the RECIRCULATION SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:

10CFR55 RO/SRO Written Exam Content
10 CFR 55.41(b) (7) Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

NRC Exam Usage
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Higher cognitive level
New
NRC Early Review
RO

Associated objective(s):
Feedwater Control (C3200)
Cognitive Enabler
State the purpose of the Feedwater Control System.

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57	K/A Importance: 3.6/3.6			Points: 1.00
R57	Difficulty: 3.00	Level of Knowledge: Higher cognitive level	Source: BANK	87930

The plant is in Mode 2 with reactor heatup in progress and:

- RPV temperature is 205°F.
- The Reactor Water Cleanup System (RWCU) is lined up to blowdown to the Main Condenser.

The following events then occur:

- 2D119, RBCCW PUMPS DIFF PRESS HIGH / LOW, alarms.
- 2D46, MOTOR TRIPPED, alarms.
- Both operating RBCCW pumps indicate TRIPPED.

How will RWCU respond to these conditions?

A. G3352-F119, RWCU Inlet Isolation Valve, closes.

B. G3300-F033, Blowdown Flow Control Valve, closes.

C. G3352-F220 and G3352-F004, RWCU Containment Isolation Valves ONLY, close.

D. G3352-F220, G3352-F004, and G3352-F001, RWCU Containment Isolation Valves, close.

Answer: A

Answer Explanation:
Loss of RBCCW cooling to NRHXR causes a high temperature which leads to this effect.

RWCU NRHXs NRHX Outlet Temp at 140°F:
G3352-F119 closes.
RWCU Pumps trip.
RWCU Demins into Hold.

Distracter Explanation:

B is incorrect since only the F119 valve auto closes in this condition. This choice is plausible since F033 auto closes under high or low pressure conditions in the blowdown line. Applicants may choose this if they confuse automatic isolation signal responses.

C is incorrect since only the F119 valve auto closes in this condition. This choice is plausible since F004 & F220 auto close when SLC is initiated. Applicants may choose this if they confuse automatic isolation signal responses.

D is incorrect since only the F119 valve auto closes in this condition. This choice is plausible since F001, F004 & F220 auto close when LOCA signals are occur. Applicants may choose this if they confuse automatic isolation signal responses.

Reference Information:

ARP 2D110

Plant Procedures
20.127.01
02D110

Question Use
Closed Reference
RO

NUREG 1123 KA Catalog Rev. 2
204000 RWCU System
204000 A3. Ability to monitor automatic operations of the REACTOR WATER CLEANUP SYSTEM including:
204000 A3.03 Response to system isolations

10CFR55 RO/SRO Written Exam Content
10 CFR 55.41(b) (7) Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

NRC Exam Usage
ILO 2015 Exam
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Bank
Higher cognitive level
RO

Associated objective(s):

Reactor Water Cleanup

Cognitive Enabler

Describe general Reactor Water Cleanup system operation, including component operating sequence, normal operating parameters, and expected system response.

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58	K/A Importance: 2.8/2.8			Points: 1.00
R58	Difficulty: 3.00	Level of Knowledge: Fundamental	Source: NEW	88067

Reactor Power is 10% with a startup in progress.

An EDGE ROD is selected for withdrawal.

What is the status of rod blocks provided by the Rod Block Monitors (RBMs)? How would this compare to the status of the rod blocks if power was 40% with the SAME rod selected?

A. All rod blocks would be active for both conditions.

B. All rod blocks would be bypassed for both conditions.

C. Rod blocks would be bypassed at the lower power and active at the higher power.

D. Rod blocks would be active at the lower power and bypassed at the higher power.

Answer: B

Answer Explanation:
The candidate should recall that when an edge rod is selected, regardless of power level, both RBM A and B are automatically bypassed.

Distractor Explanation:
Distractors are incorrect and plausible because:

A. This would be true for an interior rod selected and if the RBM activated at 10% power, which is plausible since the Rod Worth Minimizer becomes bypassed >10% power the candidate could determine that the RWM is bypassed at the same power level that the RBM is activated, which makes logical sense but is incorrect.
C. This would be true for an interior rod, but is incorrect for an edge rod.
D. This would be true for an interior rod and if the RBM was active at lower powers rather than higher powers, which is plausible because the RWM is active at lower powers so the candidate could confuse the RBM with the RWM regarding when each are active and when each are bypassed. However, this is incorrect for an edge rod.

Reference Information:

23.607 Rod Block Monitor SOP

Question Use
Closed Reference
RO

NUREG 1123 KA Catalog Rev. 2
215002 A4.03 2.8/2.8 Trip bypasses: BWR-3,4,5
215002 RBM System

10CFR55 RO/SRO Written Exam Content

10 CFR 55.41(b) (7)	Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

NRC Exam Usage
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Fundamental
New
RO

Associated objective(s):
Power Range Neutron Monitoring (C5112, C5113 & C5114)
Cognitive Enabler
Describe general Power Range Neutron Monitoring System operation, including component operating sequence, normal operating parameters, and expected system response.

59	K/A Importance: 3.7 / 4.1			**Points: 1.00**
R59	Difficulty: 3.00	Level of Knowledge: Fundamental	Source: NEW	87948

The plant is at 100% power.

Surveillance procedure 24.204.01, DIV. 1 LPCI AND SUPPRESSION POOL COOLING/SPRAY PUMP AND VALVE OPERABILITY TEST is in progress.

The surveillance is performed with E1150-F048A, Div 1 RHR Hx Bypass Vlv, fully closed.

The maximum allowed RHR flow through the RHR HX while performing this test is ___(1)___.

The minimum allowed RHRSW flow through the RHR HX while performing this test is ___(2)___.

A. (1) 10,700 gpm
(2) 500 gpm

B. (1) 10,700 gpm
(2) 5,250 gpm

C. (1) 14,000 gpm
(2) 500 gpm

D. (1) 14,000 gpm
(2) 5,250 gpm

Answer: B

Answer Explanation:
Per 24.204.01 Precautions and limitations:
2.2 When using heat exchanger as a flow path, maintain a flow rate of < 10,700 gpm. RHR Service Water also must be maintained at > 5250 gpm through heat exchanger, in this mode.

Distractor Explanation:

A. Plausible because 500 gpm is the minimum flow rate within the procedure by which the RHR/LPCI flow must rise while performing the torus spray portion of the test. A less than competent examinee may choose this value based on improper recall of which minimum value is applicable to RHRSW flow rate through the HX. Incorrect because the minimum RHRSW flow is 5,250 gpm.
C. See A & D plausiblities
D. Plausible because 14,000 gpm is the maximum allowed flow of RHR/LPCI Div 1 when being used for Drywell spray. A less than competent examinee may choose this value base on improper recall of the various Torus Cooling / Torus Spray / Drywell Spray RHR/LPCI flow limits. Incorrect because the maximum allowed RHR/LPCI flow while performing this test with E1150-F048A closed is 10,700 gpm.

Reference Information:

24.204.01

Question Use
Closed Reference
RO

NUREG 1123 KA Catalog Rev. 2
G2.2.12 Knowledge of surveillance procedures
219000 RHR/LPCI: Torus/Suppression Pool Cooling Mode

10CFR55 RO/SRO Written Exam Content
10 CFR 55.41(b)(10) Administrative, normal, abnormal, and emergency operating procedures for the facility.

NRC Exam Usage
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Fundamental
New
RO

Associated objective(s):
Residual Heat Removal (E1100)
Cognitive Enabler
State major precautions and limitations, and major safety considerations for the RHR System, and describe their bases.

60	K/A Importance: 3.2/3.3			**Points: 1.00**
R60	Difficulty: 3.00	Level of Knowledge: Higher cognitive level	Source: BANK	87874

The plant is operating at 100% power when the STA comes back from the Relay Room with the following report:

- The DC Solenoid ammeter for the A INBOARD MSIV indicates 0 amps (de-energized).
- The DC Solenoid ammeter for the D OUTBOARD MSIV indicates 0 amps (de-energized).

Which MSIV(s), if any, would close if the A RPS MG Set were to trip?

A. None

B. Only A Inboard.

C. Only D Outboard.

D. Both A Inboard and D Outboard.

Answer: B

Answer Explanation:
The MSIV solenoids are powered as follows:

- Inboard MSIVs - AC solenoid powered by RPS A 120 Vac.
- Inboard MSIVs - DC solenoid powered by Div 1 130V ESF DC.
- Outboard MSIVs - AC solenoid powered by RPS B 120 Vac.
- Outboard MSIVs - DC solenoid powered Div 2 130V ESF DC.

NSSSS MSIV Isolation Logic power is as follows:

- A logic impacts the Inboard AC and Outboard DC solenoids(Channels A and C) and is powered by A RPS 120Vac.
- B logic impacts the Outboard AC and Inboard DC solenoids(Channels B and D) and is powered by B RPS 120Vac.

The candidate must recognize that the loss of RPS A MG set will de-energize all four Inboard AC solenoids (because they lost power) and all four Outboard DC solenoids (because Channels A and C de-energized).

The candidate must then recall that de-energization of BOTH solenoids (the AC and the DC) for an MSIV is necessary to cause the MSIV to close.

Therefore, the candidate must conclude that ONLY the A Inboard MSIV will close because its DC solenoid was de-energized and then its AC solenoid de-energized with the loss of RPS A power.

Distractor Explanation:
Distractors are incorrect and plausible because:

A. This would be true if RPS B powered both the Inboard AND Outboard AC solenoids and Div 1 DC powered the Inboard AND Outboard DC solenoids. This is incorrect, however, as the MSIV solenoids are energized by the sources described above.
C. This would be true if RPS A powered the Outboard AC solenoids, and RPS B powered the Inboard AC solenoids. This is incorrect, however, as the MSIV solenoids are energized by the sources described above.
D. This would be true if RPS A powered both the Inboard AND Outboard AC solenoids and Div 2 DC powered the Inboard AND Outboard DC solenoids, which is plausible because those power supplies are diverse and independent. This is incorrect, however, as the MSIV solenoids are energized by the sources described above.

Reference Information:

23.601, Instrument Trip Sheets, Enclosure G, RPS and NSSS-MSIV Trip System Definitions.

Question Use
Closed Reference
RO

NUREG 1123 KA Catalog Rev. 2
239001 K2.01 3.2*/3.3* Main steam isolation valve solenoids
239001 Main and Reheat Steam System
239001 K2. Knowledge of electrical power supplies to the following:

10CFR55 RO/SRO Written Exam Content
10 CFR 55.41(b) (7) Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

NRC Exam Usage
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Bank
Higher cognitive level
RO

Associated objective(s):
Nuclear Boiler System (B1100, B2100, B2103, B2104, N1100 & N3017)
Cognitive Enabler
Describe the normal and alternate power supplies to Nuclear Boiler system components.

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61	K/A Importance: 2.6/2.8			Points: 1.00
R61	Difficulty: 3.00	Level of Knowledge: Fundamental	Source: NEW	88048

The plant is operating at 100% power with the following:

- ALL Reactor Building water tight doors closed.
- G11-R654, SW Corner Sump G1101-D076 Level Recorder is trending up at a very fast rate.
- Both D076 sump pumps are running.

The RB NO, who was dispatched to locally inspect the SW Corner Room, reports the following:

- Flooding is apparent in the SW Corner Room.
- Water level is above the elevation of the sump and rising.
- Source of the flooding is unknown.
- It is NOT possible to enter the SW Corner Room.

Operability of which system(s) may be impacted?

A. Division 1 RHR.

B. Division 2 Core Spray.

C. HPCI and Division 2 RHR.

D. RCIC and Division 1 Core Spray.

Answer: C

Answer Explanation:
The candidate must recall that, per 2D105, Reactor Building Corner Rooms / HPCI Room Flood Level, if flooding is apparent in the SW Corner Room, the Division 2 RHR system must be declared inoperable. Since the SW Corner Room and the HPCI Room are connected via the T4500-F601, HPCI Room Floor & Equip Drain Iso Valve, the flooding could either be from Division 2 RHR or the HPCI system. Since the SW Corner Room cannot be access, the candidate must determine that the normally open T4500-F601 valve cannot be closed since valve operation is performed locally. Therefore, the operability of HPCI could also be impacted.

Distractor Explanation:
Distractors are incorrect and plausible because:
A. This distractor is true for flooding in the NW Corner Room; however it is incorrect for flooding in the SW Corner Room.
B. This distractor is true for flooding in the SE Corner Room; however it is incorrect for flooding in the SW Corner Room.
D. This distractor is true for flooding in the NE Corner Room; however it is incorrect for flooding in the SW Corner Room.

Reference Information:

2D105, Reactor Building Corner Rooms / HPCI Room Flood Level

Question Use
Closed Reference
RO

NUREG 1123 KA Catalog Rev. 2
268000 K1.09 2.6/2.8 ECCS systems
268000 Radwaste
268000 K1. Knowledge of the physical connections and/or cause-effect relationship between RADWASTE and the following:

10CFR55 RO/SRO Written Exam Content
10 CFR 55.41(b) (7) Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

NRC Exam Usage
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Fundamental
New
RO

Associated objective(s):
Containment Systems (T2200 & T2300)
Cognitive Enabler
Discuss the Containment Systems interrelationships with other systems.

62	K/A Importance: 3.3/3.9			**Points: 1.00**
R62V2	Difficulty: 3.00	Level of Knowledge: Fundamental	Source: NEW	88867

The plant is operating at 100% power when offgas radiation monitor indications stabilize at a higher than normal value due to fuel cladding failure.

If the offgas system chillers fail to maintain temperature, then ___(1)___ because ___(2)___.

A. (1) offsite radioactivity release rate will RISE
   (2) the efficiency of the offgas charcoal beds will be reduced

B. (1) offgas oxygen concentration will RISE
   (2) the effectiveness of the hydrogen recombiners will be reduced

C. (1) offgas hydrogen concentration will RISE
   (2) the effectiveness of the hydrogen recombiners will be reduced

D. (1) offsite radioactivity release rate will RISE
   (2) the sand filter's efficiency in removing particulates will be reduced

Answer: A

Answer Explanation:
The offgas charcoal bed purpose is to provide holdup time for radioactive gasses removed from the main condenser that would be present in significant amounts following a fuel failure. This holdup time allows for decay of these gasses to reduce their activity rate prior to offsite release, thereby reducing the offsite release rate. The chillers act to reduce the temperature of the gas stream to prevent degradation of the charcoal beds due to excessive temperature. If the temperature of the gas stream entering the charcoal beds were to rise, then their efficiency at providing holdup time will be reduced, allowing radioactive gasses with higher activity to be released.

Distractor Explanation:
B is incorrect since the chillers are downstream of the recombiners and do not play a role in the effectiveness of H2 and O2 recombination. Applicants may choose this if they do not correctly recall the role of the chillers in the offgas process flow.

C is incorrect since the chillers are downstream of the recombiners and do not play a role in the effectiveness of H2 and O2 recombination. Applicants may choose this if they do not correctly recall the role of the chillers in the offgas process flow.

D is incorrect since the sand filters are upstream of the chillers and will not be affected by chiller failure. The sand filters provide particulate filtration, and their ability to do so will not be impacted by a downstream temperature rise. Applicants may choose this if they do not correctly recall the role of the chillers in the offgas process flow.

Reference Information:
OFF GAS ST-OP-315-0035-001 (pg 11-14) - ST information is from basis listed in FSAR

Question Use
Closed Reference
RO

NUREG 1123 KA Catalog Rev. 2
271000 K3.02 Off-site radioactive release rate
271000 Offgas System
271000 K3 Knowledge of the effect that a loss or malfunction of the OFFGAS SYSTEM will have on the following:

10CFR55 RO/SRO Written Exam Content
10 CFR 55.41(b)(11) Purpose and operation of radiation monitoring systems, including alarms and survey equipment.

NRC Exam Usage
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Fundamental
New
RO

Associated objective(s):
Off Gas
Cognitive Terminal
Given the Off Gas system operating conditions/parameters, evaluate those conditions/parameters for the appropriate operator response in accordance with approved plant procedures.

63	K/A Importance: 3.3/3.5			**Points: 1.00**
R63	Difficulty: 3.00	Level of Knowledge: Fundamental	Source: BANK	88789

A fire has resulted in actuation of a wet pipe automatic sprinkler system, the following conditions exist:

- Fire Water Header Pressure lowered to 125 psig and is now steady.
- General Service Water Header Pressure lowered to 105 psig and is now steady.

Which Fire Protection System pump(s) automatically initiated, if any, to maintain Fire Header Pressure?

A. None.

B. ONLY the Electric Fire Pump.

C. ONLY the Diesel Fire Pump.

D. BOTH the Electric and Diesel Fire Pumps.

Answer: B

Answer Explanation:
Per 23.501.01, Section 1.1:

Fire Pumps

The source of water for the Fire Protection (Water) System is the General Service Water Screenwell. A Fire Protection Jockey Pump supplies makeup water to the underground fire loop and keeps it pressurized to approximately 150 psig. The primary supply for the underground fire loop is a 2500 gpm Electric Motor Driven Fire Pump which takes suction from the Screenwell. When in AUTO, the pump will automatically start should system pressure drop below 130 psig. As a supplemental backup, a 2500 gpm Diesel-Driven Fire Pump is also located in the General Service Water Pump House and takes suction from the Screenwell. The Diesel Fire Pump automatically starts when fire header pressure drops to 110 psig with the local controller in AUTO.

Distractor Explanation:
Distractors are incorrect and plausible because:

A. This distractor would be true if Fire header Pressure stabilized above 130 psig since the Jockey Pump normally supplies the fire header and keeps it pressurized to approximately 150 psig. However, since header pressure dropped below 130 psig, the EFP would have started.
C. This distractor would be true if the DFP was the primary supply for the fire header and therefore started when fire header pressure drops below 130 psig. However, the EFP is the primary supply for the fire header and is capable of maintaining fire header pressure when it automatically starts. Since header pressure in the stem did not drop below 110 psig the DFP, which is the supplemental backup to the EFP, did would not have automatically started.
D. This distractor would be true if Fire header Pressure dropped below 110 psig since the EFP (at 130) and DFP (at 110) would both have automatically started to maintain the fire header and keep it pressurized. However, since header pressure stabilized above 110 psig the DFP would not have started.

Reference Information:

23.501.01, FIRE WATER SUPPRESSION SYSTEM

Plant Procedures
07D10

Question Use
Closed Reference
RO

NUREG 1123 KA Catalog Rev. 2
286000 K4.02 Automatic system initiation
286000 Fire Protection System
286000 K4 Knowledge of FIRE PROTECTION SYSTEM design feature(s) and/or interlocks which provide for the following:

10CFR55 RO/SRO Written Exam Content
10 CFR 55.41(b) (8) Components, capacity, and functions of emergency systems.

NRC Exam Usage
ILO 2009 Exam
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Bank
Fundamental
RO

Associated objective(s):
Fire Protection and Detection (P8000)

Cognitive Enabler

Describe general Fire Protection and Detection System operation, including component operating sequence, normal operating parameters, and expected system response.

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64	K/A Importance: 2.6/2.7			**Points: 1.00**
R64V2	Difficulty: 3.00	Level of Knowledge: Fundamental	Source: NEW	88947

The plant is in MODE 1 with the following:

- Division 1 CCHVAC in operation.
- Division 1 EESW is running for chemistry control of the RHRSW reservoir.

Which of the following sets of conditions will require declaring Division 1 CCHVAC INOPERABLE and why?

A. Outside air temperature drops below 68°F due to possible CCHVAC chiller surging and loss of oil pressure.

B. Division 1 EESW TCV is taken to Manual due to the inability to ensure cooling water to the CCHVAC chiller remains >68°F.

C. Outside air temperature drops below 68°F, for 40 minutes or more, due to freon boiling in the CCHVAC evaporator and condensing in the compressor.

D. Division 1 EESW TCV is taken to Manual, for 40 minutes or more, due to the potential for freon boiling in the CCHVAC evaporator and condensing in the compressor.

Answer: B

Answer Explanation:
To answer this question, the candidate must recall 23.413, CCHVAC System SOP, Precaution and Limitation 3.15, which states:

EECW TCV not in AUTO will require declaring the associated CCHVAC chiller inoperable, if operating, due to no temperature control of EECW to ensure chiller remains functional (chiller requires greater than 68°F supply cooling water). Therefore, when the EECW TCV is bypassed the associated Division of CCHVAC must be declared INOPERABLE, while in MODEs 1, 2, or 3, or during movement of recently irradiated fuel in Secondary Containment.

With the above P&L in mind, the candidate must determine that, with Div 1 CCHVAC running and the Division 1 EESW TCV not in AUTO while shutting down Division 1 EESW, it must be declared INOPERABLE.

Distractor Explanation:
Distractors are incorrect and plausible because:

A. 23.127, RBCCW/EECW SOP, P&L 3.8 warns the operator that operation of RBCCW/EECW system with temperatures below 68°F, for prolonged periods of time, may result in surging of the CCHVAC chiller and loss of oil pressure. The candidate could incorrectly relate this P&L with outside air temperature, since outside air temperature can effect RHRSW reservoir temperature and thus RBCCW/EECW temperature, and determine that CCHVAC must be declared INOP with EESW running and outside air temperature below 68°F, which is incorrect.
C. 23.127, RBCCW/EECW SOP and 23.413, CCHVAC System SOP, contain several cautions regarding operating CCHVAC with RBCCW/EECW supply temperatures below 68°F. 23.413 P&L 3.12 concerns operating the CCHVAC chillwater pump, with the chiller shutdown, for >40 minutes due to the potential for causing boiling of freon in the evaporator and condensing in the compressor. The candidate could confuse these two P&Ls in his/her mind and incorrectly determine that outside air temperature dropping below 68°F, for 40 minutes or more, will make CCHVAC INOP, which is not correct.
D. Several P&Ls in both 23.127 and 23.413 warn against operating with the EECW TCV in Manual due to its impact on CCHVAC operability. The candidate could confuse these P&Ls with P&L 3.12 of 23.413 and determine that having the Division 1 EESW TCV in Manual for 40 minutes or more will make CCHVAC INOP due to the potential for freon boiling in the CCHVAC evaporator and condensing in the compressor, which is incorrect. Although having the TCV in Manual will make CCHVAC INOP, it does not require being in Manual for 40 minutes and the reason for it being INOP is incorrect.

Reference Information:

23.413, CCHVAC System SOP
23.127, RBCCW/EECW SOP

Question Use
Closed Reference
RO

NUREG 1123 KA Catalog Rev. 2
290003 K5.03 2.6/2.7 Temperature control
290003 Control Room HVAC
290003 K5. Knowledge of the operational implications of the following concepts as they apply to CONTROL ROOM HVAC:

10CFR55 RO/SRO Written Exam Content
10 CFR 55.41(b)(10) Administrative, normal, abnormal, and emergency operating procedures for the facility.

NRC Exam Usage
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Fundamental
New
RO

Associated objective(s):
Control Center HVAC (T4102)
Cognitive Terminal
Given various controls and indications for Control Center HVAC operations, analyze that indication to determine proper plant response in accordance with approved plant procedures.

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65	K/A Importance: 2.6			**Points: 1.00**
R65V2	Difficulty: 3.00	Level of Knowledge: Fundamental	Source: BANK	89547

The reactor is operating at 100% power.

Due to a logic relay failure, the G3352-F001 RWCU Inboard Isolation valve receives an isolation signal and automatically closes.

Which one of the following describes the effect on the plant if this malfunction cannot be corrected for several weeks?

A. Increased reactor thermal power.

B. Decreased RPV bottom head thermal gradients.

C. Decreased hydrogen buildup in the offgas system.

D. Increased fouling of RPV internal heat transfer surfaces.

Answer: D

Answer Explanation:
From G33-00, RWCU Design Basis Document.
Power Generation: Design Basis of Reactor Water Cleanup is to provide a means for reducing the concentration of radioactive and corrosive species in the Reactor. In addition, Reactor Water Cleanup shall:

- Discharge excess Reactor water during startup, shutdown, and hot standby conditions.
- Minimize Reactor heat loss during system operation.
- Remove solid and dissolved impurities from recirculated Reactor coolant.
- Minimize temperature gradients in the Reactor piping and vessel during periods of low flow rates.

A. Plausible because removal of RWCU results in a reactor thermal power change. Incorrect because it causes thermal power to decrease, not increase, as described in 23.707 P&L 3.3.10.
B. Plausible because minimizing temperature gradients is a function of RWCU during periods of low flow. Incorrect because flow is not low at 100% power.
C. Plausible because RWCU helps to control chemistry in the RPV. Incorrect because RWCU only controls solid and dissolved impurities. Hydrogen is volatile and leaves with the steam.

This question is a K/A match because RO applicants must recall the long term effect of loss of RWCU on internal RPV components without reference to procedures.
References:
G33-00, RWCU Design Basis Document
23.707, RWCU operating procedure.

Question Use
Closed Reference

NUREG 1123 KA Catalog Rev. 2
290002 K6.07 2.6/2.7 RWCU
290002 Reactor Vessel Internals
290002 K6 Knowledge of the effect that a loss or malfunction ofthe following will have on the REACTOR VESSEL INTERNALS :

10CFR55 RO/SRO Written Exam Content

10 CFR 55.41(b) (5)	Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.
10 CFR 55.41(b) (7)	Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

NRC Exam Usage
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Bank
Fundamental
RO

Associated objective(s):

66	K/A Importance: 3.3 / 3.8			**Points: 1.00**
R66V2	Difficulty: 2.00	Level of Knowledge: Fundamental	Source: BANK	89588

10 CFR 55.25 states "If, during the term of the license, the licensee develops a permanent physical or mental condition that causes the licensee to fail to meet the requirements of § 55.21 of this part, the facility licensee shall notify the Commission..."

To ensure that Fermi 2 meets these requirements, MGA13, Fermi Medical Requirements, requires that licensed individuals shall be responsible to immediately notify Medical and their immediate supervisor.
What other individuals (if any) must be immediately notified?

A. Supervisor, Nuclear Licensing

B. Supervisor, Operations Training

C. Supervisor, Radiation Protection

D. No other notifications must be made immediately

Answer: B

Answer Explanation:
MGA 13 Section 2.10.1 states "Licensed individuals shall be responsible to immediately notify Medical, their immediate supervisor, and the Supervisor, Operations Training of any change in medical status."

Distracter Explanation:
A. Plausible because the Licensing department must be notified eventually and MGA13 directs this action by others, but it is incorrect since it is not the operator's responsibility to do so
C. Plausible because the Fermi Medical staff report directly to Rad Protection, and MGA13 directs RP supervision to be notified under certain circumstances, but it is incorrect since it is not the operator's responsibility to do so under these conditions.
D. Plausible because operators may incorrectly believe that no other individuals require IMMEDIATE notification, but it is incorrect since the procedure requires immediate notification of ops training supervision.

Reference Information:
MGA 13 pg 14

Plant Procedures
MGA13

Question Use
Closed Reference
RO

NUREG 1123 KA Catalog Rev. 2

G2.1.4	Knowledge of individual licensed operator responsibilities related to shift staffing, such as medical requirements, œno-solo operation, maintenance of active license status, 10CFR55, etc.

10CFR55 RO/SRO Written Exam Content

10 CFR 55.41(b)(10)	Administrative, normal, abnormal, and emergency operating procedures for the facility.

NRC Exam Usage
ILO 2015 Exam
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Bank
Fundamental
RO

Associated objective(s):

67	K/A Importance: 3.6 / 3.8			**Points: 1.00**
R67	Difficulty: 2.00	Level of Knowledge: Fundamental	Source: BANK	87856

According to MOP06, "Recordkeeping", which one of the following actions could be performed <u>without</u> a subsequent unit log entry by the CRLNO?

A. Start of the third condenser pump during plant startup

B. Changes in mode switch position directed by a plant procedure

C. Recirc Flow increase to maintain power at 100% due to fuel burnout

D. Recirc Flow increase to raise power to 100% after a preplanned turbine test

Answer: C

Answer Explanation:
Per MOP06, Section 2.2.6:

The CRLNO will normally make entries concerning specific plant equipment manipulations. Examples of entries made by the CRLNO are as follows:

1. Generator voltage changes
2. Start and stop in significant changes in Reactor Power (no entry required for adjustments made to maintain power)
3. Reactor criticality data during startup
4. System status changes (for example, startup, shutdown, or system mode changes such as Short Cycle to Long Cycle, etc.)
5. Equipment starts, stops, and trips
6. Independent verification completion not documented in another procedure
7. Changes in the position of Mode switch

Distractor Explanation:

A. Plausible if the examinee believes that logging the startup procedure is sufficient to cover the start of each pump during power ascension. Incorrect because this is required to be logged by #5 above.
B. Plausible if the examinee believes that logging the start and stop of the procedure is sufficient to cover the manipulation of the mode switch contained within the procedure. Incorrect because this is required to be logged by #7 above.
D. Plausible if an examinee believes that logging the start of the test is sufficient to cover any power manipulations made during the testing. Incorrect because this is a significant change of reactor power.

References:
MOP06

Question Use
Closed Reference
RO

NUREG 1123 KA Catalog Rev. 2
G2.1.18 3.6/3.8 Ability to make accurate, clear and concise logs, records, status boards, and reports

10CFR55 RO/SRO Written Exam Content
10 CFR 55.41(b)(10) Administrative, normal, abnormal, and emergency operating procedures for the facility.

NRC Exam Usage
ILO 2019 Retake Exam
ILO 2001 Retake Exam

NRC Question Use (ILO 2019 Retake)
Bank
Fundamental
RO

Associated objective(s):
Administrative Conduct Manuals - Licensed Operators/STA
Performance Terminal
For the following positions, list examples of events that require entry into the Fermi 2 Unit Log:

a. Shift Manager
b. Control Room Supervisor
c. Control Room Nuclear Supervising Operator

68	K/A Importance: 2.8 / 3.7			**Points: 1.00**
R68 - OPTION 1	Difficulty: 2.00	Level of Knowledge: Fundamental	Source: MODIFIED: 2019 FERMI NRC EXAM	89647

Per MOP13, Conduct of Refueling and Core Alterations, the Refuel Floor Supervisor is in charge of what on the Refuel Floor?

A. ALL activities.

B. Core Alterations ONLY.

C. ANY movement of fuel ONLY.

D. Everything EXCEPT Core Alterations.

Answer: B

Answer Explanation:
NOTE: Question was modified to make previously correct answer (D) an incorrect distractor and a previously incorrect distractor (B) the correct answer.

Per MOP 13 Section 3.2.1, the Refuel Floor Supervisor is in charge of Core Alterations.
Therefore, the examinee must recall that the Refuel Floor Coordinator is in charge of ALL refuel floor activities, with the exception of Core Alterations.

Distractor Explanation:
Distractors are incorrect and plausible because:

A. The examinee could determine that all personnel on the refuel floor report to the Refuel Floor Supervisor and thus could conclude that the Refuel Floor Supervisor is in charge of ALL refuel floor activities, which is plausible because the Refuel Floor Supervisor is an SRO who reports directly to the Shift Manager. This in incorrect because MOP 13 Section 3.2.2 states that the Refuel Floor Supervisor shall hold a Senior Reactor Operator (SRO) or a modified (limited to fuel handling) SRO license and the Refuel Floor Supervisor shall have NO other responsibilities or duties other than the supervision of core alterations. Therefore, this is incorrect because the Refuel Floor Supervisor will not be in charge of other Refuel Floor activities that are the responsibility of the Refuel Floor Coordinator.
C. The examinee could determine that any fuel movements are the responsibility of the Refuel Floor Supervisor, including fuel movements in the spent fuel pool and in the reactor, which is plausible because the Refuel Floor Supervisor is an SRO who reports directly to the Shift Manager. This is incorrect because movement of fuel in the spent fuel pool is NOT considered a Core Alteration and thus NOT the responsibility of the Refuel Floor Supervisor as per MOP 13.
D. This distractor is plausible if the candidate confuses the role of the Refuel Floor Supervisor with that of the Refuel Floor Coordinator since MOP 13, Section 3.2.6, states that the Refuel Floor Coordinator is in charge of the Refuel Floor, with the EXCEPTION of core alterations, and should be on the Refuel Floor when major evolutions are in progress. This distractor is incorrect because the Refuel Floor Supervisor is ONLY responsible for core alterations.

Reference Information:

MOP13 – Conduct of Refueling and Core Alterations.

Question Use
Closed Reference
RO

NUREG 1123 KA Catalog Rev. 2
G2.1.41 Knowledge of the refueling process

10CFR55 RO/SRO Written Exam Content
10 CFR 55.41(b)(10) Administrative, normal, abnormal, and emergency operating procedures for the facility.

NRC Exam Usage
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Bank
Fundamental
Modified

Associated objective(s):

69	K/A Importance: 3.9 / 4.3			**Points: 1.00**
R69	Difficulty: 2.00	Level of Knowledge: Fundamental	Source: BANK: 2017 FERMI NRC EXAM	87873

Per MOP05, Control of Equipment, while performing lineups, components located in a locked high radiation area are not required to be verified in correct position if:

A. Lineup is for a non-safety related system.

B. The lineup is being performed on an operating system.

C. The valve is required to be in a throttled position, and flow can be verified from the MCR.

D. Documentation is available showing no personnel entries could affect component positioning.

Answer: D

Answer Explanation:
Operations Conduct Manual MOP05, Control of Equipment, section 3.2 for performing lineups states the following:

3.2.3. For components located in locked high radiation areas, an entry is not required if:
1. Documentation is available showing no personnel entries that could affect component positioning.
2. Previous lineup exists, documenting correct position. Recent lineups are normally kept in the Control Room.

Distractor Explanation:
If an Initial Licensed candidate or Licensed Operator does not fully understand the requirements for performing lineups in a locked high radiation area (LHRA), the distractors could be considered plausible.
A. Plausible because the requirement might seem to apply to only safety related systems to a less than competent operator. Incorrect because no exception to this requirement exists based on this reason.
B. Plausible because valve lineups are normally done on systems in the shutdown (pre-startup) condition. Incorrect because the exception regarding LHRA does not apply to this condition. The method is to have the SM/CRS sign off the valve position if it had been changed by a procedure step. (3.1.1)
C. Plausible because throttle valves would be verified not fully closed by this method. Incorrect because additional requirements exist for throttle valves, which aren't met by a simple flow check.

Reference Information:
MOP05 Sedition 3.2 (pg 20)

Plant Procedures
MOP05 - Control Of Equipment

Question Use
Closed Reference
RO

NUREG 1123 KA Catalog Rev. 2
G2.2.14 Knowledge of the process for controlling equipment configuration or status.

10CFR55 RO/SRO Written Exam Content
10 CFR 55.41(b)(10) Administrative, normal, abnormal, and emergency operating procedures for the facility.

NRC Exam Usage
ILO 2017 Exam
ILO 2019 Retake Exam

NRC Question Use (ILO 2017):
New
RO
Low

NRC Question Use (ILO 2019 Retake)
Bank
Fundamental
RO

Associated objective(s):
Administrative Conduct Manuals - Licensed Operators/STA

Performance Terminal

Describe the requirements that must be followed accessing High Radiation, Locked High Radiation, or Very High Radiation Areas at Fermi 2.

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70	K/A Importance: 3.6/4.5			Points: 1.00
R70	Difficulty: 3.00	Level of Knowledge: Fundamental	Source: MODIFIED: 2019 FERMI NRC EXAM	87876

The following conditions exist:

- ALL RPV Head Closure Bolts are FULLY TENSIONED.
- Reactor Coolant System Temperature is 185°F.
- The Reactor Mode Switch is in SHUTDOWN.

Based on these conditions, which ONE of the following is the correct MODE of operation per Technical Specifications?

A. MODE 2, Startup.

B. MODE 3, Hot Shutdown.

C. MODE 4, Cold Shutdown.

D. MODE 5, Refuel.

Answer: C

Answer Explanation:
Per Technical Specifications Table 1.1-1 MODES (see below):

Table 1.1-1 (page 1 of 1)
MODES

MODE	TITLE	REACTOR MODE SWITCH POSITION	AVERAGE REACTOR COOLANT TEMPERATURE (°F)
1	Power Operation	Run	NA
2	Startup	Refuel[(a)] or Startup/Hot Standby	NA
3	Hot Shutdown[(a)]	Shutdown	> 200
4	Cold Shutdown[(a)]	Shutdown	≤ 200
5	Refueling[(b)]	Shutdown or Refuel	NA

(a) All reactor vessel head closure bolts fully tensioned.

(b) One or more reactor vessel head closure bolts less than fully tensioned.

Therefore, the examinee must recall that, with the Mode Switch in SHUTDOWN and all RPV Head Bolts fully tensioned, note (a) applies and the reactor is considered to be in MODE 4, Cold Shutdown.

Distractor Explanation:
Distractors are incorrect and plausible because:
A. Plausible because this distractor would be true if the Reactor Mode Switch was in the REFUEL or STARTUP/HOT STANDBY position. Incorrect because the correct mode for the listed conditions is Cold Shutdown.
B. Plausible because this distractor would be true if the Temperature was > 200°F. Incorrect because the correct mode for the listed conditions is Cold Shutdown.
D. Plausible because this distractor would be true if the head closure bolts were not fully tensioned. Incorrect because the correct mode for the listed conditions is Cold Shutdown.

Reference Information:

Technical Specifications Table 1.1-1 MODES.

Question Use
Closed Reference
RO

NUREG 1123 KA Catalog Rev. 2
G2.2.35 Ability to determine Technical Specification Mode of Operation

10CFR55 RO/SRO Written Exam Content
10 CFR 55.41(b)(10) Administrative, normal, abnormal, and emergency operating procedures for the facility.

NRC Exam Usage
ILO 2019 Retake Exam

NRC Question Use (ILO 2019)
Bank
Low
RO

NRC Question Use (ILO 2019 Retake)
Fundamental
Modified
RO

Associated objective(s):
Technical Specifications for Licensed Operators
Performance Enabler
Given a set of plant conditions and a copy of Section 1.1 of the Technical Specifications, state the plant MODE.

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71	K/A Importance: 3.2 / 3.7			**Points: 1.00**
R71	Difficulty: 3.00	Level of Knowledge: Fundamental	Source: BANK	87880

An operator is preparing to Inspect equipment located in a HIGH RADIATION AREA while performing rounds per 27.000.05, Operator Rounds, using a GREEN TRIP TICKET.

In accordance with plant administrative procedures, the operator must perform the inspection by which of the following methods?

A. Perform a visual inspection at the barrier to the area.

B. Obtain Radiation Protection verbal approval, then enter the area.

C. Obtain a hand-held radiation monitoring device, then enter the area.

D. Enter the area after the entry is preplanned and maintain dose ALARA.

Answer: A

Answer Explanation:
IAW 27.000.05, Operator Rounds, section 5.7:

5.7 Normal Operator Rounds method for inspecting high radiation and high contamination areas is by visual inspection at the barrier to the area. Look and listen for unusual noises, steam, or leakage.

Distractor Explanation:
B. Plausible because this would be necessary if it is required to enter the inspection area. Incorrect because entry is not allowed for day to day inspections.
C. Plausible because this practice would allow the operator to be aware of dose as he approaches hot spots within the area. Incorrect because entry is not allowed for day to day inspections.
D. Plausible because this would be necessary if it is required to enter the inspection area. Incorrect because entry is not allowed for day to day inspections.

Reference Information:
27.000.05

Question Use
Closed Reference
RO

NUREG 1123 KA Catalog Rev. 2
G2.3.12 Knowledge of radiological safety principles pertaining to licensed operator duties, such as containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.

10CFR55 RO/SRO Written Exam Content
10 CFR 55.41(b)(12) Radiological safety principles and procedures.

NRC Exam Usage
ILO 2012 Exam
ILO 2017 Audit
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Bank
Fundamental
RO

Associated objective(s):
Administrative Conduct Manuals - Licensed Operators/STA
Performance Terminal
Describe the requirements that must be followed accessing High Radiation, Locked High Radiation, or Very High Radiation Areas at Fermi 2.

72	K/A Importance: 3.4 / 3.8			Points: 1.00
R72	Difficulty: 3.00	Level of Knowledge: Higher cognitive level	Source: NEW	89667

The plant is at 100% power.

You are the Shift Foreman.

You are briefing multiple clearance applications which will require the clearance appliers to hang a single tag in each area listed below.

The appliers believe each tag will take the same amount of time to hang.

In which area is it expected that the clearance appliers will recieve the highest dose?

A. Near the standby CRD pump

B. Near the RCIC turbine, in standby

C. Near one of the in-service Condenser Pumps

D. Near one of the in-service Heater Feed Pumps

Answer: C

Answer Explanation:
Each of the pumps referenced is used to pump potentially contaminated primary water. Of these, the heater feed pumps, CRD pumps and RCIC turbine are not in shielded areas of the plant, and pump (or are near) relatively smaller sources of radiation.

The condenser pumps are behind shielded walls, pump a very large volume of water that was recently in the RPV, and are located beneath the condenser, which is a very large radiation source.

Distractor Explanation:

A. Plausible because this pump is in the Aux Building, and accessed from the Reactor Building, and there are multiple cautions within the EOPs regarding radiation exposure while performing some operations regarding the CRD system, such as venting the overpiston area or draining the scram discharge volume. Incorrect because the pump is not pumping water recently in the RPV, and is not near any large radiation sources, nor is it behind shielded walls.
B. Plausible because the RCIC turbine is in the Reactor building, and when in service uses steam released directly from the RPV. Incorrect because while in standby, it is not using, nor pumping, water that had recently been in the RPV, and is not behind shielded walls.
D. Plausible because these pumps have a high volumetric flow rate of water that had fairly recently been in the RPV. Incorrect because they pump water that has passed through the CFDs, and are not near the large radiation source of the condenser.

Reference Information:

Survey maps from surveys performed at 100% power:

- TB1 General area - Shows Heater Feed Pumps area
- AB Sub basement - Shows CRD pumps area
- RB Sub basement, NE quad - Shows RCIC turbine area
- TB Basement - Shows Condenser Pump area

Question Use
Closed Reference
RO

NUREG 1123 KA Catalog Rev. 2
G2.3.14 Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities.

10CFR55 RO/SRO Written Exam Content
10 CFR 55.41(b)(12) Radiological safety principles and procedures.

NRC Exam Usage
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Higher cognitive level
New
NRC Early Review
RO

Associated objective(s):

73	K/A Importance: 4.0 / 4.3			Points: 1.00
R73	Difficulty: 2.00	Level of Knowledge: Fundamental	Source: BANK: 2017 FERMI NRC EXAM	87887

Per MOP01, who is normally assigned the Command and Control Function in the control room during (1) NORMAL and (2) EMERGENCY conditions?

A. (1) CRS
(2) CRS

B. (1) CRS
(2) SM

C. (1) SM
(2) CRS

D. (1) SM
(2) SM

Answer: B

Answer Explanation:
Per MOP01:

3.7.4 There shall be one individual with an active Senior Reactor Operator (SRO) license for Fermi 2 assigned the Command and Control function located in the Main Control Room at-controls or back panel area (see Enclosure A) at all times. During emergency conditions this shall be the SM.

NOTE: The use of CRS in this section refers to the individual with an active SRO license assigned the Command Function.

3.7.5 The CRS should normally be in the At Controls Area or back panel area unless relieved of the command and control function.

The note prior to Section 3.7.5 establishes that the CRS is the person who is normally assigned the Command function within the MCR. Section 3.7.5 establishes that he is to remain within the MCR.

Distractor Explanation:

A. Plausible because the SRO license held by a CRS does make him eligible to be the senior person within the MCR during all conditions, including emergencies. Incorrect because the Fermi administrative procedure designates the SM as the person holding the command and control function within the MCR during an emergency.
C. Plausible because the Shift Manager is at all times the senior SRO assigned to the shift, and the crew takes normal workday direction from the SM as put out in the morning meetings. The CRS role during emergencies includes frequent and important work direction related to the emergency, and it could be thought that this frequent and important work direction means that the CRS has the command and control function. Incorrect because Fermi designates the SM as having the command and control functions during emergencies, and that (during normal operation) that this may be delegated to the CRS.
D. Plausible because the SM is the senior SRO assigned to the shift, and leads the crew. Incorrect because the ongoing command and control function is delegated to the CRS during normal operations.

Reference Information:
MOP01

Plant Procedures
MOP01 Conduct of Operations

Question Use
Closed Reference
RO

NUREG 1123 KA Catalog Rev. 2
G2.4.12 Knowledge of general operating crew responsibilities during emergency operations

10CFR55 RO/SRO Written Exam Content
10 CFR 55.41(b)(10) Administrative, normal, abnormal, and emergency operating procedures for the facility.

NRC Exam Usage
ILO 2017 Exam
ILO 2019 Retake Exam

NRC Question Use (ILO 2017):
New
RO
Low

NRC Question Use (ILO 2019 Retake)
Bank
Fundamental
RO

Associated objective(s):
Introduction to Emergency Operations Procedures
Cognitive Enabler
Explain the functions performed by the Shift Manager, Control Room Supervisor, Nuclear Supervising Operators, Nuclear Operators, and Shift Technical Advisor during implementation of the Emergency Operating Procedures

Intro to AOPs
Cycle 12-4 Objectives
Cognitive Enabler
Describe the following operations expectations as they apply to AOP implementation.
a) Communications
a) Procedure Use and Adherence
b) Control Room Monitoring
c) Conservative Decision Making
d) Control-Not Speed
e) Command and Control
f) Alarm Response
g) Peer/Self Checking
h) Place Keeping

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74	K/A Importance: 3.1			**Points: 1.00**
R74V2	Difficulty: 2.00	Level of Knowledge: Fundamental	Source: BANK: 2017 DAEC NRC EXAM	88967

What is the MINIMUM emergency event classification level in which the Emergency Response Organization (ERO) is REQUIRED to be fully activated?

A. Unusual Event

B. Alert

C. Site Area Emergency

D. General Emergency

Answer: B

Answer Explanation:
Per EP-103, Alert, step 4.3,"The Emergency Operations Facility (EOF), Operational Support Center (OSC), and Technical Support Center (TSC) emergency response facilities are activated with the declaration of an ALERT. The Fermi Emergency Call Out System (ECOS) must be activated and assembly of all personnel within the Protected Area ordered to achieve staffing of the facilities."
EP-102, Unusual Event, contains no such requirement. EP-104, Site Area Emergency, and EP-105, General Emergency, require activation of the ERO but are higher level emergencies than Alert.

Distractor explanations:

A is incorrect since activation of the ERO is not required for UE. Applicants may select this if they incorrectly recall this fact.
C is incorrect since it is not the minimum event requiring ERO activation. Applicants may select this if they incorrectly recall this fact.
D is incorrect since it is not the minimum event requiring ERO activation. Applicants may select this if they incorrectly recall this fact.

References:

EP-102, EP-103, EP-104, EP-105

Question Use
Closed Reference
RO

NUREG 1123 KA Catalog Rev. 2
G2.4.29 Knowledge of the emergency plan

10CFR55 RO/SRO Written Exam Content
10 CFR 55.41(b)(10) Administrative, normal, abnormal, and emergency operating procedures for the facility.

NRC Exam Usage
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Bank
Fundamental
RO

Associated objective(s):
EMERGENCY NOTIFICATIONS - INITIAL QUALIFICATON
Cycle 12-2 Objectives
Performance Terminal
Describe the process of obtaining Off-Site Support.

75	K/A Importance: 3.0 / 4.1			**Points: 1.00**
R75	Difficulty: 3.00	Level of Knowledge: Fundamental	Source: NEW	87888

The plant was initially at 100% power.

A hostile force has penetrated the protected area and has damaged equipment causing multiple safety systems to become degraded.

The SM has declared the appropriate Emergency Action Level, and the E-plan has been activated.

Per MOP01, who on the shift is relied upon to assess plant conditions that result, monitor challenges to the critical safety functions, and through knowledge of plant diverse and flexible coping strategies, provide valuable technical input to support key decisions.

A. SM

B. STA

C. CRS

D. CRNLO

Answer: B

Answer Explanation:
Per MOP01 section 3.1.8:

The STA can also be relied upon to assess plant conditions that result from hostile action-based events or beyond design basis external events. Through the monitoring of challenges to the critical safety functions and through knowledge of plant diverse and flexible coping strategies, individuals performing the STA function can provide valuable technical input to support key decisions.

Distractor Explanation:

A. Plausible because the SM is the senior SRO, and until relied by the TSC, he is the Emergency Director during a declared Emergency Event. Incorrect because his role is as a decision maker, and he is not relied upon to provide technical information to support key decisions.
C. Plausible because the CRS is responsible during emergency events, including a declared emergency, to be aware of challenges to the critical safety functions, and to use the EOPs to address those challenges. Incorrect because the CRS is NOT relied upon to develop or implement flexible coping strategies.
D. Plausible because the CRLNO has the continuing function to monitor the plant conditions at all times, and to make the CRS aware of challenges to safety functions. Incorrect because the CRLNO is NOT relied upon to develop or implement flexible coping strategies.

Reference Information:

MOP01

Question Use
Closed Reference
RO

NUREG 1123 KA Catalog Rev. 2
G2.4.37 Knowledge of the lines of authority during implementation of the emergency plan

10CFR55 RO/SRO Written Exam Content
10 CFR 55.41(b)(10) Administrative, normal, abnormal, and emergency operating procedures for the facility.

NRC Exam Usage
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Fundamental
New
RO

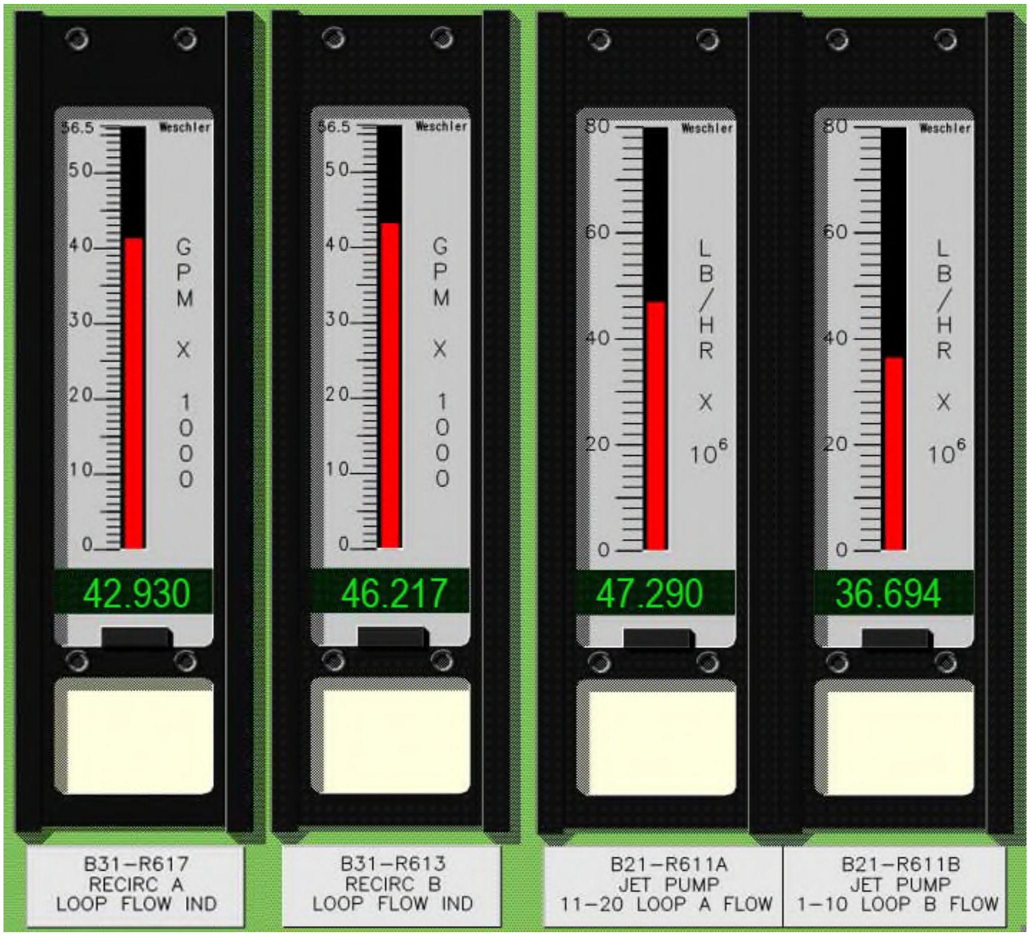
Associated objective(s):
Introduction to Emergency Operations Procedures
Cognitive Enabler
Explain the functions performed by the Shift Manager, Control Room Supervisor, Nuclear Supervising Operators, Nuclear Operators, and Shift Technical Advisor during implementation of the Emergency Operating Procedures

76	K/A Importance: 3.1			Points: 1.00
S76	Difficulty: 3.00	Level of Knowledge: Higher cognitive level	Source: BANK: 2017 FERMI NRC EXAM	87269

The plant was operating at 100% power when a transient is observed.

The plant has stabilized with the following indications:

- Generator Megawatt Output is 1047 Mwe.
- Reactor Power is 91%.
- RPV Level is 197 inches.
- Reactor Pressure is 1017 psig.
- Total core flow 84 Mlbm/hr.



Based on these conditions, (1) What has occurred and (2) which of the following action must the CRS Direct?

A. (1) Jet Pump Failure

(2) Raise South RRMG set speed to match loop jet pump flows

B. (1) Jet Pump Failure

(2) Monitor core for thermal-hydraulic instability using 24.000.01 Att.34B

C. (1) Uncontrolled Recirc Flow Change.

(2) Raise South RRMG set speed to match loop jet pump flows and verify total Core Flow is less than or equal to pre-transient Core Flow.

D. (1) Uncontrolled Recirc Flow Change.

(2) Lower North RRMG set speed to match loop jet pump flows and monitor core for thermal-hydraulic instability using 24.000.01 Att.34B

Answer: B

Answer Explanation:
The candidate must evaluate plant conditions and determine that the increased Recirc Flow and decreased Jet Pump Flow seen in the B RR Loop are consistent with a jet pump failure in the B Loop. The candidate must recall that 20.138.02 requires 24.000.01 Att. 34B be performed to Monitor core for thermal-hydraulic instability per Action A.2 of that procedure.

Distractor Explanation:
A (1) is the correct evaluation. (2) is plausible because B Loop Jet Pump Flows are significantly lower than A and Action A.5 requires the SRO to Evaluate compliance with Jet Pump loop Flow mismatch (LCO 3.4.1 and SR 3.4.1.1). This could lead the candidate to determine that the mismatched loop flows must be corrected by raising B Loop (South) RRMG set speeds. This is incorrect because no such guidance exists or is implied by Action A.5. Furthermore, the crew would evaluate declaring B loop INOPERABLE and removing it from service per Action A.6.
C (1) is plausible because B Recirc Loop Flow is significantly higher than A Recirc Loop Flow, which could lead the candidate to conclude that an uncontrolled RR Flow change (either lowering A loop or raising B loop) has occurred. This is incorrect because the candidate should evaluate further and determine that not only did B RR Loop Flow go up, but B Jet Pump Flows lowered. The deviation in flows is the key indicator which should lead the candidate to recognize a Jet Pump failure and not RR Speed Change. (2) is plausible because the candidate who focused solely on the difference in RR Loop Flows could determine that the correct action to take is from Condition A of 20.138.03, Uncontrolled Recirc Flow Change, which has the CRS direct adjusting speed of the RRMG with the locked scoop tube to match Jet Pump flows and verify total Core Flow is less than pre-transient Core Flow. Although this action is correct for an uncontrolled speed change, it is not correct for a Jet Pump Failure indicated in the stem of the question.
D (1) is plausible because B Recirc Loop Flow is significantly higher than A Recirc Loop Flow, which could lead the candidate to conclude that an uncontrolled RR Flow change (either lowering A loop or raising B loop) has occurred. This is incorrect because the candidate should evaluate further and determine that not only did B RR Loop Flow go up, but B Jet Pump Flows lowered. The deviation in flows is the key indicator which should lead the candidate to recognize a Jet Pump failure and not RR Speed Change. (2) is plausible because the candidate who focused solely on the difference in RR Loop Flows could determine that the correct action to take is from Condition B of 20.138.03, Uncontrolled Recirc Flow Change, which has the CRS direct adjusting RRMG speeds to match Jet Pump flows and monitoring for thermal hydraulic instabilities. Although this action is correct for an uncontrolled speed change, it is not correct for a Jet Pump Failure indicated in the stem of the question.

10 CFR 55.43(b)(5) SRO Justification:
This question meets ES-401 Attachment 2 requirements to be SRO-Only because answering this question requires specific knowledge of AOP 20.138.02, it is not related to immediate actions and the entry conditions are not relevant or leading to the answer. The answer to this questions is based on assessing plant conditions (jet pump failure) and then the correct mitigating action to take.

Reference Information:

20.138.02, Jet Pump Failure AOP.
20.138.03, Uncontrolled Recirc Flow Change.

Plant Procedures
20.138.02

Question Use
Closed Reference
SRO

NUREG 1123 KA Catalog Rev. 2
295001 AA2.04 Individual jet pump flows: Not-BWR-1&2
295001 AA2. Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION :

10CFR55 RO/SRO Written Exam Content
10 CFR 55.43(b) (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

NRC Exam Usage
ILO 2017 Exam
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Bank
Higher cognitive level
SRO

Associated objective(s):
Reactor Recirculation System
Cognitive Enabler
Identify abnormal and emergency operating procedures associated with the Reactor Recirculation system.

77	K/A Importance: 4.4			**Points: 1.00**
S77V4	Difficulty: 3.00	Level of Knowledge: Fundamental	Source: NEW	88887

The plant was operating at 100% power when the following occured:

- The Main Turbine Tripped due to Load Rejection.
- Turbine Stop Valve (TSV) #1 indicates OPEN.
- Turbine Control Valve (TCV) #1 indicates OPEN.

The CRS must enter ___(1)___ and direct action to ___(2)___.

A. (1) 23.109 section 7.8, On-Load Closure of a Turbine Stop Valve

(2) Close #1 TSV to prevent turbine damage due to overspeed

B. (1) AOP 20.000.21, Reactor Scram

(2) Close Outboard MSIVs to prevent turbine damage due to overspeed

C. (1) 23.109 section 7.8, On-Load Closure of a Turbine Stop Valve

(2) Close #1 TSV to prevent over-pressurizing the high pressure turbine casing

D. (1) AOP 20.000.21, Reactor Scram

(2) Close Outboard MSIVs to prevent over-pressurizing the high pressure turbine casing

Answer: B

Answer Explanation:

Answer Explanation:

Following a turbine generator trip from 100% power, a reactor scram will occur. This requires entry into 20.000.21, Reactor Scram. Condition C requires verification of all turbine steam valves closed. If it is determined that both HP steam valves in a single steam line remain open following the trip, then the CRS should direct that OUTBOARD MSIVs should be shut in accordance with action C.2. The basis for this action is to prevent over-speeding the main turbine since the generator will be disconnected.

Distractor Explanation:

A is incorrect because the scram AOP requires that the outboard MSIVs be closed. Closing the TSV using the normal operating procedure is not procedurally directed under turbine trip conditions, nor would it be effective in preventing turbine overspeed due to the time required to perform the actions. Applicants may select this if they incorrectly believe that the OP contains the appropriate direction for closing the TSV under trip conditions.

C is incorrect because the scram AOP requires that the outboard MSIVs be closed. Closing the TSV using the normal operating procedure is not procedurally directed under turbine trip conditions, nor would it be effective in preventing turbine overspeed due to the time required to perform the actions. Additionally, the basis for the action is to prevent overspeed. While the turbine casing may become pressurized over time, the overriding effect is to prevent overspeed since the generator is disconnected. Applicants may select this if they incorrectly believe that the OP contains the appropriate direction for closing the TSV under trip conditions, and incorrectly recall the basis.

D is incorrect because the basis for the action is to prevent overspeed. While the turbine casing may become pressurized over time, the overriding effect is to prevent overspeed since the generator is disconnected. Applicants may select this if they incorrectly recall the basis.

10 CFR 55.43(b)(5) SRO Justification:

This question meets ES-401 Attachment 2 requirements to be SRO-Only because answering this question requires assessment of plant conditions and then selection of a procedure or section of a procedure to mitigate or recover, or with which to proceed, and recalling the basis for that action.

The question cannot be answered solely by knowing “system knowledge”, immediate operator actions, AOP or EOP entry conditions, or the purpose and overall mitigating strategy of the AOPs.

Reference Information:

20.000.21, Reactor Scram

20.000.21 Bases

23.109, Turbine Operating Procedure

NUREG 1123 K/A Match:

295005 (APE 5) Main Turbine Generator Trip / 3

G2.2.44 – 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.

This question is a match to this K/A since the SRO must evaluate system status using control room indications, determine the effect of operator actions, and select the correct procedural direction to achieve the desired mitigating action following a main turbine trip.

Question Use
Closed Reference
SRO

NUREG 1123 KA Catalog Rev. 2

G2.2.44	Ability to interpret control room indications to verify the status and operation of system, and understand how operator actions and directives affect plant and system conditions
295005	Main Turbine Trip

10CFR55 RO/SRO Written Exam Content

10 CFR 55.43(b) (5)	Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

NRC Exam Usage
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Fundamental
New
NRC Early Review
SRO

Associated objective(s):
Turbine Steam
Cognitive Enabler
Identify abnormal and emergency operating procedures associated with the Turbine Steam System.

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78	K/A Importance: 4.1			**Points: 1.00**
S78	Difficulty: 3.00	Level of Knowledge: Higher cognitive level	Source: BANK	87288

The plant is operating at 68% power when the Control Room is evacuated. A reactor scram could not be completed prior to evacuation.

All other plant areas are accessible.

Which of the following is the PREFERRED method of scramming the reactor and determining actions were successful?

A. Open Generator output breaker CF/CM, and verify APRMs indicate decreasing Reactor Power.

B. Open CB2A (C71-P001A) and CB2B (C71-P001B) and verify RPS Testability cabinet de-energized.

C. Turn off circuits 5 & 6 on Dist Cab 2PA-2 and 2PB-2, and verify RPS white Group Scram lights indicate off.

D. Take 2 operable APRM Mode switches out of operate, and verify APRMs indicate decreasing Reactor Power.

Answer: D

Answer Explanation:
The Remote Shutdown Panel does not include reactor shutdown and turbine trip controls because it is assumed that reactor shutdown and turbine trip will be performed prior to leaving the Control Room. If this action is not possible, then reactor shutdown will be initiated by taking two APRMs out of operation per 20.000.19 SD from Outside the Control Room.

Distractor Explanation:
A is plausible; opening the breakers is an action from 20.000.18 ,Dedicated SD, and would result in a reverse power turbine trip and thus a reactor scram. However, conditions given in the stem of the question do not indicate a need to enter 20.000,18 so this method is neither specified in 20.000.19 nor is it the preferred method.
B is plausible and directed from Condition C but only if unable to scram from the preferred locations (Control Room or Relay Room). This is incorrect because the preferred method is the scram from the Relay Room.
C is plausible; action is required to remove power to SRVs to prevent a uncontrolled RPV blowdown for control of the plant from the dedicated shutdown panel based on fire in 3L zone.

10 CFR 55.43(b)(5) SRO Justification:
This question meets ES-401 Attachment 2 requirements to be SRO-Only because answering this question requires the SRO to assess plant conditions and then select the section of the procedure with which to proceed.

The question cannot be answered solely by knowing systems knowledge, immediate operator actions, AOP/EOP entry conditions, or the purpose, overall sequence of events or mitigative strategy of a procedure.

Reference Information:

20.000.19, Shutdown from Outside the Main Control Room.
20.000.18, Control of the Plant from the Dedicated Shutdown Panel.

Plant Procedures
20.000.19

Question Use
Closed Reference
SRO

NUREG 1123 KA Catalog Rev. 2
295016 AA2.01 4.1*/4.1* Reactor power
295016 Control Room Abandonment
295016 AA2. Ability to determine and/or interpret the following as they apply to CONTROL ROOM ABANDONMENT :

10CFR55 RO/SRO Written Exam Content
10 CFR 55.43(b) (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

NRC Exam Usage
ILO 2015 Exam
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Bank
Higher cognitive level
SRO

Associated objective(s):
Remote Shutdown System (C3500)

Cognitive Enabler

Identify abnormal and emergency operating procedures associated with the Remote Shutdown System.

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79	K/A Importance: 4.1			Points: 1.00
S79	Difficulty: 4.00	Level of Knowledge: Higher cognitive level	Source: NEW	87287

The plant operating at 100% power with all plant equipment available for the current MODE. A plant cleaner slips in front of the H11-P601 panel and causes the following:

- 1D80 DIV I/II EECW/EESW SYS IN MANUAL OVERRIDE alarms.
- The Division 1 key associated with this alarm breaks off inside the keylock switch.

What is the status of Division 1 RHR Suppression Pool Cooling?

A. OPERABLE; EECW can be MANUALLY initiated in an emergency.

B. OPERABLE; EECW will still AUTOMATICALLY initiate in an emergency.

C. INOPERABLE; Conditions and Required Actions of its LCO are NOT required to be entered if a loss of Safety Function does NOT exist.

D. INOPERABLE; Conditions and Required Actions of its LCO are required to be entered REGARDLESS of the result of the Safety Function Determination.

Answer: C

Answer Explanation:
The SRO candidate should recall that, per ARP 1D80, DIV I/II EECW/EESW SYS IN MANUAL OVERRIDE, when the EECW Div 1(2) Isolation Valves Keylock switch is in the OVERRD position, this prevents automatic actuation of the EECW System, which will results in a loss of Division 1 EECW OPERABILITY and will result in a loss of Div 1 EECW if an accident condition (LOCA or LOP) were to occur.
The candidate should recall that, per the Bases for SR 3.7.2.5, automatic isolation of system valves and start of EECW/EESW pumps is needed to support system OPERABILITY.
The candidate should also recall that the TS Definition of OPERABILITY states "A system, subsystem, division, component; or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, division, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s)."

Since Division 1 EECW is not capable of performing its safety function, systems supported by Division 1 EECW, such as Division 1 TC, is INOPERABLE.

The candidate should then recall that LCO 3.0.6 allows an exception to LCO 3.0.2 and, "When a supported system LCO is not met solely due to a support system LCO not being net; the Conditions and Required Actions associated with this supported system are not required to be entered. Only the support system LCO ACTIONS are required to be entered" as long as a loss of safety function does not exist.

Distractor Explanation:
Distractors are incorrect and plausible because:
A. The candidate could fail to recall that automatic initiation of EECW/EESW is required per SR 3.7.2.5, which is plausible (and SRO Only knowledge) because the bases for LCO 3.7.2 does not state that automatic initiation capability is needed for system OPERABILITY. This distractor is incorrect because SR 3.0.1 states that "Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO" thus making its support systems, like the TC mode of RHR, INOPERABLE.
B. The candidate could fail to recall the impact of operating the switch in the stem of the question and determine that the system will automatically initiate with the given conditions, which is plausible because some manual override switches (the Containment Spray Mode Select Switch in the RHR system for example) do NOT override signals until the signal is already present. This is incorrect for this switch application and EECW will not automatically initiate, as specified in ARP 1D80, rendering it INOPERABLE, thus making its support systems, like the TC mode of RHR, INOPERABLE
D. The candidate could fail to recall that LCO 3.0.6 allows an exception to LCO 3.0.2 thus allowing the supported system LCO Actions to be deferred.

10 CFR 55.43(b)(2) SRO Justification:
This question meets ES-401 Attachment 2 requirements to be SRO-Only because answering this question requires application of TS Section 3 and surveillance requirements actions in accordance with rules of application requirements in TS section 1.

The question cannot be answered by knowing <1-hour TS/TRM Actions, information "above the line", or TS safety limits.

Reference Information:

1D80, DIV I/II EECW/EESW SYS IN MANUAL OVERRIDE.
LCO 3.7.2.
LCO 3.0.6

Question Use
Closed Reference
SRO

NUREG 1123 KA Catalog Rev. 2
G2.4.31 Knowledge of annunciators alarms, indications, or response procedures
295018 Partial or Complete Loss of Component Cooling Water

10CFR55 RO/SRO Written Exam Content
10 CFR 55.43(b) (2) Facility operating limitations in the technical specifications and their bases.

NRC Exam Usage
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Higher cognitive level
New
SRO

Associated objective(s):
Reactor Building Closed Cooling Water Emergency Equipment Cooling Water (P4200/P4400)
Cognitive Enabler
Describe the Reactor Building Closed Cooling Water/Emergency Equipment Cooling Water System technical specification limiting conditions for operation, their bases, the associated surveillance requirement(s), and their relationship to operability.

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80	K/A Importance: 3.9			Points: 1.00
S80	Difficulty: 3.00	Level of Knowledge: Higher cognitive level	Source: BANK	87518

Stem information.

You are the CRS. The plant is executing RPV Control with the following conditions:

- Reactor Water Level 175 inches and rising
- Torus Pressure 1 psig
- Core Spray 1 pump injecting at 2500 gpm
- RHR 1 pump injecting at 7500 gpm
- Torus water level -90 inches
- Torus water temperature 180° F

Which one of the following actions will you direct to ensure there is no Core Spray or RHR pump damage?

A. Raise Core Spray flow to 3400 gpm and shutdown the RHR pump.

B. Raise Core spray flow to 3600 gpm and shutdown the RHR pump.

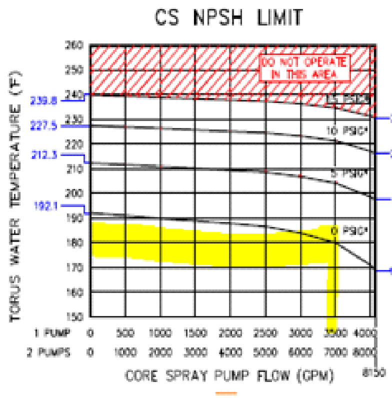
C. Raise Core Spray flow to 3400 gpm and increase RHR flow to 10,000 gpm.

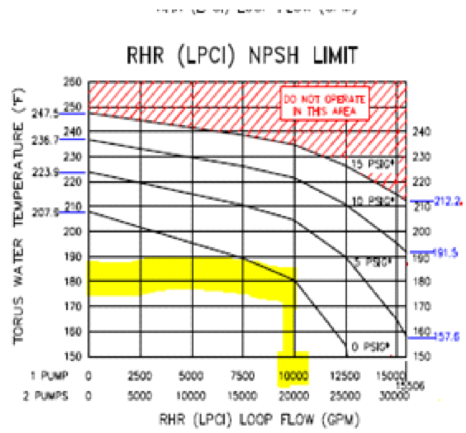
D. Raise Core spray flow to 3600 gpm and increase RHR flow to 10,000 gpm.

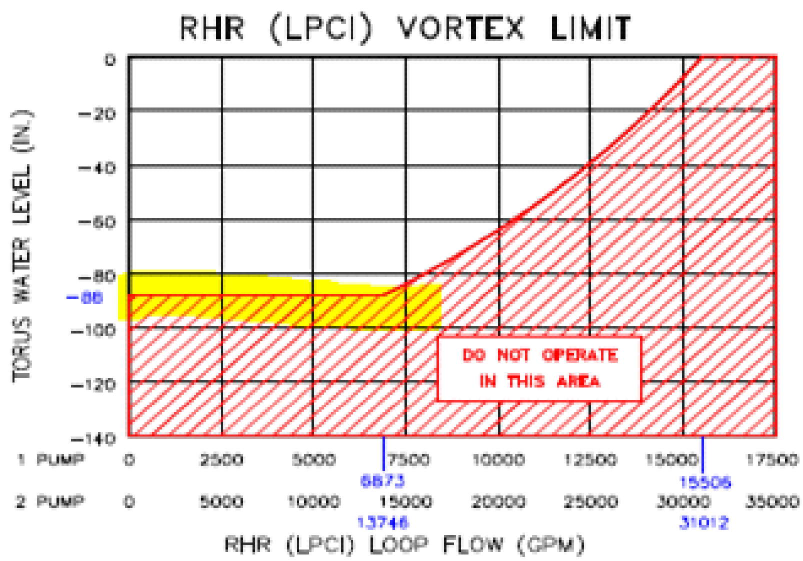
Answer: A

Answer Explanation:

Per 29.100.01, Sheet 6, Cautions, Curves and Tables, the Torus Overpressure calculation must be used to determine where on the curves to operate (1 psig + 3.5 psig + -90 inches/30)= 1.5 psig. No interpolating is allowed therefore the NPSH curves will use the 0 psig curve. The Torus water temperature and pressure results in 3,400 gpm for core spray. However a torus level of -90 inches is below the RHR vortex limit. Since water level is above TAF and RPV level is rising the CRS should direct securing the RHR pump(s).







Distractor Explanation:

Distractors are incorrect and plausible because:

B. Plausible because core spray flow of 3,600 gpm would be allowed if interpolating of the curves was used. RHR should be secured based on exceeding the operable range on the vortex limit.
C. Plausible because core spray flow should be raised to maintain or restore RPV level, this value is allowed based on using the 0 psig curve of the CS NPSH limit. RHR flow of 10,000 gpm is allowed if the vortex limit was not exceeded.
D. Plausible because core spray flow of 3,600 gpm would be allowed if interpolating of the curves was used. RHR flow of 10,000 gpm is allowed if the vortex limit was not exceeded.

10 CFR 55.43(b)(5) SRO Justification:

The question (SRO Knowledge) is not systems knowledge, is not an immediate operator action, is not an entry condition for AOP/EOP, and is not purpose or mitigative strategy of the procedure.
It is knowledge of when and how to implement attachments in the EOP network.

Reference Information:

29.100.01, Sheet 6, Cautions, Curves and Tables

ILT Retake Reference Provided:

29.100.01, Sheet 6, Cautions, Curves and Tables

Question Use
Open Reference
SRO

NUREG 1123 KA Catalog Rev. 2
295030 EA2.02 3.9/3.9 Suppression pool temperature
295030 Low Suppression Pool Water Level
295030 EA2. Ability to determine and/or interpret the following as they apply to LOW SUPPRESSION POOL WATER LEVEL :

10CFR55 RO/SRO Written Exam Content
10 CFR 55.43(b) (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

NRC Exam Usage
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Bank
Higher cognitive level
SRO

Associated objective(s):
Cautions, Curves, and Calculations
Cognitive Enabler
Describe the potential consequences of operating beyond the limits delineated in each of the following graphs/curves:

a. Boron Injection Initiation Temperature Curve
b. Core Spray Net Positive Suction Head (NPSH) Limit
c. Core Spray Vortex Limit
d. Heat Capacity Limit
e. Deleted
f. RHR Low Pressure Coolant Injection (LPCI) NPSH Limit
g. RHR LPCI Vortex Limit
h. RPV Saturation Temperature
i. SRV Tail Pipe Level Limit
j. Drywell Spray Initiation Limit
k. Pressure Suppression Pressure
l. Primary Containment Pressure Limit
m. HPCI NPSH Limit
n. RCIC NPSH Limit

81	K/A Importance: 4.5			Points: 1.00
S81	Difficulty: 3.00	Level of Knowledge: Fundamental	Source: NEW	87308

You are the Control Room Supervisor (CRS). The plant scrammed and is experiencing an ATWS with the following:

- Reactor power is 30% and lowering.
- SLC is injecting with tank level 68" and lowering.
- Manual control rod insertion is in progress.
- RPV level is 90" and steady.
- RPV pressure is 970 psig and steady.

Under which of the following conditions will you direct performance of 20.000.21, Reactor Scram AOP?

A. SLC tank indicates empty.

B. 178 Control Rods Full-In, 7 at Position 02.

C. 184 Control Rods at Position 02, 1 Full-Out.

D. Reactor power is below the Point of Adding Heat.

Answer: B

Answer Explanation:
Per 29.100.01, Sheet 1A FSQ leg, when it is determined that the Rx will remain S/D under ALL conditions w/o boron, the CRS will direct boron injection be terminated and AOP 20.000.21, Reactor Scram entered. 178 control rods full-in and 7 at position 02 satisfies the first part of the first statement in FSQ-OR1, thus requiring the CRS to direct that SLC be shutdown (terminate boron injection for reactivity) and 20.000.21 performed.

Distractor Explanation:
Distractors are incorrect and plausible because:

A. Step FSQ-20 directs shutdown of SLC pumps when SLC tank level indicates empty at FSQ-19. Since shutting down the SLC pumps (terminating boron injection) in FSQ-OR1 is closely tied to directing performance of 20.000.21, it is plausible that the SRO candidate could determine that, when SLC pumps are shut down due to an empty SLC tank, 20.000.21 is also directed. However, step FSQ-20 does not require the CRS to direct 20.000.21 when SLC is shut down for an empty tank.
C. 184 control rods at Position 02 with only one Full-Out is plausible because it seems to satisfy Shutdown Margin requirements. However, SDM at Fermi 2 is only satisfied if (1) all control rods except 1 are fully inserted or (2) all control rods are inserted to or beyond the Maximum Subcritical Banked Withdrawal Position (MSBWP) of 02. Even though 184 control rods are at this position, the 1 rod beyond this position means that shutdown margin is not satisfied.
D. The second statement of FSQ-OR1 requires the CRS to direct 20.000.21 when it is determined that the Rx is S/D with no boron injection. Since this statement is satisfied if power is <POAH at the start of an ATWS, the SRO candidate could determine that this override is taking any time power drops <POAH. However, this is incorrect because taking this override requires power <POAH on control rod insertion alone, hence the statement “with no boron injection."

10 CFR 55.43(b)(5) SRO Justification:
This question meets ES-401 Attachment 2 requirements to be SRO-Only because answering this question requires assessment of emergency plant conditions and then selection of the conditions under which a separate procedure must be entered to proceed.

The question cannot be answered solely by knowing systems knowledge, immediate operator actions, entry conditions into the EOPs or the overall mitigative strategy of the EOPs.

Reference Information:

29.100.01, Sheet 1A, RPV Control - ATWS.

Question Use
Closed Reference
SRO

NUREG 1123 KA Catalog Rev. 2

G2.4.8	Knowledge of how abnormal operating procedures are used with EOPs
295037	SCRAM Condition Present and Reactor Power Above APRM Downscale or unknown.

10CFR55 RO/SRO Written Exam Content

10 CFR 55.43(b) (5)	Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

NRC Exam Usage
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Fundamental
New
SRO

Associated objective(s):
Senior Reactor Operator
Cycle 12-2 Objectives
Performance Enabler
Identify the appropriate EOP / AOP based on plant conditions.

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82	K/A Importance: 3.6 / 4.0			**Points: 1.00**
S82V2	Difficulty: 3.00	Level of Knowledge: Higher cognitive level	Source: NEW	88907

The plant is in MODE 1.

The Main Control Room has just been notified of the following by the System Operations Center (SOC):

- Southeast Michigan is experiencing degraded grid conditions due to high winds and loss of several large power generators.
- If Fermi 2 were to trip off line, 345kV mat POST TRIP VOLTAGE would not be sufficient to sustain operability of safety-related loads.

What is the CRS required to perform?

Declare __(1)__ offsite circuit(s) INOPERABLE and enter LCO __(2)__.

A. (1) One
(2) 3.8.1, AC Sources - Operating

B. (1) Two
(2) 3.8.1, AC Sources - Operating

C. (1) One
(2) 3.8.7, Distribution Systems - Operating

D. (1) Two
(2) 3.8.7, Distribution Systems - Operating

Answer: A

Answer Explanation:
Per 20.300.GRID the candidate must recall the action of Condition B, which is required when 345kV and/or 120kV mat POST TRIP VOLTAGEs is/are not be sufficient to sustain operability of safety-related loads. The candidate must recall that Action B.1 requires appropriate offsite circuits be declared INOPERABLE. The candidate must then determine that the conditions in the stem of the question dictate that the 345kV offsite circuit be declared INOPERABLE per NOTE 2, so only 1 offsite circuit is INOPERABLE.
The candidate must then recall that condition B.2 requires entry into LCO 3.8.1 as per Notes 1 and 2 and as spelled out in ODE-12.

Distractor Explanation:
Distractors are incorrect and plausible because:

B. The candidate could determine that two offsite circuits are INOPERABLE because either (1) the conditions in the stem require declaring both the 120 kV and 345 kV offsite circuits INOPERABLE or (2) since the 345 kV mat contains two incoming offsite lines (Brownstown #2 and Brownstown #3), these TWO offsite circuits are INOPERABLE. (1) is incorrect because, as specified in TS Bases for LCO 3.8.1, the 120 and 345kV offsite circuits are "electrically and physically separated" so the condition of the 345 kV mat specified in the stem does not impact operability of the other offsite circuit. (2) is incorrect because, although the 345 kV mat is fed from 2 transmission lines, both of those lines satisfy the definition (along with the breakers, transformers, interconnecting wiring, etc.) of just one offsite power circuit.
C. The candidate could determine that inoperability of the offsite circuit renders the 4160V ESF buses supplied by that circuit INOPERABLE, therefore LCO 3.8.7 must be entered or the candidate could incorrectly recall Action B.2 and determine that LCO 3.8.7 must be entered. This is incorrect because Action B.2 requires entry into LCO 3.8.1.
D. The candidate could determine that two offsite circuits are INOPERABLE because either (1) the conditions in the stem require declaring both the 120 kV and 345 kV offsite circuits INOPERABLE or (2) since the 345 kV mat contains two incoming offsite lines (Brownstown #2 and Brownstown #3), these TWO offsite circuits are INOPERABLE. (1) is incorrect because, as specified in TS Bases for LCO 3.8.1, the 120 and 345kV offsite circuits are "electrically and physically separated" so the condition of the 345 kV mat specified in the stem does not impact operability of the other offsite circuit. (2) is incorrect because, although the 345 kV mat is fed from 2 transmission lines, both of those lines satisfy the definition (along with the breakers, transformers, interconnecting wiring, etc.) of just one offsite power circuit. The candidate could determine that inoperability of the offsite circuit renders the 4160V ESF buses supplied by that circuit INOPERABLE, therefore LCO 3.8.7 must be entered or the candidate could incorrectly recall Action B.2 and determine that LCO 3.8.7 must be entered. This is incorrect because Action B.2 requires entry into LCO 3.8.1

10 CFR 55.43(b)(5) SRO Justification:
This question meets ES-401 Attachment 2 requirements to be SRO-Only because answering this question requires assessment of plant conditions and then selection of a procedure (20.300.GRID) and section (Condition B) with which to proceed. Specifically, the question requires the SRO to implement Actions B.1 (Declare appropriate offsite circuits INOPERABLE, which includes knowing Note 2 to determine what is "appropriate") and Action B.2 to enter the correct Tech Spec.

The question cannot be answered solely by knowing systems knowledge, immediate operator actions or entry conditions into the AOP nor by knowing the purpose, overall sequence of events, or mitigative strategy of the procedure.

Reference Information:

20.300.GRID, Grid Disturbance AOP.
ODE-12.
LCOs 3.8.1 and 3.8.7 and their Bases.

Question Use
Closed Reference
SRO

NUREG 1123 KA Catalog Rev. 2

700000 AA2.07	Operational status of engineered safety features
700000	Generator Voltage and Electric Grid Disturbances
700000 AA2.	Ability to determine and/or interpret the following as they apply to GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES:

10CFR55 RO/SRO Written Exam Content

10 CFR 55.43(b) (5)	Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

NRC Exam Usage
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Higher cognitive level
New
SRO

Associated objective(s):
120/345KV Switchyards (C3600, S1400, S2000 & S3100)

Cognitive Enabler

Describe the 120/345KV Switchyards technical specification limiting conditions for operation, their bases, the associated surveillance requirement(s), and their relationship to operability.

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83	K/A Importance: 4.0			**Points: 1.00**
S83	Difficulty: 3.00	Level of Knowledge: Higher cognitive level	Source: NEW	87252

You are the Control Room Supervisor (CRS).
The plant has entered the EOPs due to a large steam leak in the Drywell. Primary Containment Conditions are:

- Drywell Temperature is 250°F and rising.
- Drywell Pressure is 4 psig and rising.

Which of the following will you direct and what is the basis for directing that action?

A. Isolate EECW to and from the Drywell to prevent a loss of EECW due to a rupture inside the Drywell.

B. Spray the Drywell to prevent exceeding component qualification and structural design temperature limits of the Drywell.

C. Emergency Depressurize due to exceeding component qualification and structural design temperature limits of the Drywell.

D. Operate all available Drywell Cooling to prevent exceeding component qualification and structural design temperature limits of the Drywell.

Answer: A

Answer Explanation:
Per reference, basis for the correct answer

EECW piping is not qualified for temperatures exceeding 242°F. At this temperature, EECW is isolated to preserve the availability of the EECW system by isolating it from the Drywell to protect from losing the system if EECW piping were to rupture inside the Drywell.

Distractor Explanation:
Distractors are incorrect and plausible because:

B. This distractor is plausible because step DWT-6 directs spraying the Drywell before 242°F is reached and Drywell temperature in the stem is 250°F. This is incorrect because the DWSIL limit has been exceeded, which will prevent operation of Drywell Sprays.
C. This distractor is plausible because Drywell Temperature is high in the stem of the question and approaching the point where ED will be required for the reason given in this distractor. However, the point has not yet been reached so step DWT-10 is not satisfied and ED is not yet required.
D. This distractor is plausible because, per step DWT-3, when it is determined that drywell temperature cannot be maintained below the higher of the drywell temperature LCO or the maximum normal operating temperature, the general direction to operate Drywell Cooling in step DWT-1 is supplemented with the explicit instruction to place into operation all available methods by which drywell cooling can be effected. This is incorrect, however, because drywell cooling cannot be placed in service since it must be isolated at step DWT-5 when DW Temperature exceeds 242°F.

10 CFR 55.43(b)(5) SRO Justification:
This question meets ES-401 Attachment 2 requirements to be SRO-Only because answering this question requires knowledge of diagnostic steps and decision points that require the SRO to assess plant conditions and then select the section of the procedure with which to proceed.

The question cannot be answered solely by knowing systems knowledge, immediate operator actions, AOP/EOP entry conditions, or the purpose, overall sequence of events or mitigative strategy of a procedure.

Reference Information:

29.100.01, Sheet 2, Primary Containment Control.
BWROG EPGs/SAGs, Appendix B section on DW/T

Question Use
Closed Reference
SRO

NUREG 1123 KA Catalog Rev. 2
G2.4.18 Knowledge of the specific bases for EOPs
295012 High Drywell Temperature

10CFR55 RO/SRO Written Exam Content
10 CFR 55.43(b) (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

NRC Exam Usage
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Higher cognitive level
New
SRO

Associated objective(s):

Reactor Building Closed Cooling Water Emergency Equipment Cooling Water (P4200/P4400)

Cognitive Enabler

Identify abnormal and emergency operating procedures associated with the Reactor Building Closed Cooling Water/Emergency Equipment Cooling Water System.

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84	K/A Importance: 3.6			Points: 1.00
S84	Difficulty: 2.00	Level of Knowledge: Fundamental	Source: NEW	87267

The plant is operating at 100% power when the running CRD Pump trips.

When would the first Control Rod Scram Accumulator(s) be considered INOPERABLE?

When CRD __(1)__ pressure drops below __(2)__ psig.

A. (1) Accumulator
(2) 940

B. (1) Accumulator
(2) 1300

C. (1) Charging Water
(2) 940

D. (1) Charging Water
(2) 1300

Answer: A

Current View

Answer Explanation:
Per TS 3.1.5, in Mode 1 as indicated in the stem of the question, all Control Rod Scram Accumulators are required to be OPERABLE. Per TS Bases for LCO 3.1.5, OPERABILITY of the scram accumulators is based on maintaining adequate accumulator pressure. The SRO candidate must recall that per SR 3.1.5.1 the accumulators are OPERABLE when accumulator pressure is ≥940 psig, making the first accumulator INOPERABLE at when its pressure drops <940 psig.

Distractor Explanation:
Distractors are incorrect and plausible because:

B. 1300 psig is the pressure at which 3D5, CRD Charging Water Pressure Low alarms and the candidate could relate this alarm, and its associated pressure, with the value that accumulator pressure must be maintained above to satisfy OPERABILITY and thus conclude that an accumulator is considered INOPERABLE below this pressure. However, this distractor is incorrect since SR 3.1.5.1 defines accumulator operability being >940 psig.
C. 940 psig is referenced in TS Required Action B.1 as the condition under which plant operators have 20 minutes to restore charging header pressure >940 psig if two or more accumulators are INOPERABLE. The candidate could recall this number from TS and determine that, when charging header pressure drops <940 psig, this is the time when the accumulators are INOPERABLE. This is incorrect because although 940 psig charging header pressure does impact Actions taken for an INOPERABLE accumulator, charging header pressure, by itself, does not define accumulator operability.
D. 1300 psig is the pressure at which 3D5, CRD Charging Water Pressure Low, alarms and the candidate could relate this alarm, and its associated pressure, with dropping below normal charging header pressure and thus conclude that all accumulators are considered INOPERABLE. This is incorrect because although charging header pressure is in Tech Specs, it does not, by itself, affect accumulator operability.

10 CFR 55.43(b)(2) SRO Justification:
This question meets ES-401 Attachment 2 requirements to be SRO-Only because answering this question requires application of required actions and SRs for TS LCO 3.1.5.

The question cannot be answered solely by knowing ≤1-hour TS/TRM Actions, or information listed “above the line,” or the TS safety limits.

Reference Information:

LCO 3.1.5 Control Rod Scram Accumulators and Bases
3D5, CRD Charging Water Pressure Low

Question Use
Closed Reference
SRO

NUREG 1123 KA Catalog Rev. 2
295022 AA2.01 3.5/3.6 Accumulator pressure
295022 AA2. Ability to determine and/or interpret the following as they apply to LOSS OF CRD PUMPS :
295022 Loss of Control Rod Drive Pumps

10CFR55 RO/SRO Written Exam Content
10 CFR 55.43(b) (2) Facility operating limitations in the technical specifications and their bases.

NRC Exam Usage
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Fundamental
New
SRO

Associated objective(s):
Control Rod Drive Hydraulics (C1150)

Cognitive Enabler

Describe the Control Rod Drive Hydraulic system technical specification limiting conditions for operation, their bases, the associated surveillance requirement(s), and their relationship to operability.

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85	K/A Importance: 3.7 / 3.9			**Points: 1.00**
S85	Difficulty: 3.00	Level of Knowledge: Fundamental	Source: NEW	87348

Failure of __(1)__ would require entry into LCO 3.3.3.1 PAM Instrumentation.

PAM instruments are provided __(2)__.

A. (1) T50-N406A PCAM PC TORUS WTR LVL MON DIV1 LVL XMTR
(2) as an input to the protection system to cause automatic actions.

B. (1) E41-N062B HPCI HI LVL SIGNAL TO SUPR POOL SUCT VLV LVL TRANSMITTER
(2) as an input to the protection system to cause automatic actions.

C. (1) T50-N406A PCAM PC TORUS WTR LVL MON DIV1 LVL XMTR
(2) to cue the operating crew to perform preplanned manual actions.

D. (1) E41-N062B HPCI HI LVL SIGNAL TO SUPR POOL SUCT VLV LVL TRANSMITTER
(2) to cue the operating crew to perform preplanned manual actions.

Answer: C

Answer Explanation:
Per 23.601, page 54, E41-N062B is a TS instrument that has input into TS 3.3.5.1 only for the high torus level HPCI suction swap. This instrument is not a PAMS instrument.

Per 44.030.156 LCO 3.3.3.1-1 function 4 is impacted by isolation of T50-N406A/B.

The PAM Torus water level instruments are T50-N406A / B.

Per TS 3.3.3.1 Bases:

The PAM instrumentation LCO ensures the OPERABILITY of Regulatory Guide 1.97, Type A variables so that the control room operating staff can:

- Perform the diagnosis specified in the Emergency Operating Procedures (EOPs). These variables are restricted to preplanned actions for the primary success path of Design Basis Accidents (DBAs), (e.g., loss of coolant accident (LOCA)), and
- **Take the specified, preplanned, manually controlled actions for which no automatic control is provided**, which are required for safety systems to accomplish their safety function.

Distractor Explanation:
Distractors are incorrect and plausible because:
A. Plausible because there are Torus water level instruments that have an input into the protection system to initiate automatic actions. Incorrect because the instruments that have that function are not PAM instruments.
B. Plausible because E41-N062B is a torus water level instrument subject to Tech Specs. Further plausible because it is a Torus water level instrument that has an input into the protection system to initiate automatic actions. Incorrect because this instrument is not a PAM instrument, and in fact has no indications available in the MCR. Also incorrect because the function of PAM instrument is to provide the indication needed for operators to perform pre-planned actions, not as an input to the protection system.
D. Plausible because E41-N062B is a torus water level instrument subject to Tech Specs. Incorrect because this instrument is not a PAM instrument, and in fact has no indications available in the MCR.

10 CFR 55.43(b)(2) SRO Justification:
This question meets ES-401 Attachment 2 requirements to be SRO-Only because it requires the SRO to have knowledge of TS Bases information required to analyze TS-required actions and terminology in that he must know the reason for an instrument to be categorized as a PAM instrument.

Reference Information:

TS 3.3.3.1 Bases
23.601
44.030.156

Question Use
Closed Reference
SRO

NUREG 1123 KA Catalog Rev. 2
G2.4.3 3.7/3.9 Ability to identify post-accident instrumentation
295029 High Suppression Pool Water Level

10CFR55 RO/SRO Written Exam Content
10 CFR 55.43(b) (2) Facility operating limitations in the technical specifications and their bases.

NRC Exam Usage
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Fundamental
New
SRO

Associated objective(s):
Containment Systems (T2200 & T2300)
Cognitive Enabler
Describe the Containment Systems technical specification limiting conditions for operation, their bases, the associated surveillance requirement(s), and their relationship to operability.

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86	K/A Importance: 3.9/4.3			Points: 1.00
S86	Difficulty: 2.00	Level of Knowledge: Higher cognitive level	Source: BANK	87349

The plant was operating at 75% power when a trip of the South RFP occurred. Current conditions are as follows:

- Recirc pump speeds lower 37%
- Reactor power 63%
- SBFW .... injecting at 600 gpm
- RPV level .... 196" and stable

HPCI logic then malfunctions causing an automatic initiation of HPCI. Which of the following actions would the CRS direct in accordance with 20.107.01, Loss of Feedwater or Feedwater Control?

A. Raise North RFP speed

B. Stop injecting with SBFW

C. Perform a Rapid Power Reduction.

D. Place the Mode Switch in Shutdown.

Answer: D

Answer Explanation:
Per 20.107.01 Loss of Feedwater or Feedwater Control:
OVERRIDE:

IF	**HPCI initiates with SBFW also in operation.**	**THEN**	**Place Reactor Mode Switch in SHUTDOWN.**

Distractor Explanation:
Distractors are incorrect and plausible because:

A. Is incorrect because of above and is plausible because it is an Action from 20.107.01 if recirc runback occurs.
B. Is incorrect because of above and is plausible because removing SBFW may allow level to be maintained with the injection from HPCI.
C. Is incorrect because of above and is plausible because it is an Action from 20.107.01; however, the override takes precedence.

10 CFR 55.43(b)(5) SRO Justification:
This question meets ES-401 Attachment 2 requirements to be SRO-Only because the question cannot be answered solely by knowing "systems knowledge". The question cannot be answered solely by knowing immediate operator actions. The question cannot be answered solely by knowing entry conditions for AOPs or plant parameters that require direct entry into major EOPs. The question cannot be answered solely by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure. It does require the SRO to assess plant conditions (HPCI initiates with SBFW also in operation during a loss of feedwater) and then direct an AOP override to mitigate those plant conditions.

Reference Information:
20.107.01

Plant Procedures
20.107.01

Question Use
Closed Reference
SRO

NUREG 1123 KA Catalog Rev. 2

206000 A2.17	HPCI inadvertent initiation: BWR-2,3,4
G2.4.49	Ability to perform without reference to procedures those actions that require immediate operation of system components and controls.
206000	HPCI System.
206000 A2.	Ability to (a) predict the impacts of the following on the HIGH PRESSURE COOLANT INJECTION SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:

10CFR55 RO/SRO Written Exam Content

10 CFR 55.43(b) (5)	Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

NRC Exam Usage
ILO 2015 Exam
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Bank
Higher cognitive level
SRO

Associated objective(s):
Reactor Feedwater (N2100)

Cognitive Enabler

Identify abnormal and emergency operating procedures associated with the Reactor Feedwater system.

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87	K/A Importance: 4.5			Points: 1.00
S87	Difficulty: 3.00	Level of Knowledge: Higher cognitive level	Source: NEW	87527

- The plant is in MODE 5 with the Reactor Vessel Head removed.
- Emergent work to replace 48/24V DC Battery 2IA is in progress.
- To support this work, a non-qualified spare battery has been connected to keep loads normally powered by this battery energized.

The temporary power supply requires IRMs to be declared inoperable within __(1)__ trip system(s).

If testing which requires IRMs to be operable were to begin now, RPS Trip Capability __(2)__ be maintained.

A. (1) both
(2) would

B. (1) both
(2) would NOT

C. (1) ONLY the A
(2) would

D. (1) ONLY the A
(2) would NOT

Answer: D

Answer Explanation:
Per 23.310, 48/24V DC Electrical System, Section 8.0, Placing Temporary Batteries in Service, while temporary batteries are in service, the following instrumentation is considered inoperable (only IRMs are listed below): With 2IA-1 or 2IA-2 supplied by temporary batteries: C51-K601A, C, E, and G, IRM A, C, E, and G. With 2IB-1 or 2IB-2 supplied by temporary batteries: C51-K601B, D, F, and H, IRM B, D, F, and H.

The SRO candidate should recall that IRMs A, C, E and G are part of RPS Trip System A, therefore all 4 IRMs supporting Trip System A are INOPERABLE.

The candidate should recognize that condition C (One or more Functions with RPS trip capability not maintained) would prevent maintenance that requires IRMs to be operable as currently configured. This requires an understanding of what it means to maintain trip capability.

- A Function is considered to be maintaining RPS trip capability when sufficient channels are OPERABLE or in trip (or the associated trip system is in trip), such that both trip systems will generate a trip signal from the given Function on a valid signal.

During fuel loading, normally installed shorting links can be removed. When removed, the coincidence on all neutron monitoring trips is changed to "one of any." If shorting links had been removed for this question, any one IRM Upscale would cause a Reactor Scram.

With all four IRMs associated with the A RPS system INOPERABLE, trip capability is not maintained since the IRM Upscale trip requires trip inputs from both the "A" and "B" trip systems to cause a Reactor Scram.

Distractor Explanation:
Distractors are incorrect and plausible because:
A. Plausible because there are four IRMs per trip system. It would be possible to have two of these instruments powered from the "A" 48/24V DC system, and the other two powered from the "B" 48/24V DC system. If this were the case, the second half of the answer would be correct as each system would still be capable of generating a reactor trip. Incorrect because only the "A" RPS instruments are affected.
B. Plausible because there are four IRMs per trip system. It would be possible to have two of these instruments powered from the "A" 48/24V DC system, and the other two powered from the "B" 48/24V DC system. Furthermore, if the definition of "RPS Trip Capability" was not understood, an SRO could determine that all three of the required instruments needed to be OPERABLE for trip capability to be maintained. Incorrect because only the "A" RPS instruments are affected.
C. Plausible because the with the shorting links removed, IRM Upscale trip is 1 of any and any of the operable "B" RPS instruments could cause a Reactor Scram, independent of the "A" RPS instruments. When shorting links are installed, this trip is 1 of 4 taken twice, similar to other trips. Incorrect because there is no information in the stem of the question to indicate that shorting links have been removed.

10 CFR 55.43(b)(2) SRO Justification:
This question meets ES-401 Attachment 2 requirements to be SRO-Only because answering this question requires knowledge of TS bases that is required to analyze
TS-required actions and terminology.

The question cannot be answered solely by knowing 1-hour or less TS/TRM actions, LCO/TRM information listed “above the line,” or TS safety limits.

Reference Information:

23.310, 48/24V DC Electrical System.
SD-2530-17 48/24V Distribution.
I-2155-06 RPS Trip System A System Relays.
I-2155-07 RPS Trip System B System Relays.

Question Use
Closed Reference
SRO

NUREG 1123 KA Catalog Rev. 2

G2.2.36	Ability to analyze the effect of maintenance activities such as degraded power sources, on the status of limiting conditions for operations
215003	IRM System

10CFR55 RO/SRO Written Exam Content
10 CFR 55.43(b) (2) Facility operating limitations in the technical specifications and their bases.

NRC Exam Usage
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Higher cognitive level
New
SRO

Associated objective(s):
Intermediate Range Monitoring (C5111)
Cognitive Enabler
Describe the normal and alternate power supplies to Intermediate Range Monitoring System components.

Reactor Protection System (C7100)
Cognitive Enabler
Identify Reactor Protection System related technical specifications, with emphasis on action statements requiring prompt actions (for example, one hour or less).

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88	K/A Importance: 3.1 / 3.2			**Points: 1.00**
S88	Difficulty: 3.00	Level of Knowledge: Higher cognitive level	Source: NEW	87447

You are the Control Room Supervisor (CRS).

The plant scrammed from 100% power due to a Station Blackout (SBO)

Plant conditions:

- RPV level is being controlled by RCIC at 193"
- HPCI is unavailable.
- Transformer #64 has just been energized by CTG 11-1.

The following then occurs:

- The low side tap to E51N003, "RCIC PUMP DISCHARGE FLOW TRANSMITTER" has ruptured, and has been isolated with the low side depressurized.
- The CRLNO attempts to take manual control of the RCIC turbine with E51-K615, "RCIC PUMP FLOW CONTROLLER", and reports he is unable to transfer the controller to manual.

(1) What is the result on ACTUAL RCIC flow due to this condition?
(2) What action will you direct?

A. (1) RCIC flow rises to maximum.

(2) Verify E5150-F045 "RCIC TURB STM INLET ISO MOV" closes at 214" per 20.000.23, "HIGH RPV WATER LEVEL."

B. (1) RCIC flow rises to maximum.

(2) Intermittently Shutdown and then Restart RCIC per 23.206, "REACTOR CORE ISOLATION COOLING SYSTEM" to control RPV level in the band 173" - 214"

C. (1) RCIC turbine lowers to idle speed, with no system flow.

(2) Start SBFW per 20.300.SBO and control RPV water level in the band 173" - 214"

D. (1) RCIC turbine lowers to idle speed, with no system flow.

(2) Emergency Depressurize the RPV per 20.100.01 Sh 3 and control level per 20.100.01 Sh 1.

Answer: C

Answer Explanation:
The fault indicated causes measured RCIC flow to increase to a maximum with maximum dP across the flow element.

As a result, the RCIC turbine control will always see actual RCIC flow above demanded flow and work to reduce the turbine speed.

With actual RCIC flow low, the SRO will have to determine the correct procedure path. With XFRM 64 energized, and SBFW required, 20.300.SBO condition K is applicable. The electrical bus feeding the SBFW pumps will be energized, and SBFW will be used to restore and maintain RPV level.

With no indication of a LOCA, it will take more than 30 minutes to reach a level which would require the crew to emergency depressurize. With CTG 11-1 already running, and XFMR 64 already energized, there is ample time available to control level with SBFW. Assumptions to the contrary would also be contrary to the bases of the SBFW system, and would thus be incorrect.

Distractor Explanation:
Distractors are incorrect and plausible because:

A. Plausible because part 1 would be correct if the high side tap had ruptured and had been isolated. The action taken would shutdown RCIC, and would restart RCIC at the L2 setpoint. Incorrect because the indicated flow rises, not the actual flow.
B. Plausible because part 1 would be correct if the high side tap had ruptured and had been isolated. The action taken in part 2 would be a valid method to control RPV water level in the event of a failed controller with RCIC flow at maximum. Incorrect because the indicated flow rises, not the actual flow.
D. Plausible because the first part is correct, and without any injection into the RPV, ED would eventually be required. There are situations in the EOPs that require ED even at RPV levels near normal, such as with two areas of secondary containment with temperature in excess of max safe. Incorrect because ED is not performed until after RPV level is < 0", which will be > 30 minutes from the conditions presented in the stem if SBFW was not used to restore RPV level control.

10 CFR 55.43(b)(5) SRO Justification:
This question meets ES-401 Attachment 2 requirements to be SRO-Only because it cannot be answered solely with systems knowledge, or solely by knowing immediate actions, or solely by knowing entry conditions for AOPs or plant parameters that require immediate entry into major EOPs, or solely by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure. It does require the SRO to assess plant conditions and then select a section of a procedure to mitigate the situation by recovering a source of injection to the RPV.

Reference Information:

20.300.SBO
23.206
20.000.23

Question Use
Closed Reference
SRO

NUREG 1123 KA Catalog Rev. 2
217000 A2.11 Inadequate system flow
217000 RCIC System
217000 A2. Ability to (a) predict the impacts of the following on the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:

10CFR55 RO/SRO Written Exam Content
10 CFR 55.43(b) (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

NRC Exam Usage
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Higher cognitive level
New
SRO

Associated objective(s):
Reactor Core Isolation Cooling
Cognitive Enabler
Discuss failure modes of RCIC System controls and vital instruments, including design features that could result in erroneous operation or indication.

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89	K/A Importance: 3.6/4.6			Points: 1.00
S89	Difficulty: 3.00	Level of Knowledge: Higher cognitive level	Source: BANK	87273

The plant is in MODE 2, STARTUP, following a Refueling Outage. Engineering has determined that ALL Inboard and Outboard MSIVs have had unqualified valve control manifolds installed during outage maintenance which has caused stroke times to EXCEED THE MAXIMUM ALLOWED VALUE.

Which ONE of the following is the MINIMUM Required Action, if any, AND the REASON for that action per Technical Specifications?

A. NO ACTIONS are required, because Primary Containment Isolation capability is OPERABLE.

B. NO ACTIONS are required, because Primary Containment Isolation capability is NOT REQUIRED to be OPERABLE in MODE 2.

C. It is REQUIRED to SHUT ONLY ONE MSIV in EACH Main Steam Line. The basis for this action is to limit the MAXIMUM Radiological Release following a Design Basis Accident.

D. It is REQUIRED to SHUT ONLY ONE MSIV in EACH Main Steam Line. The basis for this action is to limit the severity of the MAXIMUM Reactor Pressure spike following a spurious MSIV closure at power.

Answer: C

Answer Explanation:
Per 3.6.1.3 It is REQUIRED to SHUT ONLY ONE MSIV in each Main Steam Line. The basis for this action is to limit the MAXIMUM Radiological Release following a Design Basis Accident and verified by SR 3.6.1.3.5

Distractor Explanation:
Distractors are incorrect and:
A is plausible; would be true if the Maintenance error resulted in SHORTER stroke times over the cycle or the candidate assumes stroke times do not effect operability.
B is plausible; would be true for MODE 4, COLD SHUTDOWN.
D is plausible; is NOT the reason sited in TS basis for the action.

10 CFR 55.43(b)(2) SRO Justification:
This question meets ES-401 Attachment 2 requirements to be SRO-Only because answering this question requires knowledge of TS bases that is required to analyze TS-required actions and terminology.

The question cannot be answered solely by knowing 1-hour or less TS/TRM actions, LCO/TRM information listed “above the line,” or TS safety limits.

Reference Information:
TS 3.6.1.3
TSB 3.6.1.3

Objective Link: LP-OP-804-0001-0012

Question Use
Closed Reference
SRO

NUREG 1123 KA Catalog Rev. 2
G2.2.37 Ability to determine operability and/or availability of safety related equipment
223002 PCIS/NSSS

Technical Specifications
3.6.1.3 Primary Containment Isolation Valves (PCIVs)

10CFR55 RO/SRO Written Exam Content
10 CFR 55.43(b) (2) Facility operating limitations in the technical specifications and their bases.

NRC Exam Usage
ILO 2008 Exam
ILO 2017 Audit
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Bank
Higher cognitive level
SRO

Associated objective(s):
Nuclear Boiler System (B1100, B2100, B2103, B2104, N1100 & N3017)
Cognitive Enabler
Identify Nuclear Boiler system related technical specifications, with emphasis on action statements requiring prompt actions (for example, one hour or less).

90	K/A Importance: 3.2			Points: 1.00
S90	Difficulty: 3.00	Level of Knowledge: Higher cognitive level	Source: NEW	88287

You are the Control Room Supervisor (CRS). The plant is operating at 100% power with operators attempting to isolate a ground on the Division II ESF battery. The following sequence of events was performed for EVERY load supplied by Division II ESF DC electrical power distribution:

1. Breaker opened.
2. Ground verified to still exist.
3. Breaker re-closed.

The next item to be performed is to isolate the Division II ESF Battery Charger 2B-1. When 2PB-2 Ckt. 8, Div 2 130 VDC Battery Charger, is opened the ground CLEARS.

Which of the following will you direct next and within what allowable time?

A. Place Spare Battery Charger 2B1-2 in service within 4 hours.

B. Place Spare Battery Charger 2B1-2 in service within 12 hours.

C. Restore the 2PB-2 130VDC Distribution Cabinet to service within 2 hours.

D. Restore the 2PB-2 130VDC Distribution Cabinet to service within 16 hours.

Answer: A

Answer Explanation:
Per SD-2530-11, the candidate must predict the impact of opening the DC output breaker of ESF Battery Charger 2B-1 and determine that it removes charging from one half (130 VDC 2B-1) of the Division II 260 VDC ESF Battery, 2PB. The candidate must recognize that, since the ground cleared when this breaker was open, the ground is not on the battery or battery busses, but located within the battery charger itself. Therefore, the candidate must determine that the correct course of action is to direct placing spare battery charger 2B1-2 in service, which is the correct course of action per 23.309, in place of grounded charger 2B-1. Finally, the candidate must recall that this action satisfies Action A of LCO 3.8.4, DC Sources – Operating, and that it must be directed within 4 hours.

Distractor Explanation:
Distractors are incorrect and plausible because:

B. The candidate could recognize that the ground is on the battery charger and determine that the correct course of action is to direct placing spare battery charger 2B1-2 in service, in place of grounded charger 2B-1. However, the candidate could incorrectly recall that this action must be directed within 12 hours, which is plausible because that is the allowable time to complete Action C of LCO 3.8.4, which is incorrect because the action must be complete within 4 hours.
C. The candidate could incorrectly determine that the ground is on the battery or battery bus, which is plausible if the candidate fails to recall how the battery charger is tied to the bus as shown on SD-2530-11. If this determination is made, the candidate could determine that the correct action is to correct the ground and restore the distribution cabinet to operable status by reclosing the breaker in the stem of the question in 2 hours, which is plausible because if a DC distribution system is inoperable it must be restored to operable within 2 hours per Action B of LCO 3.8.7, Distribution Systems – Operating. This is incorrect because indications provided in the stem of the question are that the ground is location on the battery charger, not the distribution cabinet.
D. The candidate could incorrectly determine that the ground is on the battery or battery bus, which is plausible if the candidate fails to recall how the battery charger is tied to the bus as shown on SD-2530-11. If this determination is made, the candidate could recall that the correct action is to restore the distribution cabinet to operable status in 16 hours, which is plausible because that is the allowable time to complete Action C of LCO 3.8.7, Distribution Systems – Operating. This is incorrect because indications provided in the stem of the question are that the ground is location on the battery charger, not the distribution cabinet.

10 CFR 55.43(b)(2) and (5) SRO Justification:
This question meets ES-401 Attachment 2 requirements to be SRO-Only because answering this question requires assessment of plant conditions and selection of a procedure section (restoration of battery charger vice restoration of distribution cabinet) with which to proceed. The question then requires the candidate to apply TS required actions from TS Section 3, which is SRO Only.

The question cannot be answered solely by knowing "systems knowledge," immediate operator actions, entry conditions of EOPs/AOP, or knowing the purpose, overall sequence of events or mitigative strategies of a procedure. The question cannot be answered solely by knowing ≤1-hour TS/TRM Actions, LCO/TRM information listed “above the line,” or TS safety limits.

Reference Information:

SD-2530-11 One Line Diagram 260/130V ESS Dual Battery 2PB Distribution – Div II.
LCO 3.8.4, DC Sources – Operating.
LCO 3.8.7, Distribution Systems – Operating

Question Use
Closed Reference
SRO

NUREG 1123 KA Catalog Rev. 2
263000 A2.01 2.8/3.2 Grounds
263000 DC Electrical Distribution
263000 A2. Ability to (a) predict the impacts of the following on the DC Electrical Distribution; and (b) based on those predictions, use procedures to correct, control or mitigate the consequences of those abnormal conditions or operations:

10CFR55 RO/SRO Written Exam Content
10 CFR 55.43(b) (2) Facility operating limitations in the technical specifications and their bases.
10 CFR 55.43(b) (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

NRC Exam Usage
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Higher cognitive level
New
SRO

Associated objective(s):
DC Electrical Distribution (R3200 & S3102)
Cognitive Enabler
Describe the DC Electrical Distribution System technical specification limiting conditions for operation, their bases, the associated surveillance requirement(s), and their relationship to operability.

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91	K/A Importance: 4.7			Points: 1.00
S91	Difficulty: 2.00	Level of Knowledge: Higher cognitive level	Source: NEW	87428

Two weeks ago, while in MODE 1, control rod 18-31 was discovered to be stuck at position 48. All procedural and Tech Spec required actions were taken, and the plant continued full power operation.

It is now discovered that control rod 38-31 is stuck at position 06. The plant is still in MODE 1.

Which one of the following is the required action?

A. Be in MODE 3 within 12 hours.

B. Disarm control rod 38-31, then continue power operation.

C. Initiate action to fully insert all insertable control rods immediately.

D. Perform 24.106.01, Operable Control Rod Check, then continue power operation.

Answer: A

Answer Explanation:
Per TS 3.1.3 condition B, action B.1:
In Mode 1 or 2, If two or more withdrawn control rods are stuck, then be in MODE 3 within 12 hours.

In this instance, two withdrawn control rods are determined to be stuck while in MODE 1, therefore, action B.1 must be taken.

Distractor Explanation:

B. Disarming the stuck rod allows continued power operation if ONLY one withdrawn control rod is stuck. Applicants may select this if they incorrectly assume that multiple stuck rods may be allowed at power.

C. This action is only required by tech specs when SDM is not within limits per TS 3.1.1, and does not apply when 2 control rods are stuck. Applicants may select this if they identify the wrong TS required actions for this condition.

D. Performing the surveillance, verifying separation criteria, and disarming the stuck rod allow continued power operation if ONLY one withdrawn control rod is stuck. Applicants may select this if they incorrectly assume that multiple stuck rods may be allowed at power.

10 CFR 55.43(b)(2) SRO Justification:
This question meets ES-401 Attachment 2 requirements to be SRO-Only because answering this question requires application of required actions (TS Section 3) in accordance with rules of application requirements (TS Section 1).

The question cannot be answered by RO tech spec knowledge only.

K/A Match Justification:

This question matches the selected K/A since SRO applicants must recall and apply tech specs for the CRDM system without references.

Reference Information:

Fermi TS 3.1.3, condition B, action B.1.

Question Use
Closed Reference
SRO

NUREG 1123 KA Catalog Rev. 2
G2.2.40 Ability to apply technical specifications for a system
201003 CRDM

10CFR55 RO/SRO Written Exam Content
10 CFR 55.43(b) (2) Facility operating limitations in the technical specifications and their bases.

NRC Exam Usage
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Higher cognitive level
New
SRO

Associated objective(s):
Control Rod Drive Mechanism (B1104, C1101 & C1102)

Cognitive Enabler

Describe the Control Rod Drive Mechanism technical specification limiting conditions for operation, their bases, the associated surveillance requirement(s), and their relationship to operability.

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92	K/A Importance: 3.1			**Points: 1.00**
S92	Difficulty: 2.00	Level of Knowledge: Higher cognitive level	Source: NEW	87347

You are the CRS. The plant has scrammed due to High Drywell Pressure caused by an instrument line break inside the Drywell.

The instrument line leakage has caused temperature near the RPV Level Instrument Reference Leg vertical runs inside the Drywell to reach 170°F.

You should inform the crew that this will cause __ (1) __ level instrument(s) to indicate higher than actual and the minimum indicated level for this(these) instrument(s) is __ (2) __ inches?

A. (1) Flood Up
(2) 175

B. (1) Flood Up
(2) 191

C. (1) Wide Range
(2) 21

D. (1) Wide Range
(2) 27

Answer: B

Answer Explanation:
Per 29.100.01, Sheet 6 Curves, Cautions and Tables, the SRO candidate should recognize that the rising drywell temperature will affect the Flood Up Level Instruments. Next, the candidate should refer to Caution 1, Part 3, Table C and determine that the Minimum Indicated Level, for the Flood Up Level Instruments at 170°F, is 191".

Distractor Explanation:
Distractors are incorrect and plausible because:

A. Part (1) is correct. Part (2) is plausible because it is the MIL that is first encountered on the table and the candidate could either misread the table or see that temperature is above the MRT of 150°F and determine that the MIL is 175", which is incorrect for the given temperature.
C. The candidate could fail to recognize that the only RPV level instrument impacted by rising Drywell Temperatures is the Flood Up Level Instrument and instead go to the second table and determine that Wide Range level instruments will be affected, which is incorrect. If the second table is entered, part (2) is plausible because 21" is the MIL for the MRT of 170° given in the stem of the question.
D. The candidate could fail to recognize that the only RPV level instrument impacted by rising Drywell Temperatures is the Flood Up Level Instrument and instead go to the first table and determine that Wide Range level instruments will be affected, which is incorrect. If the first table is entered, part (2) is plausible because 27" is the MIL for the MRT of 170° given in the stem of the question.

10 CFR 55.43(b)(5) SRO Justification:
At Fermi 2, this is SRO knowledge because, although the ROs are taught the cautions, curves and their bases, actual implementation of cautions and curves that require use of the flowcharts is an SRO function.

This question meets ES-401 Attachment 2 requirements to be SRO-Only because answering this question requires knowledge of when to implement attachments to the EOPs, including how to coordinate these items with procedure steps.

The question cannot be answered by knowing systems knowledge, immediate operator actions, entry conditions, or the purpose, overall sequence of events, or mitigative strategy of the EOPs.

Reference Information:

29.100.01, Sheet 6 Cautions Curves and Tables

ILT Retake References Provided:

29.100.01, Sheet 6 Cautions Curves and Tables

Question Use
Open Reference
SRO

10CFR55 RO/SRO Written Exam Content
10 CFR 55.43(b) (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

NRC Exam Usage
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Higher cognitive level
New
SRO

Associated objective(s):
Cautions, Curves, and Calculations
Exam Objectives
RPV Instrumentation

Cognitive Enabler

For any given Reactor Pressure Vessel level instrument determine:

a. The maximum reference leg run area temperatures.
b. The minimum indicated level for each instrument.
c. Location of the variable and reference legs in the Secondary Containment.
d. Location of the transmitter rack in the Secondary Containment.

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93	K/A Importance: 4.2			Points: 1.00
S93	Difficulty: 3.00	Level of Knowledge: Higher cognitive level	Source: NEW	87350

You are the Control Room Supervisor (CRS).

The plant is operating at 80% power in preparation for performing a Rod Pattern Adjustment.

Shortly after turnover, the following conditions are observed:

- RPV Pressure rises by 3.5 psig.
- Pressure System Fault white light comes on.
- Pressure System Unhealthy white light comes on.

(1) Which alarm is consistent with these plant conditions?
(2) What action will you direct?

A. (1) 4D53, AVR General Alarm.
(2) Lower power to reduce generator field current to 2400 amps.

B. (1) 4D53, AVR General Alarm.
(2) Perform 54.000.07, Core Performance Parameter Check.

C. (1) 4D91, Electric Governor Trouble.
(2) Lower power to reduce generator field current to 2400 amps.

D. (1) 4D91, Electric Governor Trouble.
(2) Perform 54.000.07, Core Performance Parameter Check.

Answer: D

Answer Explanation:
The SRO candidate should recognize that the indications given in the stem of the question are indicative of a pressure regulator failure and determine that (1) 4D91 Electric Governor Trouble will actuate. The candidate should then recall that the alarm response procedure for 4D91 requires entry into 20.109.02, Reactor Pressure Controller Failure. The candidate should determine that, with power at 80% and the MSR and Turbine Bypass Valves in service, it is necessary to Perform 54.000.07, Core Performance Parameter Check and Notify Rx Engineering per Action F.1.

Distractor Explanation:
Distractors are incorrect and plausible because:

A. (1) 4D53 is plausible due to the close relationship between the Governor/Pressure Regulator and the Voltage Regulator in controlling turbine load and determine that failure of the controlling AVR caused the indications shown. (2) If the candidate determined that failure of the controlling AVR occurred, then it is plausible the candidate could determine that reducing power to reduce field current below 2400 amps is required since that action is required for various failures related to the AVR as specified in the NOTE on Page 1 of ARP 4D53 and the table starting on Page 8.
B. (1) 4D53 is plausible due to the close relationship between the Governor/Pressure Regulator and the Voltage Regulator in controlling turbine load and determine that failure of the controlling AVR caused the indications shown. (2) It is plausible that the candidate could determine that 4D53 alarmed but still think that it is necessary to enter 20.109.02, Reactor Pressure Controller Failure AOP, and perform Action F.1.
C. (1) 4D91 is the correct alarm for these conditions. (2) Even though the candidate determined the correct alarm in part (1), due to the close relationship between the AVR and the Electric Governor, the candidate could incorrectly determine that the correct action is to reduce generator loading to 2400 field amps to prevent a generator trip, which is incorrect for the conditions shown.

10 CFR 55.43(b)(5) SRO Justification:
This question meets ES-401 Attachment 2 requirements to be SRO-Only because answering this question requires assessment of plant conditions and then selection of a procedure and procedure section with which to proceed.

The question cannot be answered solely by knowing systems knowledge, immediate operator actions, entry conditions, or the purpose, overall sequence of events, or overall mitigative strategy of a procedure.

Reference Information:

20.109.02, Reactor Pressure Controller Failure.
4D53, AVR General Alarm.
4D91, Electric Governor Trouble

Question Use
Closed Reference
SRO

NUREG 1123 KA Catalog Rev. 2
G2.4.46 Ability to verify that the alarms are consistent with the plant conditions
241000 Reactor/Turbinie Pressure Regulating System

10CFR55 RO/SRO Written Exam Content
10 CFR 55.43(b) (6) Procedures and limitations involved in initial core loading, alterations in core configuration, control rod programming, and determination of various internal and external effects on core reactivity.

NRC Exam Usage
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Higher cognitive level
New
NRC Early Review
SRO

Associated objective(s):
Governor/Pressure Control (N3012)
Cognitive Enabler
Identify abnormal and emergency operating procedures associated with the Governor/Pressure Control System.

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94	K/A Importance: 2.9/3.9			**Points: 1.00**
S94	Difficulty: 3.00	Level of Knowledge: Fundamental	Source: MODIFIED: 2019 FERMI NRC EXAM	87248

The plant is operating at 100% power on night shift with the following shift assignments:

Date: Today	
	Nights
SM	SRO1
CRS	YOU
CRLNO	RO1
COP H11-P603	RO2
Shift Foreman	** RO3
Turbine Bldg	* NO1
Reactor Bldg	* NO2
Outside/Fermi 1	NO3
Radwaste Op-Assigned	# NO4
Dedicated Shutdown NO	NO5
Other	* NO6
Other	* NO7
Other	@ Fire Protection Inspector 1 (FB Only)

* Fire Brigade Member ** Fire Brigade Leader # CR Communicator
@ Fire Brigade Qualified Fire Protection Inspector

At 0300 Fire Protection Inspector 1 has a medical emergency and leaves site.

As the Control Room Supervisor, which of the following actions, if any, are required by MOP10, Fire Brigade?

A. Fire Brigade manning is at minimum manning, no action is required.

B. Fire Brigade manning is below minimum, inform NO5 that he is now on the fire brigade.

C. Fire Brigade manning is below minimum, however, it is acceptable to wait until turnover at 0700.

D. Fire Brigade manning is below minimum, and cannot be filled from personnel already on shift. Start calling for a Fire Brigade member to arrive by 0500.

Answer: A

Answer Explanation:
Per MOP10, Section 3.1:

3.1.1 A Fire Brigade of at least 5 members shall be maintained onsite at all times. The Fire Brigade shall not include the SM, Security, or members of the minimum shift crew necessary for safe shutdown of the unit or any personnel required for other essential functions during a fire emergency. Typical Fire Brigade should consist of:

1. One Fire Brigade Leader.
2. Four Operators (or three Operators and a Fire Brigade Qualified Fire Protection Inspector).
3. One other qualified member of plant staff as communicator.

3.1.2 The Fire Brigade composition may be less than the minimum requirements for a period not to exceed two hours to accommodate unexpected absences of Fire Brigade members if immediate action is taken to restore the Fire Brigade to its minimum requirements.

Therefore, the SRO examinee must conclude the although the fire protection inspector is qualified to be a member of the fire brigade, he was not assigned to the fire brigade before his departure, and therefore no replacement is necessary since one fire brigade leader, one communicator, and four NO are already assigned.

Distractor Explanation:
Distractors are incorrect and plausible because:

B. Plausible because the fire protection inspector has a notation that he is a qualified as a fire brigade member. If fire brigade manning was below minimum, an examinee may conclude that NO9 is eligible as a brigade member. Incorrect because the fire protection inspector is not normally assigned to the fire brigade, and in this case, the watchbill notation does not have him assigned to that position. Also incorrect because NO9 is assigned as the Dedicated Shutdown NO, and is therefore not eligible to be assigned to the fire brigade.
C. Plausible because the fire protection inspector has a notation that he is a qualified as a fire brigade member. If fire brigade manning was below minimum, a candidate may conclude that the allowed time is 4 hours to have a replacement. Incorrect because the fire protection inspector is not normally assigned to the fire brigade, and in this case, the watchbill notation does not have him assigned to that position. Also incorrect because the requirement, if the fire brigade was not fully manned, would be to have somebody to replace the member within two hours.
D. Plausible because the fire protection inspector has a notation that he is a qualified as a fire brigade member. Incorrect because the fire protection inspector is not normally assigned to the fire brigade, and in this case, the watchbill notation does not have him assigned to that position. The second half would be correct if the fire brigade manning was below minimum.

10 CFR 55.43(b)(5) SRO Justification:
This question meets ES-401 Attachment 2 requirements to be SRO-Only because answering this question involves assessing plant conditions (normal vs. abnormal shift manning) and selecting a procedure (MOP10) section (3.1, Fire Brigade Composition) to mitigate or recover shift staffing back to meet the minimum requirements. This question requires specific knowledge of the content of MOP10, Section 3.1 and not the procedure's overall purpose.

The question cannot be answered solely by knowing "systems knowledge", or solely by knowing immediate operator actions, or solely by knowing entry conditions for AOPs or plant parameters that require direct entry into major EOPs, or solely by knowing the purpose, overall sequence of events, or mitigative strategy of a procedure.

Reference Information:

MOP10, Fire Brigade

Question Use
Closed Reference
SRO

NUREG 1123 KA Catalog Rev. 2
G2.1.5 Ability to use procedures related to shift staffing, such as minimum crew complement, overtime limitations, etc.

10CFR55 RO/SRO Written Exam Content
10 CFR 55.43(b) (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

NRC Exam Usage
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Fundamental
Modified
SRO

Associated objective(s):
Administrative Conduct Manuals - Licensed Operators/STA
Performance Terminal
Describe the typical Fire Brigade composition and any exceptions to the minimum requirements.

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95	K/A Importance: 2.8 / 3.9			**Points: 1.00**
S95V2	Difficulty: 3.00	Level of Knowledge: Fundamental	Source: MODIFIED: 2015 ST LUCIE NRC EXAM	89047

To perform Refueling Operations, the Tech Spec required minimum Reactor Vessel Water Level is 20 ft 6 inches above the __(1)__.

The Technical Specification Basis for this requirement is to ensure water depth __(2)__.

A. (1) RPV flange

(2) will provide sufficient shielding to the operators to maintain dose within limits

B. (1) RPV flange

(2) is sufficient to ensure radioactive iodine released from a damaged fuel assembly is retained in the water

C. (1) top of active fuel

(2) will provide sufficient shielding to the operators to maintain dose within limits

D. (1) top of active fuel

(2) is sufficient to ensure radioactive iodine released from a damaged fuel assembly is retained in the water

Answer: B

Answer Explanation:
Per 3.9.6 Bases: The movement of irradiated fuel assemblies within the RPV cavity, or handling of new fuel or control rods within the RPV while irradiated fuel assemblies are seated in the RPV, requires a minimum water level of 20 ft 6 inches above the top of the RPV flange. During refueling, this maintains an adequate water level in the reactor vessel cavity and spent fuel pool which is necessary to retain sufficient iodine fission product activity assumed to be released in the water in the event of a fuel handling accident (Refs. 1 and 2). Sufficient iodine activity would be retained to limit offsite doses from the accident to • 25% of 10 CFR 100 limits, as provided by the guidance of Reference 3 or the TEDE doses provided in Reference 6.

The 20'6" requirement is repeated in 24.000.03, "Mode 5 Shiftly, Daily, and Weekly Surveillances."

Distractor Explanation:
Distractors are incorrect and plausible because:

A. Plausible because the water does provide shielding for the operators while moving fuel. Incorrect because limiting radiation dose to the operators is not part of the bases for the refueling water level requirement.
C. Plausible because the minimum water level could be measured from any reference point, and if an assembly is dropped, it could drop to the level of the fuel. Incorrect because the water level reference is from the RPV flange. Also plausible because the water does provide shielding for the operators while moving fuel. Incorrect because limiting radiation dose to the operators is not part of the bases for the refueling water level requirement.
D. Plausible because the minimum water level could be measured from any reference point, and if an assembly is dropped, it could drop to the level of the fuel. Incorrect because the water level reference is from the RPV flange.

10 CFR 55.43(b) (7) SRO Justification:
This question meets ES-401 Attachment 2 requirements to be SRO-Only because it relates to Fuel Handling facilities and procedures, specifically the technical specifications. It could potentially be tied to 10 CFR 55.43(b)(2) as being related to TS, but that flowchart yields an indeterminate outcome as follows:

This question cannot be answered *solely* by knowing < 1hr TS/TRM actions, nor *solely* by knowing the LCO/TRM information listed "above-the-line", nor solely by knowing the TS Safety Limits.

Knowledge of the TS bases is required to answer the question. However, the knowledge is not used to analyze TS required actions or terminology.

The result of the flowchart thus indeterminate, and ultimately it is judged to be SRO only linked to 10CFR55.43(b)(7)

Reference Information:

TS 3.9.6 Bases
TS 3.9.6
24.000.03 "Mode 5 Shiftly, Daily, and Weekly Surveillances"

Question Use
Closed Reference
SRO

NUREG 1123 KA Catalog Rev. 2
G2.1.40 Knowledge of refueling administrative requirements

10CFR55 RO/SRO Written Exam Content
10 CFR 55.43(b) (7) Fuel handling facilities and procedures.

NRC Exam Usage
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Fundamental
Modified
SRO

Associated objective(s):
Refueling
(F1500)
Cognitive Enabler
Describe the Refueling System technical specification limiting conditions for operation, their bases, the associated surveillance requirement(s), and their relationship to operability.

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96	K/A Importance: 3.9/4.3			Points: 1.00
S96	Difficulty: 3.00	Level of Knowledge: Fundamental	Source: BANK	87250

The plant is at 100% power

While performing 24.137.01, OPERABILITY OF 480V SWING BUS 72CF AUTOMATIC THROWOVER SCHEME, the throwover failed to occur after the Auto Throwover Permissive Test switch at Bus 72C, Position 1A was taken to CLOSE. The CRS has entered LCO 3.0.3 based on failure of the surveillance.

Maintenance personnel have submitted a Troubleshooting Datasheet for approval with the following information:

- The troubleshooting boundary is Bus 72C, Position 1A cubicle plane
- A flashlight must break the cubicle plane to perform an adequate visual inspection based on orientation

Based on the information given, what is the LOWEST level of approval required for the MMA26 Troubleshooting Plan?

A. SRO

B. Maintenance Director

C. Maintenance Manager

D. Troubleshooting Team Lead

Answer: A

Answer Explanation:
MMA26 designates this activity to be ‘Category C’ based on it being ‘High Risk’ and the Rigor Matrix. The Rigor Matrix allows C based on Visual Inspection **with** barrier intrusion. Category C can be approved by an SRO.

Distractor Explanation:
B. Is incorrect and plausible because Maintenance Director is not the lowest level of approval and requires the SM concurrence.
C. Is incorrect and plausible because the Maintenance Manager CAN approve the Troubleshooting data sheet, but they are not the LOWEST level. MM approval is required for Categories A/B.
D. Is incorrect because the Troubleshooting Team Lead does not have approval authority, however this is plausible because the Troubleshooting Team Lead approves revisions.

10 CFR 55.43(b)(5) SRO Justification:
This question meets ES-401 Attachment 2 requirements to be SRO-Only because answering this question requires specific knowledge of MMA26, it is not related to immediate actions and the entry conditions are not relevant or leading to the answer. The answer to this questions is based on assessing plant conditions and then the correct action to take by applying MMA26.

Reference Information:
MMA26

ILT 2019 Retake Reference Provided:
MMA26

Plant Procedures
MOP05 - Control Of Equipment
MMA26 - Troubleshooting

Question Use
Open Reference
SRO

NUREG 1123 KA Catalog Rev. 2
G2.2.14 Knowledge of the process for controlling equipment configuration or status.
G2.2.20 Knowledge of the process for managing troubleshooting activities.

10CFR55 RO/SRO Written Exam Content
10 CFR 55.43(b) (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

NRC Exam Usage
ILO 2017 Audit
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Bank
Fundamental
SRO

Associated objective(s):

Administrative Conduct Manuals - Licensed Operators/STA

Performance Terminal

List and describe other "systems" that assist Operations in the control of equipment and tracking of information in the plant, including:

a. Shiftly Responsibilities
b. Audits
c. Evolution Evaluation Program
d. Control Room Information System
e. Deleted
f. Night Orders
g. Operations Department Instructions
h. Limiting Condition for Operation

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97	K/A Importance: 4.2			Points: 1.00
S97	Difficulty: 2.00	Level of Knowledge: Fundamental	Source: MODIFIED: 2019 FERMI NRC EXAM	87427

The plant is operating at 100% when a review of valid 3D Monicore data indicates the following:

- MFLCPR …1.275
- MAPRAT… 0.812
- MFLPD...... 0.875
- MCPR ……1.05

The crew should restore compliance with all limits AND ___(1)___, because ___(2)___.

A. (1) Insert all insertable control rods.
(2) Fuel rods may reach transition boiling during a transient.

B. (1) Insert all insertable control rods.
(2) Peak centerline fuel temperature may exceed 2200°F during a LOCA.

C. (1) Reduce Recirc flow to stabilize THERMAL POWER <25%.
(2) Fuel rods may reach transition boiling during a transient

D. (1) Reduce Recirc flow to stabilize THERMAL POWER <25%.
(2) Peak centerline fuel temperature may exceed 2200°F during a LOCA.

Answer: A

Answer Explanation:
The 3DM data indicates that MCPR is below its allowable safety limit, AND MFLCPR is above its allowable tech spec limit of ≤1.00.
Since MCPR has exceeded its SL, TS actions 2.2.1 and 2.2.2 are required; restore compliance with all safety limits and insert all insertable control rods.
Additionally, MFLCPR above 1.00 requires entry into tech spec 3.2.2, Minimum Critical Power Ratio (MCPR), since this value indicates MCPR operating limit is exceeded. Condition A applies and requires that MCPR (and MFLCPR) be restored to within limits within 2 hours.
The basis for MCPR safety limit is to prevent 99.9 % of fuel rods from reaching transition boiling.

Distractor Explanation:
Distractors are incorrect and plausible because:
B. The basis for MCPR and MFLCPR is not PCT following a LOCA. This is the basis for the APLHGR limit, and in this case that limit has not been exceeded. Applicants may choose this if they incorrectly determine MAPRAT is exceeded, or they incorrectly recall the basis for the MCPR limit
C. Stabilizing power <25% with recirc flow will not comply with TS 2.2.1 requirement for violation of a Safety Limit. Reducing power <25% would be applicable if ONLY MFLCPR was exceeded and could not be restored within 2 hrs.
D. Stabilizing power <25% with recirc flow will not comply with TS 2.2.1 requirement for violation of a Safety Limit. Reducing power <25% would be applicable if ONLY MFLCPR was exceeded and could not be restored within 2 hrs. Additionally, the basis for MCPR and MFLCPR is not PCT following a LOCA. This is the basis for the APLHGR limit, and in this case that limit has not been exceeded. Applicants may choose this if they incorrectly determine that only the MCPR operating limit (MFLCPR) has been exceeded and not the MCPR SL; or they incorrectly recall the basis for the MCPR/MFLCPR limit.

10 CFR 55.43(b)(2) SRO Justification:
This question meets ES-401 Attachment 2 requirements to be SRO-Only because answering this question requires senior operator applicants to recall and apply knowledge of TS bases that are required to analyze TS-required actions and terminology.

The question cannot be answered with RO level TS knowledge.

K/A Match Justification:

This question requires SRO applicants to recall knowledge of the bases for the applicable tech spec LCOs and safety limits without reference to tech specs or procedures.

Reference Information:

Fermi TS 2.2, 3.2.2. and applicable bases.
Lesson plan BT095r4, Thermal Limits

Question Use
Closed Reference
SRO

NUREG 1123 KA Catalog Rev. 2
G2.2.25 Knowledge of bases in Technical Specifications for limiting conditions for operations and safety limits.

10CFR55 RO/SRO Written Exam Content
10 CFR 55.43(b) (2) Facility operating limitations in the technical specifications and their bases.

NRC Exam Usage
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Fundamental
Modified
SRO

Associated objective(s):
Core and Fuel (B1100 & J1100)
Cognitive Enabler
Identify Core and Fuel related technical specifications, with emphasis on action statements requiring prompt actions (for example, one hour or less).

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98	K/A Importance: 2.0/3.8			Points: 1.00
S98	Difficulty: 3.00	Level of Knowledge: Fundamental	Source: MODIFIED: 2018 EFERMI NRC EXAM	87389

A MCE06005 discharge permit has been submitted to the Shift Manager for approval. The permit includes the following information:

- The water to be discharged is not potentially contaminated.
- The discharges will occur only on dayshift.
- The discharges will require at least 7 days of pumping.
- The area containing the water is not subject to radiological restrictions.
- The source of the water is always collected ground water.

Upon deciding to approve the discharge permit, the Shift Manager should assign a maximum expiration not to exceed:

A. 4 hours

B. 24 hours

C. 5 days

D. 30 days

Answer: C

Answer Explanation:
Per MCE 06 NON-RADIOLOGICAL ENVIRONMENTAL PROTECTION, Tasks which are periodic (for example, have idle intervals during the pumping of the same source) and have a clearly defined and constant water source should be assigned an expiration not to exceed five days from the time the permit is approved.

10 CFR 55.43(b)(4) SRO Justification:
This question meets ES-401 Attachment 2 requirements to be SRO-Only because this question is about a SRO responsibility to determine a time limit that is NOT a share responsibility with an RO and this aligns with the example listed in NUREG 1021 ES401 Attachment 2 for 10 CFR 55.43(b)(4) which list "process for gaseous/liquid release approvals (i.e., release permits)"

Distractor Explanation:
A. Is plausible because 4 hours is a reasonable amount of time to pump done a barrel each day and MCE 06 does state that discharge permits are intended for the duration of the task, however this is incorrect because of the answer explanation.
B. Is plausible because 24 hours is a limit for sources within an area subject to radiological restrictions, however this is incorrect because the area is not subject to radiological restrictions,
D. Is plausible because 30 days is a reasonable amount of time for a daily barrel pumping activity to last and MCE 06 does state that discharge permits are intended for the duration of the task, however this is incorrect because of the answer explanation.

Reference Information:
MCE06, 5.1.13

ILT Retake Reference Provided:

MCE06

Plant Procedures
MCE06

Question Use
Open Reference
SRO

NUREG 1123 KA Catalog Rev. 2
G2.3.6 Ability to approve release permits

10CFR55 RO/SRO Written Exam Content
10 CFR 55.43(b) (4) Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions.

NRC Exam Usage
ILO 2019 Retake Exam

NRC Question Use (ILO 2018)
BANK
Low
SRO

NRC Question Use (ILO 2019 Retake)
Fundamental
Modified
SRO

Associated objective(s):

Administrative Conduct Manuals - Licensed Operators/STA

Performance Terminal

Describe the methods used at Fermi 2 to protect the environment, including:

a. Conditions that would indicate an NPDES Non-Compliance, and actions required in the event of a non-compliance.
b. The different Environmental Protection Plan documents used at Fermi 2.
c. Notifications required for Unusual or Important Environment Events.
d. Actions that must be taken when a Non-Radiological Environmental Spill occurs.
e. Actions taken when finding unidentified/unattended waste.

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99	K/A Importance: 3.6/4.4			**Points: 1.00**
S99	Difficulty: 3.00	Level of Knowledge: Fundamental	Source: BANK	88407

The plant is in the EOPs and AOPs due to a loss of offsite and onsite power.

As the CRS, which of the following actions should be directed first and what is the basis for this prioritization?

A. Injecting with SBFW to prevent uncovering the reactor core.

B. Closing 13.2 kV Pos A6 to prevent uncovering the reactor core.

C. Restoring EECW to service to ensure systems remain available to prevent boil down to Rx low level 1.

D. Inhibiting the RCIC/HPCI room temperature trips to ensure the systems remain available to prevent boil down to Rx low level 1.

Answer: D

Answer Explanation:
Per 20.300.SBO, Loss of Offsite and Onsite Power, Bases for Action C.1 is to disable the equipment area high temperature signals to ensure HPCI and RCIC remain available to feed the RPV. This action is directed expeditiously to ensure the RCIC/HPCI systems remain available because boil down to level 1 without RCIC/HPCI injection is expected within 25 minutes.

Distractor Explanation:
Distractors are incorrect and plausible because:

A. SBFW action is directed in Step K of 20.300.SBO. Additionally, 20.000.18, Control from the Dedicated Shutdown Panel, prioritizes injecting with SBFW because an analysis predicts that the core will be uncovered in 29 minutes with no makeup water, therefore the candidate could confuse the two priorities and determine that SBFW injection must be prioritized within 25 minutes, which is not correct in this case because HPCI and RCIC are available for injection and must be protected.
B. This action is directed in Condition G (and many other conditions in other AOPs that result from loss of Division 1 offsite 120 kV power). The candidate could determine that this action must be performed within 25 minutes to restore power to injection systems and prevent uncovering the reactor core, which is not correct in this case because the conditions to perform the step (Bus 11 de-energized and Bus 1-2 energized) were not provided in the stem and because the CRS would have to direct step G.1 (completion of Attachment 1) be completed prior to this action.
C. EECW restoration is directed by Step X of 20.300.SBO and the candidate could determine that this action must be performed within 25 minutes to restore cooling to injection systems, such as the HPCI/RCIC room coolers which makes this action similar to the correct action of disabling the high temperature trips, thus preventing boil down to Rx low level 1. This distractor is incorrect because the conditions to perform this step (SST 64 energized AND EECW restoration required) are not provided in the stem.

10 CFR 55.43(b)(5) SRO Justification:
This question meets ES-401 Attachment 2 requirements to be SRO-Only because answering the question requires assessment of plant conditions and then prioritizing the section of the procedure to be performed to mitigate the event and knowledge of the bases for this prioritization.

The question cannot be answered solely by knowing systems knowledge, immediate operator actions, entry conditions or the purpose, overall sequence of events, or overall mitigative strategy of a procedure.

Reference Information:

20.300.SBO, Loss of Offsite and Onsite Power and BASES.
20.000.18, Control from the Dedicated Shutdown Panel and BASES.

Question Use
Closed Reference
SRO

NUREG 1123 KA Catalog Rev. 2
G2.4.22 3.6/4.4 Knowledge of the bases for prioritizing safety functions during emergency operations

10CFR55 RO/SRO Written Exam Content
10 CFR 55.43(b) (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

NRC Exam Usage
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Bank
Fundamental
NRC Early Review
SRO

Associated objective(s):
Integrated Electrical Events
Cognitive Enabler
Given a copy of electrical AOPs, analyze and determine which equipment has the highest priority for restoration.

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100	K/A Importance: 2.4 / 4.4			**Points: 1.00**
S100	Difficulty: 3.00	Level of Knowledge: Fundamental	Source: NEW	87387

The plant was operating at 100% power when a plant event occurred that resulted in declaration of the following event classifications at the listed times:

- Site Area Emergency (SAE) at 1100.
- General Emergency (GE) at 1130.

(1) By what time must the Emergency Director provide an initial Protective Action Recommendation (PAR) to the appropriate offsite authorities?

(2) Once made, who is responsible for IMPLEMENTING protective actions?

A. (1) 1115
(2) state officials

B. (1) 1115
(2) the Emergency Director

C. (1) 1145
(2) state officials

D. (1) 1145
(2) the Emergency Director

Answer: C

Answer Explanation:
Per EP-545 5.2 When a General Emergency is declared:
NOTE: A PAR shall be made to appropriate offsite authorities concurrent with the initial notification of General Emergency declaration and documented using a Nuclear Plant Event Notification Form.

Per Ep-545 4.6.1 Emergency Response Organization decision-makers only recommend protective actions. State decision-makers make the final decision on what protective action(s) to implement.

A GE was declared at 1130, which requires a PAR to be made to the appropriate local and/or state authorities as appropriate. The decision to implement the PAR lies with the state officials.

Distractor Explanation:
Distractors are incorrect and plausible because:
A. The candidate could recall that a PAR is required to be formulated upon declaration of a SAE, which is plausible because EP-545 Section 5.0 (Immediate Actions) part 5.1 requires formulation of PARs when a SAE is declared, when possible, before declaration of a GE. However, the PAR is not actually made until declaration of a GE.
B. The candidate could recall that a PAR is required to be formulated upon declaration of a SAE, which is plausible because EP-545 Section 5.0 (Immediate Actions) part 5.1 requires formulation of PARs when a SAE is declared, when possible, before declaration of a GE. Plausible if the candidate believes that the Emergency Director has the authority to implement PARs. Incorrect because the PAR is only a recommendation.
D. Plausible if the candidate believes that the Emergency Director has the authority to implement PARs. Incorrect because the PAR is only a recommendation.

10 CFR 55.43(b)(5) SRO Justification:
This question meets ES-401 Attachment 2 requirements to be SRO-Only because the question cannot be answered solely by knowing "systems knowledge." The question cannot be answered solely by knowing immediate operator actions. The question cannot be answered solely by knowing the entry conditions for AOPs or plant parameters that require direct entry into major EOPs. The question cannot be answered solely by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure. The question does require knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures.

Reference Information:

EP-545

Question Use
Closed Reference
SRO

NUREG 1123 KA Catalog Rev. 2
G2.4.44 2.4/4.4 Knowledge of emergency plan protective action recommendations

10CFR55 RO/SRO Written Exam Content
10 CFR 55.43(b) (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

NRC Exam Usage
ILO 2019 Retake Exam

NRC Question Use (ILO 2019 Retake)
Fundamental
New
SRO

Associated objective(s):
INITIAL QUALIFICATION
EMERGENCY CLASSIFICATION & PROTECTIVE ACTION RECOMMENDATIONS

Performance Terminal

State who makes the Protective Action Recommendations and which agencies must receive the recommendations.