NOTE: Per NUREG 1021 ES-501 C.1.B this exam key has been annotated to indicate changes made while administering the exam.

Added text is in RED
Removed text is lined out

Questions changed:

Question: R31

Comment: Are A&C CMCs in RUN because of Surveillance or in Auto for Surveillance?
Action(s) taken: After conferring with the chief examiner, the following was added to the stem of the question: running for "Division 1 Core Spray System Pump and Valve Operability Test."

Question: S88
Comment: Candidate started to ask a question about Q S88. Candidate asked 'What MODE is the plant in?'
Action(s) taken: After conferring with the chief examiner, the candidates were informed that the plant is in MODE 1.

Question: R42
Comment: Is the question stating that it is a loss of ALL Offsite Power?
Action(s) taken: After conferring with the chief examiner, informed candidates that the question is referring to a loss of ALL Offsite Power.

THIS PAGE LEFT BLANK

1	K/A Importance: 3.5/3.6			**Points: 1.00**
R01	Difficulty: 3.00	Level of Knowledge: High	Source: BANK	73969

The plant was at rated conditions when a Main Turbine Trip and ATWS occurred.
Given the following:

- SLC is being injected.
- Reactor power is 22% and slowly lowering.
- RPV water level is 50" and intentionally being lowered.
- Lo Lo-SET SRVs are lifting and Torus temperature is 112°F and rising.

For the conditions above; which of the following is a valid reason for continuing to lower RPV water level?

A. To prevent thermal-hydraulic instabilities.

B. To further reduce feedwater inlet subcooling.

C. To reduce natural circulation and reactor power.

D. To lower the amount of voids (void fraction) within the core.

Answer: C

Answer Explanation:
As RPV water level decreases, the height of the fluid columns is reduced, thereby reducing the natural circulation driving head. This lowers the natural circulation flow which raises the void fraction, thereby adding negative reactivity and lowering reactor power.

Distractor Explanation:
Distractors are incorrect and plausible because:
A. Subcooling is minimized below 114”, therefore, thermal-hydraulic instabilities will be prevented.
B. With RPV level already below 114”, the feedwater spargers will be uncovered and further reduction in subcooling will not occur.
D. Lowering RPV level will lower natural circulation which will raise the amount of voids in the core causing void fraction to rise.

Reference Information:
Appendix B-Technical Basis, Volume 1
ST-OP-802-3003-001 Student Texts EOP RPV Control

NUREG 1123 KA Catalog Rev. 2
295001 AK1 Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION :
295001 AK1.01 Natural circulation

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 Question Use (ILO 2018)
BANK
High
RO

Associated objective(s):
RPV Control
Cycle 12-3 Objectives
Exam Objectives
Cognitive Terminal
Describe the expected plant/system response for actions directed by 29.100.01 Sh 1, 1A, 3, and 3A.

2	K/A Importance: 4.1/4.2			**Points: 1.00**
R02	Difficulty: 3.00	Level of Knowledge: Low	Source: BANK	73987

If an EDG were already running in parallel with Offsite Power, how will the EDG respond to a loss of Offsite Power?

A. The EDG output breaker will trip and the EDG will shut down.

B. The EDG output breaker will trip, Load Shed will occur, and the EDG output breaker will reclose.

C. The EDG output breaker will remain closed, the EDG will shut down, and then it will restart in isochronous mode.

D. The EDG output breaker will remain closed, the EDG will continue running and the its governor will shift to isochronous mode.

Answer: B

Answer Explanation:
When the EDG is operating in parallel with offsite power, the undervoltage relays are bypassed by a contact in the closed EDG output breaker. If an actual loss of power to the bus from offsite occurs the EDG under frequency relay will open (TRIP) the EDG output breaker and the undervoltage scheme will operate as designed (load shed, close output breaker).

Distractor Explanation:
Distractors are incorrect and plausible because:

A. Although the output breaker will trip, the EDG will not shut down.
C. The EDG will remain running, and output breaker will trip. The EDG will be in isochronous, but does not restart.
D. Output breaker trips, other parts are correct.

Reference Information:
ST-OP-315-0065 EDG Student Text
i-n-2711-36 EDG 14 Control Drawing

NUREG 1123 KA Catalog Rev. 2
295003 AK2 Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF A.C. POWER and the following:
295003 AK2.02 Emergency generators

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 Question Use (ILO 2018)
BANK
Low
RO

Associated objective(s):
Emergency Diesel Generator (R3000)
Cognitive Terminal
Given various controls and indications for Emergency Diesel Generator operations, analyze that indication to determine proper plant response in accordance with approved plant procedures.

3	K/A Importance: 3.1/3.5			**Points: 1.00**
R03	Difficulty: 2.00	Level of Knowledge: Low	Source: NEW	74089

A loss of which DC electrical distribution cabinet would result in a Reactor Scram at full power and why?

A. 2PC3-Main Turbine Trip

B. 2IA-INOP trip of SRMs/IRMs

C. 2PB2-Outboard MSIV closure

D. 2PA2-Loss of Scram Pilot Air Header

Answer: A

Answer Explanation:
Loss of BOP DC 2PC3 will cause a Main Turbine trip and subsequent reactor scram if power is greater than 30%.

Distractor Explanation:
Distractors are incorrect and plausible because:
B. Loss of 48/24VDC power would generate an INOP trip for the SRMs and IRMs but would not cause a Reactor Scram at full power.
C. Loss of DC power would cause a loss of MSIV position indication and possible MSIV closure if MSIV AC solenoids are deenergized. However, under the conditions stated in the stem, the MSIV AC solenoids will be energized, so de-energization of this DC source will not result in MSIV closure under the given plant conditions.
D. Loss of Division 1 and/or 2 ESF DC would prevent operation of the RPS Backup Scram Valves.

Reference Information:
23.309 260/130 V DC Electrical System
ST0064001

NUREG 1123 KA Catalog Rev. 2
295004 AK3. Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER :
295004 AK3.03 3.1/3.5 Reactor SCRAM: Plant-Specific

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 Question Use (ILO 2018)
Low
NEW
RO

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

4	K/A Importance: 3.6/3.6			**Points: 1.00**
R04	Difficulty: 2.00	Level of Knowledge: Low	Source: BANK	73991

Following a trip of the Main Generator from 25% power, which of the following represents the expected status of the reactor/turbine pressure regulating system?

	Turbine Stop Valves	Turbine Control Valves	Turbine Bypass Valves
A.	Open	Open	Closed
B.	Closed	Open	Open
C.	Closed	Closed	Open
D.	Open	Closed	Closed

Answer: C

Answer Explanation:
A Main Generator trip will result in a trip of the Main Turbine. In order to protect the Main Turbine, the Turbine control valves and Turbine stop valves will close to prevent steam from entering the main turbine. The Turbine Bypass Valves will open and modulate to control Reactor pressure at the pressure regulator setpoint.

Distractor Explanation:
Distractors are incorrect and plausible because:

B. The candidate could fail to understand that a Generator Trip causes a Main Turbine trip at 25% and incorrectly think the bypass valves would stay closed since the Turbine stop valves are listed as open. Closure of the stop or control valves would prevent steam addmitance to the turbine and result in bypass valves opening to control pressure
C. The candidate could fail to understand that a Generator Trip causes a Main Turbine trip and incorrectly assume the control and stop valves would stay open. This would result in the reactor/turbine pressure systems not needing the bypass valves to open to control reactor pressure
D. The candidate could incorrectly assume that the turbine control valves stay open on a turbine trip with bypass valves opening to control reactor/turbine pressure.

Reference Information:
23.109 Turbine Operating Procedure

NUREG 1123 KA Catalog Rev. 2
295005 AA1. Ability to operate and/or monitor the following as they apply to MAIN TURBINE GENERATOR TRIP:

295005 AA1.05 3.6/3.6 Reactor/turbine pressure regulating 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 Question Use (ILO 2018)
BANK
Low
RO

Associated objective(s):
Main Generator and Excitation (N3031 & N3034)
Cognitive Enabler
Discuss the Main Generator and Excitation System interrelationships with other systems.

5	K/A Importance: 3.5/3.8			**Points: 1.00**
R05	Difficulty: 2.00	Level of Knowledge: Low	Source: NEW	73995

Which of the following inputs would directly cause a Reactor Scram?

A. Condenser Vacuum 3.68 psia.

B. Wide Range Level indication of 173".

C. 3 Main Steam Isolation Valves close.

D. Closure of Turbine Stop Valves 1 and 3.

Answer: C

Answer Explanation:
MSIV closure logic is such that closure of any 1 MSIV will not cause a direct RPS actuation on either side. Closure of 2 MSIVs (A and B, C and D for RPS A; A and C or B and D for RPS B) may cause a 1/2 scram (Note: Closure of MSIVs B and C or A and D will NOT cause a half scram on either RPS trip system). Closure of any 3, of the 4, MSIVs will always cause a full reactor scram.

Distractor Explanation:
Distractors are incorrect and plausible because:
A. Vacuum of 3.68 will cause a turbine trip which will indirectly cause a Scram
B. The Scram signal for level comes from Narrow Range Instruments.
D. Closure of TSV 1&3 will cause a half Scram, not a full scram.

Reference Information:
ARP 3D73 3D74

NUREG 1123 KA Catalog Rev. 2
295006 AA2. Ability to determine and/or interpret the following as they apply to SCRAM :
295006 AA2.06 3.5/3.8 Cause of reactor SCRAM

10CFR55 RO/SRO Written Exam Content

10 CFR 55.41(b)(10)	Administrative, normal, abnormal, and emergency operating procedures for the facility.
10 CFR 55.43(b) (5)	Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

NRC Question Use (ILO 2018)
Low
NEW
RO

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

List the automatic features of Reactor Protection System operations.

6	K/A Importance: 4.3/4.4			**Points: 1.00**
R06	Difficulty: 4.00	Level of Knowledge: High	Source: NEW	74626

The plant is operating at 68% power when the Control Room is abandoned due to toxic fumes. A reactor scram could NOT be completed prior to abandonment. The ENTIRE Control Room Envelope is inaccessible due to the toxic fumes.

Which of the following is the method of performing a reactor scram for the given plant conditions?

A. Trip Turbine Trip Relay TTR1 or TTR2.

B. Take two operable APRM Mode switches out of Operate.

C. Open circuit breakers CB2A (C71-P001A) and CB2B (C71-P001B).

D. Place Main Steam Line Radiation Monitors (A or C) and (B or D) in INOP.

Answer: C

Answer Explanation:
Control Room Abandonment requires the crew to enter AOP 20.000.19, Shutdown from Outside the Main Control Room. AOP 20.000.19 requires the crew to attempt to scram the reactor from the Main Control Room and then proceed to H21-P100, Remote Shutdown Panel. 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 Condition B of 20.000.19, Shutdown from Outside the Control Room. However, this action is also not possible because the actions are taken in the Relay Room which is inside the Control Room Envelope, which is inaccessible as stated in the stem of the question.

Therefore, the candidate should determine that the actions of 20.000.19, Condition C are required, which will have the operator scram the reactor by opening CB2A and CB2B.

Distractor Explanation:
Distractors are incorrect and plausible because:

A. Tripping Turbine Trip Relay TTR1 or TTR2 is an action from 20.000.19 and would result in a reactor scram above 30% power. However, these actions are only necessary to trip the Main Turbine and should be performed after the reactor is scrammed. Additionally, these actions are performed in the Relay Room thereby requiring entry into the Control Room Envelope, which is NOT accessible.
B. Taking 2 operable APRMs out of operate is an action from 20.000.19 and would result in a reactor scram. However, these actions are performed in the Relay Room thereby requiring entry into the Control Room Envelope, which is NOT accessible.
D. Placing two Main Steam Line Radiation Monitors out of Operate (in the logic sequence specified in the distractor) is an action taken from 20.000.19 and would result in a direct reactor scram. However, these actions are performed, when Reactor Pressure is lowering too fast, in an attempt to lower the cooldown rate by closing the MSIVs (by initiating an NSSSS Group 1 Isolation). Additionally, these actions are performed in the Relay Room thereby requiring entry into the Control Room Envelope, which is NOT accessible.

Reference Information:
20.000.19, Shutdown from Outside the Main Control Room
20.000.19 BASES, Shutdown from Outside the Main Control Room Bases.

NUREG 1123 KA Catalog Rev. 2
295016 Control Room Abandonment

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

10CFR55 RO/SRO Written Exam Content

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

NRC Question Use (ILO 2018)
High
NEW
RO

Associated objective(s):
Reactor Operator
Performance Enabler
Perform proper system operations in accordance with System Operating Procedures (SOP).

7	K/A Importance: 3.2/3.5			**Points: 1.00**
R07	Difficulty: 3.00	Level of Knowledge: High	Source: BANK	74004

The plant is operating at 65% with P4300-C001 (North) and P4300-C002 (Center) TBCCW Pumps in service, when the following occurs:

- 5D5, TBCCW PUMPS DIFFERENTIAL PRESSURE HIGH/LOW, alarms
- P43-R805, TBCCW Headers Pressure Indicator, indicates a d/p of 31 psig.

With these conditions, which one of the following is the correct diagnosis and required action?

A. TBCCW Differential Pressure is too HIGH; it is required to STOP an operating TBCCW Pump.

B. TBCCW Differential Pressure is too LOW; it is required to REDUCE Main Turbine Generator Load.

C. TBCCW Differential Pressure is too LOW; it is required to START P4300-C003 (South) TBCCW Pump.

D. TBCCW Differential Pressure is too HIGH; it is required to CLOSE P43-F405, TBCCW DP Control Valve.

Answer: A

Answer Explanation:
The purpose of the TBCCW Pressure Control Valve (PCV) is to maintain a constant differential pressure between the supply and return headers of the system, thereby supplying the required amount of flow to all loads. The PCV operates in response to pressure signals to maintain a differential pressure of 20 to 30 psid between the supply and return headers. The PCV will respond to system demand by heat load TCVs. With the Pressure Indicator at 31 psig, TBCCW DP is too high. With pressure too high it is required to shut down an operating pump as directed by ARP 5D5

Distractor Explanation:
Distractors are incorrect and plausible because:

B. The candidate could incorrectly assume that a DP of 31 psig is too low and perform actions to reduce load on the MTG. This action would result in the PCV opening to adjust for the flow and keep pressure the same, this action is only performed if TBCCW temperature cannot be maintained.
C. The candidate could incorrectly assume that a DP of 31 psig is too low and perform actions to start the standby TBCCW pump. Starting an additional pump would cause the PCV to open and result in raising the DP.
D. The candidate could incorrectly assume that closing the P43-F405 TBCCW PCV would lower the DP. Closing the P43-F405 TBCCW DP PCV would result in an increase in TBCCW DP.

Reference Information:
ARP 5D5

NUREG 1123 KA Catalog Rev. 2
295018 AA2 Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER :
295018 AA2.03 3.2/3.5 Cause for partial or complete loss

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

NRC Question Use (ILO 2018)
BANK
High
RO

Associated objective(s):
Turbine Building Closed Cooling Water (P4300)
Cognitive Terminal
Given various controls and indications for Turbine Building Closed Cooling Water operations, analyze that indication to determine proper plant response in accordance with approved plant procedures.

8	K/A Importance: 3.5/3.4			**Points: 1.00**
R08	Difficulty: 3.00	Level of Knowledge: Low	Source: BANK	74686

The plant is operating at 100% power.

- Annunciator 7D53, Station Air Header Pressure Low, Alarms.
- P50-R802, Station Air Header Pressure is 90 psig, lowering.

The FIRST required action per 20.129.01, Loss of Station and / or Control Air is ___(1)___.
The BASIS for this action is to avoid conditions which will result in ___(2)___ MSIVs drifting SHUT.

A. (1) Start any available Station Air Compressors
(2) Inboard

B. (1) Start any available Station Air Compressors
(2) Outboard

C. (1) Start Div I and Div II Control Air Compressors
(2) Inboard

D. (1) Start Div I and Div II Control Air Compressors
(2) Outboard

Answer: B

Answer Explanation:
20.129.01, Loss of Station and Control Air Condition A contains the guidance for the actions in the correct answer. 20.129.01 BASES contains the reasons for taking these actions. Specifically the basis for Caution 1, which is applicable for Condition A, is what is being tested in this question to determine if the operator knows the reason for the operation of the standby Station Air Compressor.

The Candidate must know that the Station Air Compressors are started prior to Div I/Div II Control Air Compressors. The Station Air Compressor aligned for standby should have AUTO started at 95 psig. The THIRD Station Air Compressor has no auto start and requires manual action to start, per the AOP, when Station Air Pressure reaches 90 psig. The third Station Air Compressor is considered an available Station Air Compressor.
The Candidate must also know that starting the standby Station Air Compressor will stabilize the Station Air and Interruptible Air headers, thereby preventing the impacts listed in CAUTION 1. The candidate must recall that the OUTBOARD MSIVs drift closed on a loss of Interruptible Air pressure. Note: The INBOARD MSIVs have Nitrogen pneumatic supply with air backup from Division 1 NIAS.

Distractor Explanation:
Distractors are incorrect and plausible because:

A. The candidate could correctly recall that the FIRST required action of the AOP is to start any available Station Air Compressors. However, the candidate could incorrectly recall that the reason for doing so is to prevent closure of the Inboard MSIVs, which is incorrect because the Inboard MSIVs are supplied by Nitrogen and Nitrogen accumulators with Division 1 NIAS as a backup supply.
C. The candidate could incorrectly recall that the FIRST required action of the AOP is to start the Div 1 and Div 2 Control Air Compressors, which is incorrect since the Control Air Compressors do not auto start until Station Air pressure drops to 85 psig, per Condition B of the AOP. The Station Air Compressors should be started first in an attempt to stabilize the air system, preventing further degradation and automatic system actuations. This could lead the candidate to also conclude that the Control Air Compressors are started in order to prevent closure of the Inboard MSIVs, which is plausible because of the relationship between the MSIVs and Div 1 NIAS, which is a backup to the normal Nitrogen pneumatic supply inside the Drywell.
D. The candidate could incorrectly recall that the FIRST required action of the AOP is to start the Div 1 and Div 2 Control Air Compressors, which is incorrect since the Control Air Compressors do not auto start until Station Air pressure drops to 85 psig, per Condition B of the AOP. The Station Air Compressors should be started first in an attempt to stabilize the air system, preventing further degradation and automatic system actuations.

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

NUREG 1123 KA Catalog Rev. 2

295019	Partial or Complete Loss of Instrument Air
295019 AK3.	Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR:

295019 AK3.02 3.5/3.4 Standby air compressor operation

10CFR55 RO/SRO Written Exam Content

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

NRC Question Use (ILO 2018)
BANK
Low
RO

Associated objective(s):
Reactor Operator
    Performance Enabler
    Apply notes and cautions as directed by the EOP / AOP.

Reactor Operator
    Performance Enabler
    Perform other EOP / AOP actions per site procedures as directed.

THIS PAGE LEFT BLANK

9	K/A Importance: 3.5/3.5			**Points: 1.00**
R09	Difficulty: 4.00	Level of Knowledge: High	Source: MODIFIED	74746

The plant is in MODE 4 with Division 1 Residual Heat Removal (RHR) System in Shutdown Cooling (SDC) and RHR Pump A running. A valving error results in RPV water level lowering to 150" before it is corrected and the RPV level reduction is stopped.

In addition to depressing the E1150-F015A(B), Div 1(2) LPCI Inbd Iso Vlv, SDC Isolation RESET pushbuttons, of the numbered actions listed below, which are required to restore Shutdown Cooling?

1. Restore RPV level to 173 - 214".
2. Restore RPV level to 220 - 255".
3. Depress Inboard and Outboard MSIV Isolation Logic RESET pushbuttons.
4. Depress Hi Drywell Press & Reactor Low Level Logic RESET pushbuttons.
5. Depress Leak Detection Line Break Logic RESET pushbuttons.

A. 2 and 3

B. 1 and 4

C. 2, 3 and 4

D. 1, 4 and 5

Answer: A

Answer Explanation:
The actions listed are from the Loss of Shutdown Cooling AOP, 20.205.01, Condition C for a Loss of SDC Condition caused by RPV Level dropping less than Level 3 (173"). The candidate must recognize the need to perform the following actions, for the reasons listed:
2. Restore RPV Level to the SDC level band (>220") to promote natural circulation within the RPV by raising RPV level above the bottom of the Steam Separators.
3. Depress Inboard and Outboard MSIV Isolation Logic RESET pushbuttons to reset the Group 4 (RHR SDC) isolation that occurred when RPV level dropped <L3, which will allow the SDC isolation valves to be re-opened.

These actions stabilize water level, reset the isolation signals and return RHR to service. Low reactor water Level 3 is selected as the isolation level because all normal operations of the RHR SDC mode should be performed with reactor water level above Level 3. If reactor water level decreases to Level 3, the RHR System is automatically isolated to isolate potential leakage paths in the RHR System that may result in draining of the reactor vessel water inventory. If the system isolates on decreasing level before LPCI initiates (at RPV Level 2, or 110.8"), the common SDC suction (E11-F008 and F009) valves and the LPCI injection valves (E11-F015A and F015B) in both divisions close, and pumps in the loop that is operating in the SDC mode trip on loss of suction path. The LPCI loop pumps that are lined up in standby mode are not affected.

Distractor Explanation:
Distractors are incorrect and plausible because:

B. The candidate could incorrectly determine that RPV Water level should be restored to the 173-214" level band per Action 1, which is plausible because this is the normal (non-Shutdown Cooling) shutdown RPV level band. The candidate could also determine that RHR initiation logic must be reset, per Action 4, by depressing the Hi Drywell Press & Reactor Low Level Logic RESET pushbuttons, which is plausible because a low RPV Water Level (<Level 3) condition has occurred, but incorrect because LPCI initiation occurs at Level 1.
C. The candidate could correctly determine that actions 2 and 3 need to be performed, however incorrectly conclude that Action 4 also needs to be performed to reset RHR initiation logic by depressing the Hi Drywell Press & Reactor Low Level Logic RESET pushbuttons, which is plausible because a low RPV Water Level (<Level 3) condition has occurred, but incorrect because LPCI initiation occurs at Level 1.
D. The candidate could incorrectly determine that RPV Water level should be restored to the 173-214" level band per Action 1, which is plausible because this is the normal (non-Shutdown Cooling) shutdown RPV level band. The candidate could also determine that RHR initiation logic must be reset, per Action 4, by depressing the Hi Drywell Press & Reactor Low Level Logic RESET pushbuttons, which is plausible because a low RPV Water Level (<Level 3) condition has occurred, but incorrect because LPCI initiation occurs at Level 1. The candidate could also determine that RHR leak detection line break logic needs reset, which is plausible because a low RPV Water Level (<Level 3) condition has occurred, but incorrect because RHR Leak Detection Line Break (LPCI Loop Select) Logic actuates at RPV Level 2.

Reference Information:
20.205.01, Loss of Shutdown Cooling AOP.

NUREG 1123 KA Catalog Rev. 2
295021 Loss of Shutdown Cooling
295021 AA1.02 3.5/3.5 RHR/shutdown cooling

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 Question Use (ILO 2018)
High
Low
MODIFIED
RO

Associated objective(s):
Residual Heat Removal (E1100)
Cognitive Enabler
List the interlocks associated with RHR System components.

Residual Heat Removal (E1100)
Cognitive Enabler
List the interlocks associated with RHR System components.

THIS PAGE LEFT BLANK

10	K/A Importance: 3.7/3.8			Points: 1.00
R10	Difficulty: 3.00	Level of Knowledge: High	Source: BANK	74270

An irradiated fuel bundle is being removed from the core. An adjacent bundle has been lifted along with the selected bundle. When noticed, fuel movement is stopped, and the adjacent bundle falls back into the core.

When some bubbles come to the pool surface, the local Continuous Air Monitor (CAM) alarms. D21-K717, RB5 Refuel Floor Lo Range ARM Ind trip Unit, indicates 10 mR/hr and slowly rising.

(1) Due to the trip of D21-K717 what automatic actions, if any, will occur AND (2) as a result of this event 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: C

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.

Distractor Explanation:
Distractors are plausible and incorrect because:

RBHVAC does not isolate based on the stem
29.100.01, Sheet 5, Secondary Containment Control, EOP entry conditions have not been met.

Reference Information:
20.710.01 (Entry Conditions)
16D1 (Actions)

Plant Procedures
16D01

NUREG 1123 KA Catalog Rev. 2

288000 A2.	Ability to (a) predict the impacts of the following on the PLANT VENTILATION SYSTEMS ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:
288000 A2.04	High radiation: Plant-Specific
295023 AK2.	Knowledge of the interrelations between REFUELING ACCIDENTS and the following:
295023 AK2.05	Secondary containment ventilation

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 2013 Exam

NRC Question Use (ILO 2018)
BANK
High
RO

Associated objective(s):
Cycle 12-1 Objectives
Performance Objectives

Performance Terminal
Upon recognizing the entry conditions for a high radiation on the refueling floor, operators respond to control critical parameters within expected limits IAW approved plant procedures within validated completion times

11	K/A Importance: 2.9/3.1			**Points: 1.00**
R11	Difficulty: 4.00	Level of Knowledge: Low	Source: NEW	74606

The Torus Water Management System (TWMS) is in the TORUS FILL mode of operation.

The Torus is currently filling at 500 gpm via the Core Spray Test Line.

A LOCA subsequently occurs that causes a High Drywell Pressure condition.

Which of the following completes the statement below describing the impact of the High Drywell Pressure condition on Suppression Pool makeup?

Suppression Pool makeup will stop because _______

A. the North (South) Torus Water Management Pumps will trip.

B. G5100-F611, TWMS Cond to Torus Makeup Valve will close.

C. G5100-F612 (F613), TWMS Sec Cntm Inbd (Otbd) Iso Valves will close.

D. G5100-F606 (F607), TWMS Rtrn to CS Inbd (Otbd) Iso Valves will close.

Answer: D

Answer Explanation:
The G5100-F606 (607), TWMS to CS Test Line Isolation Valves allow isolation of TWMS from the CS Test Line. When in the Torus Fill mode, these valves would be open. These valves are part of the Primary Containment Isolation System (PCIS) Group 12. They close upon actuation of Group 12 logic due to either High Drywell Pressure, Low RPV Water Level (Level 2), Drywell Floor Drain sump level high-high or Reactor Building Torus sump level high-high.

Therefore, these valves will go closed upon the high Drywell pressure condition specified in the stem of the question, which is what will cause Suppression Pool makeup to be lost.

Distractor Explanation:
Distractors are incorrect and plausible because:

A. The candidate could recall that the TWMS pumps would trip, if in another TWMS mode such as Torus Fill or Torus Cleanup, due to closure of the TWMS Pump Suction Valves on a Group 12 isolation signal. This answer is incorrect, however, because the TWMS pumps do NOT run when in the system is operating in the Torus Fill mode because the fill source comes from the Condensate System.
B. The candidate could recall that the G51-F611 is in the flow path that allows clean condensate, from the Condensate Filter Demineralizers, to be sent to the Torus. The candidate could incorrectly determine that this valve would go closed, to stop this flowpath, on a High Drywell Pressure condition. This answer is incorrect, however, since the F611 only goes closed on a High/High Torus water level condition to prevent over-filling the Torus.
C. The candidate could recall that the G51-F612 (F613) are in the flow path that allows clean condensate, from the Condensate Filter Demineralizers, to be sent to the Torus. The candidate could assume that, since they are labeled as Secondary Containment Isolation Valves, they would go closed on a High Drywell Pressure condition, which is a common misconception among licensed operator candidates. However, although the F612/F613 form part of the secondary containment, they do NOT receive an automatic closure signal on High Drywell Pressure as most other Secondary Containment Isolation Valves do.

Reference Information:
23.144, TWMS System SOP
M-5713, TWMS System FOS
ST-OP-315-0069, TWMS System Student Text.

NUREG 1123 KA Catalog Rev. 2
295024 High Drywell Pressure.

295024 EK2.09 2.9/3.1 Suppression pool makeup: Plant-Specific

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 Question Use (ILO 2018)
Low
NEW
RO

Associated objective(s):
Torus Water Management System (G5100)
Cognitive Enabler
List the automatic features of Torus Water Management System operations.

Torus Water Management System (G5100)
Cognitive Enabler
List the automatic features of Torus Water Management System operations.

12	K/A Importance: 4.4*/4.7*			Points: 1.00
R12	Difficulty: 4.00	Level of Knowledge: High	Source: BANK	74273

Consider each of the conditions listed below:

A – Reactor Pressure is 90 psi above the normal pressure regulator setpoint.
B – Reactor Pressure is at the Reactor Pressure Scram setpoint.
C – Reactor Pressure is 100 psi above the highest SRV safety setpoint.
D – Reactor Pressure is 175 psi above the highest SRV safety setpoint.
E – Reactor Pressure is 225 psi above the highest SRV safety setpoint.

Based on the HIGHEST Reactor Pressure encountered during each condition, which of these would have resulted in the plant NOT being in compliance with (1) LCO 3.4.11, Reactor Steam Dome Pressure, and/or (2) SL 2.1.2, Reactor Coolant System Pressure Safety Limit?

A. (1) ALL of the listed conditions
(2) Conditions D and E only

B. (1) Conditions B through E only
(2) Conditions D and E only

C. (1) ALL of the listed conditions
(2) Conditions C through E only

D. (1) Conditions B through E only
(2) Conditions C through E only

Answer: B

Answer Explanation:
By calculation, the maximum pressures encountered for each of the conditions is as follows:

- A – 90 psi above the normal pressure regulator setpoint (944 - 949 psig) is 1034 - 1039 psig.
- B – The Reactor Pressure Scram setpoint is 1093 psig.
- C – 100 psi above the highest SRV safety setpoint (1155 psig) is 1255 psig.
- D – 175 psi above the highest SRV safety setpoint is 1330 psig.
- E – 225 psi above the highest SRV safety setpoint is 1380 psig.

(1) LCO 3.4.1.1 pressure is 1045 psig, and (2) SL 2.1.2 pressure is 1325 psig. Only condition A results in pressure <(1), and all of the other conditions are > (1). Conditions D and E would result in exceeding (2).

Distracter Explanation:
A. Answer is plausible and incorrect because the examinee could incorrectly determine that 100# above the pressure regulator setpoint would result in exceeding 1045 psig.

C. Answer is plausible and incorrect because the examinee could incorrectly determine that 100# above the pressure regulator setpoint would result in exceeding 1045 psig and/or the examinee could incorrectly determine that, since condition C resulted in exceeding 1250 psig, which is the design pressure of the RPV suction piping, that the Reactor Pressure Safety Limit was exceeded which is a common misconception. However, the SL is based on not exceeding 110% of this pressure.

D. Answer is plausible and incorrect because the examinee could incorrectly determine that, since condition C resulted in exceeding 1250 psig, which is the design pressure of the RPV suction piping, that the Reactor Pressure Safety Limit was exceeded which is a common misconception. However, the SL is based on not exceeding 110% of this pressure.

Reference Information:
S.L. 2.1
T.S. 3.4.11

NUREG 1123 KA Catalog Rev. 2

295007	High Reactor Pressure
295025 EK1.	Knowledge of the operational implications of the following concepts as they apply to High Reactor Pressure:
295025 EK1.05	Exceeding safety limits
G2.2.22	Knowledge of limiting conditions for operations and safety limits

Technical Specifications
2.1 SAFETY LIMITS (SLs)
3.4.11 Reactor Steam Dome Pressure

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 2015 Exam

NRC Question Use (ILO 2018)
BANK
High
RO

Associated objective(s):

Performance Objectives

Cognitive Enabler

Explain the operational implication as it applies to a failed safety relief valve

Performance Objectives

Performance Terminal

Upon recognizing the entry conditions for a safety relief valve failure, operators respond to control critical parameters within expected limits IAW approved plant procedures within validated completion times

THIS PAGE LEFT BLANK

13	K/A Importance: 3.5/3.8			Points: 1.00
R13	Difficulty: 3.00	Level of Knowledge: High	Source: BANK	74274

Following a transient, the following conditions exist:

- Drywell Pressure is 4.0 psig and rising.
- Drywell Temperature is 300ºF and rising.
- Torus Pressure is 3.0 psig and rising.
- Torus Temperature is 98ºF and rising.

After the LNO takes the expected manual actions for the above conditions, which one of the following reflects the expected containment response?

A. Drywell Pressure will lower due to Drywell Spray Initiation.

B. Drywell Temperature will lower due to restoring EECW to the Drywell.

C. Torus Pressure will lower due to ONLY initiating Torus Sprays.

D. Torus Pressure and Torus Temperature will lower due to operation of Torus Spray AND Torus Cooling.

Answer: D

Answer Explanation:
With Torus Temperature > 95ºF, it is required to initiate Torus Cooling. Before Torus Pressure reaches 9 psig, it is required to operate Torus Sprays.

Distractor Explanation:
A is plausible; Drywell Sprays may be used in conditions of high Drywell Pressure and Temperature, Presented conditions forbid this action due to Drywell Spray Initiation Limit violation.
B is plausible; EECW may be restored to the Drywell during conditions of High Drywell Temperature. Presented conditions forbid this action due to Drywell Temperature exceeding 242ºF.
C is plausible; Torus Spray initiation is required due to the conditions in the stem of the question which will result in a reduction in Torus Pressure. However, Torus Cooling is also required and will result in a reduction in Torus Temperature.

Reference Information:
29.100.01 SH 2
DWSIL curve to be provided.

Objective Link:
LP-OP-802-3004-0010

Plant Procedures
29.100.01 SH 2

NUREG 1123 KA Catalog Rev. 2

230000 A1.	Ability to predict and/or monitor changes in parameters associated with operating the RHR/LPCI: TORUS/SUPPRESSION POOL SPRAY MODE controls including:
230000 A1.02	Suppression pool temperature
295026 EK3.	Knowledge of the reasons for the following responses as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE:
295026 EK3.03	Suppression pool spray: Plant-Specific

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

NRC Question Use (ILO 2018)
BANK
High
RO

Associated objective(s):
Primary Containment Control
Performance Terminal
Using the graphs from the EOP Flowcharts, and parameter values for specific plant conditions, determine appropriate operator actions per the EOP Flowcharts.

14	K/A Importance: 3.9/3.8			**Points: 1.00**
R14	Difficulty: 3.00	Level of Knowledge: High	Source: BANK	74326

The reactor was operating at 100% power when a plant transient occurred. Several minutes later the following conditions are observed:

- Drywell pressure is 0.9 psig and rising very slowly.
- Drywell temperature is 155°F and slowly rising.
- Reactor water level is 28 inches.

Which of the following actions will be the MOST EFFECTIVE means of mitigating the rising drywell temperature trend?

A. Manually initiate both divisions of EECW.

B. Unisolate EECW to and from the drywell.

C. Manually start all single speed drywell cooling fans.

D. Increase cooling water flow using P42-F400, RBCCW Temp Control Vlv in AUTO or MANUAL.

Answer: C

Answer Explanation:
Rx water is less than the Level 1 setpoint (32") and results in DW single speed fans tripping. DW fans must be restarted to lower temperature and pressure.

Distractor Explanation:
A. is incorrect because all single speed fans have tripped and cooling water would have no affect on drywell temperature/pressure without fans.
B. is incorrect because a high drywell signal does not exist and with no fans, cooling water would have no impact.
D. is incorrect because all single speed fans have tripped per Level 1 signal.

Reference Information:
29.ESP.01, page 4
29.100.01 S2 DWT leg

Plant Procedures
17D41
29.ESP.01

NUREG 1123 KA Catalog Rev. 2

295028 EA1.	Ability to operate and/or monitor the following as they apply to HIGH DRYWELL TEMPERATURE :
295028 EA1.02	3.9/3.8 Drywell ventilation 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 Question Use (ILO 2018)
BANK
High
RO

Associated objective(s):
EOP Objectives
Cognitive Terminal
Given a copy of 29.100.01 sheet 2, discuss the operator's actions for Drywell Temperature Control (DWT)

15	K/A Importance: 3.7/3.9			Points: 1.00
R15	Difficulty: 4.00	Level of Knowledge: High	Source: BANK	74327

With a Torus Water Level of $\leq$ -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 requried. The need to depressuirze 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 depresurization. However per answer explanation this is incorrect.

Plant Procedures
29.100.01 SH 3

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

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

NRC Question Use (ILO 2018)
BANK
High
RO

Associated objective(s):
RPV Control
Performance Terminal
Describe the plant condition that would require use of the alternative actions contained in 29.100.01 Sh 1, 1A, 3, and 3A, including:
a. Alternate Level Control
b. Emergency RPV Depressurization
c. RPV Flooding
d. deleted
e. Steam Cooling
f. RPV Control - ATWS

16	K/A Importance: 3.9/4.6			**Points: 1.00**
R16	Difficulty: 2.00	Level of Knowledge: Low	Source: NEW	74366

If the following RPV Water Level actuations occurred at the level given, which one would require entry into Technical Specifications LCO 3.3.4.1 Anticipated Transient Without Scram Recirculation Pump Trip (ATWS-RPT) Instrumentation?

A. L1 @ 33 inches

B. L2 @ 109 inches

C. L3 @ 175 inches

D. L8 @ 215 inches

Answer: B

Answer Explanation:
Per TR 3.3.4.1 Anticipated Transient Without Scram Recirculation Pump Trip (ATWS-RPT) Instrumentation

TABLE TR3.3.4.1-1 (Page 1 of 1),
Anticipated Transient Without Scram Recirculation Pump Trip (ATWS-RPT) Instrumentation

	FUNCTION	TRIP SETPOINT
1.	Reactor Vessel Water Level - Low Low, Level 2	≥ 110.8 inches[a]
2.	Reactor Vessel Pressure - High	≤ 1133 psig

(a) As referenced to instrument zero Top of Active Fuel (TAF).

Because the L2 trip is occuring below the setpoint at least 2 instruments must not be tripping at the required setpoint (logic is (A and C) or (B and D)). Therefore at least 2 instruments are INOPERABLE.

TS LCO 3.3.4.1 Two channels per trip system for each ATWS-RPT instrumentation Function listed below shall be OPERABLE: a. Reactor Vessel Water Level -Low Low. Level 2: and b. Reactor Vessel Pressure-High. Therefore entry is required into this LCO

Distractor Explanation:
Distractors are plausible and incorrect because:

L1 is ≥ 31.8 inches so this trip is ok, however L1 is for ECCS and ADS not ATWS-PRT
L3 is ≥ 173.4 inches so this trip is ok, however L3 is for RPS and NSSSS not ATWS-PRT
L8 is ≤ 214 inches so this trip is NOT ok, and L8 is for Main Turbine Protection.

Referece Inforamation:
TS 3.3.4.1 LCO
TR 3.3.4.1 Trip Setpoint
23.601 (L1,L2,L3,L8)

Plant Procedures
23.601

NUREG 1123 KA Catalog Rev. 2

295031	Reactor Low Water Level.
G2.2.42	Ability to recognize system parameters that are entry-level conditions for Technical Specifications

Technical Specifications
3.3.4.1 Anticipated Transient Without Scram Recirculation Pump Trip (ATWS-RPT) 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 Question Use (ILO 2018)
Low
NEW
NRC Early Review
RO

Associated objective(s):
Technical Specifications for Licensed Operators

Performance Enabler

Given plant conditions that constitute non-compliance with any LCO, apply Technical Specifications and Bases to determine the applicable Condition(s), Required Action(s), and associated Completion Time(s).

THIS PAGE LEFT BLANK

17	K/A Importance: 3.6/3.6			**Points: 1.00**
R17	Difficulty: 3.00	Level of Knowledge: High	Source: NEW	74806

A reactor scram condition occurred while the plant was operating at 100% power. The Mode Switch was taken to Shutdown; however, reactor power remained above the APRM downscale trip setpoint.

The P603 Operator is attempting to determine Control Rod status from the Rod Worth Minimizer.

Under which condition, if any, would manual action be required by the P603 Operator to view the Confirm Shutdown screen on the Rod Worth Minimizer?

A. Manual action is NEVER required.

B. Manual action is ALWAYS required.

C. If BOTH RPS Trip Systems failed to actuate.

D. If BOTH RPS Trip Systems actuated but ARI failed to actuate.

Answer: C

Answer Explanation:
The Confirm Shutdown display, which is part of the Rod Worth Minimizer system, provides Rod Control and Information data to the reactor operator. The screen is informational only and provides the operator with all-rods-in or not-in status with control rod I.D. and position for any not full-in control rod(s). Per 23.608, Enclosure A, the Confirm Shutdown display is automatically displayed whenever a full scram takes place.

The full scram signal for the RWM Confirm Shutdown screen comes from RPS Backup Scram logic, via relay K21A, which can be seen on I-2115-22 at grid F-8. Relay K21A is part of RPS A backup scram logic; however, it only energizes on a Full Scram signal when relays K14A or K14C de-energize AND K14F or K14H de-energize, which can be seen on I-2155-11 at grid F-7. The K14 relays de-energize upon a half scram from RPS A (K14A/K14C) or from RPS B (K14F/K14H). Both RPS trip systems, A and B, must trip (Full Scram) to energize relay K21A and actuate the Confirm Shutdown screen on the RWM automatically.

Therefore, the candidate should recall that this screen is automatically displayed ONLY when a Full Scram takes place, which the candidate should determine is when BOTH RPS Trip Systems actuate so, conversely, the candidate should determine that manual action will be necessary if BOTH RPS Trip Systems failed to actuate. Note: The Confirm Shutdown screen is available at any time to the operator through the use of soft keys on the front of the RWM even if it fails to display automatically.

Distractor Explanation:
Distractors are incorrect and plausible because:
A. The candidate could incorrectly recall that the Confirm Shutdown screen actuates off of the Mode Switch position, which is plausible because of the multiple functions performed by the Mode Switch. However, this distractor is incorrect because relay K21A actuates on a Full Scram (trip of RPS A and B) condition to display the Confirm Shutdown Screen.
B. The candidate could fail to recall that the Confirm Shutdown screen automatically displays on a Full Scram condition and therefore determine that manual action is always required to view the Confirm Shutdown screen. This is a common misconception, and is plausible, because of the large number of RPS failure scenarios in which the candidates participate while in the Simulator. These scenarios require the candidate to manually actuate the Confirm Shutdown screen using the soft keys on the front of the RWM. The candidate could recall the numerous times that he/she had to manually actuate the Confirm Shutdown screen and conclude that it must always be done, which is incorrect since the Confirm Shutdown screen does automatically actuate on a Full Scram condition by the relay logic described above.
D. The candidate could incorrectly recall that the Confirm Shutdown screen actuates off of ARI Logic instead of RPS Logic, which is plausible because of the close interrelationships that exist between ARI and RPS. However, this distractor is incorrect because relay K21A actuates on a Full Scram (trip of RPS A and B) condition to display the Confirm Shutdown Screen and not from ARI logic.

Reference Information:
23.608, Rod Worth Minimizer System SOP, Enclosure A.
I-2115-22, RWM Schematic Diagram.
I-2155-11, RPS Scram Disch Vol Isol Valve Ind Lights, Backup Scram Valves.

NUREG 1123 KA Catalog Rev. 2

295037 SCRAM Condition Present and Reactor Power Above APRM Downscale or unknown.

295037 EA1. Ability to operate and/or monitor the following as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN :
295037 EA1.08 3.6/3.6 Rod control and information system: Plant-Specific

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 Question Use (ILO 2018)

High
NEW
RO

Associated objective(s):

Rod Worth Minimizer (C1108)

Cognitive Enabler

Discuss effective monitoring of the Rod Worth Minimizer system using local, remote, computer displays and alarms.

THIS PAGE LEFT BLANK

18	K/A Importance: 3.3/4.0			**Points: 1.00**
R18	Difficulty: 3.00	Level of Knowledge: Low	Source: BANK	74526

Why does the Radioactivity Release Control Leg of the EOPs permit the RESTART of isolated HVAC Systems?

While executing the Radioactivity Release Control Leg, restarting HVAC Systems:

A. ensures a positive pressure is maintained in the Control Room.

B. ensures accessibility is maintained inside the Secondary Containment.

C. provides filtration and adsorption of radioactivity via an elevated release path.

D. ensures accessibility is maintained in buildings outside the Secondary Containment.

Answer: D

Answer Explanation:
Continued personnel access to the turbine building, auxiliary building, or other buildings outside the secondary containment boundary may be essential for responding to emergencies or transients which may degrade into emergencies. These buildings are not always airtight structures, and radioactivity release inside the buildings would not only limit personnel access, but would eventually lead to an unmonitored ground level release. Operating HVAC preserves building accessibility and discharges radioactivity through an elevated, monitored release point.

Distractor Explanation:
Distractors are incorrect and plausible because:
A. the Control Center is outside of Secondary Containment and limiting radiation levels inside the Control Center Envelope is a primary concern during radiation release events. However, 29.ESP.25, Defeat of Radwaste and TBHVAC Radiation Isolations, does NOT include Control Center HVAC.
B. safety related equipment is located INSIDE the Secondary Containment and acting to preserve accessibility to this equipment is important during radiation release events. The candidate could confuse the action in the Secondary Containment Control EOP that permits restarting RBHVAC to preserve access to safety related equipment, however, that is not the basis behind the actions taken in the Radioactivity Release Control EOP.
C. this is the reason why the Standby Gas Treatment System is started while executing the Secondary Containment Control EOP, so the candidate could incorrectly determine that this is the basis for the action taken in the Rad Release EOP. However, although starting HVAC outside of secondary containment does provide an elevated release path, these HVAC systems (RWHVAC and TBHVAC) do NOT provide for filtration and/or adsorption of radioactivity.

Reference Information:
ST-OP-802-3005-001, Secondary Containment Control and Rad Release Student Text.
BWROG EPGs/SAGs, Appendix B, Page B-9-3.
29.ESP.25, Defeat of Radwaste and TBHVAC High Radiation Isolations.

NUREG 1123 KA Catalog Rev. 2
295038 High Off-Site Release Rate
G2.4.18 Knowledge of the specific bases for EOPs

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

NRC Question Use (ILO 2018)
BANK
Low
RO

Associated objective(s):
Secondary Containment Control and Radioactive Release
Performance Terminal
State the basis for the Override Statements contained in 29.100.01 Sh 5, Secondary Containment Control and Radioactive Release.

19	K/A Importance: 2.8/3.4			**Points: 1.00**
R19	Difficulty: 2.00	Level of Knowledge: Low	Source: NEW	74446

A plant fire is in progress. Operators are currently implementing AOP 20.000.22, Plant Fires.

The CRS then orders the Action of Condition C to PERFORM 20.000.18, Control of the Plant from the Dedicated Shutdown Panel.

What is the reason for performing this action?

A. The operators need to evacuate the Main Control Room because the fire has caused it to become uninhabitable.

B. The need to shut down the plant from the Dedicated Shutdown Panel exists due to personnel feeling the effects of smoke inhalation.

C. There is the possibility of equipment malfunctions and safe operation of the plant may be jeopardized due to a fire in one or more of the 3L zones.

D. The possibility of losing adequate core cooling exists because High Pressure Feed sources are insufficient to maintain RPV Level 3 and two or more SRVs are lifting.

Answer: C

Answer Explanation:
This question requires the candidate to know the reason why Action C of 20.000.22, Plant Fires is performed.
The reason that Action C of 20.000.22, Plant Fires is performed is stated in the 20.000.22 Bases document section for Action C, which states "The alternative shutdown system was designed to provide safe-shutdown capability separate and remote from the control center complex, when a fire in the complex or the other 3L zones is assumed to significantly damage the equipment or cabling in these zones. If there is a fire in one or more of the 3L zones, then there is the possibility of equipment malfunctions and safe operation of the plant may be jeopardized. It is for this reason that 20.000.18 is entered."

Therefore, the operator should determine that this action has been directed by the CRS because of the possibility of equipment malfunctions and safe operation of the plant may be jeopardized when a fire in a 3L zone of concern is detected.

Distractor Explanation:
Distractors are incorrect and plausible because:

A. The candidate could confuse the basis for performing Condition C with the basis for entering 20.000.19, Shutdown from Outside the Main Control Room. This could lead the candidate to conclude that, if in the Fire AOP and if the MCR becomes uninhabitable, that control would be shifted to the Dedicated Shutdown Panel. This is incorrect because control would be shifted to the Remote Shutdown Panel, NOT the Dedicated Shutdown Panel, and procedure 20.000.19 performed rather than 20.000.18 as stated in the stem of the question.
B. The candidate could recognize the condition in this distractor as being a condition (IF statement) for one of the Overrides in 20.000.22 and conclude that the correct action to take, if personnel in the MCR are feeling the effects of smoke inhalation, is to transition control to the Dedicated Shutdown Panel. This is incorrect because the Override requires all personnel to don respirators under this condition.
D. The operator could recognize the condition in this distractor as being a condition (IF statement) for one of the Overrides in 20.000.22. This could lead the candidate to conclude that the correct action to take, if the possibility of losing adequate core cooling due to a loss of adequate HP Feed sources with two open SRVs exists, is to transition control to the Dedicated Shutdown Panel. This is incorrect because the Override requires the operators to enter the EOPs for RPV Level Control.

Reference Information:
20.000.22, Plant Fires AOP
20.000.22 Bases - Plant Fires AOP BASES
20.000.19, Shutdown from Outside the Main Control Room
20.000.19, Shutdown from Outside the Main Control Room BASES

NUREG 1123 KA Catalog Rev. 2

600000	Plant Fire On Site
600000 AK3.	Knowledge of the reasons for the following responses as they apply to PLANT FIRE ON SITE:
600000 AK3.04	Actions contained in the abnormal procedure for 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 Question Use (ILO 2018)
Low
NEW
RO

Associated objective(s):
Reactor Operator
Exam Objectives
    Performance Enabler
    Explain bases for notes, cautions and overrides.

THIS PAGE LEFT BLANK

20	K/A Importance: 3.3/3.4			**Points: 1.00**
R20	Difficulty: 2.00	Level of Knowledge: Low	Source: NEW	74406

Following a Grid Disturbance and plant transient, Main Turbine Generator (MTG) conditions are as shown:

What is the current operating condition of the MTG and what action could the operator take to restore MTG MVARs to the GREEN BAND?

A. The MTG is currently over-excited;
Operate the Turbine Speed/Load Controls in the raise direction.

B. The MTG is currently over-excited;
Operate the Voltage Reg Control SW 90CS in the raise direction.

C. The MTG is currently under-excited;
Operate the Turbine Speed/Load Controls in the raise direction.

D. The MTG is currently under-excited;
Operate the Voltage Reg Control SW 90CS in the raise direction.

Answer: D

Answer Explanation:
23.118 Enclosure G indicates that the conditions shown in the stem of the question place the MTG in an Under-excited (Leading Power Factor) condition.

To cause MVARS out of the machine, and restore MTG MVARs to the GREEN BAND, the operator would raise the Voltage Reg Control SW 90CS until VARS are positive (out). This guidance can be seen on Page 27 of 23.118.

Distractor Explanation:
Distractors are incorrect and plausible because:

A. The candidate could determine that the conditions shown place the MTG in an over-excited condition, which is incorrect as can be seen on 23.118, Enclosure G. The candidate could also conclude that the Turbine Speed/Load Controls must be placed in the raise direction to restore MTG MVARs to the green band, which is incorrect as described in 23.118.
B. The candidate could determine that the conditions shown place the MTG in an over-excited condition, which is incorrect as can be seen on 23.118, Enclosure G. The candidate could correctly conclude that the Voltage Reg Control Switch must be placed in the raise direction to restore MTG MVARs in the green band.
C. The candidate could determine that the conditions shown place the MTG in an under-excited condition, which is correct as can be seen on 23.118, Enclosure G. The candidate could then conclude that the Turbine Speed/Load Controls must be placed in the raise direction to restore MTG MVARs to the green band, which is incorrect as described in 23.118.

Reference Information:
23.118, Main Generator and Generator Excitation System SOP, Enclosure G (Generator Operating Guidelines).

NUREG 1123 KA Catalog Rev. 2
700000 Generator Voltage and Electric Grid Disturbances
700000 AK1. Knowledge of the operational implications of the following concepts as they apply to GENERATOR VOLTAGE AND ELECTRICAL GRID DISTURBANCES:
700000 AK1.03 3.3/3.4 Under-excitation

10CFR55 RO/SRO Written Exam Content
10 CFR 55.41(b) (4) Secondary coolant and auxiliary systems that affect the facility.

NRC Question Use (ILO 2018)
Low
NEW
RO

Associated objective(s):
Main Generator and Excitation (N3031 & N3034)
Cognitive Enabler
Identify associated remote and local instrumentation, indications, alarms, and controls for the Main Generator and Excitation System.

Main Generator and Excitation (N3031 & N3034)
Cognitive Enabler
Identify associated remote and local instrumentation, indications, alarms, and controls for the Main Generator and Excitation System.

21	K/A Importance: 4.5/4.7			Points: 1.00
R21	Difficulty: 3.00	Level of Knowledge: Low	Source: NEW	74386

Which of the following is a Symptom for entry into Abnormal Operating Procedure 20.125.01, Loss of Condenser Vacuum?

A. Decreasing Offgas Outlet Flow.

B. Decreasing Offgas Delay Pipe Pressure.

C. Decreasing Main Generator Megawatts..

D. N62-F406A, North Ringwater Vac Pump Recirc Line PCV indicates full OPEN.

Answer: C

Answer Explanation:
20.125.01, Loss of Condenser Vacuum, lists the following as symptoms that require entry into this AOP:

- Degrading condenser vacuum as indicated on N30-R824, Main Cond Vacuum Pressure Recorder and/or N30-R823, Main Cond Vacuum Pressure Indicator
- Main Generator MW decreasing
- 4D108, CONDENSER PRESSURE HIGH
- 6D16, OFF GAS SYS MN CONDENSER PRESSURE HIGH

The candidate should recognize that, even though each of the Distractors are indications and/or components that would respond to degrading Off Gas and condenser vacuum, decreasing MTG MW is the only one that is explicitly listed as symptom.

Distractor Explanation:
Distractors are incorrect and plausible because:

A. Off Gas flow changes in response to Condenser air in-leakage and degradations in the Off Gas system can lead to a loss of condenser vacuum. The candidate could determine that the decreasing Off Gas flow is either the result of, or could be the cause of, degrading condenser vacuum and therefore conclude that AOP entry is required. However, this indication is not a symptom listed in 20.125.01 requiring entry into the AOP.
B. Off Gas Delay Pipe Pressure changes in response to Condenser air in-leakage and degradations in the Off Gas system can lead to a loss of condenser vacuum. The candidate could determine that the decreasing Delay Pipe Pressure is either the result of, or could be the cause of, degrading condenser vacuum and therefore conclude that AOP entry is required. However, this indication is not a symptom listed in 20.125.01 requiring entry into the AOP.
D. The N62-F406A changes position in response to changes in Off Gas Delay Pipe pressure. The candidate could assume that the F046A opens more to increase flow through Off Gas in response to increasing Condenser air in-leakage and, therefore, associate this indication as being a symptom of AOP 20.125.01; however, the N62-F406A going open is not a symptom listed in 20.125.01.

Reference Information:
20.125.01, Loss of Condenser Vacuum.

NUREG 1123 KA Catalog Rev. 2
295002 Loss of Main Condenser Vacuum
G2.4.4 Ability to recognize abnormal indications for system operating parameters which are entry-level conditions for emergency and abnormal operating procedures

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

NRC Question Use (ILO 2018)
Low
NEW
RO

Associated objective(s):
Reactor Operator
Performance Enabler
Appropriately recognize symptoms for entry into emergency and abnormal operating procedures.

22	K/A Importance: 3.1/3.4			Points: 1.00
R22	Difficulty: 3.00	Level of Knowledge: High	Source: NEW	74306

The plant has been operating at 100% power for 300 consecutive days, when the following events occur:

- A reactor scram causes the crew to place the Mode Switch in Shutdown.
- The MSIVs closed 10 minutes later.
- RPV Pressure is currently 1095 psig.
- The CRLNO has been directed to maintain RPV Pressure 900-1050 psig.

How many SRVs are going to be required to restore and maintain RPV pressure in the desired control band considering heat input by DECAY HEAT GENERATION ONLY?

A. One SRV opening and closing at it's Low-Low Set setpoints.

B. One open SRV will restore RPV pressure in band, then it can be closed for the duration of the event.

C. One Low-Low Set SRV will be open continuously, with the second opening and closing at it's Low-Low Set setpoints.

D. Both Low-Low Set SRVs will be open continuously with the CRLNO opening and closing one additional SRV.

Answer: A

Answer Explanation:
The conditions in the stem of the question have caused a High Reactor Pressure. The candidate will have to determine the impact of decay heat generation so as to correct the High Reactor Pressure and control RPV Pressure in the designated control band for the duration of the event.

Decay heat produced is at a level dependent on power history. From a scram at 100% power, initially the thermal output of the reactor will be about 100% power, 7% supplied by decay heat. Thermal heat output decreases rapidly to the decay heat level. Eight to ten seconds after the scram, thermal output is due mainly to decay heat and drops to 7% of rated thermal output. After approximately one minute, thermal output is 3 to 5% of rated and drops to about 2% after about ten minutes. One hour after a scram, decay heat is about 1% rated thermal output. The information given in the stem of the question indicates that the reactor has been shut down for about 10 minutes, so decay heat generation will be between approximately 2%.

SRV capacity (steam flow) is given in UFSAR Paragraph 5.2.2.3.3.4 as 87E4 lb/hr (or 870,000 lb/hr) at 1090 psig, which is approximately where the stem of the question has the plant. UFSAR Section 1.2.2, Plant Description, shows that rated steam flow at Fermi 2 is 14.9 lb/hr at 991 psia. 2% of this value is 298,000 lb/hr, which is within the capacity of one SRV. Therefore, the candidate should determine that one SRV opening and closing at its Low-Low setpoint will be enough to control the steam generated from decay heat approximately 10 minutes after plant shutdown from 100% power operation.

Distractor Explanation:
Distractors are incorrect and plausible because:

B. The candidate could determine that there is no significant decay heat generation 10 minutes after shutdown and, once RPV pressure is lowered back in band by opening an SRV, it will remain in band with no further SRV actuations. This is incorrect because decay heat, in the 2% range, exists and will continue to exist at the 1% value one hour after shutdown, so further SRV actuation must occur to control RPV pressure.
C. The candidate could determine that the decay heat generated 10 minutes after shutdown is above the capacity of one, but within the capacity of two Low-Low Set SRVs. This is a common misconception because most candidates readily remember that decay heat drops to 7% within several seconds of a plant shutdown, due to the decay of delayed neutrons and they determine that 7% is the capacity of about one and a half SRVs. However, as stated above, after approximately ten minutes, thermal output due to decay heat drops to about 2%, which is within the capacity of one Low-Low Set SRV.
D. The candidate could determine that the decay heat generated 10 minutes after shutdown is above the capacity of both Low-Low Set SRVs, which would require the operator to manually actuate one additional SRV to control RPV pressure in band. This is incorrect, however, as stated above, because after approximately ten minutes, thermal output due to decay heat drops to about 2%, which is within the capacity of one Low-Low Set SRV.

Reference Information:
BR08Sr5_Operational_Physics May 2011 - Reactor Operational Physics GFE Student Text, Page 33 description of decay heat generation after shutdown from 100% power.
UFSAR Section 1.2.2, Plant Description.
UFSAR Section 5.2.2.3.3.4, Safety/Relief Valve Characteristics.

NUREG 1123 KA Catalog Rev. 2

295007 High Reactor Pressure

295007 AK1. Knowledge of the operational implications of the following concepts as they apply to HIGH REACTOR PRESSURE:

295007 AK1.02 3.1/3.4 Decay heat generation

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 Question Use (ILO 2018)

High

NEW

RO

Associated objective(s):
Operational Physics
Cognitive Enabler

1. Explain the relationship between decay heat generation and: a. Power level history b. Power production c. Time since reactor shutdown

Nuclear Boiler System (B1100, B2100, B2103, B2104, N1100 & N3017)
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.

23	K/A Importance: 3.6/3.7			Points: 1.00
R23	Difficulty: 4.00	Level of Knowledge: High	Source: NEW	74346

The Mode Switch has been taken to shutdown at 30% power for the start of RF-19. The following plant conditions currently exist:

- Post Scram Feedwater Logic actuated on the Scram.
- Reactor Pressure has been lowered to 500 psig.
- 3D148, FW/MTG RPV H2O Level 8 Trip is in alarm.
- RPV water level is 216" and slowly rising.
- 20.000.23, High RPV Water Level has been entered.

Which of the following Reactor Water Level Control actions should be taken, in accordance with the High RPV Water Level AOP, to mitigate these conditions?

A. Lower SETPOINT on C32-R618, Master Feedwater Level Controller.

B. Lower DEMAND on C32-R620, N21-F403 RPV Startup LCV Controller.

C. Lower DEMAND on C32-R616A (B), N (S) Reactor Feed Pump Controllers.

D. Lower SETPOINT on N21-K858A (B), N (S) Reactor Feed Pump Min Flow Controllers.

Answer: B

Answer Explanation:
The conditions given in the stem of the question indicate that a high RPV water level condition has occurred. AOP 20.000.23, High RPV Water Level, provides actions that can be taken, for a high RPV water level condition, when leakage past the Startup Level Control Valve (SULCV) is suspected. For this condition, the operator should attempt to lower the demand on the SULCV Controller to minimum (-5) to attempt to fully seat the valve. The candidate should recognize that this action should be taken for the conditions given in the stem of the question.

Distractor Explanation:
Distractors are incorrect and plausible because:

A. The C32-R618, Master Feedwater Level Controller, is the controller that is normally manipulated to control components associated with the Feedwater Control System, and the candidate could assume that lowering the setpoint of this controller would have the desired impact. However, when a scram occurs, Post Scram Feedwater Logic actuates as described in ARP 3D157, Post Scram FW Logic Actuated. One of the results of this is that the C32-R616A and B (N and S RFP Controllers) get set to MANUAL mode. When this occurs, the C32-R618 also gets set to MANUAL, so it no longer controls the RFP speeds. Also, as stated in Section 1.1 of 23.107, this controller can NOT be placed in AUTO until one of the RFP controllers is placed in AUTO, so lowering its setpoint will not have any impact.
C. After a scram, the C32-R616A and B (N and S RFP Controllers) get set to MANUAL mode. When this occurs, the operator can adjust the output of these controllers to change RFP speed. The candidate could determine that lowering the output of these controllers will have the desired impact. However, the stem of the question states that RPV level is 216", which is above the High RPV Water Level Trip Setpoint of 214" (Level 8), so the RFPs will be tripped and lowering their controller outputs will have no impact.
D. The candidate could determine that leakage is occurring through the RFP Min Flow lines, which could be impacting RPV Water Level. The candidate could assume that lowering the Setpoint of these controllers will have the desired impact. However, the stem of the question states that RPV level is 220", which is above the High RPV Water Level Trip Setpoint of 214" (Level 8), so the RFPs will be tripped. When an RFP trips, its Minimum Flow Line motor operated valve (MOV) goes shut to isolate the minimum flow line, so adjusting these controllers will have no impact.

Reference Information:
20.000.32, High RPV Water Level AOP.
23.107, Reactor Feedwater and Condensate Systems, Section 1.1, System Description for the Feedwater Control System.
ARP 3D157, Post Scram FW Logic Actuated.

NUREG 1123 KA Catalog Rev. 2
295008 High Reactor Water Level
295008 AK2. Knowledge of the interrelations between HIGH REACTOR WATER LEVEL and the following:
295008 AK2.03 3.6/3.7 Reactor water level control

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 Question Use (ILO 2018)
High
NEW
RO

Associated objective(s):
Feedwater Control (C3200)
Cognitive Enabler
Discuss failure modes of Feedwater Control System controls and vital instruments, including design features that could result in erroneous operation or indication.

24	K/A Importance: 3.9/4.2			**Points: 1.00**
R24V2	Difficulty: 2.00	Level of Knowledge: Low	Source: NEW	75128

Which of the following Process Radiation Monitors, if verified to be at their high-high or upscale trip setpoints, would require entry into 29.100.01, Sheet 5, Secondary Containment Control on Secondary Containment Exhaust High Radiation?

A. D11-K601A (B) Channel A (B) Off Gas Radiation Monitors.

B. D11-K808 (K810) Div 1 (2) RB Vent Exhaust Radiation Monitors.

C. D11-K809 (K813) Div 1 (2) CCHVAC Makeup Air Radiation Monitors.

D. D11-K603A (B, C, D) Channel A (B, C, D) Main Steam Line Radiation Monitors.

Answer: B

Answer Explanation:
The D11-K808 (K810) Div 1 (2) RB Vent Exhaust Radiation Monitors have an Upscale Trip setpoint of 16,000 cpm. If the radiation monitor is at it's trip setpoint, as stated in the stem of the question, then entry into 29.100.01 Sheet 5, Secondary Containment Control EOP is required. This requires the candidate to know the interrelationship between this Process Radiation Monitor and Secondary Containment High Radiation.

Distractor Explanation:
Distractors are incorrect and plausible because:
A. The D11-K601A (B) Channel A (B) Off Gas Radiation Monitors have upscale trip setpoints of 1,000 mr/hr. Per ARP 3D12, if this radiation monitor is at it's trip setpoint, entry into 20.000.07, Fuel Cladding Failure, is required. This distractor is plausible because the candidate could determine that, since a fuel clad failure is evidenced by this alarm, and radiation levels in secondary containment could increase as a result of the fuel failure, EOP entry is required, which is incorrect.
C. The D11-K809 (K813) Div 1 (2) CCHVAC Makeup Air Radiation Monitors have upscale trip setpoints of 150 (350) cpm respectively. Per ARP 3D45, if these radiation monitors are at their trip setpoint, entry into 20.000.02, Abnormal Release of Radioactive Material, is required. This distractor is plausible because the candidate could determine that, since an abnormal release of radioactive material has occurred, then EOP entry must also be required, which is incorrect.
D. The D11-K603A (B, C, D) Channel A (B, C, D) Main Steam Line Radiation Monitors have upscale trip setpoints of 3,100 (3,300, 3,400, 3,000) mr/hr respectively. Per ARP 3D82, if these radation monitors are at their alarm setpoints, entry into AOPs 20.000.07, Fuel Clad Failure and 20.000.21, Reactor Scram is required. The candidate could determine that the radiation level sensed by these monitors, if reached, would warrant EOP entry. Or, the candidate could determine that, since a fuel clad failure is evidenced by this trip, and radiation levels in secondary containment could increase as a result of the fuel failure, that EOP entry is required, which is incorrect.

Reference Information:
3D12 Offgas Rad Monitor Rad High-High
3D36 RB Vent Exh Rad Monitor Upscale Trip
3D45 CCHVAC Makeup Air Rad Monitor Upscale Trip
3D82 Mn Steam Line Radiation Upscale/Inop Channel Trip

NUREG 1123 KA Catalog Rev. 2
295034 Secondary Containment Ventilation High Radiation
295034 EK2. Knowledge of the interrelations between SECONDARY CONTAINMENT VENTILATION HIGH RADIATION and the following:
295034 EK2.01 3.9/4.2 Process radiation monitoring system

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 Question Use (ILO 2018)
Low
NEW
RO

Associated objective(s):
Reactor Operator
Performance Enabler
Appropriately recognize symptoms for entry into emergency and abnormal operating procedures.

Process Radiation Monitoring System (D1100)
Cognitive Enabler
Identify abnormal and emergency operating procedures associated with the Process Radiation Monitoring System.

25	K/A Importance: 3.3/3.5			Points: 1.00
R25	Difficulty: 3.00	Level of Knowledge: Low	Source: NEW	74286

The control room receives annunciators 8D46 and 17D46, Div I and II Reactor Building Pressure High/Low, during a severe thunderstorm and high winds in the area.

The CRLNO observes that both the T41-R800A and B, Div 1 and 2 CR and RB Differential Pressure Recorders indicate RB differential pressure is reading +0.5 inches WC.

Which of the following indicates the response of secondary containment ventilation to these conditions as well as the reason for this response?

A. The RBHVAC Supply and Exhaust Fans will automatically trip to prevent damage to RBHVAC ductwork.

B. The RB Supply Fan blades will modulate to restore a negative differential pressure to prevent the unmonitored release of potentially contaminated air from Secondary Containment.

C. The RB Exhaust Fan inlet dampers will modulate to restore a negative differential pressure to prevent the unmonitored release of potentially contaminated air from Secondary Containment.

D. The RBHVAC Supply and Exhaust Fans will automatically trip and SGTS will automatically start to prevent damage to RBHVAC ductwork AND to restore a negative differential pressure to prevent the unmonitored release of potentially contaminated air from Secondary Containment.

Answer: C

Answer Explanation:
The Secondary Containment Atmosphere Pressure Monitoring System (which is part of the RBHVAC System) modulates the inlet dampers to the exhaust fans to maintain a differential pressure in the Reactor Building with respect to the outside air of -1/4" H2O.
With high wind conditions, RB pressure can increase and cause high RB pressure alarms to come in. The system will modulate the position of the exhaust fan dampers to restore negative RB pressure to prevent an unmonitored release of potentially contaminated air from Secondary Containment.

Distractor Explanation:

A. Is plausible because the RBHVAC Supply and Exhaust fans do trip at high RB pressure, but not until 2.5 inches WC. For a pressure this high, it is plausible to assume the fans would trip to protect the integrity of the RBHVAC ductwork.
B. Is plausible because some fans for various ventilation systems at Fermi do have, or had in the past, variable blading to control air flow. Also, the CCHVAC Compressors have variable inlet vanes to control compressor capacity. Additionally, some ventilation systems modulate the inlet dampers to their Supply Fans to maintain building pressure, but not RBHVAC. The reason for restoring RB pressure in this response is correct. Note: 2 (of 4) validators selected this response during validation, which lends further plausibility to this distractor.
D. Is plausible because the RBHVAC Supply and Exhaust fans do trip at high RB pressure, but not until 2.5 inches WC. Additionally, most RBHVAC system trips occur due to Secondary Containment Isolations, which also result in an automatic start of SGTS. The resons behind this response are plausible because it is desirable to restore negative building pressure to prevent an unmonitored release from Secondary Containment and the candidate could assume that the high RB pressure could damage the RBHVAC ductwork.

Reference Information:
23.426 RBHVAC System SOP, Section 1.1 System Description for explanation of why negative pressure is maintained in the RB as well as a list of RBHVAC fan trips.
T41-00 RBHVAC System Design Basis Document, Section 4.1.8.1.

NUREG 1123 KA Catalog Rev. 2
295035 Secondary Containment High Differential Pressure
295035 EK3. Knowledge of the reasons for the following responses as they apply to SECONDARY CONTAINMENT HIGH DIFFERENTIAL PRESSURE:
295035 EK3.02 3.3/3.5 Secondary containment ventilation response

10CFR55 RO/SRO Written Exam Content
10 CFR 55.41(b) (9) Shielding, isolation, and containment design features, including access limitations.

NRC Question Use (ILO 2018)
Low
NEW
RO

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

Reactor Building HVAC
Cognitive Enabler
Describe general Reactor Building HVAC System operation, including component operating sequence, normal operating parameters, and expected system response.

26	K/A Importance: 3.5/3.6			Points: 1.00
R26	Difficulty: 3.00	Level of Knowledge: High	Source: NEW	74272

The plant is operating at 97% power when a seismic event occurs. The following alarms and indications are received:

- 6D69, SEISMIC SYSTEM EVENT/TROUBLE
- 2D78, REAC BLDG FLR/EQUIP DRN SUMPS LVL HI-HI/LO-LO
- 2D105, REAC BLDG CORNER ROOMS/HPCI ROOM FLOOD LEVEL

The G11-R651, SE Corner Sump G1101-D074 Level Recorder, indicates the following:

- Sump Level is steady at 31.9 inches.
- Flood Level is downscale at 5.0 inches.

The G11-R654, SW Corner Sump G1101-D076 Level Recorder, indicates the following:

- Sump Level is lowering at 44.0 inches.
- Flood Level is downscale at 5.0 inches.

The G11-R655, HPCI Room Level Recorder, indicates 39.0 inches and rising.

Both G1101-D076 RB SW Floor Drn Sump Pumps are running.

Which one of the following actions is required?

A. Close the E4150-F004, HPCI CST Suction Isolation Valve.

B. Close the E2150-F036B, Div 2 Core Spray Pumps Torus Suction Valve.

C. Close the E1150-F004B (D) RHR Pump B (D) Torus Suction Isolation Valves.

D. Close the E4150-F002 (F600), HPCI Steam Supply Inboard (Outboard Bypass) Isolation Valves.

Answer: A

Answer Explanation:
Indications are given that a leak is coming from the suction (water) side of the HPCI System. 29.100.01 Sheet 5 directs the operator to isolate systems leaking into secondary containment when the Max Norm Sump Level or Area Water Level is reached. The candidate should recall that 45 inches in the D076 sump is the Max Norm Operating Water Level in that sump and 8 inches in the HPCI Room is the Max Norm Operating Water Level, therefore conditions dictate that a system must be isolated.

Next, the candidate should determine which system to isolate. The candidate should recall that the HPCI room does not have its own floor drain sump and instead drains water, via a trough and pipe, to the D076 (SW floor drain) sump. Since this sump is located in the same room as the Div 2 RHR pumps, the candidate must determine that, since Flood Level in the SW corner is steady and flood level in the HPCI room is rising, that the leak is from the HPCI system and not Div 2 RHR. Lastly, the candidate must decide between the HPCI system and Div 2 Core Spray. Since HPCI and Div 2 Core Spray are both in the SE corner of the Reactor Building (separated by a normally closed watertight door), the candidate should recognize that the D074, SE Sump and Flood Level indicators are steady, which indicates that the leak is NOT from Div 2 Core Spray.

Note: Flood Control Valve T4500-F601 exists in the line between the HPCI room trough and the SW quadrant sump (DO76) to prevent cross-flooding of the SW Quadrant and the HPCI Room. Since the HPCI Room does not have its own dedicated sump, its floor and trench drains are directed to G1101-D076 (i.e., the SW Quad).
T4500-F601 is normally open to allow the HPCI drains to flow into D076. In the event of excessive equipment leakage or a line break, the valve will close on a D076 High/High alarm (+45"). This will prevent water from backing up into the drains and flooding the unaffected area.
Therefore, the candidate must conclude that the leak is from the HPCI water side, which must be isolated.

Distractor Explanation:
Distractors are incorrect and plausible because:

B. The candidate could recall that the HPCI system and Division 2 Core Spray are both in the Southeast corner of the Reactor Building and assume that the rising sump level and floor level are due to a leak from Div 2 Core Spray. This is incorrect because HPCI and Div 2 Core Spray are separated by a normally closed watertight door. Even though this door could be open, the stem of the question indicates an increasing HPCI room and D076 sump level and NOT an increasing D074 sump level, which is the sump that would indicate a leak from Div 2 Core Spray.
C. The candidate could fail to recall that the HPCI room does not have its own Floor Drain Sump and instead drains to the SW (D076 Sump), which is the same area as Division 2 RHR, in the Southwest (SW) corner of the Reactor Building. This could lead the candidate to focus in only on the rising D076 (SW corner room) sump level and assume that it is due to a leak from Div 2 RHR. This is incorrect because the stem of the question indicates an increasing HPCI room and D076 sump level and NOT a SW Corner Room Flood Level (measured from the floor) that would indicate a leak from Div 2 RHR. Additionally, the stem indicates that D076 level is lowering, which the candidate should determine is due to the T4500-F601 going closed on high/high sump level (+45").
D. The candidate could determine that the leak is from the HPCI system and conclude that the steam supply isolation valves must be closed. This is incorrect because a leak from the HPCI steam supply, although it would cause an increase in sump activity (due to condensation of the steam on the walls and floor of the HPCI room), would not cause the changes in sump volume represented by the information given in the stem of the question.

Reference Information:
29.100.01 SH 5, Secondary Containment Control.
2D78, REAC BLDG FLR/EQUIP DRN SUMPS LVL HI-HI/LO-LO
2D105, REAC BLDG CORNER ROOMS/HPCI ROOM FLOOD LEVEL

Plant Procedures
29.100.01 SH 5
29.ESP.01

NUREG 1123 KA Catalog Rev. 2
295036 Secondary Containment High Sump / Area Water Level
295036 EA1. Ability to operate and/or monitor the following as they apply to SECONDARY CONTAINMENT HIGH SUMP/AREA WATER LEVEL:
295036 EA1.02 3.5/3.6 Affected systems so as to isolate damaged portions

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

NRC Question Use (ILO 2018)
High
NEW
RO

Associated objective(s):
Secondary Containment Control and Radioactive Release
Performance Terminal
State the criteria for isolating a system that is discharging into a flooded area of the Secondary Containment.

Secondary Containment Control and Radioactive Release
Performance Terminal
State the actions required if all the Reactor Building corner rooms were flooded to 5 feet from primary or secondary systems.

THIS PAGE LEFT BLANK

27	K/A Importance: 3.3/3.3			Points: 1.00
R27	Difficulty: 4.00	Level of Knowledge: High	Source: NEW	74906

A Design Basis Accident (DBA) Loss of Coolant Accident (LOCA) is in progress and fuel damage has occurred.
The Primary Containment Monitoring System is aligned in its NORMAL configuration.
The following primary conditions exist:

- Alarms 8D55 and 17D8, Div I and Div II Hydrogen Concentration High, are both in alarm.
- Alarms 8D54 and 17D7, Div I and Div II Hydrogen Concentration High-High, are both in alarm.
- Hydrogen Concentration in both the Drywell and Supression Chamber is 6.0% and rising.

Which of the indications below would represent a combustible mixture in the SUPPRESSION CHAMBER?

A. 8D50, Div I Oxygen Concentration High, with the T50-R806A, Div 1 Drywell Atm Analysis Recorder at 3.5% Oxygen.

B. 17D12, Div II Oxygen Concentration High, with the T50-R806B, Div 2 Drywell Atm Analysis Recorder at 3.5% Oxygen.

C. 8D51, Div I Oxygen Concentration High-High, with the T50-R806A, Div 1 Drywell Atm Analysis Recorder at 4.5% Oxygen.

D. 17D4, Div II Oxygen Concentration High-High, with the T50-R806B, Div 2 Drywell Atm Analysis Recorder at 4.5% Oxygen.

Answer: D

Answer Explanation:
From the basis for TS LCO 3.6.3.1, Primary Containment Oxygen Concentration:

"The primary containment (both drywell and suppression chamber) oxygen concentration is maintained < 4.0% to ensure that an event that produces any amount of hydrogen does not result in a combustible mixture inside primary containment."

Therefore, as long as Primary Containment Oxygen Concentration remains below 4.0%, a combustible mixture is not present.

The stem of the question gives that Hydrogren Concentration is above 4%, which is the MEL (minimum explosive limit) for Hydrogen. The candidate must recognize the significance of Oxygen concentration being above 4% in the Suppression Chamber with >4% Hydrogen.

Division I PCMS is normally alligned to monitor the Drywell atmosphere. Division II PCMS is normally aligned to monitor the Suppression Chamber, therefore only the Division II indications (alarms and indications on the H11-P817) would indicate Oxygen above 4% in the Suppression Chamber.

The setpoint for 17D12, Div II Oxygen Concentration High is 3.5%. With Oxygen at the alarm setpoint, a combustible mixture is NOT present.
The setpoint for 17D4, Div II Oxygen Concentration High-High is 4.5%. With Oxygen at the alarm setpoint, a combustible mixture IS present.

Therefore, the candidate must recognize that a combustible mixture is only present, given the High Primary Containment Hydrogen Concentration in the stem of the question, when a valid 17D12 alarm is received, as evidenced by 4.5% Oxygen Concentration.

Distractor Explanation:
Distractors are incorrect and plausible because:

A. The candidate could incorrectly determine that 3.5% Oxygen Concentration would form a combustible mixture with 6% Hydrogen. However, this is not true with only 3.5% Oxygen. Also, these are Division I indications, which are indicative of Drywell Oxygen and not the Suppression Chamber.
B. These are Division II indications, so they are indicative of Suppression Chamber Oxygen Concentration. The candidate could incorrectly determine that 3.5% Oxygen Concentration would form a combustible mixture with 6% Hydrogen. However, this is not true with only 3.5% Oxygen.
C. The indications given are above 4% Oxygen and would represent a combustible mixture given 6% Hydrogen. However, these are Division I indications, which are indicative of Drywell Oxygen Concentration and not the Suppression Chamber.

Reference Information:
Tech Spec Bases 3.6.3.1, Primary Containment Oxygen Concentration.
ARPs 8D55 and 17D8, Div I and Div II Hydrogen Concentration High.
ARPs 8D54 and 17D7, Div I and Div II Hydrogen Concentration High.
ARP 8D50, Div I Oxygen Concentration High.
ARP 8D51, Div I Oxygen Concentration High-High.
ARP 17D4, Div II Oxygen Concentration High-High.
ARP 17D12, Div II Oxygen Concentration High.

NUREG 1123 KA Catalog Rev. 2
500000 High Containment Hydrogen Concentration.
500000 EA2. Ability to determine and / or interpret the following as they apply to HIGH PRIMARY CONTAINMENT HYDROGEN CONCENTRATIONS:
500000 EA2.04 3.3/3.3 Combustible limits for wetwell

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

NRC Question Use (ILO 2018)
High
NEW
RO

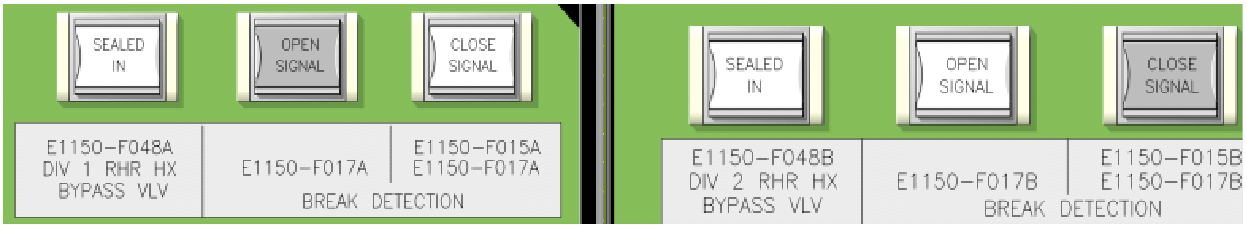
Associated objective(s):
Primary Containment Monitoring System (T5000)
Cognitive Enabler
Given various system operating parameters, relate system/equipment operation to fundamental concepts to determine proper operation/response as described in the BWR Fundamentals Catalog.

THIS PAGE LEFT BLANK

Two minutes after a small steam leak develops inside the drywell from the B Main Steam line, the following conditions exist:

- Drywell pressure is 3.0 psig.
- Reactor pressure is 750 psig and slowly lowering.
- RPV level is 170 inches and slowly rising.

Which of the following indications are correct for the current plant conditions?

A.


B.


C.


D.


A. A.

B. B.

C. C.

D. D.

Answer: D

Answer Explanation:
The conditions presented in the stem indicate that RHR LPCI Loop Select Logic will have determined the loop selected for injection since a High Drywll Pressure condition (>1.68 psig) exists. Since the source of the high drywell pressure is from a Main Steam Line, Recirc Riser DP, for both loops, will be unaffected. This will result in LPCI Loop Select Logic selecting the B Loop (Division 2 RHR) for injection because Loop A Recirc Riser DP will not be greater than Loop B Recirc Riser DP by 0.627 psig.

Since B loop is selected for injection, the A loop injection valves (E11-F015A & F017A) will get a close signal that seals in for 10 minutes, so this light will be lit on the Div 1 side. Since RPV pressure is not yet less than 461 psig (at 750 psig as stated in the stem of the question), LPCI Loop Select Logic will not have sent an open signal to the Loop B injection valves (E11-F015B & F017B) so the lights for this signal will NOT be lit.

Answer D shows the light arrangement for the conditions described above.

Distractor Explanation:
Distractors are incorrect and plausible because:

A. This distractor is plausible if the candidate recognized correctly that Loop B was selected for injection, thereby resulting in the E11-F017A & F015A receving a closed signal. The candidate could incorrectly assume that the selected loop (Loop B) injection valves (F017B & F015B) receive an open signal (as indicated by the open light lit) at the time of selection, instead of when RPV pressure drops <461 psig. This assumption makes this distractor incorrect.
B. This distractor is plausible if the candidate incorrectly determined that LPCI Loop Select would select the A Loop for injection based on the conditions given in the stem of the question. This is possible if the candidate incorrectly recalled the logic for determining the intact loop (comparison of riser DP) or possibly if the candidate misread the stem and thought the leak was from the B Recirc Loop instead of the B Main Steam Line. This assumption is incorrect as described above.
C. This distractor is plausible if the candidate incorrectly determined that LPCI Loop Select would select the A Loop for injection based on the conditions given in the stem of the question. This is possible if the candidate incorrectly recalled the logic for determining the intact loop (comparison of riser DP) or possibly if the candidate misread the stem and thought the leak was from the B Recirc Loop instead of the B Main Steam Line. This assumption is incorrect as described above. The candidate could incorrectly assume that the selected loop (Loop A) injection valves (F017A & F015A) receive an open signal (as indicated by the open light lit) at the time of selection, instead of when RPV pressure drops <461 psig. This assumption is incorrect as described above.

Reference Information:
23.205, Residual Heat Removal System, Section 9.0 Emergecy Operations
23.601, Instrument Trip Sheets, Enclosure B, Logic Sheet for LPCI Loop Select Logic.

Plant Procedures
23.205

NUREG 1123 KA Catalog Rev. 2
203000 A3. Ability to monitor automatic operations of the RHR/LPCI: INJECTION MODE (PLANT SPECIFIC) including:
203000 A3.07 4.2*/4.6* Loop selection: Plant-Specific

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 Question Use (ILO 2018)
High
MODIFIED
RO

Associated objective(s):
Cycle 15-4 Objectives

Cognitive Terminal

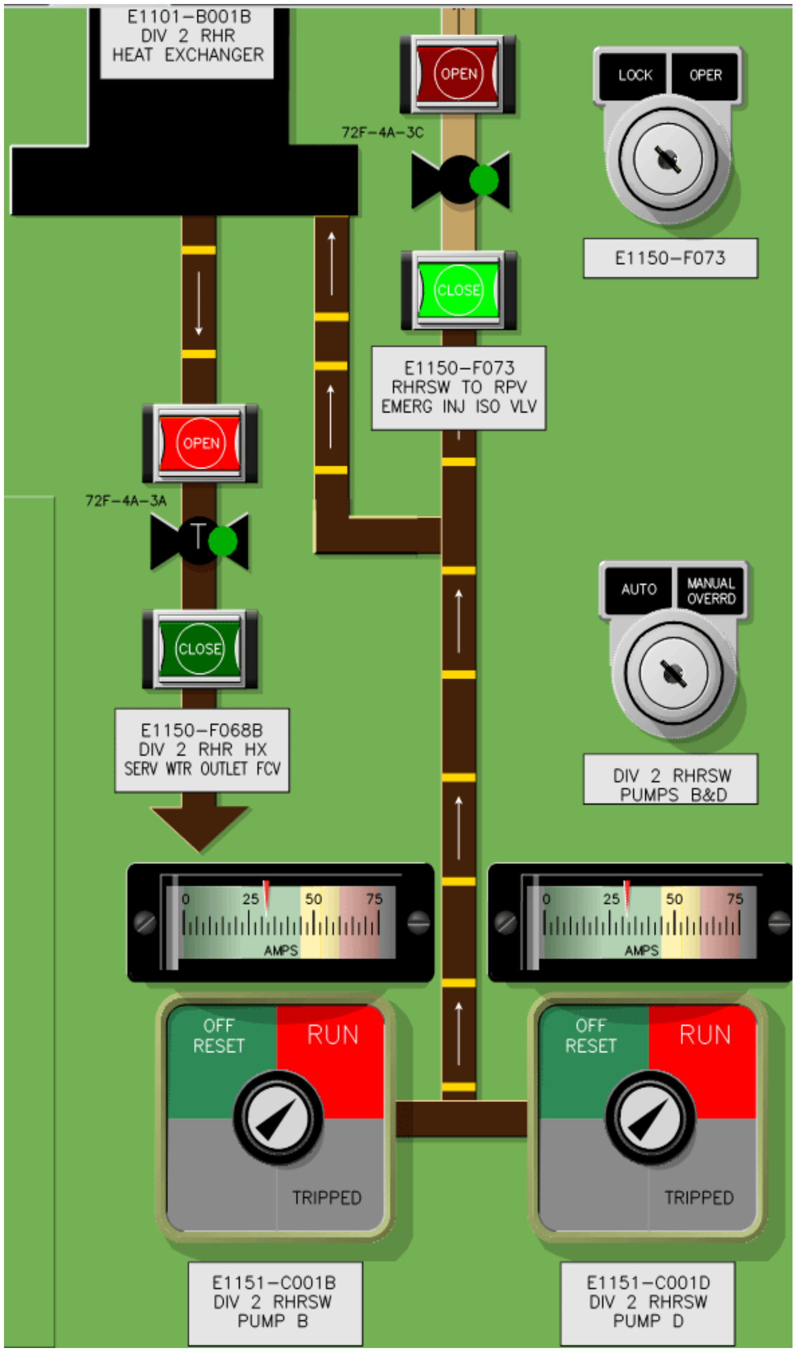
Given a copy of 23.205, discuss licensed operator actions to Force Low Pressure Coolant Injection (LPCI) Loop Select Logic to Division I(II)

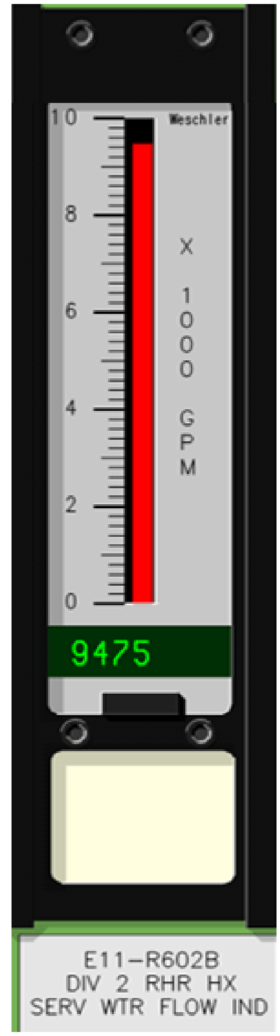
THIS PAGE LEFT BLANK

29	K/A Importance: 3.2/3.2			**Points: 1.00**
R29	Difficulty: 3.00	Level of Knowledge: High	Source: NEW	73826

The plant is shut down with the following conditions:

- RHR Pump B is operating in Shutdown Cooling Mode.

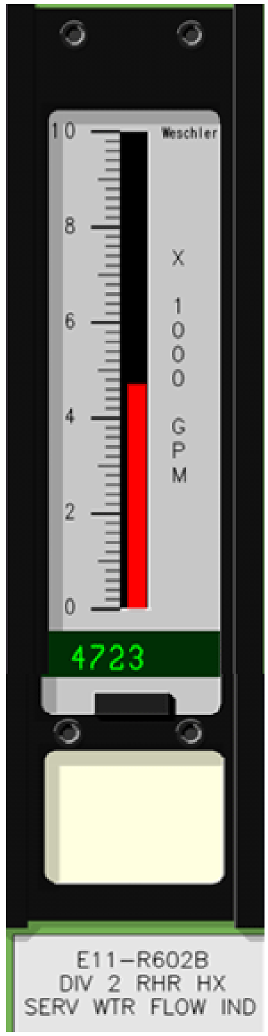




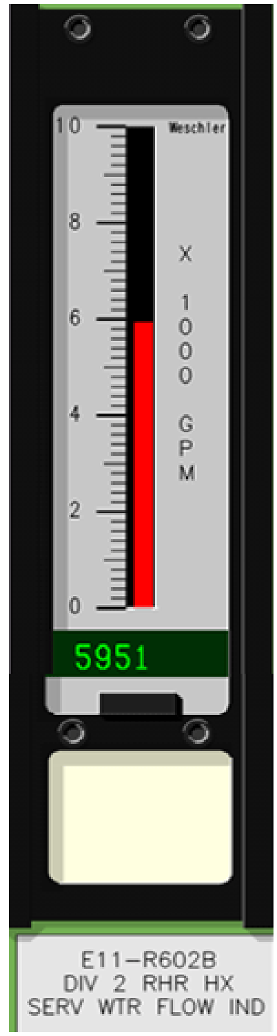
Following a trip of RHRSW Pump B the CRS directs you to establish the proper RHR Service Water flow. Which of the following flow indications meets the requirements of 23.208, RHR Complex Service Water Systems?

A.

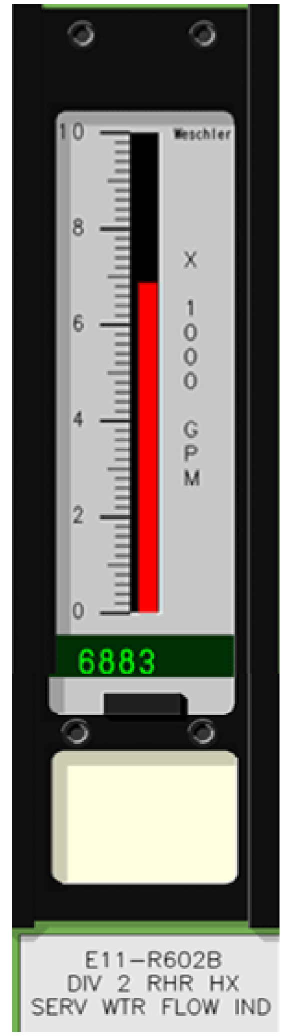
B.



C
.



D
.



A. A

B. B

C. C

D. D

Answer: C

Answer Explanation:
This question tests the candidates ability to monitor and control RHR Heat Exchanger cooling flow (RHRSW flow), within allowable limits, for conditions resulting from a tripped RHRSW pump.

23.208 has a precaution that states "When only one RHR Service Water Pump is in operation with E1150-F068A (B), Div 1 (2) RHR Hx Serv Wtr Outlet FCV open, RHR Service Water flow should be maintained 5400 gpm to 6300 gpm to prevent excessive vibration of E1150-F068A (B), and to prevent pump runout".

The conditions in the stem of the question show Division 2 RHRSW in operation, with 2 pumps running, and the E1150-F068B fully open.

Upon a trip of one RHRSW pump, the candidate should recognize the need to lower flow (by throttling the E1150-F068) to protect the pump from runout and protect the valve from excessive vibration. Answer B shows RHRSW flow lowered to within the allowable band.

Distractor Explanation:
A. This distractor is plausible because the candidate could determine that RHRSW is not permitted to operate with only one RHRSW pump. Therefore, the candidate could conclude that the RHRSW flow indicator would indicate 0 gpm after the requirements of 23.208 were met. This is incorrect because one-pump RHRSW operation is permitted as long as flow is maintained within the prescribed band specified by the precaution.
B. This distractor is plausible because the indicated flow is approximately half of the flow prior to the pump trip. If the candidate assumed that flow with one pump should be about half of what it was for two pumps, then he/she could determine this is the proper ending flow.
D. The flow shown for this distractor is the indicated flow that results from tripping one RHRSW pump and not taking any operator actions, such as throttling the E1150-F068B. This distractor is plausible if the student recognized this as being the expected flow that he/she would see and if the candidate failed to recall the flow range specified in the precaution above and therefore the need to lower flow.

Reference Information:
23.208 RHR Complex Service Water Systems SOP

NUREG 1123 KA Catalog Rev. 2
205000 RHR Shutdown Cooling Mode
205000 A4. Ability to manually operate and/or monitor in the control room:
205000 A4.11 3.2/3.2 Heat exchanger cooling 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 Question Use (ILO 2018)
High
NEW
RO

Associated objective(s):

Residual Heat Removal Service Water

Cognitive Enabler

Identify relationships between significant Residual Heat Removal Service Water System operating parameter values.

Residual Heat Removal Service Water

Cognitive Enabler

Describe general Residual Heat Removal Service Water System operation, including component operating sequence, normal operating parameters, and expected system response.

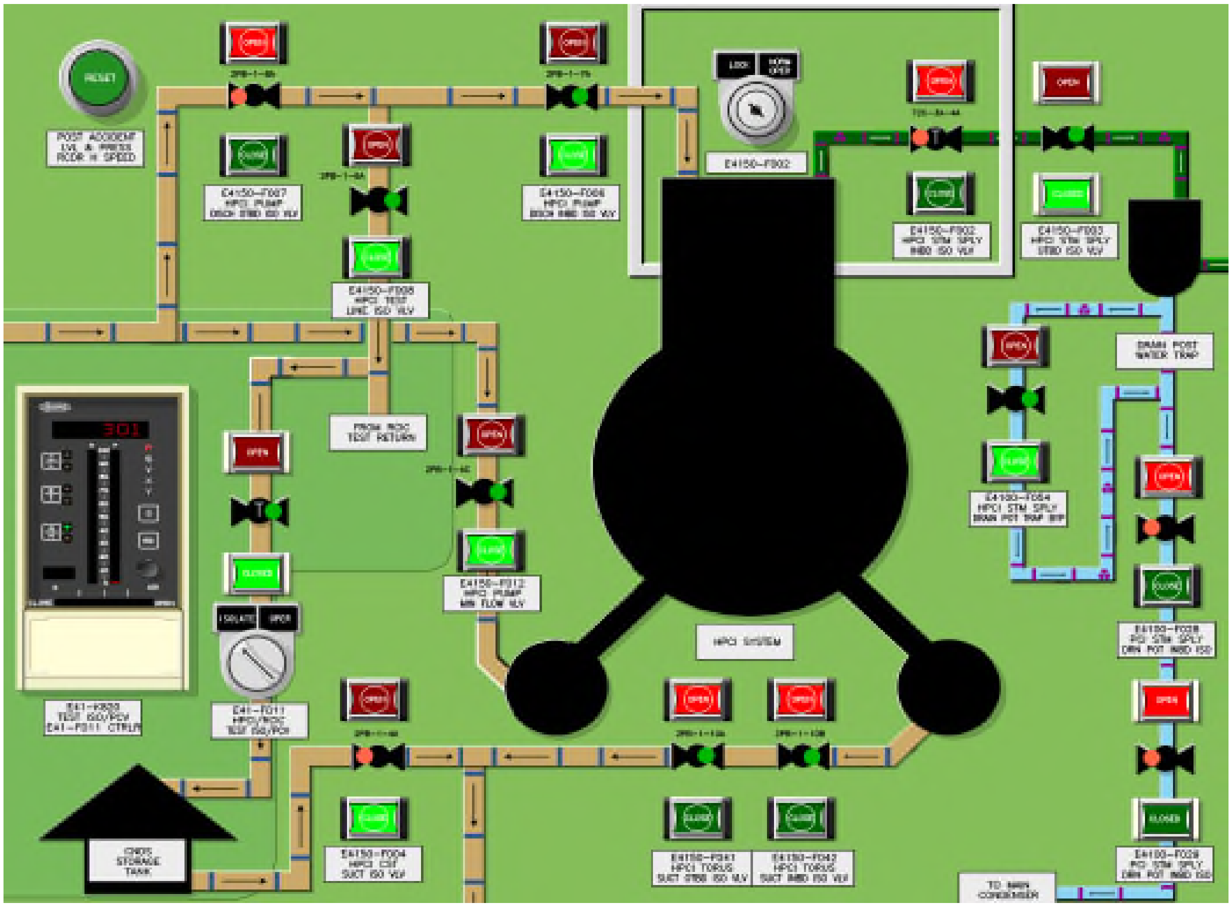
Residual Heat Removal Service Water

Cognitive Enabler

State major precautions and limitations, and major safety considerations for the Residual Heat Removal Service Water System, and describe their bases.

30	K/A Importance: 4.6/4.6			Points: 1.00
R30	Difficulty: 3.00	Level of Knowledge: High	Source: MODIFIED	73606

The HPCI system is in standby.



With the lineup shown, under which of the following conditions must HPCI be considered INOPERABLE?

A. If it remains in this configuration for more than one hour.

B. If it remains in this configuration for more than 12 consecutive hours.

C. It is currently INOPERABLE if the keep fill system is NOT in operation.

D. If it stays in this configuration for more than 12 consecutive hours, without keep fill in operation.

Answer: D

Answer Explanation:
This question is RO level of knowledge because it requires the candidate to recall a procedure step (P&L) that relates to the impact of HPCI being aligned to the non-preferred suction source, or the Torus. This is required by objective LP-OP-315-0139-C002, State major precautions and limitations, and major safety considerations for the High Pressure Coolant Injection System, and describe their bases.

23.202, Step 3.19 states "If aligned to the Torus (in standby) for more than twelve consecutive hours without HPCI Keep Fill System in operation, HPCI should be considered inoperable, due to potential drain down of system piping. Reference CARD 98-11671 and TMPE 97-0345. The twelve hour clock can be reset by valving in the CST and venting the discharge line high point vent. If aligned for more than one hour on torus suction a high point vent when suction returned to CST or keep fill restored should be performed.
The conditions (picture) in the stem of the question show the HPCI system in a standby lineup with suction aligned to the Torus. Under this condition, the HPCI system is OPERABLE, however 23.202, Step 3.19 applies. This question requires the candidate to evaluate the system lineup and then interpret and execute 23.202, Step 3.19 to determine system OPERABILITY.
Answer D is correct because, given the current HPCI system configuration, if keep fill became unavailable, the system would be INOPERABLE in 12 hours.

Distractor Explanation:
Distractors are incorrect and plausible because:
A. This distractor is plausible if the candidate did not correctly interpret and apply another step in 23.202, that relates to the configuration shown in the picture. 23.202 Step 3.21 states: When HPCI is aligned to the Torus without HPCI Keep Fill System in operation for greater than one hour:
- There is a potential for a void to form causing a pressure transient with HPCI startup.
- When HPCI Suction is realigned to CST, the discharge line high point vent should be vented.
- HPCI will not be considered inoperable in this condition (CARD 01-20890).

The candidate could incorrectly recall this step and consider HPCI to be INOPERABLE after one hour with its suction aligned to the Torus as shown.
B. This distractor is plausible if the candidate incorrectly interprets and applies 23.202, Step 3.19 and determines that HPCI is INOPERABLE if it remains in the configuration shown, for greater than 12 hours, regardless of the status of the keepfill system.
C. This distractor is plausible if the candidate incorrectly interprets and applies the steps regarding the status of the HPCI system, with its suction aligned to the Torus without keep fill in operation, and determines that HPCI is INOPERABLE any time the HPCI suction is aligned to the Torus (as shown) without keep fill in operation. This is incorrect because there are conditions that allow HPCI to remain OPERABLE, with its suction aligned to the Torus and without keep fill in operation, as long as the appropriate actions are taken as specified in steps 3.19 and/or 3.21.

Reference Information:
23.202 HPCI System, Step 3.19

NUREG 1123 KA Catalog Rev. 2
206000 HPCI System.

G2.1.20 Ability to interpret and execute procedure steps

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

NRC Question Use (ILO 2018)
High
MODIFIED
NRC Early Review
RO

Associated objective(s):
High Pressure Coolant Injection

Cognitive Enabler

State major precautions and limitations, and major safety considerations for the HPCI System, and describe their bases.

High Pressure Coolant Injection

Cognitive Enabler

State major precautions and limitations, and major safety considerations for the HPCI System, and describe their bases.

THIS PAGE LEFT BLANK

31	K/A Importance: 3.7/3.8			**Points: 1.00**
R31	Difficulty: 3.00	Level of Knowledge: High	Source: NEW	73607

The plant is operating at 100% power with Core Spray Pumps A and C running for ~~surveillance testing~~ Division 1 Core Spray System Pump and Valve Operability test.

Which of the responses below best completes the following statement to describe how the Core Spray pumps would respond to a simultaneous Loss of Offsite Power (LOP) and Loss of Coolant Accident (LOCA) under these conditions?

Core Spray Pumps A and C would ____(1)____ their respective EDG output breakers closed.
Core Spray Pumps B and D would ____(2)____ their respective EDG output breakers closed.

A. (1) start five seconds after
(2) start five seconds after

B. (1) start five seconds after
(2) immediately restart when

C. (1) immediately restart when
(2) start five seconds after

D. (1) immediately restart when
(2) immediately restart when

Answer: C

Answer Explanation:
Note: This explanation refers to the operation of Core Spray Pump A (see I-2211-01). All Core Spray Pumps operate in the same manner.
If power is available to bus 64B, the K3A relay will energize, closing contacts K3A in the CSS Pump A control string. K16A Time Delay Pickup (TDPU) relay will energize after 5 seconds, closing contacts K16A. This will energize relay K12A, which will start the CSS Pump 5 seconds AFTER the LOCA signal is received. Note: This 5 second delay in CSS pump starting comes from Core Spray logic and NOT the EDG Load Sequencer.

On a LOP/LOCA with Core Spray Pump A in standby, as soon as the EDG Automatic Load Sequencer (EDGALSS) senses the EDG output breaker closing, it energizes relay XK-34, which permits the Core Spray Pump breaker to close. The Core Spray pump breaker normally still has to wait on a signal from Core Spray Logic, via relay K12A (from I-2215-02), to close when in Auto.

However, when Core Spray Pump A is in RUN, as stated in the stem of the question, this 5-second time delay is effectively bypassed. In this case, when the LOP/LOCA occurs with Core Spray Pump A in RUN, as soon as the EDG Automatic Load Sequencer (EDGALSS) senses the EDG output breaker closing, it energizes relay XK-34, which permits the Core Spray Pump breaker to close. With the CMC in RUN, the Core Spray pump breaker will immediately close without needing to wait on a signal from Core Spray Logic (effectively bypassing the 5-second time delay).

Distractor Explanation:
Distractors are incorrect and plausible because:

A. This distractor is plausible if the candidate did not recall the difference in the response of the Core Spray system when in Automatic compared to when the pump CMC is in RUN. This is incorrect, however, as described above.
B. This distractor is plausible if the candidate recalled that the Core Spray pumps respond the same as the RHR pumps and they normally start immediately after the EDG restores power to the bus following a LOP/LOCA when in Auto. The candidate could also could recall that Core Spray logic changes when the pump is in RUN and incorrectly assume that the way it changes is to cause a 5-second time delay on start when in RUN. This is incorrect as described above.
D. This distractor is plausible if the candidate recalled that the Core Spray pumps always started immediately after the EDG restored bus power following a LOP/LOCA. This is plausible because this is how the RHR pumps respond. Since the RHR pumps restart immediately after the EDG output breakers close, following a LOP/LOCA, the candidate could incorrectly assume that the Core Spray pumps do as well since they are both low pressure injection systems. This is incorrect, however, as described above.

Reference Information:
23.203, Core Spray System
I-2215-02 Div 1 and Div 2 CSS Logic Schematic
I-2211-01 Core Spray Pump A Schematic
ST-OP-315-0040, Core Spray System Student Text

NUREG 1123 KA Catalog Rev. 2
209001 K1. Knowledge of the physical connections and/or cause effect relationships between LOW PRESSURE CORE SPRAY SYSTEM and the following:
209001 K1.10 3.7/3.8 Emergency generator

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 Question Use (ILO 2018)
High
NEW
RO

Associated objective(s):

Core Spray System

Cognitive Enabler

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

Core Spray System

Cognitive Enabler

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

THIS PAGE LEFT BLANK

32	K/A Importance: 3.1/3.2			**Points: 1.00**
R32V2	Difficulty: 4.00	Level of Knowledge: High	Source: BANK	75086

The plant is operating at 100% power. The control power fuse for Motor Control Center (MCC) 72E-5B Pos 2B, 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: C

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 72E-5B, Pos 2B has been lost, the candidate should recognize that power to the B SLC Pump, and B explosive (squib) control circuits, has been lost. This will cause a loss of continuity through the B squibs and loss of continuity indication (amber light). Upon system initiation, the B SLC pump will be unavailable. Upon starting the A SLC pump, even though a contact will close in the control circuit for the B squib valve (C4104F004B), without control power available, this squib will not fire and the valve will not open. Only the A squib will fire for the A squib valve (C4104F004A) and only the A valve will open.

Distractor Explanation:
Distractors are incorrect and plausible because:

A. The candidate could recognize that 72E-5B is the power supply to the B squib ciruit and, therefore, recognize that only the A squib continuity light will be lit. The candidate 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 candidate could apply this misconception and assume that both valves will still open upon system initiation, which is incorrect.
B. The candidate could incorrectly determine that 72E-5B is the power supply to the A squib ciruit and, therefore, determine that only the B 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 candidate could apply this misconception and assume that both valves will still open upon system initiation, which is incorrect.
D. This distractor is plausible because the operator could incorrectly determine that the MCC given in the stem of the question (72E-5B, Pos 2B) powers the A SLC Pump and A 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

NUREG 1123 KA Catalog Rev. 2
211000 SLC System
211000 K2. Knowledge of electrical power supplies to the following:
211000 K2.02 3.1*/3.2* Explosive valves

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 Question Use (ILO 2018)
BANK
High
RO

Associated objective(s):
Standby Liquid Control

Cognitive Terminal

Given various controls and indications for Standby Liquid Control, analyze that indication to determine proper plant response in accordance with approved plant procedures.

THIS PAGE LEFT BLANK

33	K/A Importance: 3.7/3.9			**Points: 1.00**
R33	Difficulty: 3.00	Level of Knowledge: High	Source: MODIFIED	73567

What is the effect that a loss of RPS Bus B will have on the RHR system with Division 1 RHR running in Shutdown Cooling on RHR Pump A?

A. RHR will continue to operate, but some Division 1 RHR instrumentation will be INOPERABLE.

B. E1150-F008, RHR SDC Otbd Suction Iso Vlv, will close resulting in a loss of Shutdown Cooling.

C. E1150-F009, RHR SDC Inbd Suction Iso Vlv, will close resulting in a loss of Shutdown Cooling.

D. E1150-F008, RHR SDC Otbd Suction Iso Vlv, AND E1150-F009, RHR SDC Inbd Suction Iso Vlv, will close resulting in a loss of Shutdown Cooling.

Answer: B

Answer Explanation:
As stated in 23.316, Enclosure B, a loss of RPS Bus B will cause the E1150-F008 to close. This will cause any RHR pumps running in SDC to trip on loss of suction interlock and an ultimate loss of SDC.

Distractor Explanation:
Distractors are incorrect and plausible because:

A. This answer is plausible if the candidate incorrectly assumed that, since RHR was running in SDC on Division 1, a loss of the RPS B power supply (a Division 2 power supply) would have no impact. Additional plausibility is lent to this distractor since the second part of the distractor is correct because the E11-F080 A & B (Div 1 & Div 2 RHR Heat Exch Outlet Sample Otbd Iso Vlvs) close, which isolates Tech Spec instruments if SDC sample pts 42a or 42b are in service.
C. This distractor is plausible if the candidate fails to recognize the relationship between RPS B and the E1150-F008 and instead incorrectly recalls that RPS B loss will cause the E1150-F009 to close. This plasibility is even greater considering the somewhat confusing numbering relationship with the RHR valves in that the inboard valve is the F009 and the outboard valve is the F008. Because of this, licensed operator candidates often confuse the numbering of these valves. If the candidate got distracted by the confusing numbering scheme, as often happens, he/she could incorrectly conclude that RPS B impacts the E1150-F009.
D. This distractor is plausible if the candidate incorrectly recalled that a loss of either RPS Bus caused both RHR SDC containment isolation valves to go closed, which is the case for some PCIS group isolation valves, such as the T4901-F602 and T4901-F468, Division 2 DW Pneumatics Inbd and Otbd Iso Valves, which both go closed on a loss of RPS B. This is incorrect, however, since only the E1150-F008 will close on a loss of RPS B.

Reference Information:
23.316, RPS 120Vac and RPS MG Sets, Enclosure B (RPS Bus B - Affected Equipment List)
Student Text ST-OP-315-0027, RPS System

NUREG 1123 KA Catalog Rev. 2

212000	RPS
212000 K3.	Knowledge of the effect that a loss or malfunction of the REACTOR PROTECTION SYSTEM will have on the following:

212000 K3.02 3.7/3.9 Primary containment isolation system/nuclear steam supply shut-off: Plant-Specific

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 Question Use (ILO 2018)
High
MODIFIED
RO

Associated objective(s):
Reactor Protection System (C7100)
Cognitive Enabler
Discuss the Reactor Protection System interrelationships with other systems.

Reactor Protection System (C7100)
Cognitive Enabler
Discuss the Reactor Protection System interrelationships with other systems.

34	K/A Importance: 3.7/3.7			**Points: 1.00**
R34	Difficulty: 2.00	Level of Knowledge: High	Source: BANK	73568

The RETRACT PERMIT light for IRM G is NOT lit. Which of the following describes the effect on IRM G?

A. It will NOT retract.

B. It can be retracted. Doing so will cause a Rod Block.

C. It can ONLY be retracted if it is on Range 1. Doing so will cause a Rod Block.

D. It can ONLY be retracted if it is on Range 1. Doing so will NOT cause a Rod Block.

Answer: B

Answer Explanation:
The IRMs are not normally withdrawn until in MODE 1. Upon entering MODE 1, the Mode Switch is taken to RUN at which time all IRM RETRACT PERMIT lights will come on. The RETRACT PERMIT light is simply information to the operator that conditions are met to withdraw a detector. The RETRACT PERMIT circuitry does not permit or block actual withdrawal of a detector. Without a RETRACT PERMIT, if an unbypassed IRM detector is not fully inserted, a Rod Block will ensue. Therefore, under the conditions provided in the stem of the question, the operator could retract the IRM detector, however, doing so will result in a Rod Block.

Distractor Explanation:
Distractors are incorrect and plausible because:
A. This distractor is plausible because the candidate could take the name of the light (RETRACT PERMIT) literally and assume that, without the light lit, withdrawal of the detector would not be possible. However, this is not true because the RETRACT PERMIT circuitry does not permit or block actual withdrawal of a detector.
C. This distractor is plausible because of an interrelationship that exists between the IRMs and SRMs. If the associated IRMs are all on Range 1, then an SRM can go below 100 cps without causing a Rod Block. If the candidate misapplied the Range 1 interrelationship to the conditions given in the stem of this question, he/she could conclude that the IRM can only be retracted if on Range 1. This is incorrect because the IRMs will retract on all ranges. The candidate may recognize that retracting the IRM, however, without the RETRACT PERMIT light will cause a Rod Block, which is true.
D. This distractor is plausible because of an interrelationship that exists between the IRMs and SRMs. If the associated IRMs are all on Range 1, then an SRM can go below 100 cps without causing a Rod Block. If the candidate misapplied the Range 1 interrelationship to the conditions given in the stem of this question, he/she could conclude that the IRM can only be retracted if on Range 1. This is incorrect because the IRMs will retract on all ranges. The candidate may also incorrectly conclude that, because the IRM is on Range 1, a Rod Block will not occur (similar to how the SRM rod block is prevented with the associated IRMs are on Range 1).

Reference Information:
I-2145-56 SRM IRM Aux Relays
I-2145-59 NMS RMCS-RPS Interlocks
23.602 SRM System
23.603 IRM System
Student Text ST-OP-315-0022, SRM System
Student Text ST-OP-315-0023, IRM System

NUREG 1123 KA Catalog Rev. 2
215003 K4. Knowledge of INTERMEDIATE RANGE MONITOR (IRM) SYSTEM design feature(s) and/or interlocks which provide for the following:
215003 K4.01 Rod withdrawal blocks

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 Question Use (ILO 2018)
BANK
High
RO

Associated objective(s):

Intermediate Range Monitoring (C5111)

Cognitive Enabler

List the interlocks associated with Intermediate Range Monitoring System components.

Intermediate Range Monitoring (C5111)

Cognitive Enabler

List the interlocks associated with Intermediate Range Monitoring System components.

THIS PAGE LEFT BLANK

35	K/A Importance: 2.6/2.6			Points: 1.00
R35	Difficulty: 3.00	Level of Knowledge: High	Source: NEW	73646

How is the Signal-to-Noise ratio of an SRM detector determined and what is the operational implication of this number?

A. $$Signal - to - Noise\ Ratio = \frac{SRM\ Counts\ Inserted - SRM\ Counts\ Withdrawn}{SRM\ Counts\ Withdrawn}$$

If it is ≥ 20:1, then the minimum allowed count rate is 0.7 cps.

B. $$Signal - to - Noise\ Ratio = \frac{SRM\ Counts\ Withdrawn - SRM\ Counts\ Inserted}{SRM\ Counts\ Withdrawn}$$

If it is ≥ 20:1, then the minimum allowed count rate is 0.7 cps.

C. $$Signal - to - Noise\ Ratio = \frac{SRM\ Counts\ Inserted - SRM\ Counts\ Withdrawn}{SRM\ Counts\ Withdrawn}$$

If it is ≥ 20:1, then the minimum allowed count rate is 3.0 cps.

D. $$Signal - to - Noise\ Ratio = \frac{SRM\ Counts\ Withdrawn - SRM\ Counts\ Inserted}{SRM\ Counts\ Withdrawn}$$

If it is ≥ 20:1, then the minimum allowed count rate is 3.0 cps.

Answer: A

Answer Explanation:
With an SRM detector fully inserted, it is in a high neutron flux region of the core. This is the core location where most of the signal produced will be the result of neutron interactions. At this location, the detector output will be a combination of both the neutron signal plus any background noise.
With an SRM detector fully withdrawn, it is in a low flux region (approximately 2.5 feet below the bottom of active fuel) where very little, if any, of the detector output will be the result of neutron interactions. At this location, the detector output will be the result of background noise only.
These two signals can be used to check for conformance with the signal-to-noise ratio requirement.

The candidate must determine that, with the detector fully inserted, the resultant signal output equals signal + noise. When withdrawn the output is just noise. If the withdrawn value is subtracted from the insert value, the result is known as the 'signal'. Then, if the resulting signal is divided by the withdrawn value (noise), the end result is known as the signal to noise ratio.
From Technical Specifications Section 3.3.1.2, a detector is considered to be OPERABLE as long as it reads at least 3.0 cps OR if it reads at least 0.7 cps as long as the detector's signal-to-noise ratio (determined above) is at least 20:1.

This information, in addition to being a TS requirement, is also a requirement of the Reactor Startup GOP (General Operating Procedure) that the RO uses when performing a Reactor Startup, specifically GOP Step 22.000.02, Step 5.2.7.

Distractor Explanation:
Distractors are incorrect and plausible because:
B. This distractor is plausible if the candidate did not fully understand SRM detector operation and assumed that the detector 'signal' is determined with the SRM withdrawn from the core. This is incorrect since the SRMs are miniature fission chambers that develop their 'signal' from interactions with thermal neutrons.
C. This distractor is plausible if the candidate determined the correct method for calculating the signal-to-noise ratio but mis-applied the Tech Spec Surveillance requirement and assumed that a ratio greater than 20:1 required a higher count rate of at least 3.0 cps, which is incorrect.
D. This distractor is plausible if the candidate did not fully understand SRM detector operation and assumed that the detector 'signal' is determined with the SRM withdrawn from the core. This is incorrect since the SRMs are miniature fission chambers that develop their 'signal' from interactions with thermal neutrons. The candidate could also incorrectly apply the Tech Spec Surveillance requirement and assume that a ratio greater than 20:1 required a higher count rate of at least 3.0 cps, which is incorrect.

Reference Information:
ST-OP-315-0022, SRM System Student Text.
24.000.01, Situational Surveillances/LCO Action Tracking, Attachment 37.
Tech Spec Bases for SR 3.3.1.2.5 and 3.3.1.2.6.

NUREG 1123 KA Catalog Rev. 2
215004 K5. Knowledge of the operational implications of the following concepts as they apply to SOURCE RANGE MONITOR (SRM) SYSTEM:
215004 K5.01 2.6/2.6 Detector operation

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 Question Use (ILO 2018)
High
NEW
RO

Associated objective(s):
Source Range Monitoring (C5110)

Cognitive Enabler

Discuss design considerations, capabilities, and limitations related to Source Range Monitoring System component operation.

Reactor Startup Certification

Performance Terminal

Perform a reactor startup in accordance with 22.000.02, Plant Startup to 25% Power, from reactor shutdown to establishing a <90°F/Hr heatup rate

Source Range Monitoring (C5110)

Cognitive Enabler

Describe general Source Range Monitoring System operation, including component operating sequence, normal operating parameters, and expected system response.

Source Range Monitoring (C5110)

Cognitive Enabler

Describe general Source Range Monitoring System operation, including component operating sequence, normal operating parameters, and expected system response.

THIS PAGE LEFT BLANK

36	K/A Importance: 3.1/3.2			**Points: 1.00**
R36	Difficulty: 3.00	Level of Knowledge: High	Source: NEW	74966

The plant was operating at 100% power when APRM #1 failed upscale.

BEFORE any operator actions are taken, the Recirculation Flow Unit for APRM #3 subsequently fails downscale.

Which of the following responses describes the impact of these malfunctions on the APRM System 2/4 Logic Modules?

A. ONLY 2/4 Logic Module #1 will have a trip from APRM #1.

B. ALL four 2/4 Logic Modules will have a trip from APRM #1 ONLY.

C. ALL four 2/4 Logic Modules will have trips from BOTH APRM #1 and APRM #3.

D. 2/4 Logic Module #1 will have a trip from APRM #1 ONLY and 2/4 Logic Module #3 will have a trip from APRM #3 ONLY.

Answer: C

Answer Explanation:
This question requires the candidate to determine the effect of multiple APRM failures on the overall APRM System by determining the trips that will be present on the APRM 2/4 Logic Modules. Note: The 2/4 Logic Modules are the interface between the NUMAC APRM System and the Reactor Protection System (RPS) and are also known as Voters.

The candidate should determine that the first APRM upscale failure will result in each of the 4 APRM Voters receiving one "vote" from APRM #1. The candidate should then determine that the subsequent downscale failure of the flow input to APRM #3 will result in APRM #3 calculating a Simulated Thermal Power (STP) trip (flow-biased trip) setpoint that is below actual APRM Power, which will result in a trip output from APRM #3 as well. This second trip will also get "voted" by all 4 APRM voters. Therefore, the candidate should conclude that all 4 APRM 2/4 Logic Modules will see trips from both APRMs #1 and #3.

Although not asked, the overall result of these two failures, and two "votes", is that all 4 Voters will go to a tripped condition, which will cause both RPS A and B to trip, resulting in a reactor scram.

Distractor Explanation:
Distractors are incorrect and plausible because:

A. The candidate could incorrectly recall that the 4 APRMs only output to their associated Voters, which could lead the candidate to determine that the upscale trip from APRM #1 will only be seen by Voter #1, which is incorrect because all 4 APRMs input into all 4 Voters. This misconception is often caused by licensed operator candidates misunderstanding the relationship between the 2/4 Logic Modules and their 'home' APRM. Each Voter has a 'home' APRM (#1 for #1, #2 for #2, and so on) that it receives bypass and self-test information from. However, this does not change the fact that each Voter receives trip input information from ALL 4 APRMs and NOT just their 'home' APRM. The candidate could also fail to determine that the downscale failure of the flow input to APRM #3 will cause it to trip on STP Upscale.
B. The candidate could correctly recall that all 4 APRMs input into all 4 Voters and determine that the Upscale Trip from APRM #1 will be seen by all 4 Voters. However, the candidate could fail to recognize the significance of the downscale failure of the flow input to APRM #3 and not conclude that APRM #3 will also be in a tripped condition due to a STP Upscale Trip, which is incorrect because the downscale failure of the flow input will result in a low STP Setpoint, compared to actual power, which will put APRM #3 in a tripped condition.
D. The candidate could incorrectly recall that the 4 APRMs only output to their associated Voters, which could lead the candidate to determine that the upscale trip from APRM #1 will only be seen by Voter #1, and the Upscale Trip from APRM #3 will only be seen by Voter #3, both of which are incorrect because all 4 APRMs input into all 4 Voters. Again, this misconception is often caused by candidates misunderstanding the relationship between the Voters and their 'home' APRMs as described in A above.

Reference Information:
23.605 APRM System SOP
Student Text ST-OP-315-0024, PRNM System

NUREG 1123 KA Catalog Rev. 2
215005 APRM/LPRM
215005 K6. Knowledge of the effect that a loss or malfunction of the following will have on the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM:
215005 K6.04 3.1/3.2 Trip units

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 Question Use (ILO 2018)
High
NEW
RO

Associated objective(s):
Power Range Neutron Monitoring (C5112, C5113 & C5114)

Cognitive Terminal

Given various controls and indications for PRNM operations, analyze that indication to determine proper plant response in accordance with approved plant procedures.

THIS PAGE LEFT BLANK

37	K/A Importance: 3.7/3.7			Points: 1.00
R37	Difficulty: 3.00	Level of Knowledge: High	Source: BANK	73706

The Reactor Core Isolation Cooling (RCIC) system has been started in accordance with 24.206.04, RCIC SYSTEM AUTOMATIC ACTUATION AND FLOW TEST.
The following conditions exist:

- RCIC Flow Controller, E51-K615, is in Auto, set at 650 gpm
- RCIC flow - 650 gpm
- RPV pressure - 185 psig

What happens to RCIC final speed and discharge pressure if E41-F011 HPCI Test Line Iso/PCV is OPENED an additional 5%?

A. RCIC speed lower
RCIC discharge pressure lower

B. RCIC speed lower
RCIC discharge pressure higher

C. RCIC speed higher
RCIC discharge pressure lower

D. RCIC speed higher
RCIC discharge pressure higher

Answer: A

Answer Explanation:
The RCIC system is tested during plant operation by recirculating condensate tank (CST) water through the HPCI/RCIC test return line. A drag type valve (E41-F011) is employed in the test return line to simulate discharge pressure conditions. E41-F011 is designed to simulate conditions where the RCIC system is supplying cooling water to the reactor vessel under high and low reactor vessel pressure. Required test differential pressure is obtained by use of a differential pressure controller which maintains the test pressure at the desired setpoint. E41-F011 provides a differential pressure to demonstrate RCIC will provide the required flow rate at simulated reactor pressures of 1135 psia and 135 psia. The simulated pressure represents reactor pressure plus pressure drop from line losses between RCIC and the reactor vessel. This valve is also used for developing the required HPCI differential test pressure.
When the valve is opened, as stated in the stem of the question, RCIC pump back pressure will lower, which will cause flow through the test line to increase and RCIC system flow to increase. With the RCIC flow controller (E51-K615) in automatic, as stated in the stem of the question, the increase in system flow will be sensed by the controller which will close down on the RCIC trip throttle valve to restore system flow back to the setpoint of 650 gpm. This will result in less steam flow to the RCIC turbine, thereby lowering turbine speed.
The end result will be system flow restored to the setpoint of 650 gpm, at a lower discharge pressure (because of less backpressure and reduced headloss by opening the drag valve) and lower turbine speed.

Distractor Explanation:
Distractors are incorrect and plausible because:

B. The candidate could determine that the test line pressure control valve directly controls RCIC pump discharge pressure (rather than acting as a backpressure control, or drag valve) and therefore assume that opening it would result in a higher discharge pressure. If this was assumed, the candidate could then determine that the higher discharge pressure would result in more system flow, so the turbine would have to slow down to bring flow back to the controlled setpoint.
C. The candidate could recognize that opening the test line PCV could lower system back pressure and, therefore, pump discharge pressure, which is correct. The candidate could assume that the turbine speed would have to increase to develop this higher pressure, which is not true. The candidate could fail to recognize the impact of the reduced backpressure on system flow and the response of the flow controller to lower turbine speed in an effort to control system flow at the setpoint.
D. The candidate could determine that the test line pressure control valve directly controls RCIC pump discharge pressure (rather than acting as a backpressure control, or drag valve) and therefore assume that opening it would result in a higher discharge pressure. This could lead the candidate to determine that the turbine speed would have to increase to develop this higher pressure, which might be true if it was a discharge pressure control valve, but it is not true as described above.

Reference Information:
24.206.04, RCIC System Automatic Actuation and Flow Test
Student Text ST-OP-315-0043, RCIC System
GE BWR Fundamentals, Components, Chapter 2, Pumps
GE BWR Fundamentals, Thermodynamics Chapter 6, Fluid Statics and Dynamics.
E51-00 HPCI System Design Basis Document (DBD). Note: The E41-F011 is a test valve shared between the RCIC and HPCI systems. The explanation in the HPCI DBD is better than in the RCIC DBD. Also, the RCIC DBD is coded as "confidential" due to proprietary information contained therin.

NUREG 1123 KA Catalog Rev. 2
217000 RCIC System
217000 A1. Ability to predict and/or monitor changes in parameters associated with operating the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) controls including:
217000 A1.05 3.7/3.7 RCIC turbine speed

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 Question Use (ILO 2018)
BANK
High
RO

Associated objective(s):
Reactor Core Isolation Cooling
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.

THIS PAGE LEFT BLANK

38	K/A Importance: 4.2/4.3			Points: 1.00
R38	Difficulty: 4.00	Level of Knowledge: High	Source: BANK	73626

Following a Loss of Offsite Power, the following conditions occur at the listed times:

- 12:00 Reactor Scram occurred, all Control Rods are inserted.
- 12:01 ONLY EDG 14 has started and loaded.
- 12:05 Drywell Pressure is 0.9 psig and stable.
- 12:10 RPV Water Level is 64 inches, lowering 4 inches per minute.

Given these conditions: (1) Which ONE of the following describes the response of the Automatic Depressurization (ADS) System and (2) the operator actions that should be taken to prevent ADS initiation when directed to do so by the CRS when in the EOPs?

A. (1) ADS will OPEN Safety Relief Valves at 12:20.
(2) Place the ADS Inhibit Sw Logic A (B) key switches in INHIBIT.

B. (1) ADS will OPEN Safety Relief Valves at 12:27.
(2) Place the ADS Inhibit Sw Logic A (B) key switches in INHIBIT.

C. (1) ADS will OPEN Safety Relief Valves at 12:20.
(2) Simultaneously depress the ADS Division I (II) Timer Logic RESET pushbuttons.

D. (1) ADS will OPEN Safety Relief Valves at 12:27.
(2) Simultaneously depress the ADS Division I (II) Timer Logic RESET pushbuttons.

Answer: B

Answer Explanation:
Correct Answer: B With NO High Drywell Pressure signal present, L1 (31.8 inches) will cause ADS Timer to initiate in 7 minutes. The ADS Timer lasts 105 seconds or 2 minutes. A total of 9 minutes later, SRVs will OPEN. L1 will be reached in 8 minutes. 8 minutes + 9 minutes = 17 minutes 12:10 + 17 minutes = 12:27.

23.201, Section 5.4 gives specific direction for preventing ADS initiation by placing the ADS Inhibit Sw Logic A and B key switches in INHIBIT. These are the actions that the operator would take if directed by the CRS to inhibit ADS per the EOPs.

Distractor Explanation:
Distractors are incorrect and plausible because:

A. This distractor is plausible because the time would be true if Drywell Presssure was above 1.68 psig. When above 1.68 psig, the 7-minute timer is bypassed so the 105 second countdown occurs when L1 is reached in 8 minutes. 8 minutes + 2 minutes = 10 minutes. 12:10 + 10 minutes = 12:20. The second part of this distractor is correct.
C. This distractor is plausible because the time would be true if Drywell Presssure was above 1.68 psig. When above 1.68 psig, the 7-minute timer is bypassed so the 105 second countdown occurs when L1 is reached in 8 minutes. 8 minutes + 2 minutes = 10 minutes. 12:10 + 10 minutes = 12:20. The second part is plausible because these actions would cause the ADS 105 second timer to reset and start couting down again. However, it is incorrect because these actions would not prevent ADS initiation nor are they the actions that an operator is expected to take when directed by the CRS, per the EOPs, to prevent ADS initiation. These push buttons are only used to RESET ADS logic, following initiation, in the ADS System Shutdown Section (Section 6.3) of 23.201.
D. The first part of this distractor (timing of ADS initiation) is correct. The second part is plausible because these actions would cause the ADS 105 second timer to reset and start couting down again. However, it is incorrect because these actions would not prevent ADS initiation nor are they the actions that an operator is expected to take when directed by the CRS, per the EOPs, to prevent ADS initiation. These push buttons are only used to RESET ADS logic, following initiation, in the ADS System Shutdown Section (Section 6.3) of 23.201.

Reference Information:
23.201, SRV and ADS System, Section 5.4 Automatic Depressurization System Operation.

NUREG 1123 KA Catalog Rev. 2
218000 ADS

218000 A2.	Ability to (a) predict the impacts of the following on the AUTOMATIC DEPRESSURIZATION SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:
218000 A2.06	ADS initiation signals present

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 Question Use (ILO 2018)
BANK
High
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.

Automatic Depressurization System (B2104)

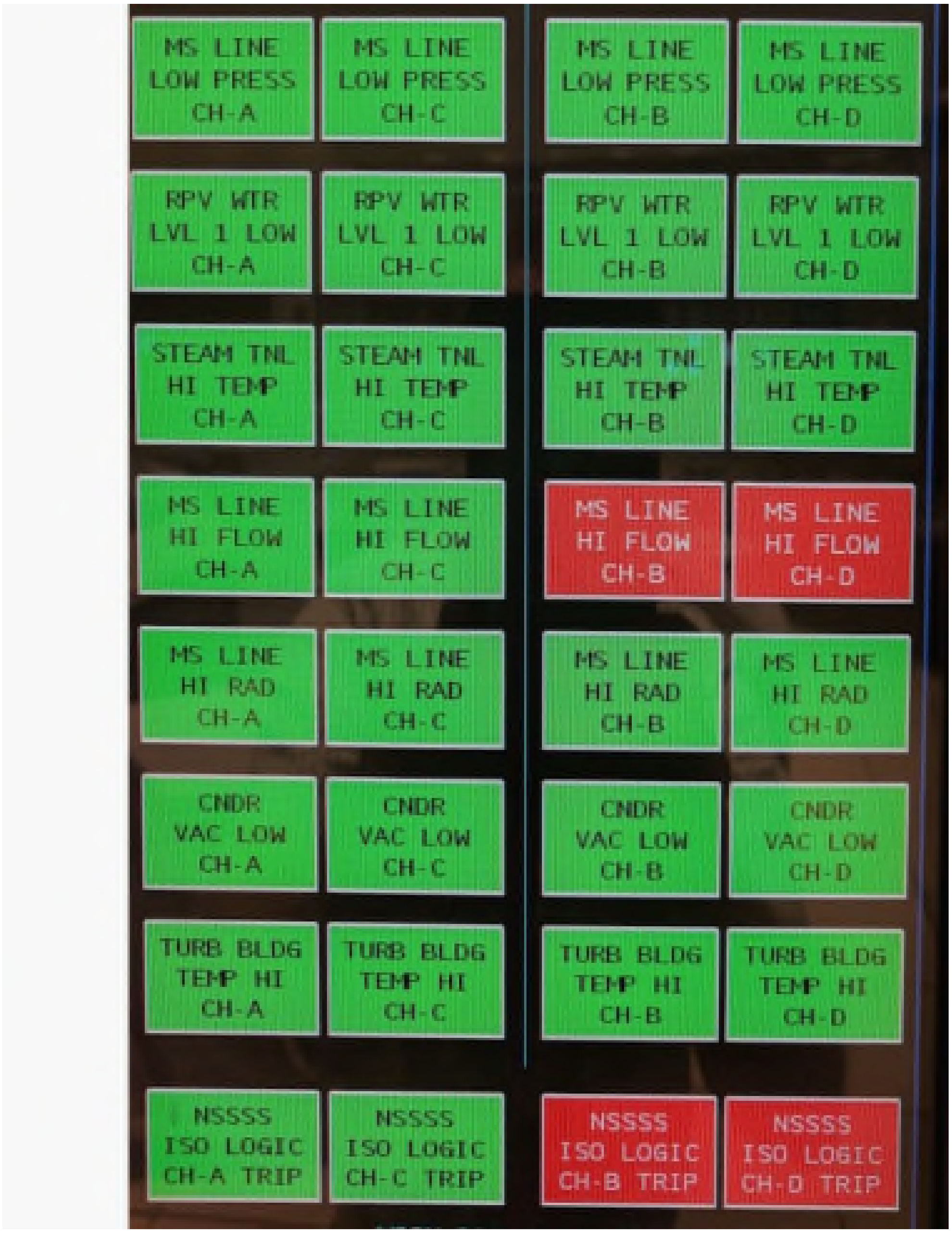
Cognitive Enabler

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

THIS PAGE LEFT BLANK

39	K/A Importance: 3.4/3.4			**Points: 1.00**
R39	Difficulty: 3.00	Level of Knowledge: High	Source: BANK	73726

The plant is at 100% power when a change in plant conditions results in the following indications:



What will the Main Steam Isolation Valve (MSIV) indicating lights show 10 seconds later?

A. All MSIV RED OPEN indicating lights will be lit.

B. All MSIV GREEN CLOSED indicating lights will be lit.

C. The B2103-F022A-D, Inboard MSIV, GREEN CLOSED indicating lights will be lit.
The B2103-F028A-D, Outboard MSIV, RED OPEN indicating lights will be lit.

D. The B2103-F022A-D, Inboard MSIV, RED OPEN indicating lights will be lit.
The B2103-F028A-D, Outboard MSIV, GREEN CLOSED indicating lights will be lit.

Answer: A

Answer Explanation:
This question requires the candidate to interpret the information provided by some indicating lights associated with the Primary Containment Isolation System (PCIS) and then determine the impact of that prediction on MSIV indicating light status.

Per ARP 2D36, with channels B or D tripped, no isolation actions will occur. IF a B/D channel were to trip AND an A/C channel were to trip, then a full MSIV isolation would occur. This is known as 1 out of 2 twice logic or [Full Isolation = (A OR C) AND (B OR D)].
Therefore, A is the correct answer since only half of the trip logic was satisfied with the B and D instruments being in trip as shown in the stem of the question. Therefore, the candidate must determine that the MSIV red open indicating lights will be lit.

Distractor Explanation:
Distractors are incorrect and plausible because:
B. The candidate could incorrectly recall MSIV isolation logic as 2 out of 2 taken once or [Full isolation = (A AND C) OR (B AND D)]. With this assumption, and with the B and D channels tripped, the candidate could determine that a full isolation has occurred.
C. The candidate could incorrectly recall MSIV isolation logic as 2 out of 2 taken once or [Otbd isolation = (A AND C)] OR [Inbd isolation = (B AND D)]. With this assumption and the B and D channels tripped, the candidate could determine that only the Inbd MSIVs would indicate closed (green).
D. The candidate could incorrectly recall MSIV isolation logic as 2 out of 2 taken once or [Inbd isolation = (A AND C)] OR [Otbd isolation = (B AND D)]. With this assumption and the B and D channels tripped, the candidate could determine that only the Otbd MSIVs would indicate closed (or green).

Reference Information:
ARP 2D36 NSSS Isolation Ch B/D Trip
ST-OP-315-0048, PCIS System Student Text
ST-OP-315-0005, Nuclear Boiler System Student Text

NUREG 1123 KA Catalog Rev. 2
223002 PCIS/NSSS
223002 A3. Ability to monitor automatic operations of the PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF including:
223002 A3.01 3.4/3.4 System indicating lights and alarms

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 Question Use (ILO 2018)
BANK
High
RO

Associated objective(s):
Primary Containment Isolation System
Cognitive Enabler
Discuss effective monitoring of Primary Containment Isolation System using local, remote, computer displays and alarms.

THIS PAGE LEFT BLANK

40	K/A Importance: 3.2/3.2			**Points: 1.00**
R40	Difficulty: 2.00	Level of Knowledge: Low	Source: NEW	73746

Following an uncomplicated reactor scram from 100% power, what is the final speed of the Reactor Recirculation Pumps and why?

A. 28%, due to DCS logic sensing both RRMG Set Field Breakers open.

B. 30%, due to the Feedwater Control System sensing less than 20% Total Feedwater Flow.

C. 37% due to the Feedwater Control System sensing RPV Level less than Level 4 with low feed pump flow .

D. 40% due to the DCS logic sensing closure of both heater drain check valves, N2200-F026A and B, for greater than 3 seconds.

Answer: B

Answer Explanation:
On a reactor scram, Post Scram Feedwater Logic reduces the speed of the operating feedwater pumps in order to prevent over-feeding the reactor vessel. This reduction in feedwater flow results in total feedwater flow dropping less than 20%. This total feedwater flow signal is developed by Feedwater DCS. The Reactor Recirculation System (RRS) receives this Total Feedwater flow signal from the Feedwater Control System (FWCS) for the purpose of generating a RR Limiter #1 signal. RR Limiter 1 is actuated when total Feedwater flow is <20% (approximately 3.15 Mlbm/hr). Recirc DCS then limits RR flow to 30% speed, which is where the RR pumps end up following a scram.

Distractor Explanation:
Distractors are incorrect and plausible because:

A. 28% speed is the minimum speed of the RR DCS controllers. When the RRMG set field breakers are open, RR DCS automatically adjusts the RRMG set speed controller setpoint to 28% in preparation for the next RRMG set startup. It is a common misconception among licensed operator candidates that the RR pumps run back to minimum (28%) speed, vice the actual 30% speed, on limiter #1 because of this design feature.
C. 37% is the nominal speed for RR Limiter #2. Limiter 2 is related to Feedwater DCS because it is generated from the receipt of an RPV Level 4 signal (low RPV water level alarm setpoint) from the Feedwater Control System in conjunction with a RFP trip OR low RFP suction flow signal. The candidate could confuse the low RFP suction flow signal (which comes in following every scram due to the reduction in Feedwater Flow) with the 20% feedwater flow signal discussed above, and conclude that the RR pumps must end up at 37% speed following a reactor scram.
D. 40% speed is the nominal speed for RR Limiter #3. Limiter 3 occurs when both Heater Drain Check Valves, N2200-F026A and B, are closed for greater than 3 seconds, which signifies a loss of heater drains. On a reactor scram, a loss of heater drain flow occurs every time, which does bring in Limiter #3. The candidate could recognize this and assume that the RR pumps stop at Limiter #3 on a reactor scram. However, the RR pumps do not stop at Limiter 3 and instead lower down to the Limiter 1 setpoint of 30%.

Reference Information:
23.107, Reactor Feedwater and Condensate Systems
23.138.01, Reactor Recirculation System

NUREG 1123 KA Catalog Rev. 2
259002 Reactor Water Level Control System
259002 K1 Knowledge of the physical connections and/or cause effect relationships between REACTOR WATER LEVEL CONTROL SYSTEM and the following:

259002 K1.08 3.2/3.2 Recirculation system: Plant-Specific

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 Question Use (ILO 2018)
Low
NEW
RO

Associated objective(s):
Feedwater Control (C3200)
Cognitive Enabler
Discuss the Feedwater Control System interrelationships with other systems.

41	K/A Importance: 3.7/4.1			Points: 1.00
R41	Difficulty: 3.00	Level of Knowledge: High	Source: NEW	73928

The plant is in MODE 5. 24.404.02 Section 5.3, Division 1 SGTS and Secondary Containment Isolation Dampers 18 Month Operability Test is in progress with the following plant conditions:

- RBHVAC is in service.
- Both divisions of SGTS are in standby.
- Division 1 CCHVAC is running in the Normal mode.
- Refueling operations are in progress on RB-5.
- Underwater touchup painting is in progress inside the Torus.

What will be the impact of the operator arming and depressing the Division 1 Manual Isolation TRIP pushbutton and what action is required because of this impact?

A. Both divisions of SGTS will start, direct the Torus painting crew to stop.

B. RBHVAC will trip, notify refuel floor personnel that RBHVAC will be lost.

C. Both divisions of SGTS will start, place the Division 2 SGTS Exhaust Fan CMC in OFF-RESET.

D. CCHVAC will shift to the Recirculation Mode, start Division 2 CCHVAC and shut down Division 1.

Answer: B

Answer Explanation:
The surveillance procedure has a precaution (2.3) and several notes that inform the operator of the impact of this surveillance on the RBHVAC system. Performance of this surveillance will cause RBHVAC to isolate, so the surveillance requires the operator to notify Refuel Floor personnel if the surveillance is performed during refuel floor activities, as specified in the stem of the question.

Distractor Explanation:
Distractors are incorrect and plausible because:

A. Normally, any Secondary Containment Isolation signal will cause both divisions of SGTS to automatically start. However, the manual isolation pushbuttons are divisional and allow for divisional testing of the logic, such as with the surveillance being run in the stem of this question. In this case, depressing the Division 1 Manual Isolation TRIP pushbutton will only cause a start of Division 1 SGTS. If the candidate thought that both divisions of SGTS would start, then directing the crew to stop Torus painting could be appropriate because the surveillance procedure has a precaution (2.1) and several notes that prohibit running the surveillance with painting activities in progress in the Reactor Building. However, Precaution 3.6 of the SOP, 23.404, has a clarification that states "torus touchup painting underwater doesn't release vapors to RB atmosphere and would be excluded from this requirement." so the test could continue with underwater Torus touchup painting in progress.
C. Normally, any Secondary Containment Isolation signal will cause both divisions of SGTS to automatically start. However, the manual isolation pushbuttons are divisional and allow for divisional testing of the logic, such as with the surveillance being run in the stem of this question. In this case, depressing the Division 1 Manual Isolation TRIP pushbutton will only cause a start of Division 1 SGTS. If the candidate thought that both divisions of SGTS would start, then placing the Division 2 Exhaust Fan in OFF-RESET would prevent this from occurring.
D. The surveillance procedure has a precaution (2.3) and several notes that inform the operator that the performance of Section 5.3 will cause CCHVAC to shift to the Recirculation Mode. However, CCHVAC logic causes BOTH divisions to shift to Recirc for any Secondary Containment Isolation, regardless of which division is running, due to the Control Room Emergency Filtration (CREF) function of CCHVAC. If the candidate thought that CCHVAC was divisional, then starting Div 2 and shutting down Div 1 CCHVAC would be a correct course of action.

Reference Information:
24.404.02, Division 1 SGTS Filter and Secondary Containment Isolation Damper Operability Test.
23.404, Standby Gas Treatment System SOP

NUREG 1123 KA Catalog Rev. 2
261000 Standby Gas Treatment System
G2.2.12 Knowledge of surveillance procedures

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

NRC Question Use (ILO 2018)
High
NEW
RO

Associated objective(s):
Standby Gas Treatment System
Cognitive Enabler
State major precautions and limitations, and major safety considerations for the Standby Gas Treatment System, and describe their bases.

42	K/A Importance: 2.9/3.2			Points: 1.00
R42	Difficulty: 2.50	Level of Knowledge: High	Source: BANK	73946

The reactor has scrammed due to a Loss of ALL Offsite Power (LOP).
ONLY EDGs 13 & 14 have started and loaded.

What is the status of DC electrical distribution?

A. Div 1 Chargers are supplying Div 1 DC loads.
Div 2 Chargers are supplying Div 2 DC loads.

B. Div 1 Chargers are supplying Div 1 DC loads.
Div 2 Batteries are supplying Div 2 DC loads.

C. Div 1 Batteries are supplying Div 1 DC loads.
Div 2 Chargers are supplying Div 2 DC loads.

D. Div 1 Batteries are supplying Div 1 DC loads.
Div 2 Batteries are supplying Div 2 DC loads.

Answer: D

Answer Explanation:
When power is lost to the ESF 4160V and 480V busses, a load shed will occur. When the EDGs restore power to the busses, EDG load sequencing will restore power to certain safety related loads. However, the ESF battery chargers do NOT get re-sequenced back on. Although the busses supplied by EDGs 13 and 14 have power available, the battery chargers remain de-energized unless manual operator action is taken. Therefore, until the Division 2 battery chargers are restored (per 20.300.Offsite, Steps AU through AY), all station DC loads will be supplied by the station batteries.

Distractor Explanation:
Distractors are incorrect and plausible because:
A. This is the normal configuration for DC electrical distribution. If the candidate failed to recognize the impact of the AC power loss, he/she could incorrectly determine that the DC electrical distribution configuration did not change for either division.
B. This is the normal configuration for Div 1 DC electrical distribution. If the candidate failed to recognize the impact of the loss of the Div 1 EDGs or if the candidate failed to recognize that all offsite power was lost and, because only Div 2 EDGs started, assumed that Div 1 offsite power was still available, he/she could incorrectly determine that the DC electrical distribution configuration did not change. The second part of this distractor is correct.
C. The first part of this distractor is correct. For the second part, the candidate could assume that the Division 2 battery chargers are restored automatically, via load sequencing, when the Div 2 EDGs (13 & 14) are loaded. This is incorrect because the battery chargers do not automatically re-sequence as described above.

Reference Information:
20.300.Offsite, Steps AU through AY for actions taken to restore the Division 1 battery chargers.
20.300.Offsite Bases, Description of actions taken for AU.1 through AY.1.

NUREG 1123 KA Catalog Rev. 2
262001 AC Electrical Distribution
262001 K3. Knowledge of the effect that a loss or malfunction of the A.C. ELECTRICAL DISTRIBUTION will have on following:
262001 K3.03 2.9/3.2 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 Question Use (ILO 2018)
BANK
High
RO

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

43	K/A Importance: 3.1/3.4			**Points: 1.00**
R43	Difficulty: 3.00	Level of Knowledge: High	Source: BANK	73968

UPS B local indication shows that the unit's rectifier has failed and has no output.

From where is the UPS B distribution cabinet being supplied power?

A. The UPS battery, via the UPS B Inverter.

B. Alternate Power Source 72M, via the UPS A Rectifier.

C. Alternate Power Source 72M, via the UPS A Inverter.

D. Normal Power Source 72R, via the 480/120VAC voltage regulator.

Answer: B

Answer Explanation:
The normal power supply to UPS Bus A and the alternate supply to UPS Bus B is from Bus 72M position 3D. The normal supply to UPS Bus B and the alternate supply to UPS Bus A is from Bus 72R position 2B.

During normal operation, each Rectifier/Charger provides regulated DC voltage to the Inverters and maintains the common battery at full charge voltage. In turn, the Inverter supplies its associated distribution panel with AC voltage. If a normal power supply is lost or a Rectifier/Charger fails, the other UPS unit's Rectifier/Charger will supply both Inverters and maintain the battery charged. If the DC power is lost (both Rectifier/Chargers and the battery), a static switch will automatically supply the distribution cabinet on that side from the alternate power supply.

Since the stem of the question states that UPS B has encountered a rectifier failure, the candidate must recall that UPS has an installed design feature that will allow the other UPS unit's (UPS A in this case) power supply (72M) to take over, via the other unit's rectifier (UPS A) and through the affected unit's (UPS B's) inverter.

Distractor Explanation:
Distractors are incorrect and plausible because:

A. The candidate could incorrectly recall the flow of power through the UPS system and assume that failure of the UPS B rectifier will block the input of the normal AND alternate AC sources, which could lead the candidate to determine that power then shifts to the output of the UPS battery through the UPS A inverter. This is not correct because the UPS A rectifier can supply both units' inverters and the UPS battery.
C. The candidate could incorrectly recall the flow of power through the UPS system and assume that the two systems are cross-connected below the two inverters. This could lead the candidate to determine that failure of the UPS B rectifier blocks the flow of power through the UPS B inverter, requiring the UPS A inverter to take over and supply 120Vac power to the UPS B distribution cabinet. This is incorrect because the systems are cross-connected BEFORE the two inverters, allowing UPS A rectifier to feed the UPS B loads through the UPS B inverter and NOT the UPS A inverter.
D. The candidate could incorrectly recall the flow of power through the UPS system and assume that loss of the UPS B rectifier blocks all flow of power through the B UPS unit. This could lead the candidate to determine that the UPS B Static Transfer Switch would transfer to allow 120Vac to the UPS B distribution cabinets from the Alternate Power Source (72M) via the 480VAC/120VAC regulated transformer. This is not correct because the UPS A rectifier can supply both units' inverters and the UPS battery, so the Static Transfer Switch will not have to throw over to the regulated transformer.

Reference Information:
23.308.01, UPS System SOP, Section 1.1 System Description and Enclosure C, UPS System Single Line Diagram.

NUREG 1123 KA Catalog Rev. 2
262002 UPS (AC/DC)

262002 K4. Knowledge of UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.) design feature(s) and/or interlocks which provide for the following:
262002 K4.01 3.1/3.4 Transfer from preferred power to alternate power supplies

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 Question Use (ILO 2018)
BANK
High
RO

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

THIS PAGE LEFT BLANK

44	K/A Importance: 3.1/3.4			Points: 1.00
R44	Difficulty: 2.00	Level of Knowledge: Low	Source: BANK	73986

What would be the result of a loss of Division I ESF DC Power?

A. The Reactor Core Isolation Cooling (RCIC) automatic start function is DISABLED.

B. The High Pressure Coolant Injection (HPCI) automatic start function is DISABLED.

C. The Reactor Water Cleanup (RWCU) Outboard Isolation Valve G3352-F004 will CLOSE.

D. The Inboard Main Steam Isolation Valves (MSIVs) B2103-F022A through D will CLOSE.

Answer: A

Answer Explanation:
Division 1 ESF DC Power supplies power to motor operated valves and RCIC initiation logic. Loss of this power source will DISABLE the automatic start function of the RCIC system, which is a major Division 1 ESF DC Load.

Distractor Explanation:
Distractors are incorrect and plausible because:

B. The candidate could incorrectly recall that HPCI is a Div 1 System and RCIC is a Div 2 system, which is incorrect. If HPCI were a Div 1 System, then the same impact on RCIC would apply to HPCI, rendering its automatic start function to be DISABLED. Also, two HPCI valves ARE powered by Division 1 DC. One is a steam isolation valve and another isolates the HPCI vacuum relief line. These valves are normally open and would remain open on a Loss of Div 1 ESF DC. If the candidate assumed one or more of these valves went shut on a loss of DC power, then he/she could determine that the HPCI automatic start function was affected.
C. The candidate could incorrectly recall that this valve, or its logic string, is powered from Div 1 ESF DC and therefore assume that it will close upon loss of Div 1 ESF DC. This valve is, in fact, powered from 2PB-1-6B, a Div 2 ESF DC source, and is impacted by RPS B logic, not Div 1 ESF DC, so the valve will stay open.
D. The Inboard MSIVs each have one 3-way DC solenoid that will re-position when they lose power from Division 1 ESF DC. The candidate could determine that, when these 3-way solenoids are de-energized, air to the MSIV actuator will be lost, causing them to close on spring force. However, the DC solenoid valves are in series with 3-way AC solenoids. To close an MSIV, both the DC and the AC solenoids have to de-energize to vent air from the actuator.

Reference Information:
23.309, 260/130VDC Electrical System (ESF and BOP), Enclosure C - ESF DC Electrical Distribution List

NUREG 1123 KA Catalog Rev. 2
263000 DC Electrical Distribution
263000 K2. Knowledge of electrical power supplies to the following:
263000 K2.01 Major D.C. loads

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 Question Use (ILO 2018)
BANK
Low
RO

Associated objective(s):
DC Electrical Distribution (R3200 & S3102)
Cognitive Enabler
Describe how the DC Electrical Distribution System assists in maintaining the critical safety functions.

45	K/A Importance: 3.4/3.5			**Points: 1.00**
R45	Difficulty: 3.00	Level of Knowledge: Low	Source: BANK	74129

The plant experienced a loss of Bus 64B.

The bus was subsequently restored to the normal lineup EXCEPT the operators neglected to reset the digital load sequencer.

Following the restoration, all power is again lost to Bus 64B.

How will the EDG and electrical distribution system respond to this event?

A. The EDG will automatically start.
The loads on bus 64B will sequence after the output breaker is closed.

B. The EDG will require a manual start.
The loads on bus 64B will sequence after the output breaker is closed.

C. The EDG will require a manual start.
The loads on bus 64B will NOT sequence after the output breaker is closed.

D. The EDG will automatically start.
The loads on bus 64B will NOT sequence after the output breaker is closed.

Answer: A

Answer Explanation:
This questions tests misconceptions on the operation of the EDG Digital Load Sequencer, specifically the operational implication of not resetting the load sequencer and how it will function if it is NOT manually reset.
From Section 3.0 of 23.321: "The associated Digital Load Sequencer must be manually reset upon restoring power to the associated 4160V EDG Bus. Failure to do so will result in the operation of the Digital Load Sequencer following a manual start of the associated EDG and the subsequent closing of its output breaker."
A common misconception is that, if the load sequencer is not reset, the EDG will require manual action to restart. That is not what this precaution says, and not how the load sequencer functions. Contributing to this another misconception regarding where the undervoltage start signal is developed.
The undervoltage start signal comes from the LOAD SHED logic for the respective 4160V ESF bus and NOT from the Load Sequencer. Therefore, if the load sequencer is not reset, the EDG will still automatically start.
Another misconception is that, if the load sequencer is not reset, then it has already actuated and it will not actuate again when the EDG output breaker closes, therefore the loads will not be sequenced back on. This is also not correct because the load sequencer resets on any subsequent undervoltage (load shed) condition. Therefore, when the EDG output breaker re-closes, the load sequencer will actuate again and re-sequence loads back on without any operator action.
The impact of the precaution is how the load sequencer responds to a MANUAL EDG start (during EDG testing, for example) if it is not reset. If the load sequencer is not reset, and the EDG is manually started, when the EDG output breaker is closed the load sequencer will actuate and go through its normal sequencing, which may be undesirable with the bus energized by the EDG in parallel with offsite power. Therefore, the correct response of the EDG to this condition is that it will still start, and load properly, if power to a 4160V bus is lost even if the EDG load sequencer was not reset.

Distractor Explanation:
Distractors are incorrect and plausible because:
B. The candidate could misunderstand the precaution discussed above and believe that the EDG will not automatically start if the load sequencer was not properly reset, which is not the case as explained above.
C. The candidate could misunderstand the precaution discussed above and believe that the EDG will not automatically start if the load sequencer was not properly reset, which is not the case, as explained above. The candidate could also believe that, if not reset, the EDG load sequencer will not actuate when its output breaker closes, so loads will not sequence on as they should, which is also incorrect as explained above.
D. The candidate could correctly recognize that the EDG will automatically start, however, incorrectly determine that, if not reset, the EDG load sequencer will not actuate when its output breaker closes, so loads will not sequence on as they should, which is incorrect as explained above.

Reference Information:
23.321, Engineered Safety Features Auxiliary Electrical Distribution System

NUREG 1123 KA Catalog Rev. 2
264000 Emergency Generators (Diesel/Jet)
264000 K5. Knowledge of the operational implications of the following concepts as they apply to EMERGENCY GENERATORS (DIESEL/JET) :
264000 K5.06 3.4/3.5 Load sequencing

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 Question Use (ILO 2018)
BANK
Low
RO

Associated objective(s):

Emergency Diesel Generator (R3000)

Cognitive Enabler

Discuss failure modes of Emergency Diesel Generator System controls and vital instruments, including design features that could result in erroneous operation or indication.

Emergency Diesel Generator (R3000)

Cognitive Enabler

Discuss failure modes of Emergency Diesel Generator System controls and vital instruments, including design features that could result in erroneous operation or indication.

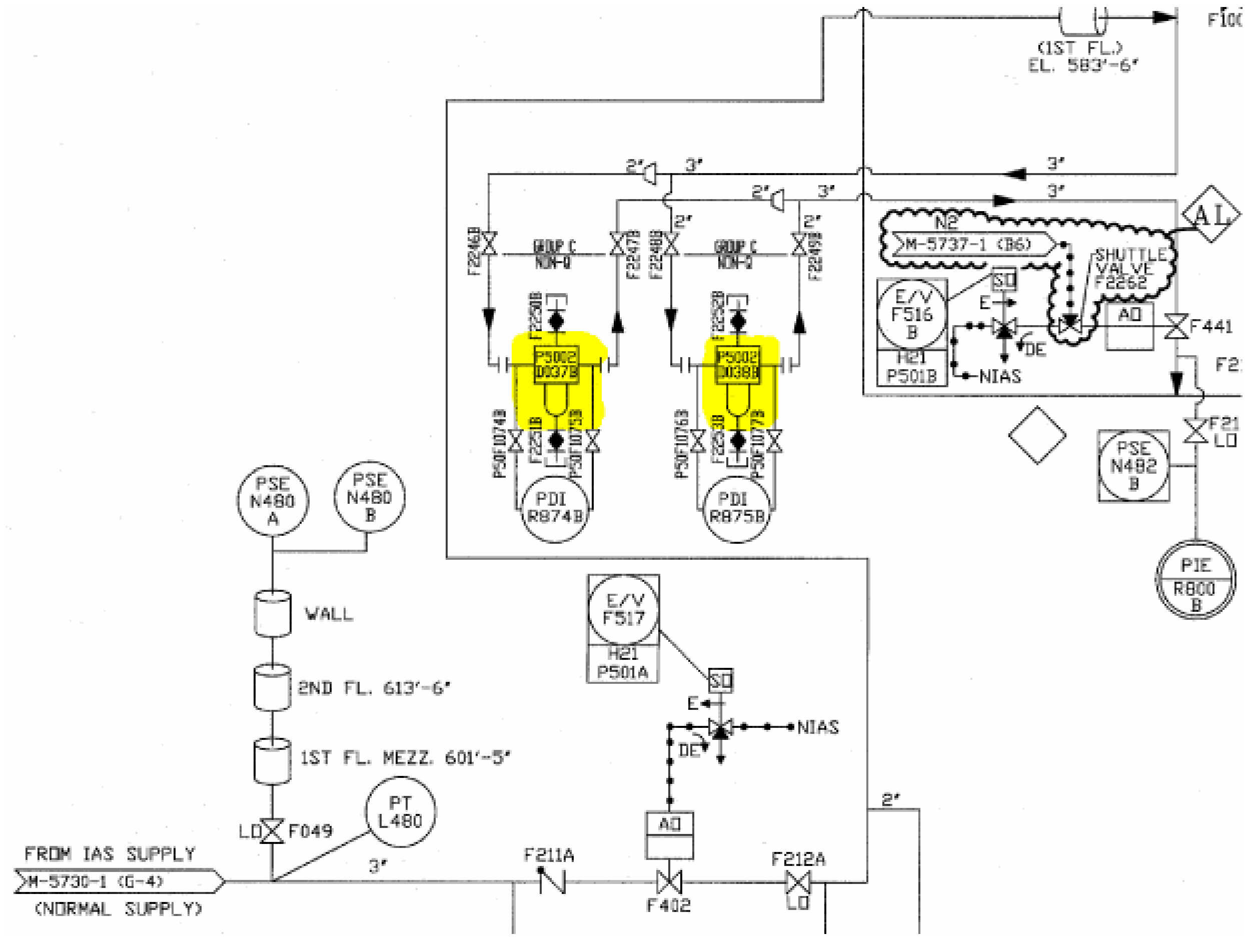
THIS PAGE LEFT BLANK

46	K/A Importance: 2.8/2.3			**Points: 1.00**
R46	Difficulty: 3.00	Level of Knowledge: High	Source: NEW	74146

A portion of the Instrument Air System is shown below.

Which of the responses below best completes the following statement to describe the effect on the system if BOTH components highlighted on the drawing were to clog with corrosion products?

Downstream pressure will lower, and when that pressure reaches _________________.



A. 95 psig, the Standby Station Air Compressor will start.

B. 85 psig, the P5000-F401, Station Air to TB Header Isolation Valve will close.

C. 85 psig, the Division 2 Control Air Compressor will start and the P5000-F441, Div 2 Control Air Iso Valve, will be OPEN.

D. 75 psig, the P5000-F441, Div 2 Control Air Iso Valve, will CLOSE AND the Division 2 Control Air Compressor will then automatically start.

Answer: C

Answer Explanation:
The components highlighted are filters in the normal air supply line from the Interruptable Air System (IAS) to the Non-Interruptable Air System (NIAS).
If these filters were to clog, air flow downstream (to the right) of the filters would lower, resulting in lowering pressure sensed by pressure switch PSE-N482B.
This pressure switch causes an auto start of the Division 2 Control Air Compressor (CAC) at 85 psig (see ARP 7D50, Control Air Compressor Auto Start). At this pressure, the P5000-F441, Div 2 Control Air Iso Valve, will be OPEN, since it does not get a signal to close until pressure sensed by PSE-N482B reaches 75 psig (See Student Text ST-OP-315-0071, Table 7).

Therefore, when pressure sensed by that switch reaches 85 psig, the CAC will auto start and the P5000-F441 will still be OPEN.

Distractor Explanation:
Distractors are incorrect and plausible because:
A. The Standby Station Air Compressors do receive a signal to start automatically at 95 psig. This is incorrect, however, because the pressure switch that causes this action (P5000-N415) is located upstream of the filters (not shown on the drawing in the stem of the question) and would not be impacted by clogging of the filters highlighted in the question.
B. The P5000-F401 does receive a signal to isolate automatically at 85 psig. This is incorrect, however, because the pressure switch that causes this action (P5000-N404) is located upstream of the filters (not shown on the drawing in the stem of the question) and would not be impacted by clogging of the filters highlighted in the question.
D. The candidate could correctly recall that the P5000-F441 valve will close at 75 psig. However, the candidate could incorrectly determine that the Division 2 CAC starts at the same time the Div 2 isolation valve closes, at 75 psig, which is not correct since the CAC gets a signal to start at 85 psig as sensed by PSE-N482B.

Reference Information:
ST-OP-315-0071, Compressed Air System Student Text.
M-5730-3, NIAS Division I & II FOS.
7D50 Control Air Compressor Auto Start.

NUREG 1123 KA Catalog Rev. 2
300000 Instrument Air System
300000 K6. Knowledge of the effect that a loss or malfunction of the following will have on the INSTRUMENT AIR SYSTEM:
300000 K6.13 2.8/2.3 Filters

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 Question Use (ILO 2018)
High
NEW
RO

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

List the automatic features of Compressed Air System operations.

Compressed Air Systems (P5001 & P5002)
Cognitive Enabler

List the automatic features of Compressed Air System operations.

THIS PAGE LEFT BLANK

47	K/A Importance: 2.7/2.7			Points: 1.00
R47	Difficulty: 2.00	Level of Knowledge: Low	Source: NEW	74166

The plant is at 100% power. The crew is preparing to alternate running RBCCW pumps, which will require them to start the South and then shut down the North.

Which of the following describes the response that the CRLNO will observe when the South RBCCW Pump is started and why?

A. The P42-F403, RBCCW DP Control Valve, will OPEN to maintain a constant DP between the RBCCW and GSW headers.

B. The P42-F403, RBCCW DP Control Valve, will CLOSE to maintain a constant DP between the RBCCW and GSW headers.

C. The P42-F403, RBCCW DP Control Valve, will OPEN to maintain a constant DP between the RBCCW supply and return headers.

D. The P42-F403, RBCCW DP Control Valve, will CLOSE to maintain a constant DP between the RBCCW supply and return headers.

Answer: C

Answer Explanation:
Per a NOTE in section 6.1 of 23.127, when the standby (South) RBCCW Pump is started, the P42-F403 RBCCW DP Control Valve, will OPEN to compensate for the increased flow and pressure.
This recirculation control valve maintains a constant pressure differential across the supply and return headers, based on RBCCW Pump d/p. By doing this, the flowrate through the various loads will not change thereby minimizing temperature fluctuations on RBCCW cooled components.

The candidate must recognize that, when the system controls are manipulated to start the standby RBCCW Pump, then RBCCW Supply Pressure would tend to rise. However, the candidate must then determine that the P42-F403 is designed to counteract this rise in pressure and will OPEN in response to this as it attempts to maintain the d/p between the RBCCW supply and return headers constant.

Distractor Explanation:
Distractors are incorrect and plausible because:
A. The candidate could correctly recall that the P42-F403 will OPEN in this situation, but incorrectly recall that the valve maintains d/p between RBCCW and GSW, which it does not.
B. The candidate could incorrectly recall that the P42-F403 responds to maintain d/p between the RBCCW and GSW headers and conclude that, if an additional pump is started, this valve would have to CLOSE in order to maintain this differential pressure. This is plausible because other systems at Fermi 2 (Main Turbine Lube Oil, Hydrogen Seal Oil) have pressure control valves that would CLOSE, in response to an additional pump start, in an attempt to maintain system pressure or differential pressure.
D. The candidate could correctly recall that the P42-F403 maintains the d/p between the RBCCW supply and return headers. However, the candidate could incorrectly determine that this valve would CLOSE in response to starting the standby RBCCW Pump. This is plausible because other systems at Fermi 2 (Main Turbine Lube Oil, Hydrogen Seal Oil) have pressure control valves that would CLOSE, in response to an additional pump start, in an attempt to maintain system pressure or differential pressure.

Reference Information:
23.127, RBCCW/EECW System SOP, Section 6.1 Alternating RBCCW Pumps.
ST-OP-315-0067, RBCCW/EECW System Student Text.

NUREG 1123 KA Catalog Rev. 2
400000 Component Cooling Water System
400000 A1. Ability to predict and / or monitor changes in parameters associated with operating the CCWS controls including:
400000 A1.03 2.7/2.7 CCW Pressure

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 Question Use (ILO 2018)
Low
NEW
RO

Associated objective(s):

Reactor Building Closed Cooling Water Emergency Equipment Cooling Water (P4200/P4400)

Cognitive Enabler

List the automatic features of Reactor Building Closed Cooling Water/Emergency Equipment Cooling Water System operations.

Reactor Building Closed Cooling Water Emergency Equipment Cooling Water (P4200/P4400)

Cognitive Enabler

List the automatic features of Reactor Building Closed Cooling Water/Emergency Equipment Cooling Water System operations.

THIS PAGE LEFT BLANK

48	K/A Importance: 3.2/3.3			Points: 1.00
R48	Difficulty: 3.00	Level of Knowledge: High	Source: NEW	74206

A LOCA has occurred. RHR Pump D is operating in the LPCI Mode, at 10,000 gpm, maintaining RPV Level steady at 197 inches.

2D26, DIV II RHR SYSTEM LOW FLOW BYPASS INITIATED alarms due to a flow instrument failure and the RHR system responds as if this were a valid condition.

Which one of the following describes the impact of this failure and the action that should be taken to mitigate its consequences?

A. RHR Pump D will trip due to suction valve closure; START RHR Pump B.

B. RHR Pump D will be operating with NO discharge flow path; STOP RHR Pump D.

C. RPV level will rise due to water bypassing the RHR Heat Exchanger; CLOSE E1150-F048B, Div 2 RHR HX Bypass Valve.

D. RPV level will lower due to water being pumped to the Torus; CLOSE the E1150-F007B, Div 2 RHR Pumps Minimum Flow Valve.

Answer: D

Answer Explanation:
Alarm 2D26 comes in when instrument E1100-N021B senses low flow (<3,000 gpm) through the RHR system. The impact of this will be an automatic opening of the E1150-F007B, Div 2 RHR Pumps Min Flow Valve. When open, this valve will pass 3,000 gpm of water directly from the RHR Pump discharge back to the Torus, which will result in a lowering RPV Water Level. The correct action, per ARP 2D26, is to attempt to CLOSE the E1150-F007B.

Distractor Explanation:
Distractors are incorrect and plausible because:

A. The candidate could assume that the alarm was the result of a low flow condition or the alarm could result in an automatic closure of the RHR Suction Valve(s). If the suction valve did close, this would result in a trip of the running RHR Pump, so starting another RHR Pump to maintain RPV level would be the correct course of action. This is incorrect, however, because the alarm results in opening of the Minimum Flow Bypass Valve.
B. The candidate could assume that the alarm will result in automatic closure of a valve in the RHR Pump discharge path, either directly or upon interlock from opening of the Minimum Flow Valve. Also, if an RHR Pump were operating with no discharge flow path, alarm 2D26 would occur, which also makes this distractor plausible. The operator could determine the correct course of action is to stop the running RHR Pump under this condition. This distractor is incorrect, however, because the alarm results in opening of the Minimum Flow Bypass Valve.
C. The candidate could assume that the alarm resulted in opening of the Heat Exchanger BYPASS and not the Minimum Flow BYPASS valve, thereby causing RPV temperature and level to rise, due to thermal expansion of the water during heatup, and also reduced headloss. The candidate could then determine that closing the HX Bypass Valve is the correct course of action to take. This answer is incorrect, however, because the alarm will cause the Minimum Flow Bypass Valve to open, thereby lowering RPV water level.

Reference Information:
2D26, Div II RHR System Low Flow Bypass Initiated.

NUREG 1123 KA Catalog Rev. 2
203000 RHR/LPCI: Injection Mode

203000 A2.	Ability to (a) predict the impacts of the following on the RHR/LPCI: INJECTION MODE (PLANT SPECIFIC) ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:

203000 A2.13 3.2/3.3 Valve openings.

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

NRC Question Use (ILO 2018)
High
NEW
RO

Associated objective(s):
Residual Heat Removal (E1100)
Cognitive Enabler
Describe general RHR System operation, including component operating sequence, normal operating parameters, and expected system response.

Residual Heat Removal (E1100)
Cognitive Enabler
Describe general RHR System operation, including component operating sequence, normal operating parameters, and expected system response.

THIS PAGE LEFT BLANK

49	K/A Importance: 3.2/3.2			Points: 1.00
R49	Difficulty: 3.00	Level of Knowledge: High	Source: NEW	74207

The Plant is in Cold Shutdown. RHR pump B is aligned and operating in the Shutdown Cooling Mode. A rise in RPV pressure is detected as follows:

- B31-N111A, Division 1 Reactor Vessel Shutdown Cooling Cut-in Permissive Interlock Pressure Instrument - 95 psig
- B31-N111B, Division 2 Reactor Vessel Shutdown Cooling Cut-in Permissive Interlock Pressure Instrument - 85 psig

You observe the following RHR System conditions:

- E1150-F009, RHR SDC Inbd Suction Iso Valve is CLOSED.
- E1150-F008, RHR SDC Otbd Suction Iso Valve is OPEN.
- E1150-F004B, RHR Pump B Torus Suct Iso Valve is CLOSED.
- E1150-F006B, RHR Pump B SDC Suct Iso Valve is OPEN.
- E1102-C002B, Div 2 RHR Pump B is RUNNING.

Which of the following automatic operations FAILED to occur and requires manual operator action to correct?

A. E1150-F009, failed to OPEN.

B. E1102-C002B failed to TRIP.

C. E1150-F008, failed to CLOSE.

D. E1150-F006B, failed to CLOSE.

Answer: B

Answer Explanation:
The Reactor Vessel Shutdown Cooling Cut-In Permissive Interlock setpoint is 89.5 psig. There are two instruments that sense this pressure to enforce this interlock, they are the B31-N111A and B31-N111B. The logic is such that the A instrument supplies Division 1 and the B instrument Division 2. If the A instrument senses pressure above 89.5 psig, then the E1150-F009 will close. Conversely, if the B instrument senses pressure above 89.5 psig, then the E1150-F008 will close. Therefore, the valves in the stem of the question have responded correctly.

The RHR Pumps have interlocks that cause them to trip on loss of suction path from either the RPV or the Torus. An RHR Pump will trip if either its Torus Suction path (E1150-F004 not fully open) AND its SDC Suction path (F009, F008 and/or F006 are not fully open) is not aligned. With the F009 and the F004B closed, the candidate should recognize a condition that would require an automatic pump trip.

Distractor Explanation:
Distractors are incorrect and plausible because:
A. The candidate could incorrectly recall the SDC Cut-In Permissive is met when pressure is ABOVE the setpoint, and not below, and therefore conclude that the Division 1 (Inboard) permissive is met, so the E1150-F009 should be open, which is incorrect as described above.
C. The candidate could incorrectly recall the logic for automatic closure of the RHR SDC Suction Valves on an overpressure condition and conclude that the one instrument above the setpoint should have caused both valves to close, which is incorrect per the logic described above.
D. The candidate could incorrectly determine that the combination of high pressure on the Division 1 instrument, along with closure of the F009, would result in closure of the SDC Pump Suction valve, which is incorrect as described above.

Reference Information:
23.601, Instrument Trip Sheets
ST-OP-315-0041, RHR System Student Text, Table 4 Control Functions and Interlocks.

NUREG 1123 KA Catalog Rev. 2
205000 RHR Shutdown Cooling Mode
205000 A3. Ability to monitor automatic operations of the SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE) including:
205000 A3.02 3.2/3.2 Pump trips

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 Question Use (ILO 2018)
High
NEW
RO

Associated objective(s):
Residual Heat Removal (E1100)
Cognitive Enabler
List the interlocks associated with RHR System components.

Residual Heat Removal (E1100)
Cognitive Enabler
List the interlocks associated with RHR System components.

50	K/A Importance: 4.0/3.9			**Points: 1.00**
R50	Difficulty: 2.00	Level of Knowledge: Low	Source: NEW	74227

The plant was operating at 100% power when an inadvertent HPCI initiation occurred. The CRLNO has evaluated plant conditions and determined that HPCI injection is not needed.

Which of the following contains the actions that are needed to trip the HPCI Turbine and keep it in a tripped condition?

A. Turn the arming collar to ARMED, depress then release the TRIP Pushbutton.

B. Turn the arming collar to ARMED, depress the TRIP Pushbutton, then release the TRIP Pushbutton when turbine speed indicates zero rpm.

C. Turn the arming collar to ARMED, depress the TRIP Pushbutton, place the Aux Oil Pump in OFF when turbine speed indicates zero rpm, then release the TRIP Pushbutton.

D. Turn the arming collar to ARMED, depress the TRIP Pushbutton, place the Aux Oil Pump in OFF when turbine speed indicates zero rpm, depress the HPCI Initiation Signal RESET pushbutton, then release the TRIP Pushbutton.

Answer: C

Answer Explanation:
Per 23.202, Section 8.1, the actions listed are required to trip the HPCI Turbine. To summarize:

- The Operator must ARM the Trip Pushbutton.
- The Operator must depress and HOLD the Trip Pushbutton.
- The Operator will observe various HPCI System indications to verify proper system shutdown.
- When Turbine RPM reaches 0, the Operator must place the Aux Oil Pump in OFF to prevent HPCI from re-starting.
- Then, the Operator can release the Trip Pushbutton.

Distractor Explanation:
Distractors are incorrect and plausible because:

A. The candidate could confuse the HPCI system with the RCIC system and determine that all that is required is to arm, then depress the HPCI Trip Pushbutton. For RCIC, this will unlatch the Trip Throttle Valve, which will keep the system in a tripped condition (see 23.206, Section 8.1 RCIC Shutdown). For HPCI, if the Trip Pushbutton is release, HPCI will restart. It is plausible, but incorrect, for the student to assume that the HPCI Trip controls work the same as RCIC due to the similarity between the systems.

B. The candidate could recall that the Trip Pushbutton needs to be armed and depressed and recall that the operator must verify that HPCI stops rotating by observing turbine RPM. The candidate could then incorrectly determine that, once turbine RPM is zero, oil pressure developed by the installed turbine driven oil pump is too low to support turbine restart, which is true. However, the candidate would fail to recall that HPCI has a motor driven oil pump that will start on low oil pressure, when the turbine driven oil pump speed is too low, and the motor driven oil pump develops sufficient pressure to support turbine restart.

D. The candidate could determine that the initiation signal must be reset to prevent HPCI from restarting, which is not correct. Further down the actions of section 8.1, resetting the initiation signal is only necessary if the initiating condition is clear and before restoring HPCI to standby. The stem of the question asked what it takes to TRIP the HPCI system and NOT restore it to standby.

Reference Information:
23.202, HPCI System SOP, Section 8.1 HPCI Shutdown.
23.206, RCIC System SOP, Section 8.1 RCIC Shutdown.

NUREG 1123 KA Catalog Rev. 2
206000 HPCI System.

206000 A4. Ability to manually operate and/or monitor in the control room:
206000 A4.12 4/3.9 Turbine trip controls: BWR-2,3,4

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 Question Use (ILO 2018)
Low
NEW
RO

Associated objective(s):
High Pressure Coolant Injection
Cognitive Enabler
Identify associated remote and local instrumentation, indications, alarms, and controls for the HPCI System.

High Pressure Coolant Injection
Cognitive Enabler
Identify associated remote and local instrumentation, indications, alarms, and controls for the HPCI System.

THIS PAGE LEFT BLANK

51	K/A Importance: 3.7/3.9			**Points: 1.00**
R51	Difficulty: 3.00	Level of Knowledge: High	Source: NEW	74228

Below is a numbered list of instruments associated with monitoring the status of the Core Spray System when it is aligned to the Torus:

1. G51-R402, Narrow Range Torus Level Indicator
2. T50-R804A(B), Div 1(2) Torus Level Recorder
3. T23-R800, Torus Water Temperature Recorder
4. T50-R800A(B), Div 1(2) PC Air and Water Temperatures Recorder

The plant is in a lowering Torus Water Level event in which Core Spray is needed to maintain RPV Water Level.

With Torus Level below -11", which of the above are the Post-Accident Instruments that CAN BE USED to verify Core Spray is within its Vortex and NPSH Limits?

A. ONLY 2 and 4 can be used.

B. Either 1 OR 2 can be used WITH 4.

C. ONLY 2 can be used with either 3 OR 4.

D. ALL of the above instruments can be used.

Answer: A

Answer Explanation:
The candidate must first recall that the Core Spray Vortex and NPSH Limits are impacted by a combination of Torus Water Level, and Torus Water Temperature, among others. The candidate must then determine that instrument 1 is NOT a post-accident instrument and, for the conditions given in the stem of the question, cannot be used to verify CS is within NPSH and Vortex limits.

Also, with Torus Water Level below -11", the candidate must recall that instrument 3 is unreliable (per Caution 6 of the EOPs) so, although it IS a post-accident instrument, it cannot be used under these conditions to verify CS is within limits.

Therefore, the candidate must conclude that only instruments 2 and 4 are post-accident instruments that can be used under the conditions given in the stem of the question.

Distractor Explanation:
Distractors are incorrect and plausible because:
B. The candidate could incorrectly determine that instrument 1 is a post-accident instrument that can be used under these conditions, which is incorrect as described above.
C. The candidate could incorrectly determine that both instruments 3 and 4 are post-accident instruments (which is true) that can be used to determine CS system status. However, instrument 3 cannot be used with Torus Water Level below -11" per Caution 6 of the EOPs.
D. The candidate could incorrectly recall that all instruments listed are post-accident instruments that can be used to determine CS system status under the conditions given in the stem of the question, which is incorrect as described above.

Reference Information:
29.100.01, EOP Curves, Cautions and Tables.

NUREG 1123 KA Catalog Rev. 2
209001 Low Pressure Core Spray System.

G2.4.3 3.7/3.9 Ability to identify post-accident 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 Question Use (ILO 2018)
High
NEW
RO

Associated objective(s):
Core Spray System
Cognitive Enabler
Discuss effective monitoring of the Core Spray System using local, remote, computer displays and alarms.

Core Spray System
Cognitive Enabler
Discuss effective monitoring of the Core Spray System using local, remote, computer displays and alarms.

52	K/A Importance: 2.7/2.7			Points: 1.00
R52	Difficulty: 2.00	Level of Knowledge: Low	Source: NEW	74248

Which of the following instruments would be adversely affected by a break in the SLC injection piping within the Reactor Pressure Vessel?

A. CRD Drive Water Pressure.

B. Core Spray Discharge Pressure.

C. Core Plate Differential Pressure.

D. Calibrated Jet Pump Differential Pressure.

Answer: C

Answer Explanation:
Note: Refer to M-5701-2, Nuclear Boiler System FOS, Grid D5 for this explanation.

The SLC system penetration and piping provides above core plate pressure for the Core Spray line break detection instrumentation from the outside pipe. The same penetration is also used for Core Plate DP, Jet Pump DP: and CRD Drive Water DP, from the inside pipe (SLC discharge sparger pipe).

A break in the SLC injection piping would adversely affect the Core Plate Differential Pressure instrument, which is the correct answer for this question.

This break would also adversely affect the CRD Drive Water Differential Pressure, Core Spray Break Detection and non-calibrated Jet Pump Differential Pressure instruments.

Distractor Explanation:
Distractors are incorrect and plausible because:
A. The candidate could incorrectly recall that the SLC penetration impacts the CRD Drive Water Pressure instrument, which is incorrect but plausible because this SLC penetration provides the RPV Pressure (above Core Plate pressure) input to the CRD Drive Water Differential Pressure instrument and not the Drive Water Pressure instrument.
B. The candidate could incorrectly recall that the SLC penetration impacts the Core Spray Discharge Pressure instrument, which is incorrect but plausible because this SLC penetration provides the RPV Pressure (above Core Plate pressure) input to the Core Spray Break Detection (differential pressure) instrument and not the Core Spray Discharge Pressure instrument.
D. The candidate could incorrectly recall that the SLC penetration impacts the Calibrated Jet Pump Differential Pressure instrument, which is incorrect but plausible because this SLC penetration provides the high-pressure input, i.e. diffuser pressure, from the below Core Plate pressure tap to the NON calibrated Jet Pumps. The Calibrated Jet Pumps (Jet Pumps 5, 10, 15 & 20) have a separate tap directly in their diffuser section that provides the high-pressure input to the Calibrated Jet Pump Differential Pressure instruments.

Reference Information:
ST-OP-315-0014 SLC System Student Text, System Interrelations section on the Core Spray System.
M-5701-2 Nuc Boiler Instrumentation FOS.

NUREG 1123 KA Catalog Rev. 2
211000 SLC System
211000 K1. Knowledge of the physical connections and/or cause-effect relationships between STANDBY LIQUID CONTROL SYSTEM and the following:
211000 K1.02 2.7/2.7 Core plate differential pressure indication

10CFR55 RO/SRO Written Exam Content
10 CFR 55.41(b) (2) General design features of the core, including core structure, fuel elements, control rods, core instrumentation, and coolant flow.

NRC Question Use (ILO 2018)
Low
NEW
RO

Associated objective(s):
Standby Liquid Control
Cognitive Enabler
Discuss the Standby Liquid Control System interrelationships with other systems.

Standby Liquid Control
Cognitive Enabler
Discuss the Standby Liquid Control System interrelationships with other systems.

53	K/A Importance: 3.5/3.7			Points: 1.00
R53	Difficulty: 3.00	Level of Knowledge: Low	Source: NEW	74269

The plant is operating at 96% Power.

A Pressure Control System malfunction results in indicated RPV pressure rising to 1100 psig. During the transient, the B21-N678A, Reactor Vessel Steam Dome Pressure - High Trip Unit, remains UNTRIPPED.

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: A

Answer Explanation:
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 failure given in the stem of the question, the B, C and D instruments will trip, resulting in Trip Channels A2, B1 and B2 tripping, which will result in BOTH RPS Trip Systems A and B tripping, which will result in a FULL reactor scram.

Distractor Explanation:
Distractors are incorrect and plausible because:
B. The candidate could incorrectly recall RPS Trip Logic and determine the instrument failure will prevent the B RPS Trip System from tripping, which is incorrect as can be seen from the Trip Logic described above.
C. The candidate could incorrectly recall RPS Trip Logic and determine the instrument failure will prevent 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 all four pressure instrument must trip and therefore determine that the instrument failure 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.

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

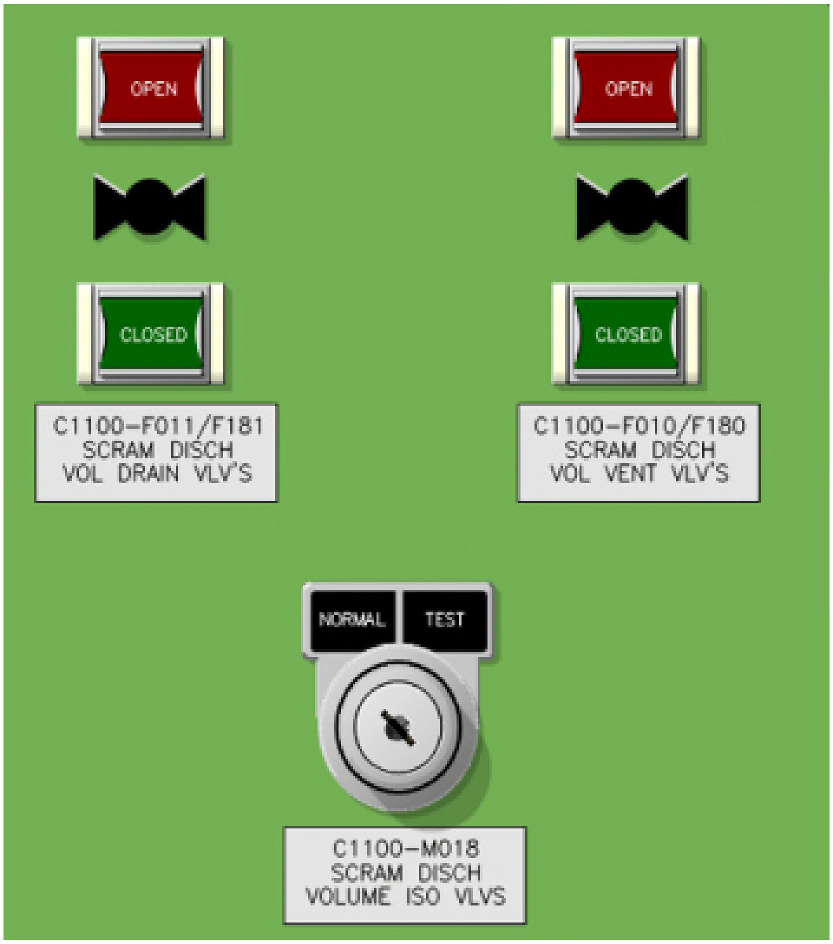
NRC Question Use (ILO 2018)
Low
NEW
RO

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.

Reactor Protection System (C7100)
Cognitive Enabler
Describe general Reactor Protection System operation, including component operating sequence, normal operating parameters, and expected system response.

THIS PAGE LEFT BLANK

54	K/A Importance: 2.8			**Points: 1.00**
R54	Difficulty: 3.00	Level of Knowledge: Low	Source: NEW	73327



NOTE: The drawing above is shown with all lights OFF (not lit).

When the switch shown is rotated to the TEST position, what occurs to allow for testing of the Scram Discharge Volume isolation valves?

A. Air is isolated and vented to test closure of the valves.

B. Air is isolated and vented to test opening of the valves.

C. Air is supplied to the actuators to test closure of the valves.

D. Air is supplied to the actuators to test opening of the valves.

Answer: A

Answer Explanation:
The candidate must recognize that the design feature that has been installed to allow testing of the Scram Discharge Volume (SDV) isolation valves is a keylock switch on the H11-P603 panel that, when taken to TEST, energizes a valve which isolates and vents air from the SDV vent and drain valves to test their closing ability.

Distractor Explanation:

B. Distractor Is incorrect because the valves are normally open and rotating the keylock switch to TEST isolates and vents air to close them, not open them. This distractor is plausible if the candidates incorrectly remembered that the keylock switch tested the valves in the open direction with a scram signal in, rather than how we actually test them, which is in the close direction. This distractor is also plausible if the candidate incorrectly recalled the normal lineup of the valves.
C. Distractor is incorrect because rotating the keylock switch to TEST isolates and vents air to close them, it does not supply air to the actuator to close them. This distractor is plausible because some valves in the Scram Air Header close when air is supplied to their actuators and some open when air is supplied to their actuators. If a candidate incorrectly thought that these valves need air to close, then this distractor could be selected.
D. Distractor is incorrect because the valves are normally open and rotating the keylock switch to TEST isolates and vents air to close them, not open them. Distractor is plausible if the candidate incorrectly remembered that the keylock switch tested the valves in the open direction with a scram signal in and if the candidate thought that air was needed to open the valves. This distractor is also plausible if the candidate incorrectly recalled the normal lineup of the valves.

Reference Information:
M-5703-2, CRD Scram Discharge Volume Functional Operating Sketch (FOS).
Student Text ST-OP-315-0010

NUREG 1123 KA Catalog Rev. 2

201001 K4.	Knowledge of CONTROL ROD DRIVE HYDRAULIC SYSTEM design feature(s) and/or interlocks which provide for the following: Testing SCRAM discharge volume isolation valves.
201001 K4.07	Testing SCRAM discharge volume isolation valves

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 Question Use (ILO 2018)
Low
NEW
RO

Associated objective(s):
Control Rod Drive Hydraulics (C1150)
Cognitive Enabler
Identify associated remote and local instrumentation, indications, alarms, and controls for the Control Rod Drive Hydraulic system.

55	K/A Importance: 4.1			**Points: 1.00**
R55	Difficulty: 3.00	Level of Knowledge: High	Source: NEW	73346

The plant is operating at 100% power when a reactor scram occurs.

What is the (1) impact of the scram on Control Rod Drive Mechanism (CRDM) temperatures and (2) action that can be taken to mitigate this impact?

A. (1) They will rise
(2) Reset the reactor scram.

B. (1) They will lower
(2) Reset the reactor scram.

C. (1) They will rise
(2) Open the CRD Flow Control Valve

D. (1) They will lower
(2) Close the CRD Flow Control Valve.

Answer: A

Answer Explanation:
Following a scram, after the CRD accumulators discharge, the scram stroke is completed by RPV pressure unseating the internal ball check valve of the CRDM, thereby allowing reactor water into the under-piston volume. Reactor water temperature overcomes the cooling effect of CRDH cooling water flow (nominally 63 gpm spread out over 185 Control Rod Drive Mechanisms) and results in rising CRDM internal temperatures. Furthermore, the normal scram response of the CRD FCV is to CLOSE, which lowers cooling water flow even more (to about 15 gpm spread out over 185 CRDMs).

ARP 3D13, CRD Hydraulic Temperature High, has a note which states "This alarm will nearly always accompany a Reactor Scram. Resetting the scram should result in lowering temperatures".

Therefore, a reactor scram causes CRDM temperatures to rise, which is corrected by resetting the scram.

Distractor Explanation:
Distractors are incorrect and plausible because:

B. The candidate could incorrectly conclude that the HCU Accumulator discharge, caused by the Scram action, will cool the CRDMs. However, this is incorrect because CRDM temperatures will rise as evidenced by the note in ARP 3D13. Resetting the scram is the correct action to take but it is taken to correct a rising temperature condition (per the note in ARP 3D13) and not a lowering temperature condition.
C. This distractor is incorrect because, even though CRDM temperatures will rise, there is no procedural guidance to perform open the CRD FCV under these conditions. To the contrary, AOP 20.106.03, CRD FCV Failure, has a note which directs CLOSING the CRD FCV on a scram condition, with the FCV in manual, to allow for limiting pump flow due to runout considerations and to allow for faster recharging of the scram accumulators. Since the normal scram action of this valve is to close, opening it to correct the rising temperature condition is not warranted.
D. The candidate could incorrectly conclude that the HCU Accumulator discharge, caused by the Scram action, will cool the CRDMs so closing the CRD FCV is the correct action. However, this is incorrect because CRDM temperatures will rise as evidenced by the note in ARP 3D13. Also, the action to close the CRD FCV is not necessary because the valve's normal scram function is to close automatically.

Reference Information:
ARP 3D13, CRD Hydraulic Temperature High
AOP 20.106.03, CRD Flow Control Valve Failure
Student Text ST-OP-315-0009

NUREG 1123 KA Catalog Rev. 2
201003 A2. Ability to (a) predict the impacts of the following on the CONTROL ROD AND DRIVE MECHANISM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:
201003 A2.05 4.1*/4.1 Reactor Scram

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 Question Use (ILO 2018)
High
NEW
RO

Associated objective(s):
Control Rod Drive Mechanism (B1104, C1101 & C1102)

Cognitive Enabler

Describe general Control Rod Drive Mechanism operation, including component operating sequence, normal operating parameters, and expected system response.

Control Rod Drive Mechanism (B1104, C1101 & C1102)

Cognitive Enabler

Describe general Control Rod Drive Mechanism operation, including component operating sequence, normal operating parameters, and expected system response.

THIS PAGE LEFT BLANK

56	K/A Importance: 2.9			**Points: 1.00**
R56	Difficulty: 3.00	Level of Knowledge: Low	Source: NEW	73366

See the attached Rod Pull Sheet to answer this question.

A startup is in progress with count rate approaching criticality. What is the operational implication of the P603 Operator selecting control rod 26-39 for movement?

A. A Select Error (SE) would be generated. ONLY withdrawal of this control rod would be prevented.

B. A Select Error (SE) would be generated. ANY movement of this control rod would be prevented.

C. No error would be generated. Rod movement could occur in either direction at which time a rod block, due to Insert Error (IE) or Withdrawal Error (WE), would occur.

D. A Select Error (SE) would be generated. Rod movement could occur in either direction at which time a rod block, due to Insert Error (IE) or Withdrawal Error (WE), would occur.

Answer: B

Answer Explanation:
RWM sequence enforcement restricts movement of control rods that are not selected and moved in compliance with the selected sequence below the LPSP. Rod motion that would result in an insert or withdraw error is not permitted. Sequence enforcement is active between all rods full-in and the LPSP. Select Errors (SE) will be indicated upon selection of any rod which is not in sequence order. Rod movement must not only conform to the RWM step, but also the rod order within the step.
Since the selected control rod is NOT the next rod in the sequence, a Select Error will occur that results in an insert block (IB) and withdrawal block (WB). Therefore, movement of this control rod, in either direction, will be blocked due to selecting this control rod out of sequence.

Distractor Explanation:
Distractors are incorrect and plausible because:

A. The candidate could recognize that the selected rod is not the next one in the sequence, resulting in a Select Error. However, the candidate could incorrectly assume that all rods in a group (on a page of the pull sheets) can move inward to their group insertion limit but the rod could not move outward until the previous rods in the group were at their group withdrawal limit. This is plausible because all the rods in a group are on the same RWM step, but it is incorrect because rod movement must not only conform to the RWM step but also the rod order within the step.
C. The candidate could fail to recognize that selecting a rod not in sequence will result in a Select Error. This could lead the candidate to determine that the rod is free to move out of its current position and, when it does, the Rod Worth Minimizer would THEN impose either rod block (due to either IE or WE, depending on direction of rod travel).
D. The candidate could recognize that the selected rod is not the next one in the sequence, resulting in a Select Error. The candidate could fail to recognize that a Select Error (SE) results in both a Insert Block (IB) and Withdrawal Block (WB). This could lead the candidate to determine that the rod could move out of its current position and, as soon as it does, the resulting Insert Error (IE) or Withdrawal Error (WE) is what would THEN generate the rod block (due to either IB or WB, depending on direction of rod travel). This is incorrect because the SE will generate both an IB and WB as described above.

Reference Information:
SOP 23.608, Rod Worth Minimizer System
Student Text ST-OP-315-0013

NUREG 1123 KA Catalog Rev. 2
201006 K5. Knowledge of the operational implications of the following concepts as they apply to ROD WORTH MINIMIZER SYSTEM (RWM) (PLANT SPECIFIC):
201006 K5.08 2.9/2.9 Operating sequence: P-Spec(Not-BWR6)

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 Question Use (ILO 2018)
Low
NEW
RO

Associated objective(s):
Rod Worth Minimizer (C1108)
Cognitive Enabler
List the interlocks associated with Rod Worth Minimizer system components.

57	K/A Importance: 3.4/3.5			Points: 1.00
R57V2	Difficulty: 3.00	Level of Knowledge: Low	Source: BANK	73386

The Plant is operating at 100% power with both RWCU Pumps and Filter/Demineralizers in service.

Alarm 2D110, RWCU NON REGEN HX OUTLET TEMP HIGH, annunciates due to a sensed temperature of 142°F.

In what way, if any, will the RWCU system automatically respond?

A. No automatic system response will occur.

B. G3352-F119, RWCU Supply Suct Iso Vlv, will shut and the RWCU Pumps will trip on low flow.

C. G3352-F119, RWCU Supply Suct Iso Vlv, will shut and the RWCU Pumps will remain running.

D. G3352-F001, RWCU Inboard Containment Iso Vlv, G3352-F004, RWCU Outboard Containment Iso Vlv, and G3352-F220, RWCU Return Iso Valve will shut and the RWCU Pumps will trip on interlock.

Answer: B

Answer Explanation:
The G3352-F119 will close on the sensed high temperature, which will result in an indirect trip of the RWCU Pumps on Low Flow.
The NRHX Outlet Temperature High Isolation has no accident mitigation function. The basis for this isolation is to protect the ion exchange resin from deterioration due to high temperature. The low flow trip protects the pumps from damage due to running on low flow.

Distractor Explanation:
Distractors are incorrect and plausible because:
A. The candidate could determine that the temperature given is not valid for the alarm condition and determine that automatic closure of the G3352-F119 would not occur. Or, the candidate could recognize the alarm and temperature as being valid, but not correctly recall the automatic system response that would take place as described above. Either of these reasons could lead the candidate to determine that no automatic system response will occur, which is not correct as described above.
C. The candidate could recognize the initiating condition as a signal that causes the G3352-F119 to close. However, the candidate could fail to recognize that closure of the G3352-F119 will result in a low flow condition in the system that will in turn cause the RWCU pumps to trip.
D. The candidate could incorrectly recall the high temperature trip as an isolation signal and conclude that the RWCU system isolation valves (G3352-F001, F004 and F220) would close and cause the RWCU pumps to immediately trip, since the RWCU pumps do immediately trip (direct trip) on closure of any of these valves.

Reference Information:
ARP 2D110, RWCU Non-Regen HX Outlet Temp High
23.707, RWCU System SOP
Student Text ST-OP-315-0008

NUREG 1123 KA Catalog Rev. 2
204000 A3. Ability to monitor automatic operations of the REACTOR WATER CLEANUP SYSTEM including:

204000 A3.04 3.4/3.5 Response to interlocks and trips designed to protect system components

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 Question Use (ILO 2018)
BANK
Low
RO

Associated objective(s):
Reactor Water Cleanup
Cognitive Enabler
Discuss effective monitoring of the Reactor Water Cleanup system using local, remote, computer displays and alarms.

Reactor Water Cleanup
Cognitive Enabler
Discuss effective monitoring of the Reactor Water Cleanup system using local, remote, computer displays and alarms.

58	K/A Importance: 3.1/3.4			**Points: 1.00**
R58	Difficulty: 3.00	Level of Knowledge: Low	Source: NEW	73406

The plant is operating in the Power Range with a Traversing In-Core Probe (TIP) run in progress.

A LOCA occurs resulting in a reactor scram on high Drywell pressure.

A malfunction of the Primary Containment Isolation System (PCIS) results in Group 15 - Traversing In-core Probe System, **NOT** receiving an isolation signal.

What is the effect of this malfunction on the TIP system?

A. The TIP detector WILL automatically withdraw, however the Ball Valve WILL NOT automatically close. The Ball Valve can be manually actuated to establish containment integrity.

B. The TIP detector WILL automatically withdraw, however the Ball Valve WILL NOT automatically close. The Shear Valve can be manually actuated to establish containment integrity.

C. The TIP detector WILL NOT automatically withdraw and the Ball Valve WILL NOT automatically close. The Ball Valve can be manually actuated to establish containment integrity.

D. The TIP detector WILL NOT automatically withdraw and the Ball Valve WILL NOT automatically close. The Shear Valve can be manually actuated to establish containment integrity.

Answer: D

Answer Explanation:
Failure of the PCIS to detect and/or initiate an isolation signal to Group 15 will result in the TIP detector remaining at its in-core location and failure of the TIP Ball Valve to close. Containment Integrity can be restored by manually actuating the TIP Shear Valve assembly, and is procedurally driven by 23.606, Section 6.2, Step 6.2.3.

Distractor Explanation:
Distractors are incorrect and plausible because:

A. The candidate could incorrectly assume that, since the Ball Valve is a Primary Containment Isolation Valve, that it is the only component in the system that will not operate due to failure of the PCIS. This assumption is plausible and could lead the candidate to determine that the detector drive assembly will retract the detector and the Ball Valve will remain open, requiring manual action to close the Ball Valve and restore Containment Integrity.
B. The candidate could incorrectly assume that, since the Ball Valve is a Primary Containment Isolation Valve, that it is the only component in the system that will not operate due to failure of the PCIS. This assumption is plausible and could lead the candidate to determine that the detector drive assembly will retract the detector and the Ball Valve will remain open, requiring manual action to actuate the Shear Valve and restore Containment Integrity.
C. The candidate could correctly determine that the failure of the PCIS impacts both the detector retract circuit and the Ball Valve automatic isolation logic. However, the candidate could incorrectly determine that the Ball Valve could still be manually closed to restore Containment Integrity, which is plausible since the Ball Valve normally provides Containment Integrity when the TIP detector retracts into the shield chamber.

Reference Information:
SOP 23.606, Traversing In-Core Probe (TIP) System
Student Text ST-OP-315-0025

NUREG 1123 KA Catalog Rev. 2
215001 K6. Knowledge of the effect that a loss or malfunction of the following will have on the TRAVERSING IN-CORE PROBE:
215001 K6.04 3.1/3.4 Primary containment isolation system: Mark-I&II(Not-BWR1)

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.

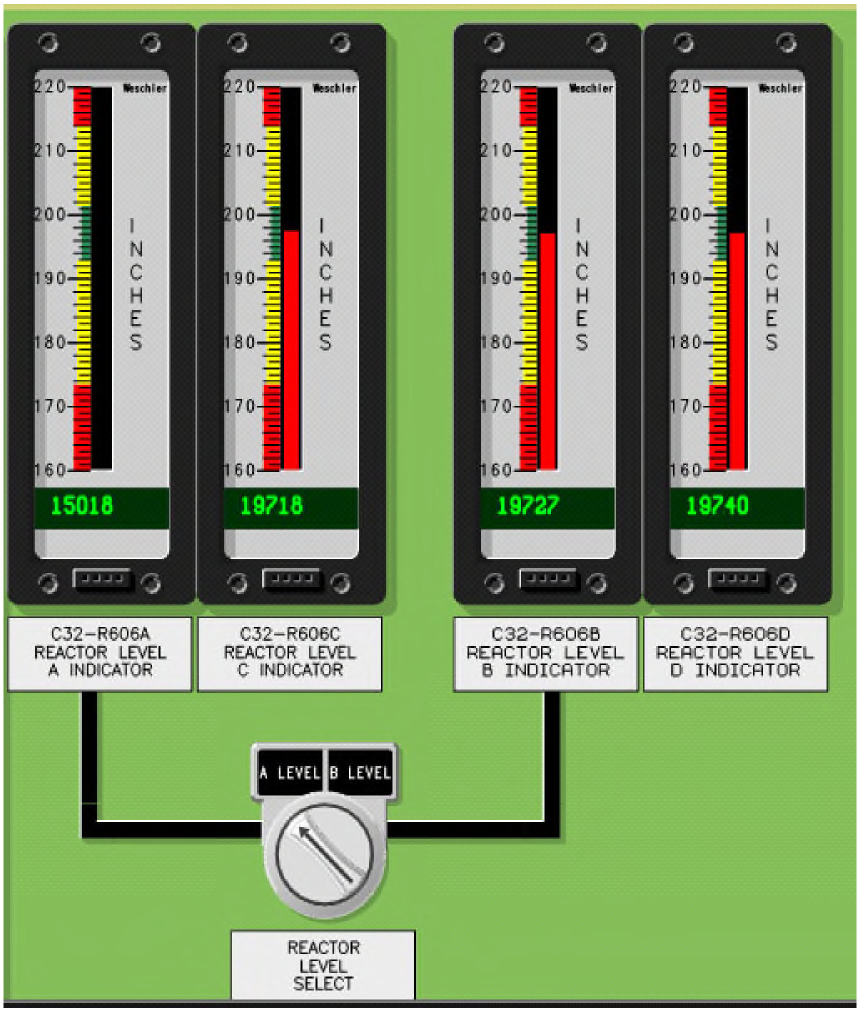
NRC Question Use (ILO 2018)
Low
NEW
RO

Associated objective(s):
Traversing In-Core Probe (C5102)
Cognitive Enabler
Discuss the Traversing Incore Probe interrelationships with other systems.

Traversing In-Core Probe (C5102)
Cognitive Enabler
Discuss the Traversing Incore Probe interrelationships with other systems.

59	K/A Importance: 3.3/3.1			Points: 1.00
R59	Difficulty: 3.00	Level of Knowledge: High	Source: NEW	73426

The plant at 100% power when the following is observed by the P603 Operator:



What is the impact to DCS logic, and what would the operator accomplish by taking the Reactor Level Select switch to B LEVEL?

A. Reactor Feed Pump speed will increase. The action could prevent a Level 8 trip if taken before the trip occurs.

B. Water level control will transfer to the B level instrument. The action would ensure the panel switch is aligned with DCS.

C. DCS will transfer to 1-element level control. The action would cause DCS to automatically transfer back to 3-element control.

D. Water level control will transfer to the median signal from the three remaining signals. The action would force DCS to use B as the lead level instrument.

Answer: D

Answer Explanation:
Four narrow-range RPV water level signals, A, B, C, and D, are supplied to DCS logic. Operation of Reactor Level Select switch selects the lead RPV water level signal A or B, to control RPV water level and input to C32-R614, RPV Narrow Range Level Recorder. If the lead selected RPV water level signal, A or B fails, DCS logic will automatically transfer RPV water level control and the input to C32-R614, to the median signal from the three remaining RPV water level signals, preventing an RPV water level transient. When a single RPV water level signal fails, the operator can restore control to a lead-selected RPV level instrument by changing the position of the selector switch. The Reactor Level Select switch position may be changed without causing an RPV water level transient.

Distractor Explanation:
Distractors are incorrect and plausible because:

A. This answer is plausible because the candidate could assume that the low RPV water level sensed by the lead-selected level signal would cause a level error (indicated level lower than level setpoint of 197) that would result in DCS logic sending a signal to the RFP speed control circuit to raise RFP speed to correct the level error. This is an incorrect assumption because of the design of DCS logic described above. If DCS logic behaved as described in the first part of this distractor, then selecting the B level to terminate the transient would be the correct action. Note that this action IS required if three level instruments were to fail and the remaining level signal can be selected as the lead signal.
B. This answer is plausible because there are only two level signals capable of being selected by DCS (A or B). The candidate could incorrectly assume that a failure of the lead-selected level signal would result in DCS logic transferring control to the non-selected level signal. However, this assumption is incorrect because of the design of DCS logic described above. If DCS responded as stated in this distractor, then placing the switch to B would be correct so that the switch position aligned properly with the signal being used by DCS for level control.
C. The answer is plausible because there are numerous failures (many of which have been on previous NRC exam questions) that would cause DCS to automatically transfer to 1-element (or single-element) control. The candidate could incorrectly recall this failure as being one of those. However, this is incorrect because of the design of DCS logic described above. IF DCS responded as stated in this distractor, then the correct action would be to change the lead-selected RPV water level signal to restore the preferred 3-element control.

Reference Information:
23.107, Reactor Feed and Condensate System, section 1.1 Feedwater System Description.
Student Text ST-OP-315-0046, Page 15, Instrumentation and Controls section for RPV Level

NUREG 1123 KA Catalog Rev. 2
216000 A4. Ability to manually operate and/or monitor in thecontrol room:
216000 A4.02 3.3/3.1 Channel select controls

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 Question Use (ILO 2018)
High
NEW
RO

Associated objective(s):
Reactor Pressure Vessel Instrumentation (B2100)
Cognitive Terminal
Given various controls and indications for Reactor Pressure Vessel Instrumentation system operations, analyze that indication to determine proper plant response in accordance with approved plant procedures.

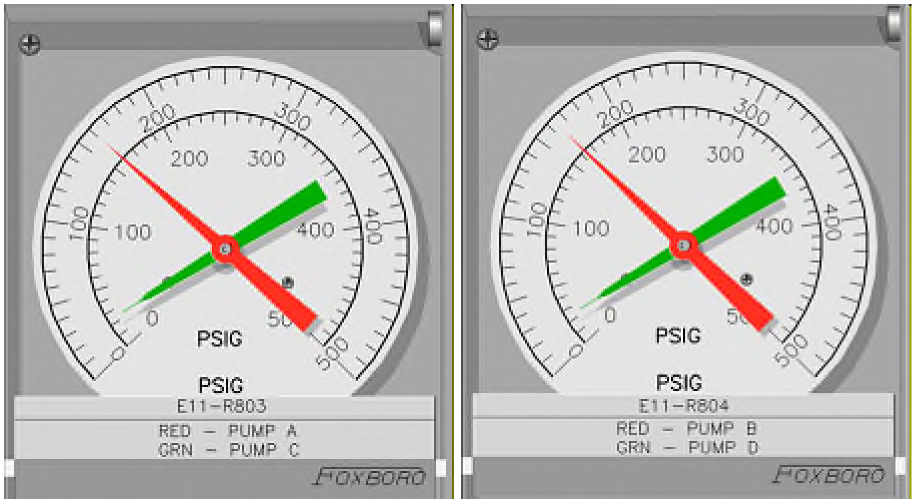
The plant has been Emergency Depressurized due to a LOCA / Loss of injection.

- RHR Pumps A and B are injecting to the RPV at 12,500 gpm each through Div 2 RHR.
- The E1150-F010, RHR Crosstie Header Valve, is OPEN.
- Division 1 and 2 RHR Pump Discharge Pressures are as indicated.

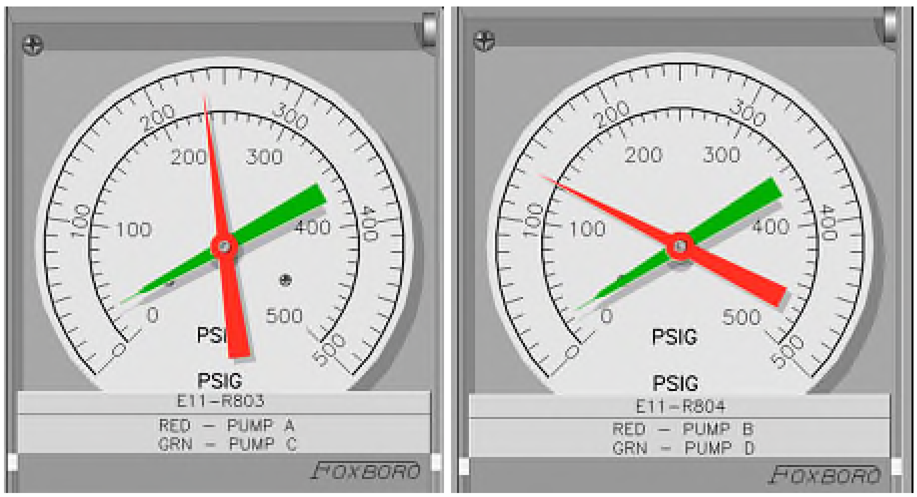
Which of the following indicates the change in RHR system pressure that will occur due to initiating Drywell Sprays with Division 1 RHR?

A.

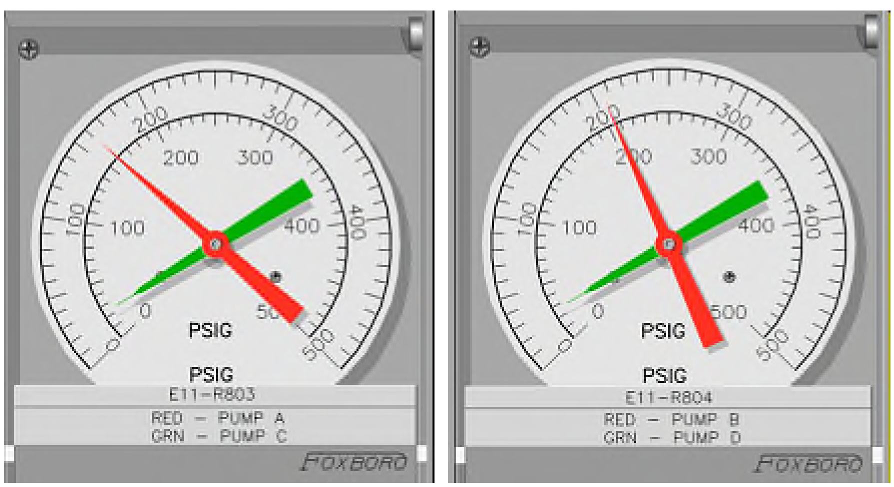
B.


PSIG
PSIG
E11−R803
RED − PUMP A
GRN − PUMP C
FOXBORO
PSIG
PSIG
E11−R804
RED − PUMP B
GRN − PUMP D
FOXBORO

C.


PSIG
E11−R803
RED − PUMP A
GRN − PUMP C
FOXBORO
PSIG
PSIG
E11−R804
RED − PUMP B
GRN − PUMP D
FOXBORO

D.


PSIG
PSIG
E11−R803
RED − PUMP A
GRN − PUMP C
FOXBORO
PSIG
PSIG
E11−R804
RED − PUMP B
GRN − PUMP D
FOXBORO

Answer: B

Answer Explanation:
Note: The information used to develop the graphics in this question was obtained from 29.ESP.01, Supplemental Information, Section 16.0, Pump Capacities Table, surveillance paperwork (such as 24.204.01 and 24.206.06, Div 1 and Div 2 LPCI and Suppression Pool Cooling/Spray Pump and Valve Operability Test) and the RHR system student text.

The RHR pumps develop 230 psig discharge pressure at 10,000 gpm flow. As pump flow increases, discharge pressure decreases. The starting conditions in the stem of the question were 12,500 gpm per pump, which results in a discharge pressure of ~205 psig.

With the E1150-F010 open, system flow (for any use) will be supported by both divisions of RHR. When an additional flow path is established by aligning Drywell Sprays with Division 1 RHR, because the E1150-F010 is open, both Division 1 and 2 pump discharge pressures will lower as pump pressure is converted to increased flow. The conditions shown in Answer B result from both pumps carrying ~17,500 gpm of flow at the shown discharge pressure of ~165 psig.

Distractor Explanation:
Distractors are incorrect and plausible because:

A. This distractor shows Pump A and B discharge pressures staying the same as in the initial conditions. This distractor is plausible because the candidate could fail to recall the relationship between increasing system flow, by opening another flow path through the Drywell Spray line, and pump discharge pressure. The candidate could incorrectly conclude that pump discharge pressure is a fixed value and will remain at the value shown in the stem (~205 psig) when Drywell Sprays is initiated.
C. This distractor shows Pump A discharge pressure increasing to ~235 psig. This indication was developed by closing the E1150-F010, RHR Crosstie Valve, and then placing Division 1 RHR in Drywell Sprays, by itself. This would result in 10,000 gpm flow at 235 psig. This distractor is plausible because the SRO has the option of closing the E1150-F010 to split out divisions of RHR to use one division for containment cooling while leaving the other division in the LPCI mode. If the candidate assumes that the E1150-F010 was closed, contrary to the information stated in the stem of the question, or if the candidate assumes that the E1150-F010 is always closed when spraying the drywell, which is not the case, then the candidate may recognize this discharge pressure shown as being consistent with one pump at 10,000 gpm (nominal drywell spray flow).
D. This distractor shows Pump B discharge pressure staying the same as in the initial conditions. This distractor is plausible because the candidate could fail to recall the relationship between increasing system flow, by opening another flow path through the Drywell Spray line, and pump discharge pressure. This distractor is also plausible because the candidate could assume that, since Division 1 RHR is being placed in Drywell Sprays, then division 2, or Pump B, would remain unaffected. Or, the candidate could have missed the fact that the E1150-F010 is open and incorrectly assume that Div 2 RHR would remain unaffected by Division 1 being placed in Drywell Sprays.

Reference Information:
E11-00, RHR Design Basis Document (DBD).
29.ESP.01, Supplemental Information, Section 16.0, Pump Capacities Table.
24.204.01, Div 1 LPCI and Suppression Pool Cooling/Spray Pump and Valve Operability Test.
24.204.06, Div 2 LPCI and Suppression Pool Cooling/Spray Pump and Valve Operability Test.

NUREG 1123 KA Catalog Rev. 2

226001 A1. Ability to predict and/or monitor changes in parameters associated with operating the RHR/LPCI: CONTAINMENT SPRAY SYSTEM MODE controls including:

226001 A1.07 3.1/3.1 System pressure

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 Question Use (ILO 2018)

High
NEW
RO

Associated objective(s):

Residual Heat Removal (E1100)

Cognitive Terminal

Given various controls and indications for RHR system operations, analyze that indication to determine proper plant response in accordance with approved plant procedures.

61	K/A Importance:			Points: 1.00
R61	Difficulty: 0.00	Level of Knowledge: Low	Source: BANK	75927

Select the response that correctly completes the following statement.

The refueling interlocks ensure that __(1)__ does not occur during fuel handling operations by preventing __(2)__ with the Mode Switch in the Refuel position.

A. (1) inadvertent criticality; (2) control rod withdrawal whenever loaded fuel equipment is over the core

B. (1) inadvertent criticality; (2) control rod withdrawal whenever fuel loading equipment is energized

C. (1) excessive iodine gas release; (2) refueling hoist movement when RPV Water Level is <20' 6" above the top of the RPV flange

D. (1) excessive iodine gas release; (2) withdrawl of any control rods when RPV Water Level is <20' 6" above the top of the RPV flange

Answer: A

Answer Explanation:
The refueling interlocks are explicitly assumed in the UFSAR SAFETY ANALYSES for the control rod removal error during refueling and the fuel assembly insertion error during refueling. These analyses evaluate the consequences of control rod withdrawal during refueling and also fuel assembly insertion with a control rod withdrawn. A prompt reactivity excursion during refueling could potentially result in fuel failure with subsequent release of radioactive material to the environment. Criticality and, therefore, subsequent prompt reactivity excursions are prevented during the insertion of fuel, provided all control rods are fully inserted during the fuel insertion. The refueling interlocks accomplish this by preventing loading of fuel into the core with any control rod withdrawn or by preventing withdrawal of a rod from the core during fuel loading. The refueling platform location switches activate at a point outside of the reactor core such that, considering switch hysteresis and maximum platform momentum toward the core at the time of power loss with a fuel assembly loaded and a control rod withdrawn, the fuel is not over the core.
Therefore, the candidate should recognize that the refuleing interlocks (1) prevent inadvertent criticality by (2) preventing control rod withdrawal whenever loaded fuel equipment is over the core.
Distractor Explanation:
Distractors are incorrect and plausible because:

B. The candidate could correctly recall that the refueling interlocks (1) prevent inadvertent criticality. However, the candidate could incorrectly recall HOW this interlock is imposed by the logic and (2) conclude that it monitors status of power to the fuel loading equipment.
C. TS LCO 3.9.6, RPV Water Level during Refueling Operations requires that, during 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, a minimum water level of 20 ft 6 inches above the top of the RPV flange be maintained to ensure 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. The candidate could incorrectly determine that this (1) TS requirement for RPV Water Level is also interlocked with (2) refueling hoist equipment when the Mode Switch is in the Refuel position, which is incorrect.
D. TS LCO 3.9.6, RPV Water Level during Refueling Operations requires that, during 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, a minimum water level of 20 ft 6 inches above the top of the RPV flange be maintained to ensure 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. The candidate could incorrectly determine that this (1) TS requirement for RPV Water Level is also interlocked with (2) control rod movement when the Mode Switch is in the Refuel position, which is incorrect.

Reference Information:
TS Bases 3.9.1, 3,9.6 (for Distractors C and D).

NUREG 1123 KA Catalog Rev. 2
234000 Fuel Handling
G2.1.28 4.1/4.1 Knowledge of the purpose and function of major system components and controls

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 Question Use (ILO 2018)
BANK
Low
RO

Associated objective(s):
Refueling
(F1500)
Cognitive Enabler
Discuss the function and purpose of Refueling System components, including their importance to nuclear safety.

Refueling
(F1500)
Cognitive Enabler
List the interlocks associated with Refueling System components.

Refueling
(F1500)
Cognitive Enabler
Discuss the function and purpose of Refueling System components, including their importance to nuclear safety.

THIS PAGE LEFT BLANK

62	K/A Importance: 2.7/2.7			**Points: 1.00**
R62	Difficulty: 3.00	Level of Knowledge: Low	Source: NEW	73526

A ground on the BOP DC electrical distribution system occurs and causes the following:

- 2PC3-17 Position 1 to TRIP
- 4D49, AVR 125V DC Supply Trouble to ALARM

What is the effect of this DC loss on the Main Turbine Generator?

A. DC voltage supplied to the Main Generator field will be lost.

B. The ability to regulate the Main Generator field will be disabled.

C. If a Main Turbine trip occurred, the generator field breaker would fail to trip.

D. The Main Turbine would coast down without an adequate lube oil supply if the AC Lube Oil pumps were lost.

Answer: B

Answer Explanation:
2PC3-17, Position 1 is the DC electrical distribution power supply to the Main Turbine Generator Excitation Control Cubicle. Loss of this source results in a loss of power to circuits Q80 and Q40. Circuit Q40 supplies power to the binary inputs for the Main Turbine Generator Voltage Regulator (AVR) which regulates the amount of DC voltage applied to the generator rotor for supplying and regulating the DC field and, therefore, DC output voltage or reactive load. Loss of these controls renders the voltage regulator inoperable so that automatic and manual voltage regulation is lost.

Distractor Explanation:
Distractors are incorrect and plausible because:

A. This distractor is plausible if the candidate thought that this DC circuit supplied voltage to the generator field. This is incorrect since generator field DC voltage comes from the output of the Power Convertor Thyristor Bank. This distractor is also plausible if the candidate confused this power supply loss with a loss of all of 2PC3-17, which would cause a loss of the generator field and a Main Generator trip.
C. This distractor is plausible if the candidate assumed that this circuit also powered the generator field breaker control circuit. This is incorrect because the Field Breaker DC Control circuit is powered from 2PC3-16.
D. This distractor is plausible if the candidate assumed that this circuit also powered the Main Turbine Emergency Oil Pump or its control circuit. This is incorrect because the Emergency Oil Pump is powered from 2PC-1, Pos 4A, which is the same power supply to the pump's DC control circuit.

Reference Information:
ARP 4D49, AVR 125V DC Supply Trouble
Student Text ST-OP-315-0055

NUREG 1123 KA Catalog Rev. 2
245000 Main Turbine Generator and Auxiliary System
245000 K1. Knowledge of the physical connections and/or causeeffect relationships between MAIN TURBINE GENERATOR AND AUXILIARY SYSTEMS and the following:
245000 K1.09 2.7/2.7 D. C . electrical distribution

10CFR55 RO/SRO Written Exam Content
10 CFR 55.41(b) (4) Secondary coolant and auxiliary systems that affect the facility.

NRC Question Use (ILO 2018)
Low
NEW
RO

Associated objective(s):
Main Generator and Excitation (N3031 & N3034)
Cognitive Enabler
Describe the normal and alternate power supplies to Main Generator and Excitation System components.

Main Generator and Excitation (N3031 & N3034)
Cognitive Enabler
Describe the normal and alternate power supplies to Main Generator and Excitation System components.

63	K/A Importance: 3.3/3.3			Points: 1.00
R63	Difficulty: 3.00	Level of Knowledge: High	Source: BANK	73546

The plant was operating at rated conditions when a STATION BLACKOUT occurred.
Shortly after the transient, the CRLNO reported that the Bus 1-2 red power available light was lit.

Which of the following describes the AVAILABILITY of power, and/or the QUICKEST method to restore power, to the SBFW system to allow injection to the RPV?

A. Power CAN NOT be restored to the SBFW pumps.

B. Power CAN be restored to the SBFW pumps by closing the A6 breaker for Transformer 64.

C. Power CAN be restored to the SBFW pumps by starting CTG 11 Unit 1 and restoring power to Transformer 64.

D. Power is CURRENTLY AVAILABLE to the SBFW pumps, no further actions are necessary to restore power for injection.

Answer: B

Answer Explanation:
The SBFW pumps are motor driven feedwater pumps that serve as a High-Pressure injection source of water to the RPV. The pumps are powered from 4160V Busses 64V (East, or A) and 65W (West, or B). Busses 64V and 65W can be cross-tied such that both pumps can be powered from either the Division 1 (120 kV) or Division 2 (345kV) offsite circuits.

The conditions given in the stem of the question indicate that at least one 120kV offsite circuit is available and currently powering Bus 1-2, as indicated by the red Bus 1-2 red power available light being lit. With power available at Bus 1-2, the operator can readily power transformer 64 by closing in the A6 breaker. Powering transformer 64 restores power to bus 64V, which restores power to the SBFW pumps.

Note: At Fermi 2, the Station Blackout procedure has a unique situation that differentiates it from other stations. At Fermi 2, it is possible to be in the Station Blackout procedure but still have power available at Bus 1-2. If this is the case, power can be restored to the SBFW pumps as specified in the correct answer.

Distractor Explanation:
Distractors are incorrect and plausible because:

A. This distractor is plausible if the candidate fails to correctly recall the flow of power from Division 1 (120kV) offsite power, through the distribution system, to transformer 64 and eventually busses 64V and 65W. The information in the stem of the question could lead the candidate to believe that power is not available, and therefore cannot be restored, to the SBFW pump motors.
C. This distractor is plausible if the candidate fails to correctly recall the flow of power from Division 1 (120kV) offsite power, through the distribution system, to transformer 64 and eventually busses 64V and 65W. Starting CTG 11-1 is an alternative to closing in the A6 position. However, starting CTG 11-1 is not necessary because Bus 1-2 is already energized, as stated in the stem of the question.
D. This distractor is plausible if the candidate fails to correctly recall the flow of power from Division 1 (120kV) offsite power, through the distribution system, to transformer 64 and eventually busses 64V and 65W. The information in the stem of the question could lead the candidate to believe that, with Bus 1-2 energized, power is already available to the SBFW pumps. This is not true, however, without first closing the A6 breaker.

Reference Information:
SD-2500-01, One Line Diagram Plant 4160V & 480V System Service
Student Text ST-OP-315-0018, SBFW System

NUREG 1123 KA Catalog Rev. 2
259001 Reactor Feedwater System
259001 K2. Knowledge of electrical power supplies to the following:
259001 K2.01 Reactor feedwater pump(s): Motor-Driven-Only

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 Question Use (ILO 2018)
BANK
High
RO

Associated objective(s):
Standby Feedwater (N2103)
Cognitive Enabler
Describe the normal and alternate power supplies to Standby Feedwater System components.

Standby Feedwater (N2103)
Cognitive Enabler
Describe the normal and alternate power supplies to Standby Feedwater System components.

64	K/A Importance: 2.9/3.3			Points: 1.00
R64	Difficulty: 4.00	Level of Knowledge: High	Source: NEW	73806

The plant is operating at 100% power when the following sequence of events takes place:

- 0700 Indications of a fire in the Cable Spreading Room are received in the Main Control Room. As a result, the following radiation monitors fail downscale at the times specified:

- 0730 K11-K609A, Div 1 Fuel Pool East Vent Exh Duct Rad Monitor.
- 0800 K11-K609C, Div 1 Fuel Pool West Vent Exh Duct Rad Monitor.
- 0810 K11-K609B, Div 2 Fuel Pool East Vent Exh Duct Rad Monitor.
- 0830 K11-K609D, Div 2 Fuel Pool West Vent Exh Duct Rad Monitor.

Which of the following identifies the EARLIEST TIME that CCHVAC will shift to the Recirculation Mode?

A. 0730

B. 0800

C. 0810

D. 0830

Answer: C

Answer Explanation:
Per ARP 3D27, A downscale failure of channels A and B or C and D will cause CCHVAC to shift to the Recirculation Mode. The earliest time that this logic is satisfied is at 0810 when the A and B (and C, although it is not needed to satisfy the logic) instruments have failed.

Distractor Explanation:
Distractors are incorrect and plausible because:

A. The candidate could incorrectly recall the logic for secondary containment isolation on a Fuel Pool radiation monitor failure. CCHVAC will also shift to Recirc if just ONE Fuel Pool rad monitor goes UPSCALE. If the candidate assumed that the downscale logic was the same, then this distractor could be chosen.
B. The candidate could incorrectly recall the logic for secondary containment isolation on a Fuel Pool radiation monitor failure. The candidate could assume that the logic was a downscale failure of channels A and C or B and D. Since both A and C instruments are Division 1 instruments, this line of thinking is plausible so this distractor could be chosen.
D. The candidate could incorrectly recall the logic for secondary containment isolation on a Fuel Pool radiation monitor failure. The candidate could assume that the logic required a downscale failure of ALL Fuel Pool radiation monitors to shift CCHVAC to Recirc. This line of thinking is plausible because the logic does require BOTH rad monitors in the same duct (East or West) to fail downscale before the logic actuates. The candidate could choose this distractor by carrying this line of thinking to the entire system (all 4 rad monitors) rather than just one duct.

Reference Information:
ST-OP-315-0050, Process Radiation Monitoring System Student Text.
ARP 3D27, FP Vent Exh Rad Monitor Downscale/INOP

NUREG 1123 KA Catalog Rev. 2
272000 Radiation Monitoring System
272000 K3. Knowledge of the effect that a loss or malfunction of the RADIATION MONITORING System will have on following:
272000 K3.10 Control room ventilation: Plant-Specific

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 Question Use (ILO 2018)
High
NEW
RO

Associated objective(s):
Process Radiation Monitoring System (D1100)
Cognitive Enabler
Discuss the Process Radiation Monitoring System interrelationships with other systems.

65	K/A Importance: 3.5/3.8			**Points: 1.00**
R65	Difficulty: 2.00	Level of Knowledge: Low	Source: NEW	74986

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

- You are the CRLNO.
- Movement of material is taking place from the Reactor Building into the Railroad Airlock.
- 8D24, Rail Airlock Inner Door Seal Pressure Low is IN ALARM.
- 17D43, Rail Airlock Outer Door Seal Pressure Low is CLEAR.

A contractor calls you on the radio and says that he is in the Airlock going from the Reactor Building Fifth Floor to the Auxiliary Building and CANNOT open the Auxiliary Building airlock door.

What is the cause for his not being able to exit into the Auxiliary Building?

A. Power to the door interlock is de-energized.

B. The door from the Reactor Building is not closed.

C. The seals for the Reactor Building Door are not pressurized.

D. The seals for the Auxiliary Building Door are not depressurized.

Answer: B

Answer Explanation:
The correct answer describes the airlock and door arrangement that someone would use to access AB-5, from RB-5, as asked in the stem of the question. The airlock consists of two airtight doors separated by a vestibule. Electrical interlocks prevent both doors from being simultaneously open. Entry to and from Secondary Containment, through this airlock, is accomplished without uncontrolled release of Secondary Containment atmosphere to the environment.
The candidate should recognize this type of door/interlock arrangement and determine that the RB door may not be closed, thereby preventing operation of the AB door.

Distractor Explanation:
Distractors are incorrect and plausible because:

A. This distractor is plausible if the candidate recalled that the door interlocks were in effect if power to them was lost, which is incorrect. If de-energized, the interlocks are defeated meaning both doors can be open at the same time, as can be seen by Precaution and Limitation 3.2 in 23.428, Secondary Containment Airlocks and Penetrations.
C. This distractor is plausible if the candidate confused the door in the stem of the question with the type of airlock used for equipment access to and/or from the Reactor Building via the Rail Airlock doors. This door type is equipped with inflatable seals that prevent the free movement of air in/out of the Reactor Building. It is plausible that the candidate could confuse 8D24 being in alarm with the door in the stem of the question and determine that the reason the outer (Auxiliary Building) door cannot be opened is because the inner (Reactor Building) door seal is depressurized.
D. This distractor is plausible if the candidate confused the door in the stem of the question with the type of airlock used for equipment access to and/or from the Reactor Building via the Rail Airlock doors. This door type is equipped with inflatable seals that prevent the free movement of air in/out of the Reactor Building. It is plausible that the candidate could confuse 17D43 being clear with the door in the stem of the question and determine that the reason the outer (Auxiliary Building) door cannot be opened is because the Auxiliary Building door seal is still pressurized.

Reference Information:
23.428, Secondary Containment Airlocks and Penetrations
Student Text ST-OP-315-0016, Containment Systems

NUREG 1123 KA Catalog Rev. 2
290001 Secondary Containment
290001 K4. Knowledge of SECONDARY CONTAINMENT design feature(s) and/or interlocks which provide for the following:
290001 K4.01 3.5/3.8 Personnel access without breaching secondary containment: Plant-System

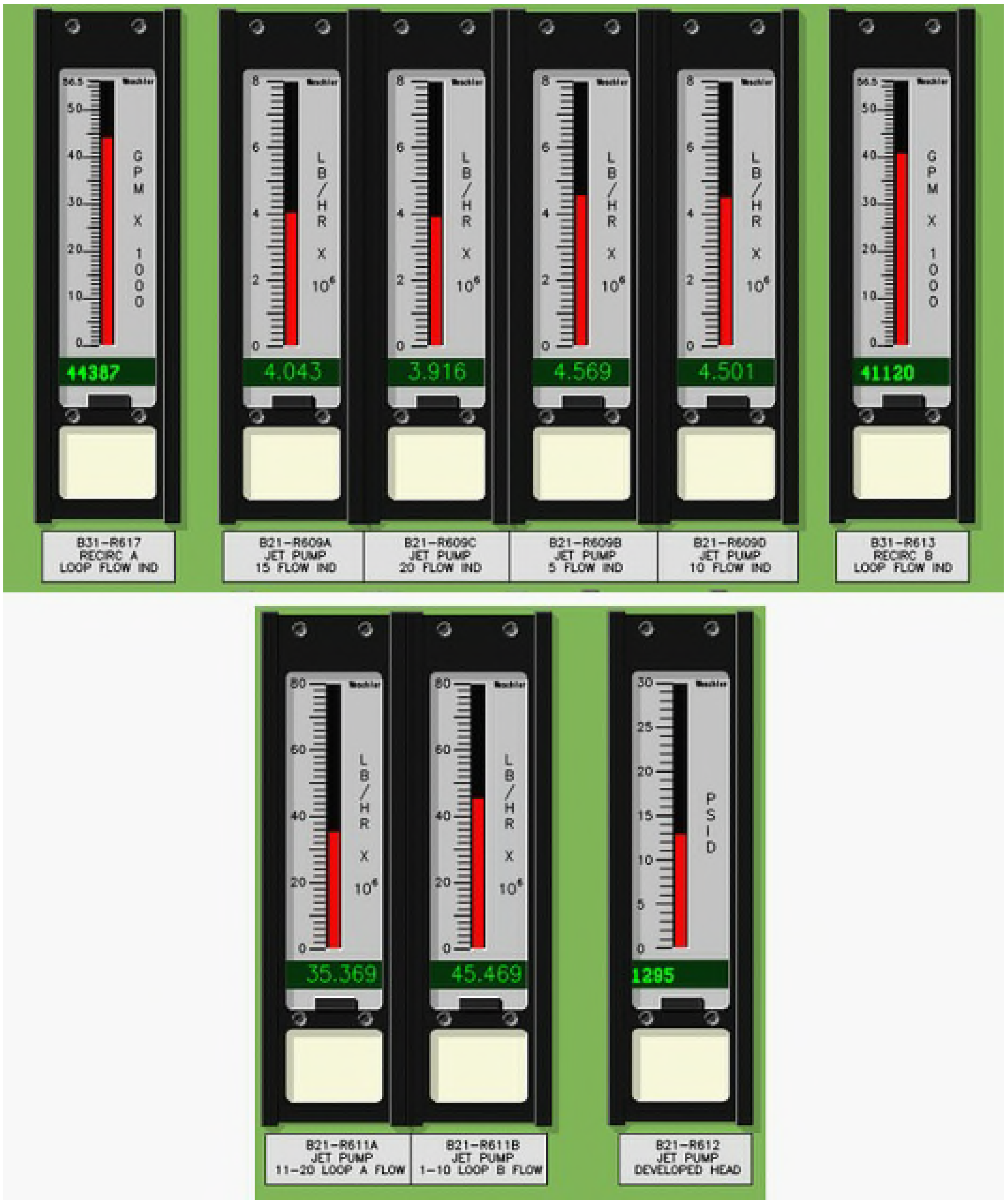
10CFR55 RO/SRO Written Exam Content
10 CFR 55.41(b) (7) Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

NRC Question Use (ILO 2018)
Low
NEW
RO

Associated objective(s):
Containment Systems (T2200 & T2300)
Cognitive Terminal
Given various controls and indications for Containment Systems operations, analyze that indication to determine proper plant response in accordance with approved plant procedures.

66	K/A Importance: 4.4/4.7			Points: 1.00
R66	Difficulty: 3.00	Level of Knowledge: High	Source: MODIFIED	73829

The plant was operating at 100% power with Jet Pump Total Flow of 91.7 Mlbm/hr when reactor power lowered. The following flow indications were observed:



Based ONLY on the indications above, what event has occurred?

A. Jet Pump Failure on Loop A

B. Jet Pump Failure on Loop B

C. Uncontrolled Recirc Flow change of 5% on A Reactor Recirc Pump (lowering)

D. Uncontrolled Recirc Flow change of 5% on B Reactor Recirc Pump (rising)

Answer: A

Answer Explanation:
20.138.02 Jet Pump Failure Symptoms:

- Unexplained change in indicated Core Flow
- Unexplained change in Recirc Loop Flow
- Unexplained decrease in Core D/P
- Jet Pump Percent Differential Pressure deviates excessively from the average of the remaining Jet Pump Percent Differential Pressures

The increase in Recirc Loop Flow in the A Loop compared to B Loop (B31-R617 vs B31-R613) and the decrease in Jet Pump Flow in the A Loop vs the B loop (B21-R611A vs B21R611B) indicates that a break in a Jet Pump in the A Loop has occurred, which has caused an increase in Recirc Loop Flow (due to a reduction in back pressure on the RR A RR Pump) and a decrease in Jet Pump Flow (due to flow bypassing the jet pumps).

Distracter Explanation:

B. This distractor is plausible because the candidates could misread the Jet Pump Flow indications and apply the deviation in Jet Pump Flows to the B Loop, vice the A Loop, escpecially since, at Fermi 2, Jet Pumps 1 through 10 are in the A Loop and Jet Pumps 11 through 20 are in the B Loop. This answer is incorrect because the difference between driving (B31-R617) and driven (B21-R611) flow in the A loop indicates that a break has occurred in the A loop. B loop Jetpump flows as indicated by B21-R609B, B21-R609D and B21-R611B all indicate consistent flows and do not provide indication of a Jet Pump failure in the B Loop.

C. This distractor is plausible because Jet Pump Flow on the A loop has decreased and is about 10% lower than the B loop (B21-R611A vs B21R611B), which could lead the candidate to conclude that A Loop Recirc Flow has lowered. This is incorrect, however, because the imbalance between Jet Pump Flows and Recirc Loop Flows as well as the deviation between Calibrated Jet Pump Flows (B21-R609A & C vs B21-R611A) indicate that a Jet Pump Failure has occurred, and not a Recirc Flow decrease.

D. This distractor is plausible because Jet Pump Flow on the B loop is about 10% higher than Jet Pump Flow on the A Loop (B21-R611A vs B21R611B), which could lead the candidate to conclude that B Loop Recirc Flow has increased. This is incorrect, however, because the stem of the question states that reactor power has lowered, which would not occur on a Recirc Flow increase. Also, the imbalance between Jet Pump Flows and Recirc Loop Flows in the A Loop, as well as the deviation between Calibrated Jet Pump Flows (B21-R609A & C vs B21-R611A) indicate that a Jet Pump Failure has occurred, and not a Recirc Flow increase.

Reference Information:

AOP 20.138.02 Jet Pump Failure (pg 5) Jet Pump Failure Symptoms

Plant Procedures
23.138.02

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

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 2015 Exam

NRC Question Use (ILO 2018)
High
MODIFIED
RO

Associated objective(s):
Cycle 15-3 Objectives
Cognitive Terminal
Given a copy of 20.138.02, discuss licensed operator actions for Jet Pump Failure

67	K/A Importance: 4.6/4.3			Points: 1.00
R67	Difficulty: 4.00	Level of Knowledge: High	Source: NEW	74987

NOTE: Use the SBFW System Mimic handout to answer this question.

The plant is at 100% power in MODE 1. The East SBFW Pump was just shut down following the performance of 24.107.03, SBFW Pump and Valve Operability and Lineup Verification Test.

You have been tasked with verifying that the system is in the standby lineup.

Which of the responses below accurately completes the following statement to reflect the standby availability of the SBFW system?

If the CMC Switches for the SBFW Pumps are taken to RUN, __(1)__ will start and __(2)__ inject into the RPV.

A. (1) ONLY the West
(2) WILL

B. (1) BOTH
(2) WILL

C. (1) BOTH
(2) NOT

D. (1) ONLY the East
(2) NOT

Answer: D

Answer Explanation:
Following completion of the flow verification test, the SBFW Auxiliary Lube Oil pump is left running for 5 minutes. During this time, however, the Mode Selection switch should be in NORM. With the switch in TEST, as shown in the picture of the system mimic, starting logic for the SBFW pumps and injection valve is impacted. When the Mode Selection switch is in TEST, the N2103-F001, SBFW Disch to RPV Iso Valve, will NOT open. Normally, this valve would open automatically when the first pump is started. Additionally, when the Mode Selection switch is in TEST, the SBFW pumps are prevented from running unless an oil pressure permissive is met by running the pump's Auxiliary Lube Oil Pump.

With this information, the candidate should recognize that ONLY the East SBFW Pump will run when it's CMC is taken to RUN. The candidate should also recognize that the N2103-F001 will not open preventing injection into the RPV when in TEST.
Note: The Mode Selection switch being in TEST does NOT generate any alarms in the Main Control Room to inform the operator of the switch being out of position, so it is important for the candidate to recognize this switch being out of its normal standby position and to understand the switch's impact on system logic.

Distractor Explanation:
Distractors are incorrect and plausible because:

A. The candidate could incorrectly apply precaution 3.2 of the SBFW System SOP, which states that "When the SBFW System is aligned in the Test mode, only one pump is to be tested at a time". The candidate could assume that this precaution describes logic that is enforced by the Mode Selection switch being in TEST that would only allow one pump to start when in TEST, which is not correct. The candidate could also fail to recall the impact of the Mode Selection switch on the logic for the N2103-F001 valve or fail to recognize that the switch is in TEST.
B. The candidate could fail to recognize that the Mode Selection switch is in TEST, which would lead the candidate to believe that both pumps will start and the N2103-F001 will open on pump start. Also, the candidate could recognize the switch being in TEST but fail to recall the impact of this on the pump start logic and on the logic for the N2103-F001.
C. The candidate could recognize that the Mode Selection Switch is in the TEST position, but fail to recall the impact of this on pump start logic and conclude that both pumps will start when their CMCs are taken to RUN, which is not correct. The candidate could correctly recall the impact of the switch being in TEST on the logic for the N2103-F001.

Reference Information:
23.107.01, SBFW System
24.107.03, SBFW Pump and Valve Operability and Lineup Verification Test
ST-OP-315-0018, Standby Feedwater System Student Text, Table 4 SBFW Control Functions and Interlocks

NUREG 1123 KA Catalog Rev. 2
G2.1.31 Ability to locate control room switches, controls and indications and to determine that they are correctly reflecting the desired plant lineup

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

NRC Question Use (ILO 2018)
High
NEW
RO

Associated objective(s):
Standby Feedwater (N2103)
Cognitive Terminal
Given various controls and indications for SBFW operations, analyze that indication to determine proper plant response in accordance with approved plant procedures.

68	K/A Importance: 4.3/4.3			**Points: 1.00**
R68	Difficulty: 2.00	Level of Knowledge: Low	Source: NEW	73847

While operating at power, a reactor scram occurs.
Following the scram, alarm 2D28, HPCI System Actuated, is received.

With the RPV still pressurized to 1000 psig, which of the following RPV level indications should you use to validate that the HPCI system is responding to an actual plant condition versus a spurious actuation?

A. B21-R605, RPV Flood Up Level Indicator, reading 160 inches.

B. C32-R606A(B), Reactor Level A(B) Indicator(s), reading 160 inches.

C. B21-R604A(B), Div 1(2) Reactor Water Level Indicator(s), reading 100 inches.

D. B21-R610(615), Div 1(2) RPV Core Level Recorder(s), reading 100 inches.

Answer: C

Answer Explanation:
The HPCI system is actuated on a low RPV water level condition when RPV level drops less than 110.8" as sensed by the Wide Range RPV Level instruments. Per ARP 2D28, HPCI System Actuated, this is verified by observing the B21-R604A(B) Div 1(2) Reactor Water Level Indicators showing less than 110.8". These indicators are driven by Wide Range level instruments. The candidate should recognize that the indicators and value listed in the correct answer are driven by Wide Range instruments and are therefore indicators than can be used to validate proper response of the instruments that input into HPCI initiation logic.

Distractor Explanation:
Distractors are incorrect and plausible because:

A. The B21-R605, Flood Up Level indicator is low off scale and the candidate could interpret this as being a valid indication to use to verify HPCI system response. However, the lower end of the indicating range for this instrument is 160", so using it to validate an actuation at 110.8" is not acceptable. Additionally, this instrument is not related to the Wide Range level instruments that input into HPCI initiation logic and is not listed in ARP 2D28.
B. The C32-R606A(B), Reactor Level A(B) Indicators are low off scale and the candidate could interpret this as being a valid indication to use to verify HPCI system response. However, the lower end of the indicating range for these instruments is 160", so using them to validate an actuation at 110.8" is not acceptable. Additionally, these instruments are not related to the Wide Range level instruments that input into HPCI initiation logic and they are not listed in ARP 2D28.
D. The B21-R610(615), Div 1(2) RPV Core Level Recorder is reading below the initiation setpoint for the HPCI logic. However, the Core Level instruments are calibrated for a usable range of -150 to +50 inches. Although the recorder will display information above +50", it should not be used for validating HPCI system response. Additionally, these instruments are not related to the Wide Range level instruments that input into HPCI initiation logic and are not listed in ARP 2D28.

Reference Information:
ARP 2D28, HPCI System Actuated.
ST-OP-315-0021, RPV Instrumentation Student Text, Page 17 Monitoring Instrumentation.

NUREG 1123 KA Catalog Rev. 2
G2.1.45 4.3/4.3 Ability to identify and interpret diverse indications to validate the response of another indication

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 Question Use (ILO 2018)
Low
NEW
RO

Associated objective(s):
Reactor Pressure Vessel Instrumentation (B2100)
Cognitive Enabler
Discuss effective monitoring of Reactor Pressure Vessel Instrumentation using local, remote, computer displays and alarms.

69	K/A Importance: 3.7/4.1			**Points: 1.00**
R69V2	Difficulty: 2.00	Level of Knowledge: Low	Source: NEW	75106

A Licensed Nuclear Operator (LNO) is attempting to perform an Operations department Technical Specification surveillance procedure. He has encountered a non-conditional step that cannot be performed under the current plant conditions and the LNO believes that all the requirements of MGA03, Procedure Use and Adherence, are met to use Not Applicable (N/A) for this step.

Who can approve marking the step N/A?

A. Shift Manager

B. Operations Engineer

C. Field Support Supervisor

D. Control Room Supervisor

Answer: A

Answer Explanation:
MGA03, step 5.1.10.3.c states that the Shift Manager must approve the use of N/A for all non-conditional steps for all Operations procedures and Technical Specification and Technical Requirements Manual Surveillances.

Therefore, the candidate must determine that Shift Manager approval is required in this instance since the individiual is an LNO who is attempting to perform an Operations department Tech Spec surveillance.

Distractor Explanation:
B. Is plausible if the candidate believes that a higher level of approval, beyond the Shift Manager, is required to waive non-conditional steps in Tech Spec related surveillances. This is plausible because other Operations Department administrative requirements, such as the approval required to not perform system lineups in 22.000.01, Plant Startup Master Checklist, does require the Operations Engineer to N/A steps.
C. Is plausible because the use of N/A in non-Operations department procedures can be approved by the individual's immediate supervisor which, for LNOs assigned to the Research Tagging Center, is the Field Support Supervisor.
D. Is plausible because the use of N/A in non-Operations department procedures can be approved by the individual's immediate supervisor which, for LNOs assigned to the Main Control Room, is the Control Room Supervisor.

Reference Information:
MAG03, Procedure Use and Adherence (5.1.10)
22.000.01, Plant Startup Master Checklist

NUREG 1123 KA Catalog Rev. 2
G2.2.12 Knowledge of surveillance procedures

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

NRC Question Use (ILO 2018)
Low
NEW
RO

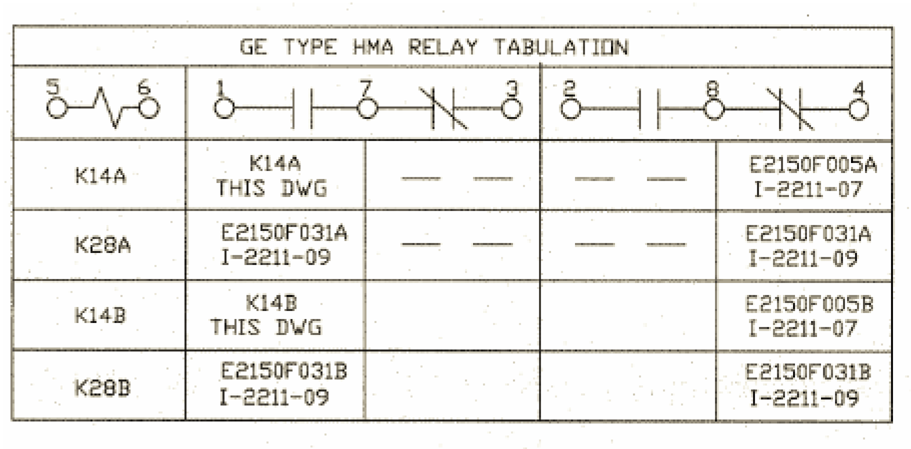
Associated objective(s):
Administrative Conduct Manuals - Licensed Operators/STA
Performance Terminal
Describe when partial revisions may be made to procedures and any requirements that must be met.

Administrative Conduct Manuals - Licensed Operators/STA
Cognitive Terminal
Discuss the administrative requirements and responsibilities of the Licensed Operator role

Administrative Conduct Manuals - Licensed Operators/STA
Performance Terminal
Describe the differences between Continuous Use and Reference Use procedures.

70	K/A Importance: 3.9/4.3			**Points: 1.00**
R70	Difficulty: 2.50	Level of Knowledge: Low	Source: NEW	75866

You are reviewing an electrical print. The review takes you to the relay tabulation shown below:



GE TYPE HMA RELAY TABULATION				
5 6	1 7	3	2 8	4
K14A	K14A THIS DWG	— —	— —	E2150F005A I-2211-07
K28A	E2150F031A I-2211-09	— —	— —	E2150F031A I-2211-09
K14B	K14B THIS DWG			E2150F005B I-2211-07
K28B	E2150F031B I-2211-09			E2150F031B I-2211-09

What is physically located between Terminals 5-6 and what is the state of the component located between Terminals 1-7 if component 5-6 is ENERGIZED?

A. (1) Relay coil; (2) Open

B. (1) Relay contact; (2) Closed

C. (1) Relay coil; (2) Closed

D. (1) Relay contact; (2) Open

Answer: C

Answer Explanation:
From the I&C Print Reading student text: "The most common symbol in a control schematic is a relay. Its symbol is shown in Figure 3 [not provided, but the symbol is shown at the top left of the relay tabulation between terminals 5-6] with some representative contacts. Relays and their contacts are shown in their shelf state (de-energized). When the relay is energized, the contacts switch to the opposite state."

The candidate should determine that the K14A component located between Terminals 5-6 is the relay's coil. The relays shown in the tabulation given in the stem of the question are all the GE HMA relay type. When this type of relay, or specifically relay K14A, is in its shelf (de-energized) state, the contacts are as shown in the tabulation. The contacts across relay terminals 1-7 and 2-8 are open and the contacts across terminals 7-3 and 8-4 are closed. If the relay were to become energized, as stated in the stem of the question, the contacts would change state thus causing the contacts across relay terminals 1-7 to close.

Distractor Explanation:
Distractors are incorrect and plausible because:
A. The candidate could correctly determine that the relay's coil is between Terminals 5-6. The candidate could then incorrectly determine that relay on this relay tabulation are shown in the energized state and conclude that contact 1-7 would be open when the relay is energized. This is incorrect because relays are shown in their de-energized state in the relay tabulation.
B. The candidate could incorrectly determine that a relay contct is shown between Terminals 5-6 instead of the relay coil.
D. The candidate could incorrectly determine that a relay contct is shown between Terminals 5-6 instead of the relay coil. The candidate could then incorrectly determine that relay on this relay tabulation are shown in the energized state and conclude that contact 1-7 would be open when the relay is energized. This is incorrect because relays are shown in their de-energized state in the relay tabulation.

Reference Information:
I-2215-02, Grid F-8, GE Type HMA Relay Tabulation
ST-GN-175-0003-001, Print Reading Operations - I&C.

Objective Information:
ST-GN-175-0003-001, Print Reading Operations - I&C
Objective 0008 Describe the standard method by which relays, contacts, and switches are electrically pictured on schematic drawings.
Objective 0009 Use a relay tabulation to determine the function of each contact on a given relay.

NUREG 1123 KA Catalog Rev. 2
G2.2.15 3.9/4.3 Ability to determine the expected plant configuration using design and configuration control documentation, such as drawings, line-ups, tag-outs, etc.

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

NRC Question Use (ILO 2018)
Low
NEW
RO

Associated objective(s):

71	K/A Importance: 3.9/4.5			**Points: 1.00**
R71	Difficulty: 2.00	Level of Knowledge: Low	Source: BANK	73887

The Plant is in Mode 5 with movement of recently irradiated fuel in progress on the Refueling Floor. The following alarms are received:

- 8D46, DIV I REACTOR BLDG PRESSURE HIGH/LOW
- 17D46, DIV II REACTOR BLDG PRESSURE HIGH/LOW

Reactor Building (RB) Pressure is - 0.120 (minus 0.120) inches WC.

What Technical Specification action MUST be taken?

A. Suspend the fuel movement immediately.

B. Verify RBHVAC supply and exhaust fans operating within 1 hour.

C. Verify at least one door in each RB access is closed within 1 hour.

D. Start both divisions of Standby Gas Treatment System immediately.

Answer: A

Answer Explanation:
The stem of the question shows RB pressure high (not enough vacuum) and outside of the allowable range for LCO 3.6.4.1, which requires ≥ 0.125 inch of vacuum water gauge.
The actions taken are verbatim from Condition D of the LCO and the Immediate Action is required RO level of knowledge.

Distractor Explanation:
Distractors are incorrect and plausible because:

B. The candidate could recognize the need to verify proper operation of the RBHVAC system and lose site of the Immediate TS action. The action specified is also an action required by the ARP, however it is not a Tech Spec required action. The 1 hour time limit given is consistent with the knowledge level required of RO candidates.
C. The candidate could recognize the need to verify that each one door in each RB access is closed, per the ARP, and lose site of the Immediate TS action. Although the action specified is an action required by the ARP, it is not a Tech Spec required action. The 1 hour time limit given is consistent with the knowledge level required of RO candidates.
D. The candidate could recognize starting SGTS as an action of the ARP and a viable response to high RB pressure and lose site of the Immediate TS action. Although the action specified is an action required by the ARP, it would only be required if RBHVAC was not operating (more information needed than provided in the stem of the question) and is not a Tech Spec required action. The immediate time limit given is consistent with the knowledge level required of RO candidates.

Reference Information:
Technical Specifications 3.6.4.1
ARPs 8D46 (17D46) Div I (II) Reactor Building Pressure High/Low

NUREG 1123 KA Catalog Rev. 2

G2.2.39	Knowledge of less than one hour technical specification action statements for systems.

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 Question Use (ILO 2018)
BANK
Low
RO

Associated objective(s):
Containment Systems (T2200 & T2300)

Cognitive Enabler
Identify Containment Systems related technical specifications, with emphasis on action statements requiring prompt actions (for example, one hour or less).

72	K/A Importance: 3.2/3.7			Points: 1.00
R72	Difficulty: 4.00	Level of Knowledge: High	Source: NEW	73906

The plant is operating at 100% power with Division 2 CCHVAC in service. A DBA LOCA occurs which results in an offsite radiation release and the following:

- CCHVAC has automatically shifted to the Recirculation mode.
- The Division 2 CCHVAC Emergency Makeup Fan is NOT running.
- Neither the North nor the South Emergency Air intakes has received a Hi-Hi radiation trip signal.
- The Division 1 and Division 2 Emergency Air Intake Selector switches are in AUTO.

An operator sent to the Relay Room reports the following:

- D11-K836A, Div 1 S CCHVAC Emerg Air Inlet Rad Monitor is reading 48,000 cpm.
- D11-K837A, Div 1 N CCHVAC Emerg Air Inlet Rad Monitor is reading 33,000 cpm.
- D11-K836B, Div 2 S CCHVAC Emerg Air Inlet Rad Monitor is reading 45,000 cpm.
- D11-K837B, Div 2 N CCHVAC Emerg Air Inlet Rad Monitor is reading 31,000 cpm.

What action is required?

A. Start Division 1 and shut down Division 2 CCHVAC.

B. Shut down the Division 1 CCHVAC Emergency Makeup Fan.

C. Place the Emergency Air Intake Selector switches in NORTH.

D. Maintain the Emergency Air Intake Selector switches in AUTO.

Answer: C

Answer Explanation:
When in the Recirc Mode with the Emergency Air Intake Selector switch in AUTO, the Control Center Emergency Air Inlet Radiation Monitor system logic will auto-isolate one air inlet path (N/S) which has a Hi-Hi trip following a five-minute sampling period (in effect for selecting the air inlet with the lowest radiation level). Dampers on the tripped side will be closed for one hour. Both sides will then reopen for five minutes while inlet air is resampled. Inlet selection based on trip/no-trip will then be repeated. If no Hi-Hi trip is present, both inlets will remain open since there is no need (or signal present) to isolate one side or the other.

However, per 23.413 Section 7.3 as well as ARPs 3D38 and 3D39, with CCHVAC shifted to the Recirc Mode, and with an offsite radiation release in progress, the operator should place the Emergency Air Intake Selector switches to either NORTH or SOUTH, selecting the intake with lowest indicated radiation level.

In this instance, the operator should select NORTH based on the radiation levels given in the stem of the question.

Note: This concept is significant because it is a time critical operator action that must be performed within 30 minutes (see M-5988).

Distractor Explanation:

A. The candidate could recognize that both Division 1 radiation monitors are reading higher than both Division 2 radiation monitors. The candidate could also recognize that only the Division 1 CCHVAC Emergency Makeup Fan is running. Either of these could lead the candidate to determine that CCHVAC needs realigning to Division 1 in service; however, plant conditions do not dictate shifting divisions of CCHVAC.
B. The candidate could incorrectly determine that the CCHVAC Emergency Makeup Fan in the non-operating division (Division 1) needs to be shut down. This is plausible because, when CCHVAC shifts to the Recirc Mode, both Emergency Makeup Fans start and, if both fans are running, the normal course of action is to shut down the fan in the non-operating division. However, per 23.413 Section 7.3, the operator should leave the one makeup fan (Div 1) running, even if it is in the non-operating division.
D. The candidate could incorrectly recall the actions that need to be taken if neither CCHVAC Emergency Makeup intake receives a Hi-Hi radiation signal when a release is in progress. This is plausible since neither CCHVAC Emergency Makeup intake radiation monitor is above the Hi-Hi trip setpoint, and in the case the logic would keep both inlets open, which could lead the candidate to believe that the intake selector switch needs to stay in AUTO. However, this is incorrect as per 23.413 and ARPs 3D38 and 3D39, which both require the operator to select the intake with the lower reading.

Reference Information:
23.413 CCHVAC System SOP, Section 7.3
ARPs 3D38(39) CCHVAC Emerg Air Div I(II) Rad Mon Trouble
M-5988 Operator Time Critical Actions and Design Basis Sheet 1

NUREG 1123 KA Catalog Rev. 2
290003 Control Room HVAC
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 Question Use (ILO 2018)
High
NEW
RO

Associated objective(s):

Control Center HVAC (T4102)

Cognitive Enabler

List the interlocks associated with Control Center HVAC system components.

Reactor Operator

Performance Enabler

Perform proper system operations in accordance with System Operating Procedures (SOP).

THIS PAGE LEFT BLANK

73	K/A Importance: 3.4/3.8			Points: 1.00
R73	Difficulty: 3.00	Level of Knowledge: Low	Source: NEW	73926

The plant is in an ATWS condition and you are performing 29.ESP.03, Alternate Control Rod Insertion Methods. You are evaluating the following procedure sections:

- Section 3.0, Manual Control Rod Insertion.
- Section 4.0, Scram Reset and Manual Scram Re-initiation.
- Section 7.0, Vent Scram Air Header.
- Section 8.0, Vent CRD Over Piston Volumes.

Per 29.ESP.03, which of these sections, if performed, may result in the release of pressurized radioactive steam or water from the RPV?

A. Sections 3.0 and 7.0.

B. Sections 7.0 and 8.0.

C. Sections 3.0 and 4.0.

D. Sections 4.0 and 8.0.

Answer: D

Answer Explanation:
29.ESP.03, Sections 4.0 and 8.0 contain cautions informing the operator that the performance of the steps associated with these sections may result in the release of pressurized radioactive steam or water from the RPV. No other sections of 29.ESP.03 contain this caution.

Distractor Explanation:

Note: 29.ESP.03, Sections 3.0, 7.0, and 8.0 all contain Cautions warning the operator of the potential for higher than normal radiation levels that may be experienced during their performance.

Distractors are incorrect and plausible because:

A. The candidate could recall the caution regarding higher than normal radiation levels from Sections 3.0 and 7.0 and assume that the reason these sections contain this caution is because they could result in the release of radioactive steam or water from the RPV, which is incorrect.
B. The candidate could recall the caution regarding higher than normal radiation levels from Section 7.0 and assume that the reason this section contains this caution is because it could result in the release of radioactive steam or water from the RPV, which is incorrect. Section 8.0 is one of the sections that has the potential for releasing radioactive steam or water from the RPV.
C. The candidate could recall the caution regarding higher than normal radiation levels from Section 3.0 and assume that the reason this section contains this caution is because it could result in the release of radioactive steam or water from the RPV, which is incorrect. Section 4.0 is one of the sections that has the potential for releasing radioactive steam or water from the RPV.

Reference Information:
29.ESP.03, Alternate Control Rod Insertion Methods.
ST-OP-802-3006-001, Emergency Support Procedures Student Text.

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 Question Use (ILO 2018)
Low
NEW
RO

Associated objective(s):
Emergency Support Procedures
Performance Terminal
Describe the methods of manually inserting control rods during an Anticipated Transient Without Scram.

74	K/A Importance: 4.0/4.2			**Points: 1.00**
R74	Difficulty: 3.00	Level of Knowledge: Low	Source: BANK	73927

Which of the following describes the basis behind operation of the discharge valve for a tripped Reactor Recirculation Pump as required by the subsequent actions of 20.138.01, Recirculation Pump Tripped AOP?

A. The valve is CLOSED to prevent thermal binding or pressure locking of the valve.

B. The valve is RE-OPENED to prevent thermal binding or pressure locking of the valve.

C. The valve is CLOSED to prevent the pump from fully seating on its thrust bearing following the trip.

D. The valve is RE-OPENED to prevent the pump from fully seating on its thrust bearing following the trip.

Answer: B

Answer Explanation:
From the Reactor Recirculation Pump Trip AOP (20.138.01):

"The recirculation pump discharge valve for the pump that is tripped is closed to allow for flow coast down and to prevent the possibility of reverse flow through the pump. It also ensures that the pump will have seated on its thrust bearing following the trip

In order to prevent any thermal binding or pressure locking, the discharge valve should be opened after approximately five minutes from the time that it was closed. This will ensure that enough time has elapsed with the valve closed to address the flow concerns and the seating on the thrust bearing, yet be re-opened prior to any thermal binding or pressure locking occurring."

The correct answer is the only one that contains the correct basis for the action being taken in that it states the valve is re-opened to prevent thermal binding or pressure locking.

Distractor Explanation:
Distractors are incorrect and plausible because:

A. The valve is closed per the AOP, however, the reason that it is closed is NOT to prevent thermal binding or pressure locking. The reason the valve is closed is to allow for flow coast down and ensure the pump will seat on its thrust bearing. The valve is re-opened to prevent thermal binding or pressure locking.
C. The valve is closed per the AOP, however, the reason that it is closed is NOT to prevent the pump from fully seating on its thrust bearing but, in fact, to ENSURE the pump does seat on its thrust bearing.
D. The valve is re-opened per the AOP, however, the reason that it is re-opened is NOT to prevent the pump from fully seating on its thrust bearing. The reason it is re-opened is to prevent the valve from thermally binding or pressure locking.

Reference Information:
20.138.01, Reactor Recirculation Pump Trip AOP
20.138.01 Bases, Reactor Recirculation Pump Trip AOP Bases

NUREG 1123 KA Catalog Rev. 2
G2.4.11 Knowledge of abnormal condition procedures.

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

NRC Question Use (ILO 2018)
BANK
Low
RO

Associated objective(s):
Reactor Operator
Exam Objectives
Performance Enabler
Explain bases for notes, cautions and overrides.

75	K/A Importance: 3.6/4.0			**Points: 1.00**
R75	Difficulty: 4.00	Level of Knowledge: High	Source: BANK	74546

The plant is currently operating at 100% power in a black board condition when the CRLNO notices that the VAS display shows 3D19 as red, along with VAS Hardware System Trouble and MUX A & C FAILED.

GIVEN a copy of the Alarm Response Procedure for 3D19, which of the responses below correctly completes the following statement regarding the impact of these conditions on the VAS system?

This information indicates that the VAS system will incur ____(1)____of functionality and will result in _____(2)_____.

A. (1) loss
(2) loss of redundancy for one half side of VAS I/O

B. (1) no loss
(2) loss of P601 through P805 window failures

C. (1) loss
(2) loss of P601 through P805 window failures

D. (1) no loss
(2) loss of redundancy for one half side of VAS I/O

Answer: C

Answer Explanation:
ARP 3D19 includes a flowchart on Page 6 that the candidate should refer to based on the information provided in the stem of the question.
Due to the VAS System Hardware Failed and the blackboard condition, the operator should answer N to the question "have a number of false alarm windows turned on?" This should lead the candidate to review the table to determine Data Acquisition Status. Given the failure of MUX A and C, the operator should review Action C, which is located on the next page (page 7). This page describes the impact of a loss of the redundant pair of MUX A&C.
A failure of this pair (or any redundant pair, which includes A&C or B&D) will result in a loss of VAS functionality and loss of alarm windows on the P601 through P805 panels.

Distractor Explanation:
Distractors are incorrect and plausible because:
A. The candidate could incorrectly read the flowchart in the ARP and read Action B, which would lead the candidate to determine that the loss MUX A/C means that redundancy for one half side of VAS was lost, which is incorrect because the redundant pair has been lost (A and C MUX are redundant to one another).
B. The candidate could incorrectly read the flowchart in the ARP and determine that no loss of functionality occurred, which is incorrect as stated on Action C.
D. The candidate could incorrectly read the flowchart in the ARP and determine that no loss of functionality occurred, which is incorrect as stated on Action C.

Reference Information:
ARP 3D19, Annunciator System Trouble (provided as reference)

NUREG 1123 KA Catalog Rev. 2
G2.4.32 3.6/4.0 Knowledge of operator response to loss of all annunciators

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

NRC Question Use (ILO 2018)
BANK
High
NRC Early Review
RO

Associated objective(s):
Integrated Plant Computer Systems (IPCS) / Annunciator and Sequence Recorder System
Cognitive Enabler
Identify alarm response procedures associated with the Integrated Plant Computer System / Annunciator and Sequence Recorder System.

Integrated Plant Computer Systems (IPCS) / Annunciator and Sequence Recorder System
Cognitive Enabler
Identify alarm response procedures associated with the Integrated Plant Computer System / Annunciator and Sequence Recorder System.

76	K/A Importance: 3.3/4.0			**Points: 1.00**
S76V2	Difficulty: 3.00	Level of Knowledge: High	Source: BANK	75046

You are the Control Room Supervisor (CRS) when the following conditions occur:

- The station experiences a Complete Loss of AC Power.
- The HPCI system is the ONLY available RPV injection source.
- Primary Containment conditions dictate that an Emergency Depressurization (ED) be performed.

Once the ED is commenced, (1) at what point, if any, will you direct the ED be terminated and (2) what is the basis for this action?

A. (1) ED is never terminated until the SRVs close on low steam pressure.
(2) To ensure the RPV is at the lowest possible energy state.

B. (1) When RPV pressure reaches ~200 psig.
(2) To ensure sufficient steam pressure to allow continued operation of HPCI for Adequate Core Cooling (ACC).

C. (1) When less than 5 SRVs are open and RPV pressure is greater than 50 psig above torus pressure.
(2) To ensure adequate steam flow through open SRVs to maintain Adequate Core Cooling (ACC).

D. (1) When RPV pressure reaches ~75 psig.
(2) To prevent recieving a low pressure isolation for HPCI.

Answer: B

Answer Explanation:
ED is terminated when RPV pressure reaches ~200 psig to ensure sufficient steam pressure to allow continued operation of HPCI since the HPCI low pressure isolation occurs at 100 psig.

The candidate should determine the correct response, as directed by the EOPs for the given plant conditions, per ED-OR1, which states IF "It is anticipated that RPV depress will result in loss of inj required for ACC" THEN "Terminate RPV depress" AND "Control RPV press as low as practicable while maintaining injection required for ACC. Defeat isolations and exceed Off-Site release rate limits if necessary."

Note: The override that this question is based on was added as a result of the Extended Loss of AC Power (ELAP) event encountered at Fukishima.

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 EOP actions and 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 (ED during a loss of AC power with only a steam driven feed source available) and then taking action based on the requirements of the EOPs.

Distractor Explanation:
Note: Distractor Structure is NOT 2 of 2, each is a separate statement. The use of the (1) and (2) is to help the examinee understand the answers and distractors.

A. "ED is never terminated until the SRVs close on low steam pressure, to ensure the RPV is at the lowest possible energy state" is a valid distractor since this is the normal end state of ED in conditions not involving a loss of AC power with only a steam driven feed source available.
C. "When less than 5 SRVs are open and RPV pressure is greater than 50 psig, To ensure adequate steam flow through open SRVs to maintain Adequate Core Cooling (ACC)." is a logical distractor because this another override in the ED leg, ED-OR2, which is an override that requires that ED be performed (or re-performed) if the RPV were to become re-pressurized. This answer is incorrect because it does not ensure ACC in conditions 0involving a loss of AC power with only a steam driven feed source available.
D. "ED is terminated when RPV pressure reaches ~75 psig To prevent receiving a low-pressure isolation for HPCI" is a logical distractor since 62 psig is the isolation setpoint for the RCIC system and the candidate could confuse the RCIC and HPCI low pressure isolations. Additionally, the candidate could assume that the HPCI low pressure is defeated while under these conditions, which it is not. Even if it were isolated, per the HPCI Design Basis Document, the isolations are based on the minimum pressure for HPCI turbine operation, so going below 100 psig would result in a loss of the HPCI turbine and a loss of ACC.

Reference Information:
29.100.01 SH 3 ED-OR1
HPCI DBD E41-00

Plant Procedures
29.100.01 SH 3

NUREG 1123 KA Catalog Rev. 2

295003 AK1.	Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER :
295003 AK1.06	Station blackout: Plant-Specific
G2.4.18	Knowledge of the specific bases for EOPs

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 Question Use (ILO 2018)
BANK
High
SRO

Associated objective(s):
EOP Objectives

Cognitive Terminal

Given a copy of 29.100.01 sheet 3, discuss the operator's actions for Reactor Pressure Vessel Emergency Depressurization (ED)

THIS PAGE LEFT BLANK

77	K/A Importance: 4.3/4.4			**Points: 1.00**
S77	Difficulty: 3.00	Level of Knowledge: Low	Source: BANK	74251

The reactor has scrammed and the plant has been stabilized. The CRS has directed the scram to be RESET per 20.000.21, Reactor Scram.

What action must be taken if a Control Rod does not settle to the 00 position following reset of the scram?

A. Disarm the Control Rod per 23.106 Instrument Trip Sheets.

B. Verify rod position on the Rod Worth Minimizer.

C. Verify the Full-In light is lit on the Full Core Display.

D. Attempt to insert the Control Rod to 00 per 23.623 Reactor Manual Control System.

Answer: D

Answer Explanation:
For control rods that did not settle to 00, the Reactor Scram AOP states to attempt to insert CR to 00 per 23.623.

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 20.00.21, 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 (SCRAM and reset) and then taking action based on the requirements of AOP 20.000.21.

Distractor Explanation:
A. is plausible, but would be only be performed if manual insertion of the CR was unsuccessful and it is not a high friction rod.
B. is plausible but the RWM would only display number of rods out and not position.
C. is plausible, but would not tell the position of the control rod.

Reference Information:
20.000.21 Cond L

Plant Procedures
20.000.21

NUREG 1123 KA Catalog Rev. 2
295006 AA2. Ability to determine and/or interpret the following as they apply to SCRAM :
295006 AA2.02 Control rod position

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 2013 Exam

NRC Question Use (ILO 2018)
BANK
Low
SRO

Associated objective(s):
AOP Objectives
Cognitive Terminal
Given a copy of 20.000.21, discuss licensed operator actions for a reactor scram.

78	K/A Importance: 3.3/3.1			**Points: 1.00**
S78	Difficulty: 3.00	Level of Knowledge: High	Source: NEW	75107

The plant is operating at 100% power when an air leak on the Station Air Header downstream of the P5000-F402, Station Air to NIAS Iso Valve, occurs. Station and Control Air equipment status is:

- P5000-F440, Div 1 Control Air Iso Valve is CLOSED.
- P5000-F441, Div 2 Control Air Iso Valve is CLOSED.
- P5002-D001, Div 1 Control Air Compressor is RUNNING.
- P5000-D002, Div 2 Control Air Compressor is TRIPPED.

Following isolation of the leak on the air header, Station and Control Air pressures are:

- P50-R802, Station Air Header Pressure is 100 PSIG.
- P50-R870, IAS Header Pressure is 100 PSIG.
- Div 1 on P50-R801, NIAS Header Pressure Recorder, is 100 PSIG.
- Div 2 on P50-R801, NIAS Header Pressure Recorder, is 20 PSIG.

Which of the following will the CRS direct the CRLNO to perform from 20.129.01, Loss of Station and/or Control Air AOP?

A. Open P5000-F403, IAS to Div 2 NIAS Iso Vlv.

B. Open P5000-F441, Div 2 Control Air Iso Valve.

C. Cross-Tie NIAS, with Div 1 supplying, in accordance with 23.129, Station and Control Air System SOP.

D. Consult ARP 17D23, Div 2 Drywl Pneu Supply Iso Valves Closed for approximate time until MSIV isolation.

Answer: C

Answer Explanation:
The candidate should interpret the conditions in the stem of the question as indicating a loss of Division 2 Non-Interruptible Air System (NIAS), or Division 2 safety related Instrument Air, pressure. Per AOP 20.129.01, with the Div 1 and Div 2 air headers intact, the candidate should determine that the correct course of action is to direct the NIAS headers to be cross-tied, with Division 1 supplying. This is per Condition E and F of the AOP.

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 20.129.01, 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 (Instrument Air System pressure) and then directing action based on the requirements of AOP 20.129.01.

Distractor Explanation:
A. This distractor is plausible because it is also directed from Condition E of AOP 20.129.01 and, since IAS is available, the candidate could determine that opening of the P5000-F403 is warranted. However, per **NOTE 1** of Condition E, with Div 2 NIAS pressure below 80 psig, the P5000-F403 will NOT open so this action is not correct and should not be directed by the CRS. Note: The keylock switch bypasses closure of the F403 on loss of power and low Station Air Header Pressure but it does NOT defeat closure of the F403 on low (<80 psig) Div 2 NIAS Header Pressure.
B. This distractor is plausible since re-opening of this valve is directed from Condition K of AOP 20.129.01 and the candidate could determine that, with the leak isolated and IAS header pressure restored to normal (100 PSIG), re-opening of the P5000-F441 is warranted in order to recover Div 2 NIAS. However, although the Low Station Air Pressure isolation (75 psig per Condition C of the AOP) for the P5000-F441 is clear, the valve also recieves an isolation on low Div 2 NIAS header pressure (sensed downstream of the P5000-F441 itself) at 75 PSIG so the valve is incapable of being opened until Div 2 NIAS header pressure is restored via another means, such as cross-tying NIAS headers.
D. This distractor is plausible because, with Div 2 NIAS Header pressure at 20 psig, Division 2 Drywell Pneumatics would have isolated, and 17D23 would be in, due to loss of air pressure to the pneumatic actuators for the T4901-F468 and F469, Div 2 Pneumatic Supply and Outboard Isolation Valves. The candidate could confuse Division 1 Drywell Pneumatics, which supplies the Inboard MSIVs with Nitrogen, with Division 2. This could lead the candidate to determine that Condition E of the AOP has a step similar to step D.1, which has the crew review ARP 8D66 when Division 1 Drywell Pneumatics isolates. This distractor is incorrect, however, because Division 2 Drywell Pneumatics does not supply nitrogen to the MSIVs and the AOP does not include this action.

Reference Information:
20.129.01
20.129.01 BASIS

Plant Procedures
20.129.01

NUREG 1123 KA Catalog Rev. 2
295019 AA2. Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR :
295019 AA2.01 Instrument air system pressure

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 Question Use (ILO 2018)
High
NEW
SRO

Associated objective(s):
AOP Objectives

Cognitive Terminal

Given a copy of 20.129.01, discuss licensed operator actions for a Loss of Station and/or Control Air.

THIS PAGE LEFT BLANK

79	K/A Importance: 3.8/4.0			Points: 1.00
S79	Difficulty: 3.00	Level of Knowledge: High	Source: MODIFIED	74247

The plant was operating at rated power when a partial loss of drywell cooling condition occurred.

- T48-R808, Primary Containment Pressure Rec is indicating 18 inches of water and is increasing at 0.1 inches every 2 minutes.
- T47-R803A, Average Drywell temperature is 137°F and rising 0.5°F every 10 minutes.

(1) What action(s) will the CRS direct first and (2) what is the basis for this action?

A. (1) Lower drywell pressure in accordance with 23.406, Primary Containment Nitrogen and Purge system using RBHVAC.
(2) Maintaining the expected initial conditions to ensure that, in the event of a DBA, equipment in the primary containment is able to perform its design function.

B. (1) Lower drywell temperature in accordance with EOP 29.100.01, Sheet 2, Primary Containment Control.
(2) Maintaining the expected initial conditions to ensure that, in the event of a DBA, equipment in the primary containment is able to perform its design function.

C. (1) Lower drywell pressure in accordance with 23.406, Primary Containment Nitrogen and Purge system using RBHVAC.
(2) Maintaining the expected initial conditions to ensure that the peak LOCA primary containment internal pressure does not exceed 62 psig.

D. (1) Lower drywell temperature in accordance with EOP 29.100.01, Sheet 2, Primary Containment Control.
(2) Maintaining the expected initial conditions to ensure that the peak LOCA primary containment internal pressure does not exceed 62 psig.

Answer: C

Answer Explanation:
Drywell temperature and pressures are elevated, but they are not at EOP entry conditions. The correct action is to lower drywell pressure using either SBGTS or RBHVAC in accordance with 23.406. P&L 3.10.3 Directs Maintain pressure 5 to 19 inches of water as read on T48-R808, Primary Containment Pressure Rec

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 SOP P&L and the basis behind the P&L, 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 (approaching P&L limit) and then selecting a SOP with which to proceed.

Distractor Explanation:
(1) "Lower drywell temperature in accordance with EOP 29.100.01, Sheet 2, Primary Containment Control." is plausible because drywell temperature is high, but not high enough to enter the EOPs. Also, the high pressure condition was caused by a partial loss of drywell cooling.
(2) "Maintaining the expected initial conditions and ensures that so that In the event of a DBA, equipment in the primary containment is able to perform its design function." is the TS reason for maintaining DW temperture. While this answer is CORRECT when paired with distractor (1) for temperture, it is INCORRECT when paired with the answer for (1) pressure.

Reference Information:
TSB 3.1.6.4
23.406 P&Ls

Plant Procedures
03D081
23.406

NUREG 1123 KA Catalog Rev. 2

295024	High Drywell Pressure.
G2.1.32	Ability to explain and apply system limits and precautions

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 Question Use (ILO 2018)
High
MODIFIED
SRO

Associated objective(s):
Nitrogen Purge, Inerting and Venting
Performance Terminal
Analyze Nitrogen Purge, Inerting and Venting response using control room indications and annunciators.

80	K/A Importance: 3.2/4.2			**Points: 1.00**
S80V3	Difficulty: 3.00	Level of Knowledge: High	Source: NEW	75006

Suppression pool average temperature is > 111°F and rising. Per 3.6.2.1 Suppression Pool Average Temperature, when the suppression pool average temperature is > 121°F, a required action is to have reactor pressure less than 200 psig within 12 hours and be in MODE 4 within 36 hours.
The reason for this action is:

A. To bring the plant to a MODE in which the LCO does not apply.

B. To preserve the heat absorption capability of the suppression pool.

C. To ensure that licensing bases initial conditions are met for the LCO.

D. Because the LCO is not met and an associated Required Action and Completion Time is not met and no other Condition applies.

Answer: A

Answer Explanation:
Per TS Basis for 3.6.2.1
A limitation on the suppression pool average temperature is required to assure that the containment conditions assumed for the safety analyses are met. This limitation subsequently ensures that peak primary containment pressures and temperatures do not exceed maximum allowable values during a postulated DBA or any transient resulting in heat up of the suppression pool. The LCO requirements are:
a. Average temperature s 95°F with THERMAL POWER > 1% RATED THERMAL POWER (RTP), and no testing that adds heat to the suppression pool is being performed. This requirement ensures that licensing bases initial conditions are met.
b. Average temperature g 105 °F with THERMAL POWER > 1% RTP and testing that adds heat to the suppression pool is being performed. This required value ensures that the unit has testing flexibility, and was selected to provide margin below the 110°F limit at which reactor shutdown is required. When testing ends, the temperature must be restored to • 95°F within 24 hours according to Required Action A.2. Therefore, the time that the temperature is > 95°F is short enough not to cause a significant increase in unit risk.
c. Average temperature • 110°F with THERMAL POWER, 1% RTP. This requirement ensures that the unit will be shut down at > 110°F. The pool is designed to absorb decay heat and sensible heat but could be heated beyond design limits by the steam generated if the reactor is' not shut down.

E.1 and E.2
If suppression pool average temperature cannot be maintained at < 120°F, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the reactor pressure must be reduced to < 200 psig within 12 hours, and the plant must be brought to at least MODE 4 within 36 hours. The allowed Completion Times is reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. Continued addition of heat to the suppression pool with suppression pool temperature > 120°F could result in exceeding the design basis maximum allowable values for primary containment temperature or pressure. Furthermore, if a blowdown were to occur when the temperature was > 120°F, the maximum allowable bulk and local temperatures could be exceeded very quickly.

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 specific knowledge of applying a TS and the basis for the actions. The answer is not related to < 1 hour TS/TRM Action. The answer cannot be answered LCO/TRM information listed “above-the-line” and cannot be answered solely by knowing the TS Safety Limits. The SRO must assess plant conditions and then evaluate operability then choose the correct action and identify the correct basis.

Distractor Explanation:
B. "preserve the heat absorption capability of the suppression pool" incorrect because the temperature is high enough that there is not enough capacity left to perform the function of the suppression pool, and is plausible because it is a reason for an action required by this TS.
C. "ensures that licensing bases initial conditions are met for the LCO" incorrect because the action does not restore suppression pool ability to absorb energy, and is plausible because this is the basis for the LCO.
D. "To take T.S required actions because an LCO is not met and An associated Required Action and Completion Time is not met and no other Condition applies." is saying use LCO 3.03. This is incorrect because an LCO action applies. This is a logical distractor for TS basis.

Reference Information:
TS 3.6.2.1 and TSB 2.6.2.1 (not provided)

NUREG 1123 KA Catalog Rev. 2
295026 Suppression Pool High Water Temperature
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 Question Use (ILO 2018)
High
NEW
NRC Early Review
SRO

Associated objective(s):
Technical Specifications for Licensed Operators
Performance Enabler
Given plant conditions that constitute non-compliance with any LCO, apply Technical Specifications and Bases to determine the applicable Condition(s), Required Action(s), and associated Completion Time(s).

THIS PAGE LEFT BLANK

81	K/A Importance: 3.8/3.9			Points: 1.00
S81	Difficulty: 3.00	Level of Knowledge: High	Source: BANK	74128

Following a transient, the following conditions exist:

- Drywell Temperature is 280ºF and stable.
- Drywell Pressure is 15 psig.
- Reactor Pressure is 20 psig.
- B21-R610 RPV CORE LEVEL RECORDER indication is oscillating between -20 inches and +30 inches
- All other RPV level indicators are off scale low or failed.
- ALL RHR Pumps are injecting.

Which one of the following actions is required?

A. Enter 29.100.01 Sheet 3, RPV Flooding, and inject until the RPV is flooded to the Main Steam Lines.

B. Open Turbine Bypass Valves and rapidly depressurize the reactor per 29.100.01 Sheet 1, RPV Control.

C. Enter 29.100.01 Sheet 3, Emergency Depressurization, and inject until RPV Pressure is > 64 psig above Torus Pressure.

D. Wait until RPV Water Level lowers to -43 inches and then Emergency Depressurize per 29.100.01 Sheet 3, Steam Cooling.

Answer: A

Answer Explanation:
The candidate must determine that the RPV Saturation Temperature curve is exceeded with the given values of RPV Pressure and Drywell Temperature. With the Saturation Curve exceeded, and level indicators oscillating, RPV Water Level is unknown. With Level Unknown, it is required to Enter 29.100.01 Sheet 3, RPV Flooding, and inject until the RPV is flooded to the Main Steam Lines. (L-OR1)

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 EOP directed RPV Flooding, 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 selecting a section of the EOPs with which to proceed.

Distractor Explanation:
B. Is plausible; and would be true for other conditions where ED is anticipated.
C. Is plausible; >64 psig above Torus pressure with <4 SRVs open leads to use of Alternate Emergency Depressurization Systems.
D. Is plausible; and is required action for Steam Cooling.

Reference Information:
LP-OP-802-3003-0013C
29.100.01 SH3
29.100.01 SH1

Plant Procedures
29.100.01 SH 3

NUREG 1123 KA Catalog Rev. 2
295028 EA2. Ability to determine and/or interpret the following as they apply to HIGH DRYWELL TEMPERATURE :
295028 EA2.02 Reactor pressure

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 Question Use (ILO 2018)
BANK
High
SRO

Associated objective(s):
EOP Objectives
Cognitive Terminal
Given a copy of 29.100.01 sheets 3 and 3A, discuss the operator's actions for Reactor Pressure Vessel Flooding (RF)

82	K/A Importance: 4.1*/4.5*			Points: 1.00
S82	Difficulty: 3.00	Level of Knowledge: High	Source: NEW	74126

A fuel failure has occurred and currently a Radiation Release in progress. The Offsite rad release rate is higher than the ALERT offsite release rate and less than the GENERAL EMERGENCY release rate. Use of HPCI is required to maintain Reactor Water Level. HPCI room temperature is 111°F and slowly rising.

Which of the following systems would the CRS direct to be isolated, per the Rad Release section of the EOPs, to mitigate the High Offsite Release Rate?

A. HPCI

B. Off Gas System

C. Main Steam Lines

D. Standby Gas System

Answer: C

Answer Explanation:
per 29.100.01 SH 5 RR-2
Isolate primary systems that are discharging into areas outside the primary and secondary containment, except systems req'd by the EOPs.

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 EOPs, 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 applying the requirements of the EOPs.

Distractor Explanation
Distractor are incorrect and plausible because:
A. HPCI could be a source that meets the requirements for isoloation however it is neccesary for the EOPs so it is incorrect.
B. Off Gas System is not a primary system, however it is a system that discharges to the enviroment.
D. Standby Gas is not a primary system, however it is a system that discharges to the enviroment.

Reference Information:
29.100.01 SH 5

Plant Procedures
29.100.01 SH 5

NUREG 1123 KA Catalog Rev. 2
295038 EA2. Ability to determine and/or interpret the following as they apply to HIGH OFF-SITE RELEASE RATE :
295038 EA2.04 Source of off-site release

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 Question Use (ILO 2018)
High
NEW
SRO

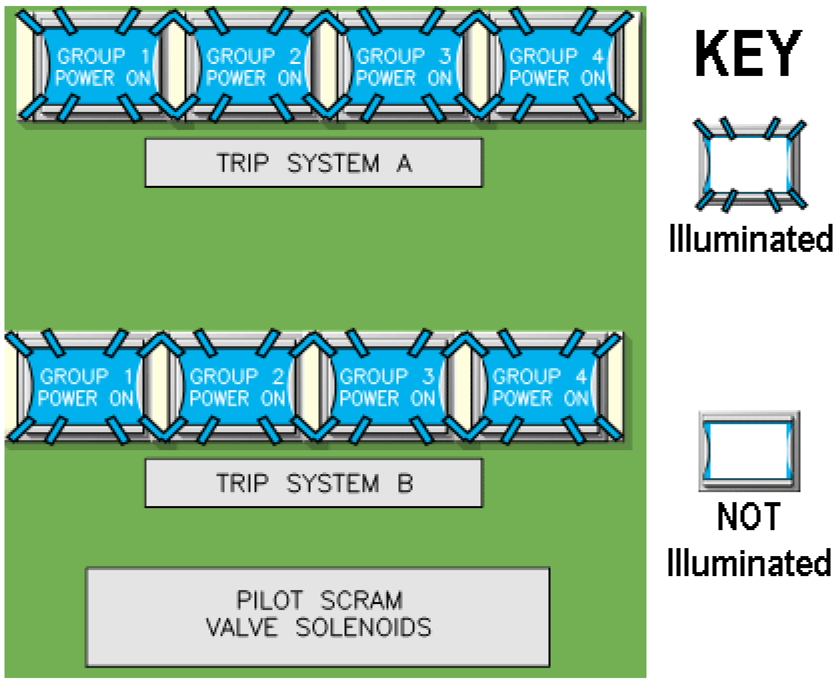
Associated objective(s):
EOP Objectives
Cognitive Terminal
Given a copy of 29.100.01 sheet 5, discuss the operator's actions for Radioactivity Release Control (RR)

83	K/A Importance: 4.4/4.0			**Points: 1.00**
S83V2	Difficulty: 3.00	Level of Knowledge: High	Source: NEW	75906

Due to plant condtions, the MODE Switch has been placed in SHUTDOWN. A SCRAM does NOT occur:

Based on this the CRS would observe the reactor operator __(1)__ both C7100-M606A MANUAL SCRAM TRIP A2 SWITCH and C7100-M606B MANUAL SCRAM TRIP B2 SWITCH located on the __(2)__ panel.

When FSQ1-8 is complete TRIP SYSTEM A/B indicate:



Based on this, the CRS will direct __(3)__

A. (1) Rotating
(2) P602
(3) 29.ESP.10 DEFEAT OF ARI LOGIC TRIPS

B. (1) Depressing
(2) P603
(3) 29.ESP.09 DEFEAT OF RPS AUTOMATIC LOGIC TRIPS

C. (1) Rotating
(2) P602
(3) 29.ESP.09 DEFEAT OF RPS AUTOMATIC LOGIC TRIPS

D. (1) Depressing
(2) P603
(3) 29.ESP.10 DEFEAT OF ARI LOGIC TRIPS

Answer: D

Answer Explanation:
Per ODE 10 If all control rods are not fully inserted after the Reactor Mode Switch has been placed in Shutdown, the P603 will depress the RPS Manual Scram pushbuttons and monitor for proper response. - This action should hace made the 8 blue RPS lights turn off.
Per 29.ESP.03 Step 4.1 If the 8 blue RPS lights are off defeat RPS logic trips in accordance with 29.ESP.09,
"Defeat of RPS Automatic Logic Trips." Based on condtions 29.ESP.10 is required.

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 actions required by the EOPs and sub procedures (ESPs), it is not related to immediate actions and the entry conditions are not relevant or leading to answer. The SRO must assess the plant conditions (ATWS) and then diagnose the status of RPS per 29.ESP.03 and then ensure 29.ESP.09 is used to mitigate / recover.

Distractor Explanation:

(1) "Rotate" and (2) "P602" is incorrect and plausible because the label for these push buttons is "SWITCH" and the P602 is in the front arc of Main Control Room.

(3) 29.ESP.03 allows for 29ESP.09, based condtions this is incorrect.

Reference Information:
ODE 10 Pg 8
29.ESP.03
EOPs

Plant Procedures
29.ESP.03

NUREG 1123 KA Catalog Rev. 2
295015 Incomplete SCRAM
G2.1.30 Ability to locate and operate components, including local controls

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 Question Use (ILO 2018)
High
NEW
NRC Early Review
SRO

Associated objective(s):
Emergency Support Procedures (ESP)
Cognitive Terminal
Given a copy of 29.ESP.03, discuss Licensed Operator actions for Alternate Control Rod Insertion

84	K/A Importance: 3.5*/3.9*			Points: 1.00
S84	Difficulty: 3.00	Level of Knowledge: High	Source: BANK	74047

The plant is operating when a small break LOCA and ATWS occurs. ATWS conditions continue for 2 hours before the crew is able to shutdown the reactor (all rods are full in). The following plant conditions are now present:

- RPV water level is 10" above TAF and steady.
- SBFW is the only injection source.
- Reactor Pressure is 600 psig and rising.
- MSIVs are closed.
- Drywell pressure is 12 psig and steady.
- Drywell Temperature is 280°F and steady.
- Suppression pool temperature is 152°F and steady.
- Torus water level is 16” and slowly rising, and all efforts to lower level are unsuccessful.

If the current trend in primary containment water level persists, what action is required and why?

A. Emergency Depressurize to protect the integrity of the SRV tailpipes.

B. Emergency Depressurize to preserve pressure suppression capability.

C. Reduce reactor pressure (<90°F/hr) to protect the integrity of the SRV tailpipes.

D. Reduce reactor pressure (<90°F/hr) to preserve pressure suppression capability.

Answer: A

Answer Explanation:
Per EPG (B-17-80) the SRVTPLL (STPLL in EPG) bases is:
The SRV Tail Pipe Level Limit (STPLL) is the lesser of:
• The Maximum Pressure Suppression Primary Containment Water Level.
• The highest suppression pool water level at which opening an SRV will not result in exceeding the code allowable stresses in the SRV tail pipe, tail pipe supports, quencher, or quencher supports.
The STPLL is a function of RPV pressure. SRV operation with suppression pool water level above the STPLL could damage the SRV discharge lines. This, in turn, could lead to containment failure from direct pressurization and damage to equipment inside the containment (ECCS piping, RPV water level instrument runs, wetwell-to-dry well vacuum breakers, etc.) from pipe-whip and jet-impingement loads.

Based on the stem of the question
Operation is currently in the unsafe area of the SRVTPLL. Efforts to lower level have so far been unsuccessful. ED is required to protect the SRV tail pipes per TWL-11. This requires the SRO to execute P-OR1-3, ED is REQ'D

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 actions required by the EOPs, it is not related to immediate actions and the entry conditions are not relevant or leading to answer. The SRO must assess the plant conditions (SRVTPLL) and then selecting the step of the EOP to mitigate / recover.

Distractor Explanation:
B. Is incorrect because the reason ED is required is to protect the SRV tail pipe. This answer is plausible based on not understanding the bases of the SRVTPLL
C. Is incorrect because an ED is required and taking P-OR1-3 requires exiting pressure leg. This answer is plausible based on not assessing the need to ED.
D. Is incorrect because an ED is required and taking P-OR1-3requires exiting pressure leg. This answer is plausible based on not assessing the need to ED.

Reference Information:
BWROG EPG/SAGs Appendix B Bases B-17-80

Plant Procedures
29.100.01 SH 1A
BWROG EPG App B

NUREG 1123 KA Catalog Rev. 2

295029 EA2.01	Suppression pool water level
295029 EK3.	Knowledge of the reasons for the following responses as they apply to HIGH SUPPRESSION POOL WATER LEVEL :
295029 EK3.01	Emergency depressurization
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
LOR 2009 Exam
LOR 2015 Exam

NRC Question Use (LOR 2017)
SRO ONLY

NRC Question Use (ILO 2018)
BANK
High
SRO

Associated objective(s):

Cautions, Curves, and Calculations

Cognitive Enabler

Discuss the definition of and reason for the shape 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

Deleted

f. Deleted
g. RHR Low Pressure Coolant Injection (LPCI) NPSH Limit
h. RHR LPCI Vortex Limit
i. RPV Saturation Temperature
j. SRV Tail Pipe Level Limit
k. Drywell Spray Initiation Limit
l. Pressure Suppression Pressure
m. Primary Containment Pressure Limit
n. HPCI NPSH Limit
o. RCIC NPSH Limit

EOP Objectives

Cognitive Terminal

Given a copy of 29.100.01 sheet 1A, discuss the operator's actions for Anticipated Transient Without Scram (ATWS) to control Reactor Pressure Vessel pressure (FSP)

THIS PAGE LEFT BLANK

85	K/A Importance: 3.8/3.9			Points: 1.00
S85	Difficulty: 2.00	Level of Knowledge: Low	Source: BANK	74011

During the execution of 29.100.01 Sheet 5, Secondary Containment and Radiation Release, the operability of equipment required to perform a safe shutdown is assured by Emergency Depressurization for which of the following conditions?

A. Radiation Level in any ONE AREA EXCEEDS the MAX NORMAL value.

B. Radiation Levels in MORE THAN ONE AREA EXCEED the MAX SAFE value.

C. Radiation Levels in MORE THAN ONE AREA EXCEED the MAX NORMAL value.

D. Radiation Level in any ONE AREA is CONFIRMED by a MAX SAFE Area Temperature OR MAX SAFE Water Level in the SAME AREA.

Answer: B

Answer Explanation:
Emergency Depressurization is required when the Radiation Levels in MORE THAN ONE AREA exceed the MAX SAFE value.

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 EOP directed Emergency Depressurization, 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 applying the requirements of the EOPs.

Distractor Explanation:
A is plausible; and is an entry condition for 29.100.01 Sheet 5.
C is plausible; does NOT require Emergency Depressurization.
D is plausible; does NOT require Emergency Depressurization.

Reference Information:
EOP 29.100.01 Sheet 5

Objective Link: LP-OP-802-3005-0009

Plant Procedures
29.100.01 SH 1

NUREG 1123 KA Catalog Rev. 2

295033 EA2.	Ability to determine and/or interpret the following as they apply to HIGH SECONDARY CONTAINMENT AREA RADIATION LEVELS :
295033 EA2.01	Area radiation levels
295033 EA2.02	Equipment operability

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 Question Use (ILO 2018)
BANK
Low
SRO

Associated objective(s):
Emergency Support Procedures (ESP)
Cognitive Enabler
Given a copy of 29.100.01 SH1/1A, discuss Licensed Operator actions to Rapidly depressurize the Reactor Pressure Vessel using the Main Steam Line Drains

86	K/A Importance: 3.4/3.5			Points: 1.00
S86VB	Difficulty: 3.50	Level of Knowledge: High	Source: BANK	76086

The plant just entered Mode 4 in preparation for a Refueling Outage. Div 1 RHR aligned for SDC. A leak develops in the RWCU system which isolates on high differential flow. The following alarms and indications are observed:

- 1D23, DIV I RHR PUMP A/C MOTOR TRIPPED
- 3D79 REAC VESSEL WATER LEVEL L3
- RPV water level is 170 inches and steady.
- RPV temperature is 150°F and rising slowly.

As CRS, (1) what procedure(s) would be selected to mitigate this event and (2) what actions are the minimum required to be performed to allow restoration of Div 1 RHR to Shutdown Cooling?

A. (1) 20.205.01 LOSS OF SHUTDOWN COOLING
(2) Reset NSSS logic, and depress E1150-F015A Seal In Reset pushbutton.

B. (1) 20.205.01 LOSS OF SHUTDOWN COOLING, 29.100.01 SH 5 SEC CONT/RAD RELEASE
(2) Reset NSSS logic, and depress E1150-F015A Seal In Reset pushbutton.

C. (1) 20.205.01 LOSS OF SHUTDOWN COOLING, 29.100.01SH 1 RPV CONTROL
(2) Raise level above 220”, reset NSSS logic, and depress E1150-F015A Seal In Reset pushbutton.

D. (1) 20.205.01 LOSS OF SHUTDOWN COOLING, 29.100.01 SH 1 RPV CONTROL, 29.100.01 SH 5 SEC CONT/RAD RELEASE
(2) Raise level above 220”, reset NSSS logic, and depress E1150-F015A Seal In Reset pushbutton.

Answer: C

Answer Explanation:
Plant conditions do not meet the entry conditions for Sheet 5, but meet those in 20.205.01 and Sheet 1 because of RPV Level 3. Actions are contained in condition C of 20.205.01.

A. is incorrect because you meet entry conditions for sheet 1. This directs you to raise level as does the AOP.
B. is incorrect because you do not meet entry conditions for sheet 5, you meet entry conditions for sheet 1. This directs you to raise level as does the AOP.
D. is incorrect because you do not meet entry conditions for sheet 5.

Reference: 20.205.01, page 5

Plant Procedures
20.205.01
29.100.01 SH 1
29.100.01 SH 5

NUREG 1123 KA Catalog Rev. 2

205000 A2.	Ability to (a) predict the impacts of the following on the SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLINGMODE) ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:
205000 A2.06	SDC/RHR pump trips

NRC Exam Usage
ILO 2012 Exam

NRC Question Use (ILO 2018)
BANK
High
SRO

Associated objective(s):
Cycle 12-1 Objectives
Performance Objectives

Performance Terminal

Upon recognizing the entry conditions for an inadvertent loss of RHR shutdown cooling operation, operators respond to control critical parameters within expected limits IAW approved plant procedures within validated completion times

87	K/A Importance: 3.4/3.6			Points: 1.00
S87V2	Difficulty: 3.00	Level of Knowledge: High	Source: NEW	75907

The plant has scrammed.

- The Drywell Pneumatics supply header is depressurized due to a rupture.
- Conditions exist that require Emergency Depressurization per the EOPs.
- Drywell pressure is 5 psig.
- Torus water level is 2 inches.

What action will the CRS direct?

A. Open 5 ADS SRVs.

B. Open 5 Divison 2 SRVs

C. Open Main Turbine bypass valves, ignoring cooldown rates.

D. Blowdown using RWCU, install defeats as necessary from 29.ESP.18 Defeat of RWCU Isolations.

Answer: A

Answer Explanation:
Per EOP Open 5 SRVs, ADS preferred

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 EOP directed Emergency Depressurization, 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 applying the requirements of the EOPs.

Distracter Explanation:
Distracters are incorrect and plausible because
B. All Drywell Pneumatics is depressurized due to a rupture, therefore the misconception is that Division 1 SRVs not available. However per EOP ADS SRV are used.
C. Open Main Turbine bypass valves, ignoring cooldown rates is an override for anticipate ED and a possible option if SRVs are not available per table 18 of EOPs. It is not an authorized option because 5 ADS SRVs are available
D. RWCU blowdown is only used if SRVs are not available and is per table 18 of EOPs.

Reference Information:
29.100.01.SH3 ED-4
M-5007

Plant Procedures
29.100.01 SH 3

NUREG 1123 KA Catalog Rev. 2

218000 A2.	Ability to (a) predict the impacts of the following on the AUTOMATIC DEPRESSURIZATION SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:
218000 A2.03	Loss of air supply to ADS valves: Plant-Specific

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 Question Use (ILO 2018)
High
NEW
NRC Early Review
SRO

Associated objective(s):
EOP Objectives
Cognitive Terminal
Given a copy of 29.100.01 sheet 3, discuss the operator's actions for Reactor Pressure Vessel Emergency Depressurization (ED)

88	K/A Importance: 3.4/4.7			Points: 1.00
S88	Difficulty: 4.00	Level of Knowledge: High	Source: NEW	73909

Review the table of as found SRV lift pressures. The Plant is in MODE 1.

SRV	#	SRV	#	SRV	#
A	1080	D	1134	E	1140
B	1122	F	1182	H	1193
C	1075	L	1080	J	1158
G	1099	M	1165	P	1116
K	1120	N	1155	R	1139

Based on T.S. 3.4.3 how many SRV setpoint(s) must be adjusted to restore the minimum number of required OPERABLE SRVs?

A. 0

B. 1

C. 2

D. 3

Answer: B

Answer Explanation:

Per SR 3.4.3.1

SRVs
3.4.3

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.3.1	Verify the safety function lift setpoints of the required SRVs are as follows: Number of SRVs — Setpoint (psig) 5 — 1135 ± 34.05 5 — 1145 ± 34.35 5 — 1155 ± 34.65 Following testing, lift settings shall be within ± 1%.	In accordance with the Inservice Testing Program
SR 3.4.3.2	Verify each required SRV is capable of being opened.	In accordance with the Surveillance Frequency Control Program

Per BASIS for SR 3.4.3.1

This Surveillance requires that the required SRVs will open at the pressures assumed in the safety analysis of Reference 1. The demonstration of the SRV safe lift settings must be performed during shutdown, since this is a bench test, to be done in accordance with the Inservice Testing Program. The lift setting pressure shall correspond to ambient conditions of the valves at nominal operating temperatures and pressures. The SRV setpoint is ± 3% for OPERABILITY, however, the valves are reset to ± 1% during the Surveillance to allow for drift. The SR gives set pressures for all 15 SRVs installed. However, since only 11 SRVs are required, the SR is met if 11 SRVs are set properly.

SRV DATA:

SRV	AS FOUND	Required		IN RANGE
A	1080	1135		
B	1122	1135		1135
C	1075	1135		
G	1099	1135		
K	1120	1135		1135
D	1134	1145		1145
F	1182	1145		1155
L	1080	1145		
M	1165	1145		1145
N	1155	1145		1145
E	1140	1155		1155
H	1193	1155		
J	1158	1155		1155
P	1116	1155		1145
R	1139	1155		1155
			@1135	2

@1145	4
@1155	4
Total	10

Therefore 1 must MUST be adjusted

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 specific knowledge of applying SR 3.4.3.1. The answer is related to < 1 hour TS/TRM Action however it cannot be soley answered with this information because OPERABILITY must first be determined using a SR. The answer cannot be answered LCO/TRM information listed “above-the-line” and cannot be answered by solely by knowing the TS Safety Limits. The SRO must assessing plant conditions and then evaluate operability based on SR 3.4.3.1 and then choose the correct action.

Distractor Explanation:
Because 2 of the as found values are not in the accepable band, but still in an acceptable range for tech spec, and at least 1 must be adjusted a max of 3 is plausible.

Reference Information:
TS for 3.4.3 and SR 3.4.3.1 (provided)

NUREG 1123 KA Catalog Rev. 2
239002 SRVs
G2.2.40 Ability to apply technical specifications for a system

10CFR55 RO/SRO Written Exam Content
10 CFR 55.43(b) (2) Facility operating limitations in the technical specifications and their bases.

NRC Question Use (ILO 2018)
High
NEW
NRC Early Review
SRO

Associated objective(s):
Technical Specifications for Licensed Operators
Performance Enabler
Given plant conditions that constitute non-compliance with any LCO, apply Technical Specifications and Bases to determine the applicable Condition(s), Required Action(s), and associated Completion Time(s).

89	K/A Importance: 4.4/4.7			**Points: 1.00**
S89V2	Difficulty: 4.00	Level of Knowledge: High	Source: MODIFIED	75008

The plant is operating at 100% power. A review of the Quarterly Battery Check surveillance, for Division II shows the following results:

- Electrolyte Level 1/4" above maximum
- Lowest Specific Gravity 1.192
- Lowest Float Voltage 2.06 (3 cells)

Which of the following action(s), if any would be required for the given conditions?

A. T.S. 3.8.6 ACTION B.1 ONLY

B. T.S. 3.8.6 ACTION A.1, A.2 and A.3 ONLY

C. T.S. 3.8.6 ACTION A.1, A.2, A.3
AND
T.S. 3.8.6 ACTION B.1 ONLY

D. T.S. 3.8.6 ACTION A.1, A.2 and A.3
AND
T.S. 3.8.6 ACTION B.1
AND
T.S. 3.8.4 ACTION B.1

Answer: D

Answer Explanation:
TS 3.8.6 Table 3.8.6.1 Category C states float voltage must be >2.07V for each connected cell. This requires entry into TS 3.8.6 Condition B to declare associated battery inoperable. This also requires entry into TS 3.8.4 Condition B for a DC electrical subsystem inoperable with a completion to operable status of 2 hours

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 specific knowledge of applying the SRs in 3.8.6 (Table 3.8.6-1). The answer is related to < 1 hour TS/TRM Action however it cannot be soley answered with this information because OPERABILITY must first be determined using a SR. The answer cannot be answered LCO/TRM information listed “above-the-line” and cannot be answered by solely by knowing the TS Safety Limits. The SRO must assessing plant conditions and then evaluate operability based on TS 3.8.6 and TS 3.8.4. and then choose the correct action

Distractor Explanation:
A. "T.S. 3.8.6 Cond B.1 ONLY" is incorrect because of the ONLY (see answer explanation). This distractor is plausible if the examinee reviews T.S. 3.8.4 and finds the DC subsystem to be OPERABLE.
B. "T.S. 3.8.6 Cond A.1, A.2 and A.3 ONLY" is incorrect because of answer explanation. This distractor is plausible if the examinee reviews T.S 3.8.6 can find the battery cells to be on meeting Category A or B limits.
C. See B & C

Reference Information:
TS & TSB for 3.8.6 and 3.8.4

NUREG 1123 KA Catalog Rev. 2
263000 DC Electrical Distribution
G2.1.7 Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation

G2.2.40 Ability to apply technical specifications for a system

Technical Specifications
3.8.4 DC Sources Operating
3.8.6 Battery Cell Parameters

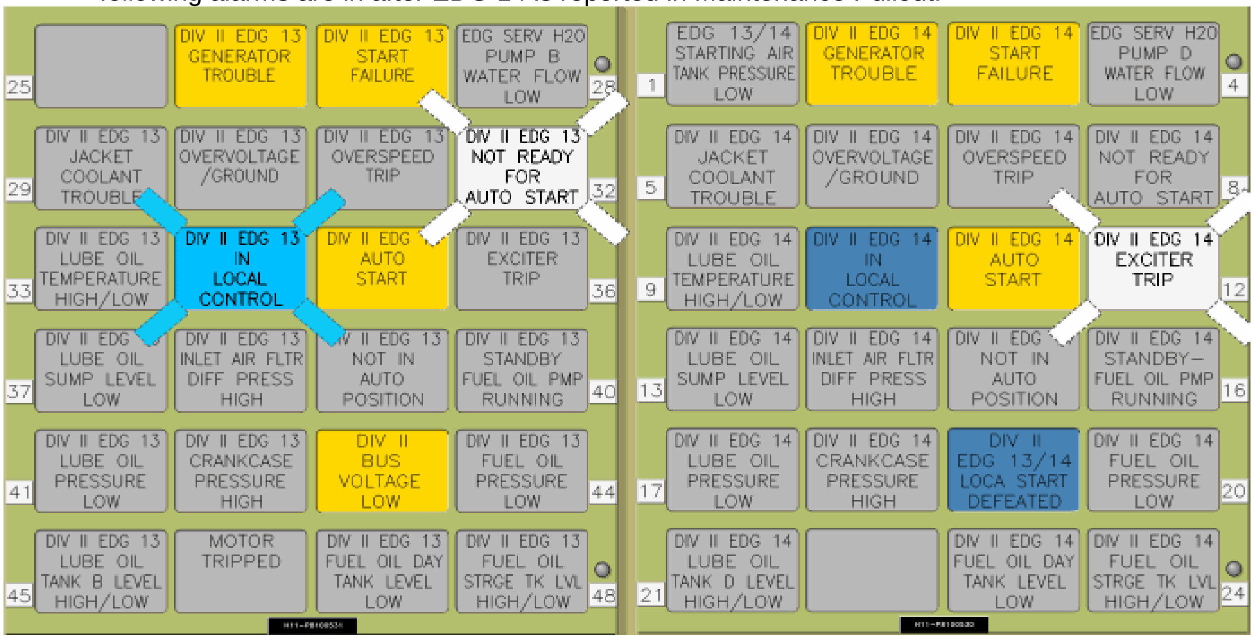
10CFR55 RO/SRO Written Exam Content
10 CFR 55.43(b) (2) Facility operating limitations in the technical specifications and their bases.

NRC Question Use (ILO 2018)
High
MODIFIED
SRO

Associated objective(s):
Technical Specifications for Licensed Operators
Performance Enabler
Given plant conditions that constitute non-compliance with any LCO, apply Technical Specifications and Bases to determine the applicable Condition(s), Required Action(s), and associated Completion Time(s).

90	K/A Importance: 4.2/4.2			**Points: 1.00**
S90V2	Difficulty: 2.00	Level of Knowledge: High	Source: NEW	75127

The plant is at 100% power when a report about the material condition of EDG 14 requires it be declared INOP. A Nuclear Operator is dispatched to place EDG 14 in Maintenance Pullout. The following alarms are in after EDG 14 is reported in Maintenance Pullout.



(1) Are these alarms consistent with the expected plant conditions after placing EDG 14 in Maintenance Pullout?

(2) What is the most limiting time to restore OPERABILITY per TS 3.8.1?

A. (1) No
(2) 14 Days

B. (1) No
(2) 72 Hours

C. (1) Yes
(2) 72 Hours

D. (1) Yes
(2) 14 Days

Answer: B

Answer Explanation:
Per 10D34 and 10D32, EDG 13 is in MPO(Maintenance Pullout).
The Nuclear Operator put the wrong EDG in MPO. This means BOTH EDG 13 and 14 are INOP

Per TS 3.8.1 B.4 - 72 hrs to restore one EDG.

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 specific knowledge of T.S and how to apply actions. The answer is NOT related immediate actions. The examinee must determine operability first to then apply the correct actions. The answer is not related to TS Safety Limits. The answer is relevant to the entry conditions but is not leading to the answer. While information found above the double line is needed, the correct understanding of the bases for the condition is required. The answer to this questions is based on assessing plant conditions (Wrong EDG in MPO) and evaluating conditions and then applying information from the T.S Bases.

Distractor Explanation
"Yes" is incorrect and plausible if the candidate cannot verify that the alarms are consistent with the plant conditions.
"14" days is incorrect and plausible because it is the answer for only 1 EDG INOP

Reference Information:
10D34 and 10D32
TS/B 3.8.1 (Provided)

NUREG 1123 KA Catalog Rev. 2

264000	Emergency Generators (Diesel/Jet)
G2.4.46	Ability to verify that the alarms are consistent with the plant conditions

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 Question Use (ILO 2018)
High
NEW
SRO

Associated objective(s):
Technical Specifications for Licensed Operators
Performance Enabler
Given plant conditions that constitute non-compliance with any LCO, apply Technical Specifications and Bases to determine the applicable Condition(s), Required Action(s), and associated Completion Time(s).

91	K/A Importance: 3.4/3.4			Points: 1.00
S91	Difficulty: 3.00	Level of Knowledge: High	Source: MODIFIED	73107

Reactor power at 74% when the B31005-F031A (A RR Pump Discharge Valve) open position switch failed causing a loss of its full open position indication.

(1) What is the expected plant response and (2) what shall the CRS direct?

A. (1) DCS Logic will run South RR MG Set back to the Limiter #1 setting.
(2) Direct LNO to:
- Take local manual control of North RR MG Set.
- Coordinate with the P603 opertor to match recirc flows.

B. (1) DCS Logic will run South RR MG Set back to the Limiter #4 setting.
(2) Direct P603 to:
- Verify Reactor Power < 66.1% or insert the Cram Array to lower Reactor Power to < 66.1%
- Increase core monitoring for instability.

C. (1) DCS Logic will run South RR MG Set back to the Limiter #1 setting.
(2) Direct P603 to:
- Verify Reactor Power < 66.1% or insert the Cram Array to lower Reactor Power to < 66.1%
- Increase core monitoring for instability.

D. (1) DCS Logic will run South RR MG Set back to the Limiter #4 setting.
(2) Direct LNO to:
- Take local manual control of North RR MG Set.
- Coordinate with the P603 opertor to match recirc flows.

Answer: B

Answer Explanation:
RR Limiter 4 is actuated by the North (South) RR MG Set drive motor or North (South) RR MG Set generator field breakers opening. The A RR MG set breaker will trip on interlock because of the discharge valve not being full open. Once the logic is cleared (fixed), the MG set can be restarted, and the limiter must be manually reset.

Distractor Explanation:
Distractor (1) Plausible because the examinee may not understand that the MG tripped off. RR Limiter 1 is actuated when B3105-F031A (B), N (S) RR Pump Discharge Vlv, is not fully open or total Feedwater flow is less than 20% of original rated value (approximately 3.15 Million lb./hr). It is however incorrect because this distractor is the South RR pump.
Distractor (2) Plausible and incorrect because the examinee may not understand that the MG tripped off. Based on this, matching flows is required by plant procedures and TS and therefore would be a priority for the CRS.

Reference Information:
23.138.01 Section 1.0 (pg 4-12)
20.138.01 Condition A (pg 3)

Plant Procedures
20.138.01
23.138.01

NUREG 1123 KA Catalog Rev. 2
202002 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:
202002 A2.01 Recirculation pump trip

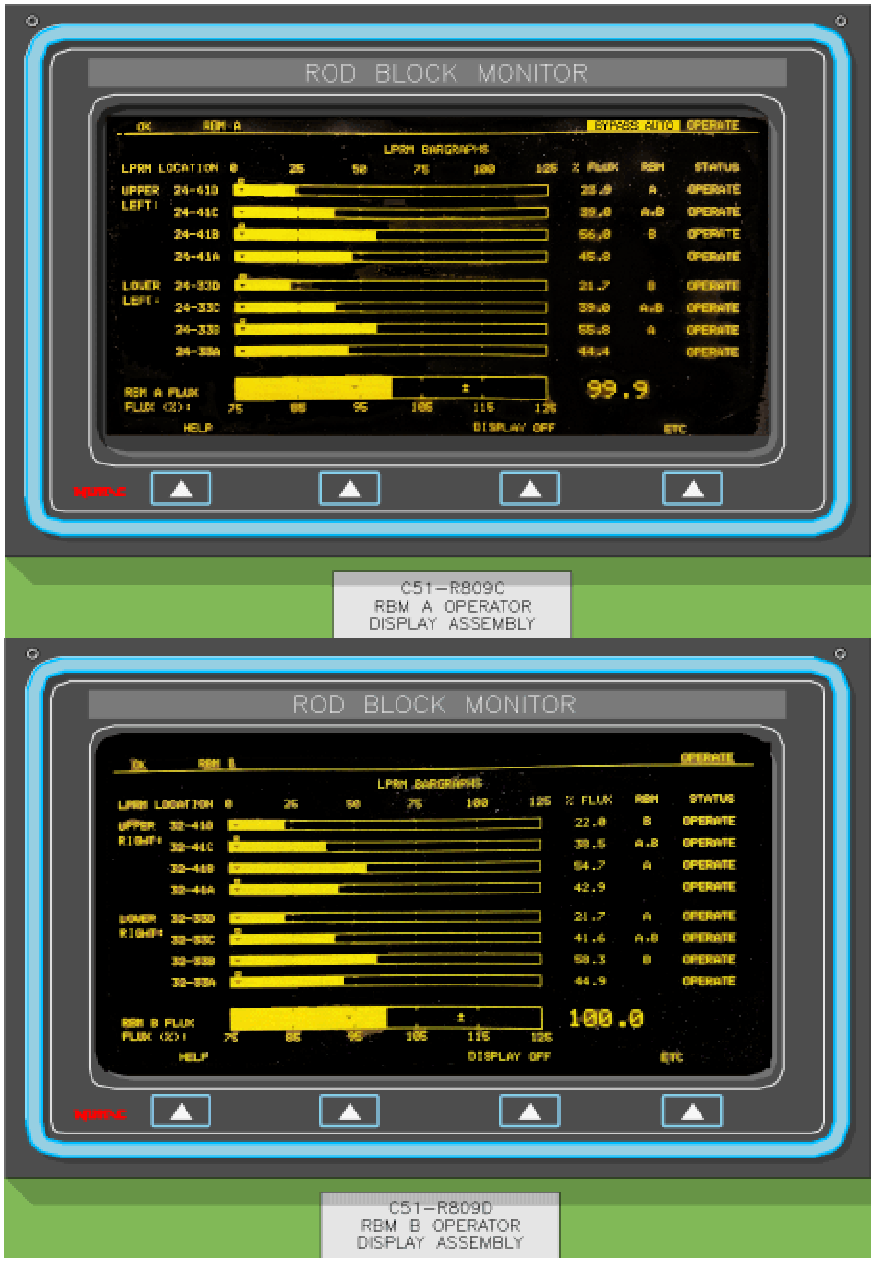
10CFR55 RO/SRO Written Exam Content
10 CFR 55.43(b) (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

NRC Question Use (ILO 2018)
High
MODIFIED
SRO

Associated objective(s):
Cycle 15-3 Objectives
Cognitive Terminal
Given a copy of 20.138.01, discuss licensed operator actions for a Recirculation Pump Trip

92	K/A Importance: 3.6/4.6			Points: 1.00
S92	Difficulty: 3.00	Level of Knowledge: High	Source: NEW	73166

All APRMs indicate 31% power. An interior control rod is currently selected.



Per Tech Specs which action, if any, is required?

	Action	Completion Time
A.	No action required.	
B.	Restore RBM channel A to OPERABLE status.	24 hours.
C.	Restore RBM channel B to OPERABLE status.	24 hours.
D.	Place one RBM channel in trip.	1 hour.

Answer: B

Answer Explanation:
Per SR 3.3.2.1.5 Verify the RBM is not bypassed when THERMAL POWER is > 30% RTP.
Per TS 3.3.2.1 Condition A Restore RBM channel to OPERABLE status. 24 hours.

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 specific knowledge of applying SR 3.3.2.1.5. The answer is related to < 1 hour TS/TRM Action however it cannot be solely answered with this information because OPERABILITY must first be determined using a SR. The answer cannot be answered LCO/TRM information listed "above-the-line" and cannot be answered by solely by knowing the TS Safety Limits. The SRO must assessing plant conditions and then evaluate operability based on SR 3.3.2.1.5. and then choose the correct action.

Distractor Explanation:
Distractors are incorrect and plausible based on answer explanation.
"No action required." is incorrect per TS 3.3.2.1 and plausible if candidate does not understand status of the RBM A
"Restore RBM channel B to OPERABLE status. 24 hours." is incorrect and because RBM is OPERABLE and plausible if the candidate does not understand status of the RBM A
"Place one RBM channel in trip. 1 hour." is incorrect and plausible because it is the correct action from T.S

Reference Information:
TS 3.3.2.1 & TSB 3.3.2.1

NUREG 1123 KA Catalog Rev. 2
215002 RBM System
G2.2.37 Ability to determine operability and/or availability of safety related equipment

Technical Specifications
3.3.2.1 Control Rod Block Instrumentation

10CFR55 RO/SRO Written Exam Content
10 CFR 55.43(b) (2) Facility operating limitations in the technical specifications and their bases.

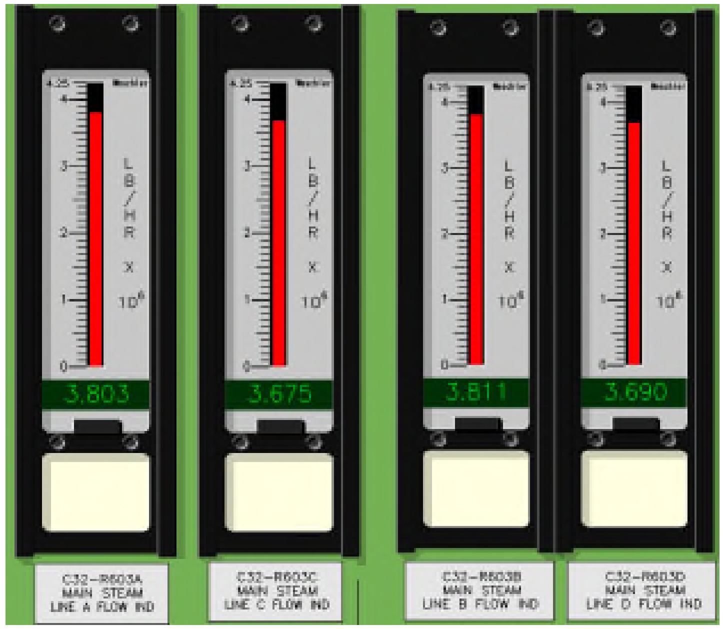
NRC Question Use (ILO 2018)
High
NEW
NRC Early Review
SRO

Associated objective(s):
Technical Specifications for Licensed Operators
Performance Enabler
Given plant conditions that constitute non-compliance with any LCO, apply Technical Specifications and Bases to determine the applicable Condition(s), Required Action(s), and associated Completion Time(s).

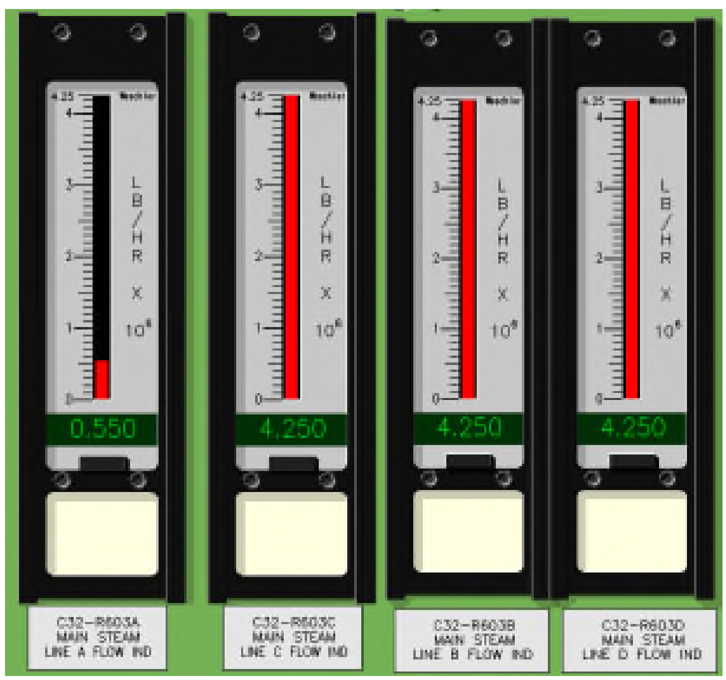
THIS PAGE LEFT BLANK

93	K/A Importance: 3.8/3.9			Points: 1.00
S93	Difficulty: 4.00	Level of Knowledge: High	Source: NEW	73788

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



The inboard MSIV for Main Steam Line A slow closes and plant indications are:



(1) What is the total steam flow going to the 52" manifold and (2) based on the MSIV configuration, the CRS will direct a change of reactor power to?

A. (1) ~14.98 E 06 LB/HR (2) 90%

B. (1) ~12.75 E 06 LB/HR (2) 90%

C. (1) ~12.75 E 06 LB/HR (2) 59%

D. (1) ~14.98 E 06 LB/HR (2) 59%

Answer: D

Answer Explanation:
A SLOW closure of a MSIV at power will result in a high reactor pressure. 3D168 will Alarm. Because the pressure regulator will maintain the same pressure in the 52" manifold, the Main Turbine continues with the same power output. This means that the same mass flow rate of steam is flowing through the turbine does not change To achieve this with 3 mainsteam lines pressure in the reactor vessel would have to rise to make up for the increase in steam velocity/mass flow rate going down 3 lines vice 4.

In this case the flow indications have a max value of 4.25 E 06, so they are indication high off scale.

Per 23.137 Reactor power must be less than 60% for a main steam line to be isolated.

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 operating limits of the Main Steam Lines, 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 applying the requirements of the SOPs.

Distractor Explanation:
~12.75 E 06 LB/HR is incorrect based on answer explaination. It is however the INDICATED flow based on the flow indication provided.

90% power is based on being lower than 91.5% power which is required per 23.109 Section 7.1 for closing off on a line to the High Pressure Turbine. However the answer is for a Main Steam line so this in incorrect.

Reference Information:
23.137 (Pg 15)

Plant Procedures
23.137

NUREG 1123 KA Catalog Rev. 2

239001 A2.	Ability to (a) predict the impacts of the following on the MAIN AND REHEAT STEAM SYSTEM ; and (b) based on those predictions, use procedures to correct, control,or mitigate the consequences of those abnormal conditions or operations:
239001 A2.10	Closure of one or more MSIV's at power

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 Question Use (ILO 2018)
High
NEW
NRC Early Review
SRO

Associated objective(s):
Nuclear Boiler System (B1100, B2100, B2103, B2104, N1100 & N3017)
Cognitive Terminal
Given various controls and indications for Nuclear Boiler System operations, analyze that indication to determine proper plant response in accordance with approved plant procedures.

THIS PAGE LEFT BLANK

94	K/A Importance: 3.9/4.2			**Points: 1.00**
S94	Difficulty: 2.00	Level of Knowledge: High	Source: BANK	73106

Following an Emergency Depressurization under ATWS conditions the following conditions exist:

- RPV Water Level indications are UPSCALE due to High Drywell and Reactor Building Temperatures.
- Injection has been Terminated and Prevented.
- 4 Safety Relief Valves are OPEN.
- Reactor Pressure is 300 psig.
- Reactor Power is 20%.

Under these conditions, which one of the following describes the status of Core Cooling?

Adequate Core Cooling is:

A. ASSURED by core submergence.

B. ASSURED by sufficient steam flow through OPEN Safety Relief Valves.

C. NOT ASSURED because injection is insufficient to cool the core.

D. NOT ASSURED because an inadequate number of Safety Relief Valves are OPEN.

Answer: B

Answer Explanation:
Per EOP/EPG, 29.100.01, Sheet 3A (provided to the candidates) with Reactor Pressure above 290 psig with 4 Safety Relief Valves OPEN, Adequate Core Cooling is ASSURED. This information is the basis for the Minimum Steam Cooling Pressure (MSCP), which is found on Table 8.

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 EOP Adequate Core Cooling conditions, 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 applying the requirements of the EOPs and knowing the bases of the EOPs/EPGs.

Distractor Explanation:
Distractors are incorrect and plausible because:
A. Would be true if UPSCALE RPV Water Level Instruments were valid.
C. If submergence were the only acceptable method of assuring Adequate Core Cooling, this would be correct. Terminated injection, long term, will not provide Adequate Core Cooling at 20% power. Long term issues are not addressed, the question pertains to “Under these conditions…”.
D. Would be true if Reactor Pressure were under 290 psig.

Reference Information:
BWROG EPGs/SAGs, Appendix B
(EOP) 29.100.01 SH 3A (provided)

Plant Procedures
29.100.01 SH 3

NUREG 1123 KA Catalog Rev. 2
216000 Nuclear Boiler System
G2.1.25 Ability to interpret station reference materials such as graphs, curves, tables, 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 Question Use (ILO 2018)
BANK
High
NRC Early Review
SRO

Associated objective(s):
EOP Objectives
Cognitive Terminal
Given a copy of 29.100.01 sheet 3, discuss the operator's actions for Steam Cooling (STC)

95	K/A Importance: 4.3/4.6			**Points: 1.00**
S95	Difficulty: 2.00	Level of Knowledge: Low	Source: BANK	73009

The plant is operating at 100%. The SM is reviewing the Plan of the Day (POD) for the upcoming week. Which one of the following evolutions would require coverage from a Reactivity Management SRO (RMSRO) who is not the STA/IA?

A. HPCI surveillance.

B. Turbine Valve Testing surveillance.

C. Power change required for CRD Operability surveillance.

D. Down power to support removal of Reactor Feed Pump from service.

Answer: D

Answer Explanation:
Per MOP-19, Large reactivity changes requiring a Reactivity Management Senior Reactor Operator (RMSRO) are planned changes to Control Rod position or Recirculation pump speed. MOP-19 then states that the RMSRO role is normally fulfilled by the on shift STA/IA with an active SRO license. MOP-19 goes on to list exceptions to the use of the STA/IA as the RMSRO. The candidate should recognize that the correct answer will involve a load drop of greater than 30% and restoration with major BOP equipment manipulations as being an exception to the use of the on shift STA/IA and therefore conclude that another individual will have to fulfill the RMSRO role.

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 Reactivity changes requiring a Reactivity Management Senior Reactor Operator (RMSRO), 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 a planned reactivity change and implementating a station administrative procedure correctly.

Distracter Explanation:
A/B/C are all incorrect because they are examples of Minor Reactivity Adjustments, generally defined to be less than 10%. They are all plausible because they are listed in MOP19-100 as reactivity adjustments and the candidate could incorrectly recall the list as requiring the use of a RMSRO other than the on-shift STA/IA. However, the examples identified in MOP19-100, section 5.4.2, will either not require a RMSRO or will allow the RMSRO function to be fulfilled by the on-shift STA/IA.

Reference Information:
MOP19, Section 4.3
MOP19-100, Section 5.4.2.

Plant Procedures
MOP19
ODE-01 Reactivity Management

NUREG 1123 KA Catalog Rev. 2

G2.1.37	Knowledge of procedures, guidelines, or limitations associated with reactivity management

NRC Question Use (ILO 2018)
BANK
Low
SRO

Associated objective(s):
Conservative Reactivity Management
Cognitive Terminal
Discuss Detroit Edison expectations on Conservative Reactivity Management as outlined in MOP19.

96	K/A Importance: 2.3/3.4			Points: 1.00
S96V2	Difficulty: 3.00	Level of Knowledge: Low	Source: NEW	76026

A surveillance testing work order and associated I&C surveillance procedure are being reviewed by a Shift's Field Support Supervisor (FSS), per MWC03 Surveillance/Performance Package Control. This review shall include, as a minimum, a review of:

A.
1. Plant configuration and mode
2. Required prerequisites
3. Impact statement (if required)

B.
1. Qualification of test personnel.
2. Required prerequisites
3. Impact statement (if required)

C.
1. Plant configuration and mode
2. Required prerequisites
3. Qualification of test personnel.

D.
1. Plant configuration and mode
2. Qualification of test personnel.
3. Impact statement (if required)

Answer: A

Answer Explanation:
Per MWC03
4.2.5 Review the surveillance procedure to ensure that it can be performed under the existing plant configuration without any adverse impact on the plant. This review shall include, as a minimum:
1.Plant configuration and mode
2.Required prerequisites
3.Impact statement (if required)

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 maintenance work order requirements from MWC03, it is not related to immediate actions, and the entry conditions are not relevant or leading to the answer. To answer this question, the examinee must know when to implement attachments and appendices, including how to coordinate these items with procedure steps. In this case MWC03 and surveillance testing work order review requirements.

Distractor Explanation:
Work hours and number of test personnel to complete is incorrect and plausible because, while it is reviewed, this review is done by planning. Work hour limits are a valid item that has federal limits.

Reference Information:
MWC03 4.2.5 (pg 11)

Plant Procedures
MWC03

NUREG 1123 KA Catalog Rev. 2
G2.2.19 Knowledge of maintenance work order requirements

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 Question Use (ILO 2018)
Low
NEW
SRO

Associated objective(s):
Administrative Conduct Manuals - Licensed Operators/STA
Performance Terminal
Describe the general requirements for controlling maintenance and modification activities, including:
a. Whose signature is required in order to begin maintenance.
b. What to do if special requirements are involved such as heavy loads, scaffolds, welding, grinding, fire permits, challenges to containment, or barriers.

97	K/A Importance: 3.2/4.2			Points: 1.00
S97	Difficulty: 3.00	Level of Knowledge: High	Source: BANK	73007

The plant is operating at 75% power with B RRMG Scoop Tube LOCKED, with the following recirculation parameters:

	Loop A	Loop B
RRMG set speed	63%	63%
Jet Pump Flow	36 Mlbm/hr	36 Mlbm/hr
Total Core Flow	72 Mlbm/hr	

Following a recirculation flow adjustment, power has lowered to 65% with the following recirculation parameters:

	Loop A	Loop B
RRMG set speed	49%	63%
Jet Pump Flow	27 Mlbm/hr	38 Mlbm/hr
Total Core Flow	63 Mlbm/hr	

Which of the responses below correctly completes the following statement?

These conditions ___(1)___ the Technical Specification 3.4.1, Recirculation Loops Operating LCO, which is based on assumptions which ___(2)___ .

A. (1) MEET
(2) prevent excessive component vibration in the reactor vessel

B. (1) DO NOT MEET
(2) preserve post accident core flow coastdown characteristics

C. (1) DO NOT MEET
(2) prevent excessive component vibration in the reactor vessel

D. (1) MEET
(2) preserve post accident core flow coastdown characteristics

Answer: B

Answer Explanation:
Tech Specs 3.4.1 Require:
SR 3.4.1.1 States:
Verify recirculation loop jet pump flow mismatch with both recirculation loops in operation is:

a. < 10% of rated core flow when operating at < 70% of rated core flow;
and
b. < 5% of rated core flow when operating at > 70% of rated core flow.

Following the adjustment core flow is <70% there fore the < 10% limit applies. Mismatch is > 10%, therefore LCO is NOT MET.

Therefore the conditions do not meet the LCO. The bases state its for post-accident core flow coast down. The internal vibrations is a concern when operating with one loop and the other pump speed is greater than 75%.

100% of rated core flow is $100x10^6$lb/hr. Therefore flows in the stem are:

	Loop A	Loop B
Jet Pump Flow	36 Mlbm/hr 36% of rated Core Flow	36 Mlbm/hr 36% of rated Core Flow
Jet Pump Flow	27 Mlbm/hr 27% of rated Core Flow	38 Mlbm/hr 38% of rated Core Flow

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 specific knowledge of T.S and how to apply actions. The answer is related immediate actions. However, to respond the examinee must determine operability first. The answer is not related to TS Safety Limits. The answer is relevant to the entry conditions but is not leading to the answer. While information found above the double line is needed, the correct understanding of the bases for the condition is required. The answer to this questions is based on assessing plant conditions (recirc flow) and evaluating conditions and then applying information from the T.S Bases.

Distractor Explanation:
Distractors are incorrect and plausible based on Answer Explanation and T.S Basis.

Distractor "prevent excessive component vibration in the reactor vessel" is a basis for another recirculation design requirement.

Reference Information:
TS & TSB 3.4.1

NUREG 1123 KA Catalog Rev. 2
202002 Recirculation Flow Control System
G2.2.25 Knowledge of bases in Technical Specifications for limiting conditions for operations and safety limits.

Technical Specifications
3.4.1 Recirculation Loops Operating

10CFR55 RO/SRO Written Exam Content
10 CFR 55.43(b) (2) Facility operating limitations in the technical specifications and their bases.

NRC Question Use (ILO 2018)
BANK
High
SRO

Associated objective(s):
Reactor Recirculation System
Cognitive Terminal
Analyze a component status change and describe its effect on Reactor Recirculation system operation. (understanding).

THIS PAGE LEFT BLANK

98	K/A Importance: 2.0/3.8			Points: 1.00
S98V2	Difficulty: 3.00	Level of Knowledge: Low	Source: BANK	75026

A discharge permit has been submitted to the Shift Manager for approval. The permit indicates the water to be discharged is potentially contaminated and will be discharged from an area subject to radiological restrictions. The actions required by MRP18, Release of Potentially Clean Fluids, have been completed.

Upon deciding to approve the discharge permit, the Shift Manager should assign an expiration not to exceed:

A. 4 hours

B. 24 hours

C. 5 days

D. 30 days

Answer: B

Answer Explanation:
Per MCE 06 NON-RADIOLOGICAL ENVIRONMENTAL PROTECTION, pumping of sources within an area subject to radiological restrictions should be assigned an expiration not to exceed 24 hours 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 radiation values used for a SRO responsibility to determine a time limit that is NOT a share responsibility with an RO or are values that are required to be known by the RO for radiological safety stay times.

Distractor Explanation:
A/C/D. Incorrect because of answer explanation and plausible because times listed are acceptable for other conditions in MCE 06.

Reference Information:
MCE06, 5.1.13

Plant Procedures
MCE06

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 2012 Exam

NRC Question Use (ILO 2018)
BANK
Low
SRO

Associated objective(s):
Senior Reactor Operator Qualification Card
Performance Terminal
Approve an effluent pump permit including determination of permit duration

99	K/A Importance: 2.4/4.4			**Points: 1.00**
S99	Difficulty: 3.00	Level of Knowledge: Low	Source: BANK	72987

An ALERT Emergency Action Level has been declared. Which facilities MUST be activated?

A. TSC ONLY.

B. TSC and OSC ONLY.

C. TSC, OSC, and EOF ONLY.

D. TSC, OSC, EOF, and JIC.

Answer: C

Answer Explanation:
Per EP-103, TSC, OSC, and EOF must be activated on step 4 of the Alert - Checklist for Immediate Actions.

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 Emergency Classification from EP-101, it is not related to immediate actions, and the entry conditions are not relevant or leading to the answer. To answer this question, the examinee must have knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps. In this case EP-103 and applicable ALERT Checklist.

Distractor Explanation:
Distractors are incorrect and plausible because:
A/B. The EOF must be activated for an ALERT per EP-103, and not for other classifications
D. The JIC is not activated until a SAE.

Reference Information:
EP-103, page 3, section 4.3
EP-103001, ALERT Checklist

Plant Procedures
EP-103

NUREG 1123 KA Catalog Rev. 2
G2.4.38 Ability to take actions called for in the facility emergency plan, including supporting or acting as emergency coordinator if required

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 2012 Exam

NRC Question Use (ILO 2018)
BANK
Low
SRO

Associated objective(s):
Emergency Procedures/ Plan
Cycle 12-1 Objectives
Cognitive Enabler
Explain the actions for an Alert

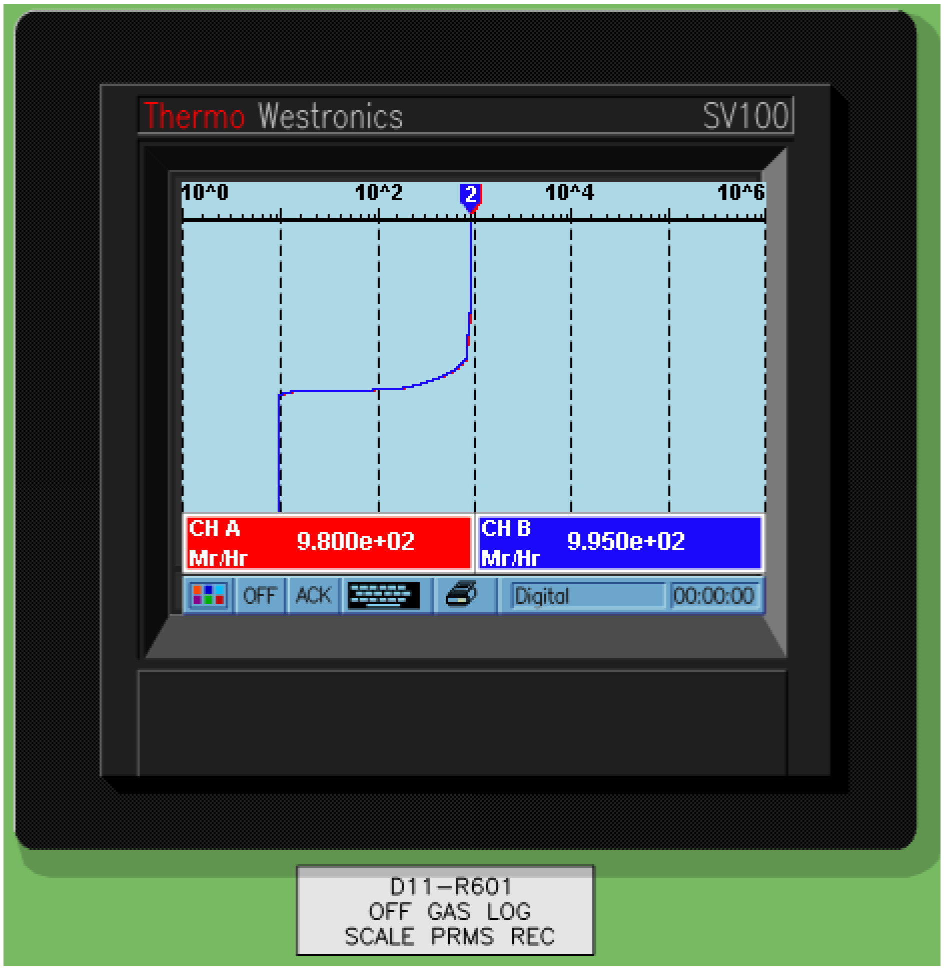
SRO Objectives
Cognitive Terminal
Given a copy of EP-103, discuss SRO actions for an Alert

100	K/A Importance: 4.2/4.2			Points: 1.00
S100V2	Difficulty: 2.00	Level of Knowledge: High	Source: MODIFIED	75926

While operating at 100% power.

- 3D8 DIV I/II OFF GAS RADN MONITOR UPSCALE alarms.
- Off gas flow rate is 105 SCFM.

And



Which of following is the CRS REQUIRED to direct?

A. Preform Rapid Power Reduction

B. Reduce Reactor Power using the GOP.

C. Reset the affected Radiation Monitor(s) in the Relay Room to clear 3D8.

D. Place the Mode Switch in Shutdown
AND
Close or Verify Closed MSIVs and main steam line drains.

Answer: B

Answer Explanation:
The calculated release rate using provided conversion sheet is 278 mCi/sec / 251 mCi/sec and RISING. Based on this indication levels are CURRENTLY bellow TS limits. Entry into the AOP 20.000.07 Fuel Cladding Failure is required by ARP 3D8.
The candidate must diagnose the trend and recognize that there is a small fuel cladding failure.
The only Actions statement that is met for AOP 20.000.07 is to reduce power using 22.000.03 (Condition B)

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 actions for a small fuel cladding failure, 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 applying the requirements of the AOP.

Distractor Explanation:
Distractors are incorrect and plausible because:
B. The ARP requires that affected Radiation Monitor(s) must be reset in the Relay Room to clear 3D8 only if less than the setpoint.
C. This is the Immediate action for a gross fuel failure. This is not a gross failure as per NOTE 1 on page 4 of 20.000.07
D. This is the Immediate action for a gross fuel failure with a few parts of conditions A. This is not a gross failure as per NOTE 1 on page 4 of 20.000.07

Reference Information:
AOP 20.000.07 pg 5
ARP 3D8

Plant Procedures
03D008
20.000.07

NUREG 1123 KA Catalog Rev. 2

G2.4.47	Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material

Technical Specifications
3.7.5 Main Condenser Offgas

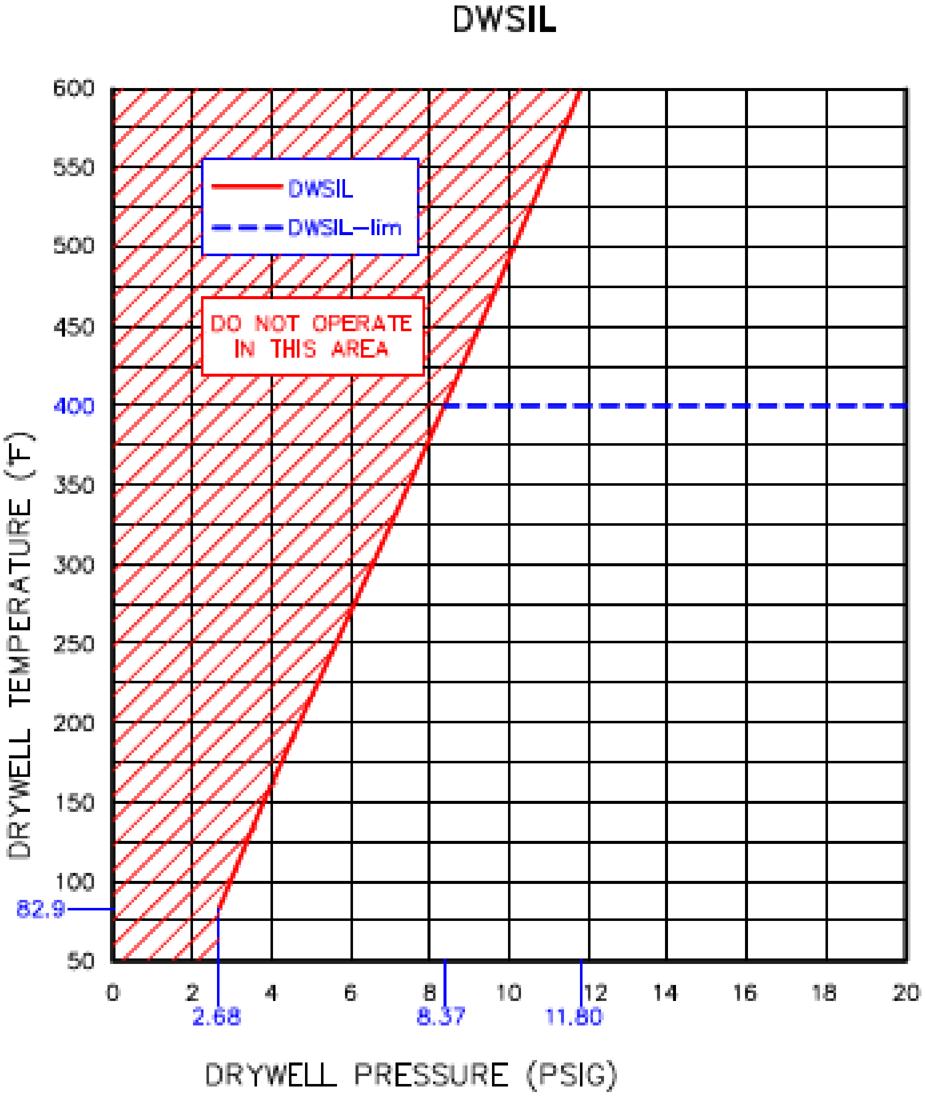
10CFR55 RO/SRO Written Exam Content

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 2012 Exam

NRC Question Use (ILO 2018)
High
MODIFIED
SRO

Associated objective(s):
Off Gas
Performance Terminal
Analyze Off Gas response using control room indications and annunciators.


DWSIL
DWSIL
DWSIL–lim
DO NOT OPERATE IN THIS AREA
DRYWELL TEMPERATURE (°F)
600
550
500
450
400
350
300
250
200
150
100
82.9
50
0
2
2.68
4
6
8
8.37
10
11.80
12
14
16
18
20
DRYWELL PRESSURE (PSIG)

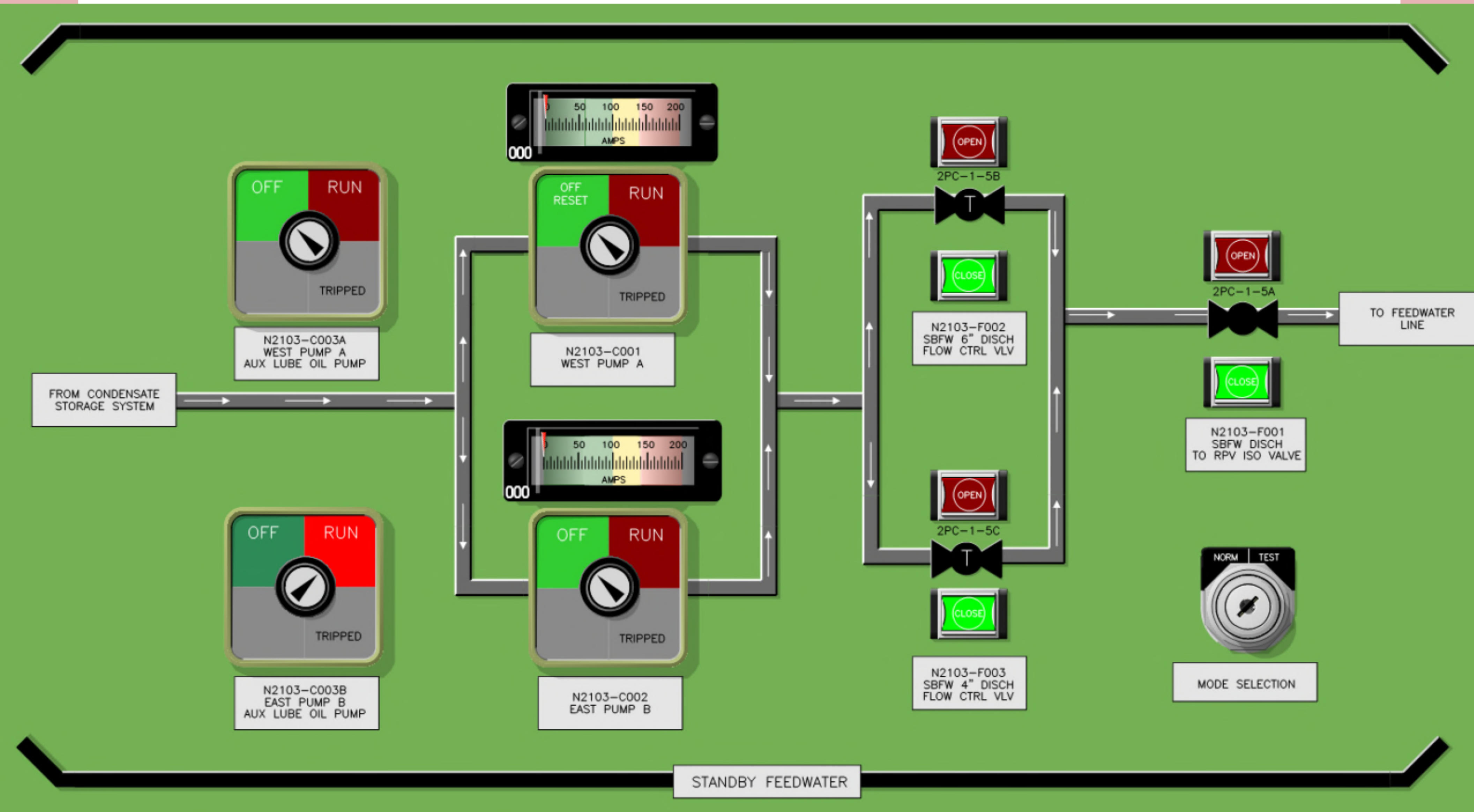
ROD PULL SHEET

Previous
Step No. 44

Sequence: **A002**

BPWS-Group/Subgroup: **2/0** RWM-Step: **03**

Step	ROD	Insert-Withdraw	P603 Initials	Verifier Initials	Notes
45	**42-07**	**08** to **12**	SMS	JVB	
46	**50-15**	**08** to **12**	SMS	JVB	IRM-G
47	**58-23**	**08** to **12**	SMS	JVB	
48	**58-39**	**08** to **12**	SMS	JVB	
49	**50-47**	**08** to **12**	SMS	JVB	
50	**42-55**	**08** to **12**	SMS	JVB	
51	**26-55**	**08** to **12**	SMS	JVB	
52	**18-47**	**08** to **12**	SMS	JVB	SRM-A
53	**10-39**	**08** to **12**	SMS	JVB	
54	**02-31**	**08** to **12**	SMS	JVB	
55	**10-23**	**08** to **12**	SMS	JVB	
56	**18-15**	**08** to **12**	SMS	JVB	IRM-H
57	**26-07**	**08** to **12**	SMS	JVB	
58	**34-15**	**08** to **12**	SMS	JVB	
59	**42-23**	**08** to **12**	SMS	JVB	SRM-C
60	**50-31**	**08** to **12**	SMS	JVB	
61	**42-39**	**08** to **12**	SMS	JVB	
62	**34-47**	**08** to **12**			
63	**26-39**	**08** to **12**			IRM-C
64	**18-31**	**08** to **12**			
65	**26-23**	**08** to **12**			
66	**34-31**	**08** to **12**			IRM-E



FROM CONDENSATE STORAGE SYSTEM
OFF
RUN
TRIPPED
N2103–C003A WEST PUMP A AUX LUBE OIL PUMP
OFF
RUN
TRIPPED
N2103–C003B EAST PUMP B AUX LUBE OIL PUMP
50 100 150 200
AMPS
000
OFF RESET
RUN
TRIPPED
N2103–C001 WEST PUMP A
50 100 150 200
AMPS
000
OFF
RUN
TRIPPED
N2103–C002 EAST PUMP B
OPEN
2PC–1–5B
T
CLOSE
N2103–F002 SBFW 6" DISCH FLOW CTRL VLV
OPEN
2PC–1–5C
T
CLOSE
N2103–F003 SBFW 4" DISCH FLOW CTRL VLV
OPEN
2PC–1–5A
CLOSE
N2103–F001 SBFW DISCH TO RPV ISO VALVE
TO FEEDWATER LINE
NORM
TEST
MODE SELECTION
STANDBY FEEDWATER

[redacted]

AUTO ACTIONS

NOTE: The system that had the critical failure will not automatically restart.

1. Failure of any software module designated as critical will cause an automatic failover of the system.

NOTE: A total loss of VAS indication on H11-P603 **OR** a loss of data acquisition MUX A and C may meet the loss of monitoring criteria for EP-101 EAL SU3.1, or SA3.1.

INITIAL RESPONSE

NOTE: Visual Annunciator System (VAS) Multiplexer (MUX) failure may occur that results in a continuous rebooting of the MUX or false alarms in the Control Room. In either case the MUX not working properly needs to be identified promptly. Upon receipt of alarm 3D19, ANNUNCIATOR SYSTEM TROUBLE, prompt identification of the associated MUX source can be performed using VAS screens as described.

1. Upon receipt of [redacted], alarm MUX related events will be noted by:

   - At least 2 SOERs will be generated VAS HARDWARE SYSTEM TROUBLE (848C97), VAS SOFTWARE SYSTEM TROUBLE (849C97).
   - Alarm Screen (TOC=ALARM) will display at the top part (VAS SYSTEM) specific alarm as to which MUX failed.
   - Alarm Screen (TOC=ALARM) will display at the bottom part (INPUT HEALTH) 1280 points in alarm (UNHEALTHY).
   - Annunciator Status screen (TOC=STATUS) will show the status of the failed MUX (realtime)
   - VAS System Software Trouble Overview (TOC=SWTRBL) will show the status for the failed MUX processor (realtime)
   - VAS System Hardware Trouble Overview (TOC=HWTRBL) will show the status of the Failed MUX and its associated Watchdog Timer Card status.

   If the initiating condition is a MUX failure and the system does not auto recover or fails again and does not recover within 15 minutes and Action A fails to recover the MUX, remove the failed MUX from service in accordance with 23.621, "Main Control Room Annunciator And Sequence Recorder."

[redacted]

[redacted]

NOTE: If the following steps do not correct the problem contact I&C / System Engineer for assistance.

2. Review the Annunciator System Trouble Overview (TOC=TROUBLE), Hardware System Trouble (TOC=HWTRBL), and Software System Trouble (TOC=SWTRBL) displays to determine system status. The attached Flow Chart may assist with Alarm diagnosis.

3. **IF** the initiating condition is Ground Fault Detection, perform 23.621 Attachment 4, Ground Isolation Procedure for VAS.

4. **IF** the initiating condition is a critical failure of a software module, restart the computer that automatically shutdown per 23.621, "Main Control Room Annunciator And Sequence Recorder."

5. **IF** the initiating condition is a non-critical failure of any software module, perform the following:

   - If the system is operating in the prime/back mode, perform a system failover per 23.621, "Main Control Room Annunciator And Sequence Recorder."

   - Verify software module restarted and is running properly on the new prime computer.

6. IF the initiating condition is a high temperature on any H11-P827 cabinet perform the following:

   - Open the cabinet doors (both front and rear)

   - Close or verify closed circuit breaker RL01-CB1.2 in the effected cabinet.

   - Verify the cabinet cooling fan is operating

7. IF the initiating condition is a Watchdog Timer Card perform the following:

   - Verify all circuit breakers in the effected cabinet are closed.

   - Verify all chassis in the effected cabinet are turned on

   - Verify at the affected MUX (G2) that power is on and the green light above the network connection is lit.

8. IF the initiating condition is a chassis tickle failure, reset the tickle lock-in per 23.621, "Main Control Room Annunciator And Sequence Recorder."

[redacted]

SUBSEQUENT ACTIONS

1. Document all conditions and actions on a CARD for tracking or correction.

INITIATING DEVICE/SETPOINTS

Ground Fault Detectors:

C97K051	C97K056
C97K052	C97K057
C97K053	C97K058
C97K054	C97K059
C97K055	C97K060

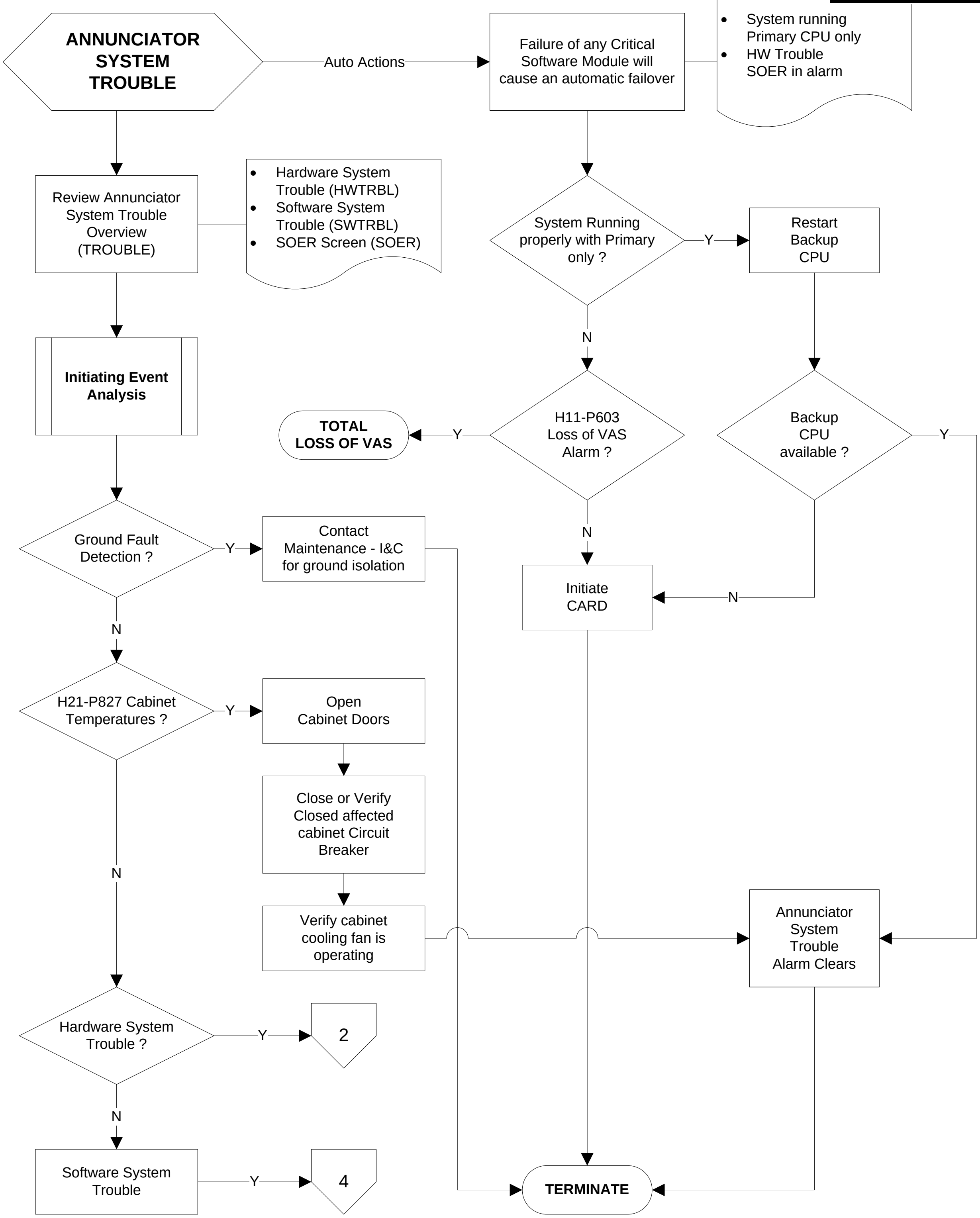
Cabinet High Temperature Detectors:

C97K062 (A1-HB01)	C97K132 (A8-HB01)
C97K070 (A2-HB01)	C97K078 (A9-HB01)
C97K110 (A3-HB01)	C97K086 (A10-HB01)

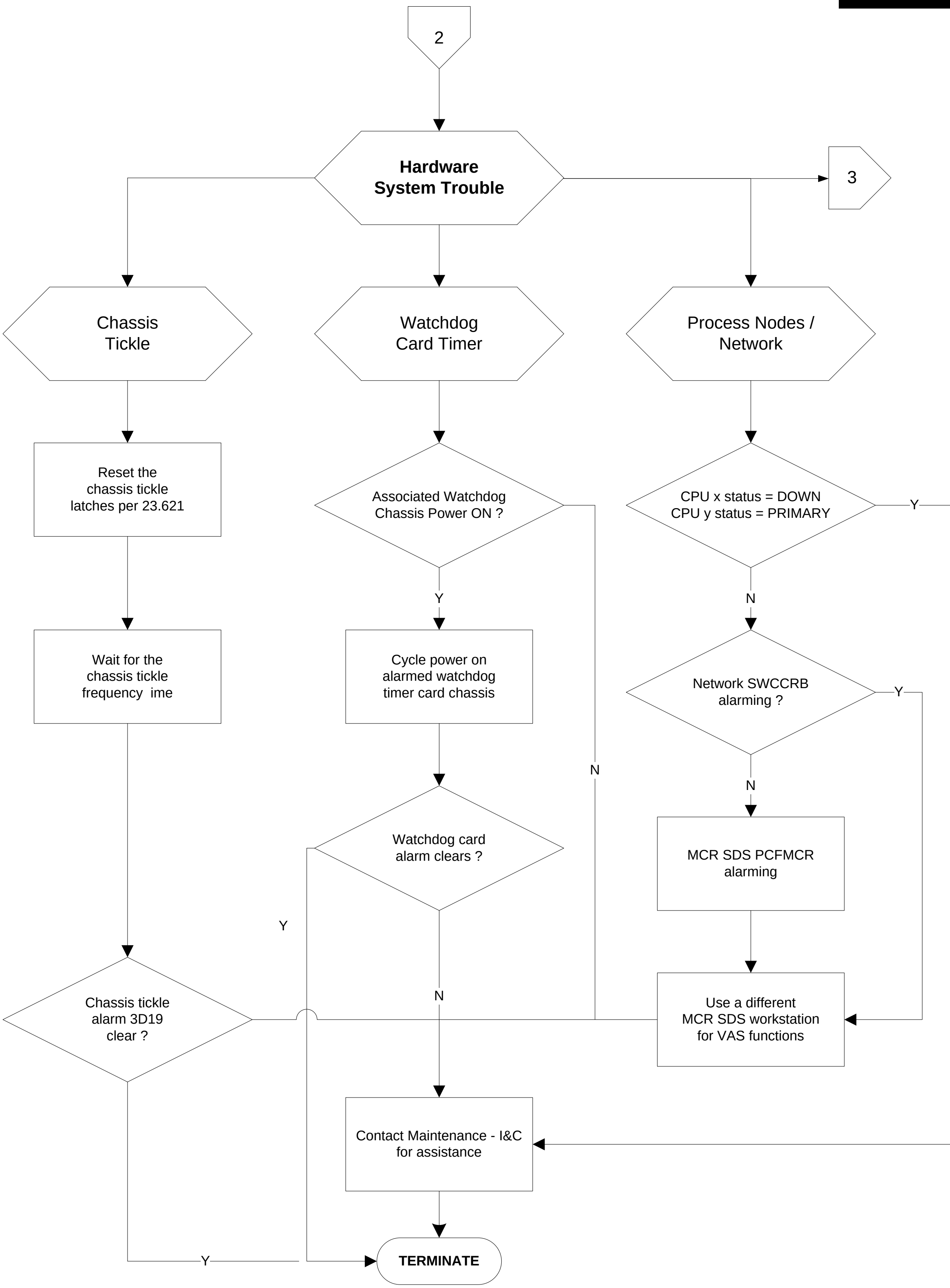
Internally generated see displays Hardware Trouble (TOC=HWTRBL) and Software Trouble (TOC=SWTRBL)

[redacted]

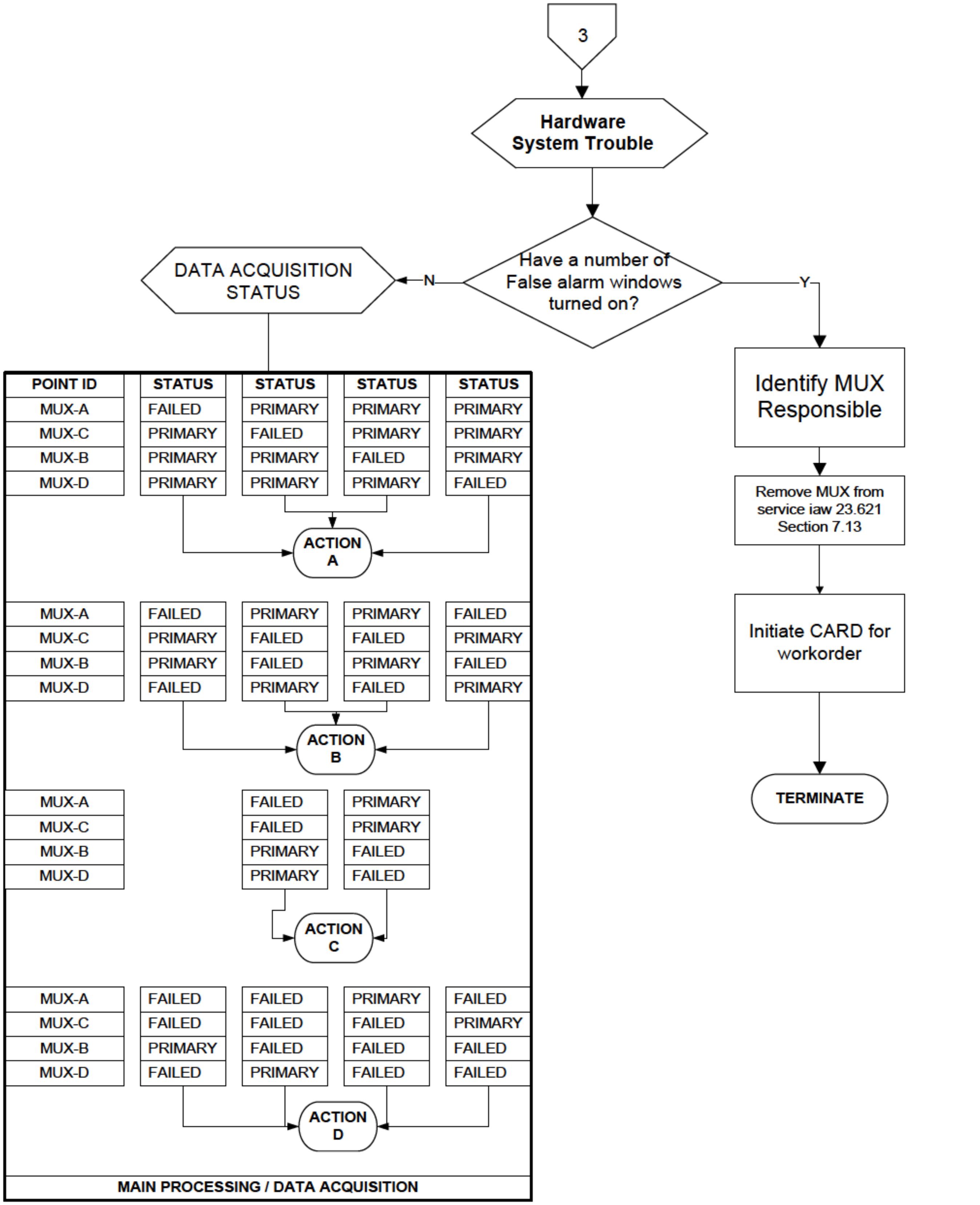
[redacted]


ANNUNCIATOR SYSTEM TROUBLE
Auto Actions
Failure of any Critical Software Module will cause an automatic failover
• System running Primary CPU only
• HW Trouble SOER in alarm
Review Annunciator System Trouble Overview (TROUBLE)
• Hardware System Trouble (HWTRBL)
• Software System Trouble (SWTRBL)
• SOER Screen (SOER)
System Running properly with Primary only ?
Y
Restart Backup CPU
N
Initiating Event Analysis
TOTAL LOSS OF VAS
Y
H11-P603 Loss of VAS Alarm ?
Backup CPU available ?
Y
Ground Fault Detection ?
Y
Contact Maintenance - I&C for ground isolation
N
Initiate CARD
N
N
H21-P827 Cabinet Temperatures ?
Y
Open Cabinet Doors
Close or Verify Closed affected cabinet Circuit Breaker
N
Verify cabinet cooling fan is operating
Annunciator System Trouble Alarm Clears
Hardware System Trouble ?
Y
2
N
Software System Trouble
Y
4
TERMINATE

[redacted]


2
Hardware
System Trouble
3
Chassis
Tickle
Watchdog
Card Timer
Process Nodes /
Network
Reset the
chassis tickle
latches per 23.621
Associated Watchdog
Chassis Power ON ?
CPU x status = DOWN
CPU y status = PRIMARY
Y
Y
N
Wait for the
chassis tickle
frequency ime
Cycle power on
alarmed watchdog
timer card chassis
Network SWCCRB
alarming ?
Y
N
N
Watchdog card
alarm clears ?
MCR SDS PCFMCR
alarming
Y
Chassis tickle
alarm 3D19
clear ?
N
Use a different
MCR SDS workstation
for VAS functions
Contact Maintenance - I&C
for assistance
Y
TERMINATE

[redacted]


3
Hardware System Trouble
Have a number of False alarm windows turned on?
N
Y
DATA ACQUISITION STATUS
Identify MUX Responsible
Remove MUX from service iaw 23.621 Section 7.13
Initiate CARD for workorder
TERMINATE
POINT ID
STATUS
STATUS
STATUS
STATUS
MUX-A
MUX-C
MUX-B
MUX-D
FAILED
PRIMARY
ACTION A
ACTION B
ACTION C
ACTION D
MAIN PROCESSING / DATA ACQUISITION

[redacted]

ACTIONS - A

ONE MUX FAILURE

- NO LOSS OF FUNCTIONALITY.
- LOSS OF REDUNDANCY FOR ONE HALF SIDE OF VAS I/O (MUX A/C or B/D pairs).
- ALARM SCREEN WILL SHOW LARGE AMOUNT OF HEALTH POINTS (~1280) ALARMING.
- IF THIS FAILURE IS AN INTERMITTENT COMMUNICATION OR NETWORK ISSUE, THE MUX SHOULD AUTOMATICALLY RECOVER (1 MINUTE). FAILURE OF THIS AUTO ACTION AFTER SEVERAL RE-TRIES WILL CAUSE THE MUX STATUS TO BE DELETED FROM PROCESSING.
- A HARD MUX FAILURE MAY BE RESET AS FOLLOWS:
  (1) PRESS THE RESET BUTTON ON MUX IN THE RR.
  (2) AFTER A MINUTE, FROM THE ANNSTAT, PRESS THE ASSOCIATED FAILED MUX “RESTORE MUX” BUTTON.
- INITIATE CARD AND DOCUMENT ACTIONS TAKEN.

ACTION- B

TWO MUX FAILURE – ONE PER REDUNDANT PAIR

- NO LOSS OF FUNCTIONALITY.
- LOSS OF REDUNDANCY FOR EACH SIDE OF VAS I/O (MUX A/C or B/D pairs).
- ALARM SCREEN WILL SHOW LARGE AMOUNT OF HEALTH POINTS (~2560) ALARMING.
- IF THIS FAILURE IS AN INTERMITTENT COMMUNICATION OR NETWORK ISSUE, THE MUX SHOULD AUTOMATICALLY RECOVER (1 MINUTE). FAILURE OF THIS AUTO ACTION AFTER SEVERAL RE-TRIES WILL CAUSE THE MUX STATUS TO BE DELETED FROM PROCESSING.
- A HARD MUX FAILURE MAY BE RESET AS FOLLOWS:
  (1) PRESS THE RESET BUTTON ON MUX IN THE RR.
  (2) AFTER A MINUTE, FROM THE ANNSTAT, PRESS THE ASSOCIATED FAILED MUX “RESTORE MUX” BUTTON.
- INITIATE CARD AND DOCUMENT ACTIONS TAKEN.

ACTION- C

TWO MUX FAILURE – REDUNDANT PAIR

- LOSS OF FUNCTIONALITY.
- ALARM SCREEN WILL SHOW LARGE AMOUNT OF HEALTH POINTS (~1280) ALARMING.
- A HARD MUX FAILURE MAY BE RESET AS FOLLOWS:
  (1) PRESS THE RESET BUTTON ON MUX IN THE RR.
  (2) AFTER A MINUTE, FROM THE ANNSTAT, PRESS THE ASSOCIATED FAILED MUX “RESTORE MUX” BUTTON.
- LOSS OF MUX A/C PAIR CAUSES P601 THROUGH P805 WINDOW FAILURES. LOSS OF MUX B/D PAIR CAUSES P806 THROUGH P811 WINDOW FAILURES.
- REVIEW OF INPUTS AFFECTED AND THEIR IMPACT ON THE REMAINING OPERATING WINDOWS CAN BE DETERMINED FROM THE “LOOKUP” VIEW USING *THE INPUT HEALTH PROBLEM* BUTTON.
- CONTACT SYSTEM ENGINEER AND I&C
- INITIATE CARD AND DOCUMENT ACTIONS TAKEN.

ACTION- D

THREE MUX FAILURE – SINGLE MUX AVAILABLE

- LOSS OF FUNCTIONALITY.
- ALARM SCREEN WILL SHOW LARGE AMOUNT OF HEALTH POINTS (~2560) ALARMING.
- A HARD MUX FAILURE MAY BE RESET AS FOLLOWS:
  (1) PRESS THE RESET BUTTON ON MUX IN THE RR.
  (2) AFTER A MINUTE, FROM THE ANNSTAT, PRESS THE ASSOCIATED FAILED MUX “RESTORE MUX” BUTTON.
- LOSS OF MUX A/C PAIR CAUSES P601 THROUGH P805 WINDOW FAILURES. LOSS OF MUX B/D PAIR CAUSES P806 THROUGH P811 WINDOW FAILURES.
- MUX A AND B, MUX C AND D ARE FED FROM SEPARATE POWER SOURCES.
- CONTACT SYSTEM ENGINEER AND I&C
- INITIATE CARD AND DOCUMENT ACTIONS TAKEN.

NOTES

- ANNUNCIATOR SYSTEM ALARM ACKNOWLEDGE AND SILENCE PUSHBUTTONS ARE PROCESS BY MUX PAIR B/D.
- ANNUNCIATOR SYSTEM HORNS ARE PROCESS BY MUX PAIR A/C.
- 3D19 – VAS SYSTEM TROUBLE ALARM IS PROCESSED BY MUX PAIR A/C.
- LOSS OF ALL FOUR MUXES (A,B,C,D) WILL RESULT IN A TOTAL LOSS OF VAS ALARM ON H11P603.

[redacted]

4

Software System Trouble

POINT ID	DESCRIPTION	ACTION
AMLOGSOEZ	SOER LOG MANAGER	CONT-SE
LRLOGSOEZ	SOER LOG DAILY MANAGER	CONT-SE
LRVSANAZ	SOER DAILY ANALYSIS REPORT	CONT-SE
EXSORECVZ	SOE READ DATA SENT TO BACK	CONT-SE
EXSOSENDZ	SOE SEND DATA TO BACK	CONT-SE
EXSRSENDZ	SOE SEND TEXT BUFFERS TO BACK	CONT-SE
ALEVENTZ	EVENT MESSAGE PROCESSING	CONT-SE
AMLOGZ	ALARM LOG MANAGER	CONT-SE
AMMGRZ	ALARM PROCESSOR	CONT-SE
AMPRINTZ	ALARM SCREEN REPORT MANAGER	CONT-SE
AMPRTLOGZ	ALARM LOG DAILY REPORT	CONT-SE
SOER / ALARM LOG		

POINT ID	DESCRIPTION	ACTION
DAPPARCZ	POINT PROCESSOR	DUAL-FAIL
DATRNZ	TRANSFORM PROCESSOR	CONT-SE
DACHNARCZ	SHORT TERM ARCHIVE	CONT-SE
DADSMGRZ	DEMAND SCAN MANAGER	CONT-SE
EXTIMERDZ	SYSTEM TIME SYNCHRONIZER	CONT-SE
DAMUXZ001	DAS MUX-A PROCESSOR	HDWR
DAMUXZ004	DAS MUX-C PROCESSOR	HDWR
DAMUXZ002	DAS MUX-B PROCESSOR	HDWR
DAMUXZ005	DAS MUX-D PROCESSOR	HDWR
DAFRENDZ	DAS MAIN TASK	AUTO
DAOUTZ	DIGITAL OUTPUT PROCESSING	AUTO
DAG2DBZ	DAS G2 DATABASE GENERATOR	CONT-SE
EXCKDASZ	DAS POINT DIAGNOSTIC	CONT-SE
EXG2PWRZ	G2 DO CHASSIS AUTO POWER CYCLE	CONT-SE
MAIN PROCESSING / DATA ACQUISITION		

POINT ID	DESCRIPTION	ACTION
DAFOZ	ALARM FIRST OUT	CONT-SE
LRVFRELZ	FREE FORMAT LOG MANAGER	CONT-SE
EXTHRBOUZ	FAILOVER / BOUNCE THRU MANAGER	CONT-SE
DABACRECZ	DAS BACK HOST RECOVERY	CONT-SE
EXHEALTHZ	SYSTEM HEALTH MONITORING	FAILOVER
EXSU1SENZ	VAS TO PPC DATA TRANSFER	FAILOVER
EXSYSAVLZ	SYSTEM AVAILABILITY CALCS	CONT-SE
EXARSVBAZ	SHORT TERM ARCHIVE CATCH UP	CONT-SE
EXPINGERZ	SYSTEM DEV / NETWORK MONITOR	CONT-SE
EXALIVEZ	SYSTEM 10-CPS KEEP ALIVE	CONT-SE
EXABSENDZ	PRIM SEND DATA TO BACK	CONT-SE
EXQU2BACZ	PRIM SEND / RECEIVE DATA QUEUE	CONT-SE
EXBACPRIZ	PRIM / BACK DATA EXCHANGE	CONT-SE
EXDT2BACZ	PROCESS REQUESTS FROM BAC/PRI	CONT-SE
EXGATEBUZ	PRIM GATEWAY DATA TO BACK	CONT-SE
EXGETPRIZ	PRIM / BACK PROC FIELD DATA	CONT-SE
EXHOTQUEZ	PRIM STAGNANT ENTRY MONITOR	CONT-SE
EXPRIBACZ	PRIM / BACK STATE EXCHANGE	CONT-SE
EXPUTBACZ	PRIM / BACK DATA SERVER	CONT-SE
EXABRECVZ	RECEIVE DATA FROM EXABSENDZ	CONT-SE
EXQLSERVZ	QUICKLOOK DATA MANAGER	CONT-SE
LRDBCRZ	DATABASE CHANGE LOG REPORT	CONT-SE
LRQUEFLZ	PRINTER PRIM / ALT FAILOVER	CONT-SE
SUPPORTING FUNCTIONS		

ACTIONS
AUTO: AUTOMATIC SYSTEM FAILOVER.
FAILOVER: PERFORM A SYSTEM FAILOVER (TOC=FAIL).
CONT-SE: MINIMAL IMPACT TO SYSTEM FUNCTIONS. INITIATE CARD.
HDWR: USE HARDWARE SYSTEM TROUBLE FLOW CHART FOR DETERMINING ACTIONS
DUAL-FAIL: DAPPARCZ FAILURE COULD BE COMMON TO BOTH PRIMARY AND BACKUP SYSTEMS. A FAILOVER (TOC=FAIL) SHOULD BE INITIATED IMMEDIATELY. IF FOLLOWING THAT FAILOVER, THE POINT IS STILL ALARMING, A SECOND FAILOVER SHOULD BE PERFORMED WHEN THE BACKUP CPU BECOMES AVAILABLE.

END

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.3 Safety Relief Valves (SRVs)

LCO 3.4.3 The safety function of 11 SRVs shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required SRVs inoperable.	A.1 Be in MODE 3.	12 hours
	AND	
	A.2 Be in MODE 4.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.3.1	Verify the safety function lift setpoints of the required SRVs are as follows: Number of SRVs — Setpoint (psig) 5 — 1135 ± 34.05 5 — 1145 ± 34.35 5 — 1155 ± 34.65 Following testing, lift settings shall be within ± 1%.	In accordance with the Inservice Testing Program
SR 3.4.3.2	Verify each required SRV is capable of being opened.	In accordance with the Surveillance Frequency Control Program

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.6 Battery Cell Parameters

BASES

BACKGROUND

This LCO delineates the limits on electrolyte temperature, level, float voltage, and specific gravity for the DC electrical power subsystems batteries. A discussion of these batteries and their OPERABILITY requirements is provided in the Bases for LCO 3.8.4, "DC Sources – Operating," and LCO 3.8.5, "DC Sources – Shutdown."

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in UFSAR, Chapter 6 (Ref. 1) and Chapter 15 (Ref. 2), assume Engineered Safety Feature systems are OPERABLE. The DC electrical power subsystems provide normal and emergency DC electrical power for the emergency diesel generators (EDGs), emergency auxiliaries, and control and switching during all MODES of operation.

The OPERABILITY of the DC subsystems is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the unit. This includes maintaining at least one division of DC sources OPERABLE during accident conditions, in the event of:

a. An assumed loss of all offsite AC or all onsite AC power; and

b. A worst case single failure.

Since battery cell parameters support the operation of the DC electrical power subsystems, they satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Battery cell parameters must remain within acceptable limits to ensure availability of the required DC power to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence or a postulated DBA. Electrolyte limits are conservatively established, allowing continued DC electrical system function even with Category A and B limits not met.

BASES

APPLICABILITY

The battery cell parameters are required solely for the support of the associated DC electrical power subsystem. Therefore, battery electrolyte is only required when the DC power source is required to be OPERABLE. Refer to the Applicability discussions in Bases for LCO 3.8.4 and LCO 3.8.5.

ACTIONS

A.1, A.2, and A.3

With parameters of one or more cells in one or more batteries not within limits (i.e., Category A limits not met or Category B limits not met, or Category A and B limits not met) but within the Category C limits specified in Table 3.8.6-1, the battery is degraded but there is still sufficient capacity to perform the intended function. Therefore, the affected battery is not required to be considered inoperable solely as a result of Category A or B limits not met, and continued operation is permitted for a limited period.

The pilot cell electrolyte level and float voltage are required to be verified to meet the Category C limits within 1 hour (Required Action A.1). This check provides a quick indication of the status of the remainder of the battery cells. One hour provides time to inspect the electrolyte level and to confirm the float voltage of the pilot cells. One hour is considered a reasonable amount of time to perform the required verification.

Verification that the Category C limits are met (Required Action A.2) provides assurance that during the time needed to restore the parameters to the Category A and B limits, the battery is still capable of performing its intended function. A period of 24 hours is allowed to complete the initial verification because specific gravity measurements must be obtained for each connected cell. Taking into consideration both the time required to perform the required verification and the assurance that the battery cell parameters are not severely degraded, this time is considered reasonable. The verification is repeated at 7 day intervals until the parameters are restored to Category A and B limits.

BASES

ACTIONS (continued)

Continued operation is only permitted for 31 days before battery cell parameters must be restored to within Category A and B limits. Taking into consideration that, while battery capacity is degraded, sufficient capacity exists to perform the intended function and to allow time to fully restore the battery cell parameters to normal limits, this time is acceptable for operation prior to declaring the DC batteries inoperable.

B.1

When any battery parameter is outside the Category C limit for any connected cell, sufficient capacity to supply the maximum expected load requirement is not ensured and the corresponding DC electrical power subsystem must be declared inoperable. Additionally, other potentially extreme conditions, such as not completing the Required Actions of Condition A within the required Completion Time or average electrolyte temperature of representative cells falling below 70°F, also are cause for immediately declaring the associated DC electrical power subsystem inoperable.

SURVEILLANCE REQUIREMENTS

SR 3.8.6.1

This SR verifies that Category A battery cell parameters are consistent with IEEE-450 (Ref. 3), which recommends regular battery inspections including voltage, specific gravity, and electrolyte temperature of pilot cells. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.8.6.2

The inspection of specific gravity and voltage is consistent with IEEE-450 (Ref. 3). The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. In addition, within 24 hours of a battery discharge < 105 V or a battery overcharge > 145 V, the battery must be demonstrated to meet Category B limits. Transients, such as motor starting transients, which may momentarily cause battery voltage to drop to ≤ 105 V, do not constitute a battery discharge provided the battery terminal voltage and float current

BASES

SURVEILLANCE REQUIREMENTS (continued)

return to pre-transient values. This inspection is also consistent with IEEE-450 (Ref. 3), which recommends special inspections following a severe discharge or overcharge, to ensure that no significant degradation of the battery occurs as a consequence of such discharge or overcharge.

SR 3.8.6.3

This Surveillance verification that the average temperature of representative cells (i.e., selection of 10 connected cells) is within limits is consistent with a recommendation of IEEE-450 (Ref. 3) that states that the temperature of electrolytes in representative cells should be determined. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

Lower than normal temperatures act to inhibit or reduce battery capacity. This SR ensures that the operating temperatures remain within an acceptable operating range. This limit is based on manufacturer's recommendations.

Table 3.8.6-1

This table delineates the limits on electrolyte level, float voltage, and specific gravity for three different categories. The meaning of each category is discussed below.

Category A defines the normal parameter limit for each designed pilot cell in each battery. The cells selected as pilot cells are those whose temperature, voltage, and electrolyte specific gravity approximate the state of charge of the entire battery.

The Category A limits specified for electrolyte level are based on manufacturer's recommendations and are consistent with the guidance in IEEE-450 (Ref. 3), with the extra ¼ inch allowance above the high water level indication for operating margin to account for temperature and charge effects. In addition to this allowance, footnote (a) to Table 3.8.6-1 permits the electrolyte level to be above the specified maximum level during equalizing charge, provided

BASES

SURVEILLANCE REQUIREMENTS (continued)

it is not overflowing. These limits ensure that the plates suffer no physical damage, and that adequate electron transfer capability is maintained in the event of transient conditions. IEEE-450 (Ref. 3) recommends that electrolyte level readings should be made only after the battery has been at float charge for at least 72 hours.

The Category A limit specified for float voltage is ≥ 2.13 V per cell. This value is based on the recommendation of IEEE-450 (Ref. 3), which states that prolonged operation of cells below 2.13 V can reduce the life expectancy of cells. The Category A limit specified for specific gravity for each pilot cell is ≥ 1.195 (0.015 below the manufacturer's fully charged nominal specific gravity or a battery charging current that had stabilized at a low value). This value is characteristic of a charged cell with adequate capacity. According to IEEE-450 (Ref. 3), the specific gravity readings are based on a temperature of 77°F (25°C).

The specific gravity readings are corrected for actual electrolyte temperature and level. For each 3°F (1.67°C) above 77°F (25°C), 1 point (0.001) is added to the reading; 1 point is subtracted for each 3°F below 77°F. The specific gravity of the electrolyte in a cell increases with a loss of water due to electrolysis or evaporation. Level correction will be in accordance with manufacturer's recommendations.

Category B defines the normal parameter limits for each connected cell. The term "connected cell" excludes any battery cell that may be jumpered out.

The Category B limits specified for electrolyte level and float voltage are the same as those specified for Category A and have been discussed above. The Category B limit specified for specific gravity for each connected cell is ≥ 1.190 (0.020 below the manufacturer's fully charged, nominal specific gravity) with the average of all connected cells 1.200 (0.010 below the manufacturer's fully charged, nominal specific gravity). These values are based on manufacturer's recommendations. The minimum specific gravity value required for each cell ensures that the effects of a highly charged or newly installed cell do not mask overall degradation of the battery.

BASES

SURVEILLANCE REQUIREMENTS (continued)

Category C defines the limits for each connected cell. These values, although reduced, provide assurance that sufficient capacity exists to perform the intended function and maintain a margin of safety. When any battery parameter is outside the Category C limits, the assurance of sufficient capacity described above no longer exists, and the battery must be declared inoperable.

The Category C limit specified for electrolyte level (above the top of the plates and not overflowing) ensures that the plates suffer no physical damage and maintain adequate electron transfer capability. The Category C Allowable Value for voltage is based on IEEE-450 (Ref. 3), which states that a cell voltage of 2.07 V or below, under float conditions and not caused by elevated temperature of the cell, indicates internal cell problems and may require cell replacement.

The Category C limit on average specific gravity ≥ 1.190, is based on manufacturer's recommendations (0.020 below the manufacturer's recommended fully charged, nominal specific gravity). In addition to that limit, it is required that the specific gravity for each connected cell must be no less than 0.020 below the average of all connected cells. This limit ensures that the effect of a highly charged or new cell does not mask overall degradation of the battery.

Footnote (b) to Table 3.8.6-1 is applicable to Category A, B, and C specific gravity. Footnote (b) of Table 3.8.6-1 requires the above mentioned correction for electrolyte level and temperature.

Because of specific gravity gradients that are produced during the recharging process, delays of several days may occur while waiting for the specific gravity to stabilize. A stabilized charger current is an acceptable alternative to specific gravity measurement for determining the state of charge of the designated pilot cell. This phenomenon is discussed in IEEE-450 (Ref. 3). Footnote (c) to Table 3.8.6-1 is applicable to Category A and C, and allows the float charge current to be used as an alternate to specific gravity.

BASES

REFERENCES

1. UFSAR, Chapter 6.

2. UFSAR, Chapter 15.

3. IEEE Standard 450, 1987.

3.8 ELECTRICAL POWER SYSTEMS

3.8.4 DC Sources-Operating

LCO 3.8.4 The Division I and Division II DC electrical power subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One battery charger inoperable.	A.1 Restore battery charger to OPERABLE status.	4 hours
B. One DC electrical power subsystem inoperable for reasons other than Condition A.	B.1 Restore DC electrical power subsystem to OPERABLE status.	2 hours
C. Required Action and Associated Completion Time not met.	C.1 --------NOTE-------- LCO 3.0.4.a is not applicable when entering MODE 3. -------------------- Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.4.1	Verify battery terminal voltage is ≥ 125.7 V on float charge.	In accordance with the Surveillance Frequency Control Program
SR 3.8.4.2	Verify no visible corrosion at battery terminals and connectors. OR Verify each battery: a. Cell-to-cell and terminal connection resistance is ≤ 1.5E-4 ohm; and b. Total cell-to-cell and terminal connection resistance is ≤ 2.7E-3 ohm.	In accordance with the Surveillance Frequency Control Program
SR 3.8.4.3	Verify battery cells, cell plates, and racks show no visual indication of physical damage or abnormal deterioration that could degrade battery performance.	In accordance with the Surveillance Frequency Control Program
SR 3.8.4.4	Remove visible corrosion and verify battery cell to cell and terminal connections are coated with anti-corrosion material.	In accordance with the Surveillance Frequency Control Program
SR 3.8.4.5	Verify each battery: a. Cell-to-cell and terminal connection resistance is ≤ 1.5E-4 ohm; and b. Total cell-to-cell and terminal connection resistance is ≤ 2.7E-3 ohm.	In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.4.6	Verify each required battery charger supplies ≥ 100 amps at ≥ 124.7 V for ≥ 4 hours.	In accordance with the Surveillance Frequency Control Program
SR 3.8.4.7	-------------------NOTE------------------ The performance discharge test in SR 3.8.4.8 may be performed in lieu of the service test in SR 3.8.4.7. -- Verify battery capacity is adequate to supply, and maintain in OPERABLE status, the actual or simulated emergency loads for the design duty cycle when subjected to a battery service test.	In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.4.8	-------------------NOTE------------------- This Surveillance shall not be performed in MODE 1, 2, or 3. However, credit may be taken for unplanned events that satisfy this SR. --	
	Verify battery capacity is ≥ 80% of the manufacturer's rating when subjected to a performance discharge test.	In accordance with the Surveillance Frequency Control Program AND 12 months when battery shows degradation or has reached 85% of expected life with capacity < 100% of manufacturer's rating AND 24 months when battery has reached 85% of the expected life with capacity ≥ 100% of manufacturer's rating

3.8 ELECTRICAL POWER SYSTEMS

3.8.6 Battery Cell Parameters

LCO 3.8.6 Battery cell parameters for the Division I and Division II batteries shall be within limits.

APPLICABILITY: When associated DC electrical power subsystems are required to be OPERABLE.

ACTIONS

------------------------------------NOTE------------------------------------
Separate Condition entry is allowed for each battery.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more batteries with one or more battery cell parameters not within Table 3.8.6-1 Category A or B limits.	A.1	Verify pilot cells electrolyte level and float voltage meet Table 3.8.6-1 Category C limits.	1 hour
	<u>AND</u>		
	A.2	Verify battery cell parameters meet Table 3.8.6-1 Category C limits.	24 hours <u>AND</u> Once per 7 days thereafter
	<u>AND</u>		
	A.3	Restore battery cell parameters to Table 3.8.6-1 Category A and B limits.	31 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time of Condition A not met. <u>OR</u> One or more batteries with average electrolyte temperature of the representative cells not within limits. <u>OR</u> One or more batteries with one or more battery cell parameters not within Table 3.8.6-1 Category C values.	B.1 Declare associated battery inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.6.1	Verify battery cell parameters meet Table 3.8.6-1 Category A limits.	In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.6.2	Verify battery cell parameters meet Table 3.8.6-1 Category B limits.	In accordance with the Surveillance Frequency Control Program AND Once within 24 hours after battery discharge < 105 V AND Once within 24 hours after battery overcharge > 145 V
SR 3.8.6.3	Verify average electrolyte temperature of representative cells is > 70°F.	In accordance with the Surveillance Frequency Control Program

Table 3.8.6-1 (page 1 of 1)
Battery Cell Parameter Requirements

PARAMETER	CATEGORY A: LIMITS FOR EACH DESIGNATED PILOT CELL	CATEGORY B: LIMITS FOR EACH CONNECTED CELL	CATEGORY C: ALLOWABLE LIMITS FOR EACH CONNECTED CELL
Electrolyte Level	> Minimum level indication mark, and ≤ ¼ inch above maximum level indication mark[(a)]	> Minimum level indication mark, and ≤ ¼ inch above maximum level indication mark[(a)]	Above top of plates, and not overflowing
Float Voltage	≥ 2.13 V	≥ 2.13 V	> 2.07 V
Specific Gravity[(b)]	≥ 1.195[(c)]	≥ 1.190 AND Average of all connected cells > 1.200	Not more than 0.020 below average of all connected cells AND Average of all connected cells ≥ 1.190[(c)]

(a) It is acceptable for the electrolyte level to temporarily increase above the specified maximum level during equalizing charges provided it is not overflowing.

(b) Corrected for electrolyte temperature and level.

(c) A battery charging current of < 2 amps when on float charge is acceptable for meeting specific gravity limits.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.4 DC Sources—Operating

BASES

BACKGROUND

The DC electrical power system provides the AC emergency power system with control power. It also provides both motive and control power to selected safety related equipment. As required by 10 CFR 50, Appendix A, GDC 17 (Ref. 1), the DC electrical power system is designed to have sufficient independence, redundancy, and testability to perform its safety functions, assuming a single failure. The DC electrical power system also conforms to the recommendations of Regulatory Guide 1.6 (Ref. 2) and IEEE-308 (Ref. 3).

The DC power sources provide both motive and control power to selected safety related equipment, as well as circuit breaker control power for the nonsafety related 480 V loads that are connected to 480 V ESF buses. Two center-tapped 260 VDC batteries are provided for Class 1E loads. They are designated 2PA for Division I and 2PB for Division II. Each 260 VDC battery is divided into two 130 VDC batteries connected in series. Each 130 VDC battery section has a battery charger connected in parallel with their respective battery. Each 260 VDC battery has a spare battery charger that can replace either of the normal 130 VDC connected chargers. Each division's two 130 VDC batteries and their chargers are the source of DC control power for that respective division, including the respective EDG. Each 260 VDC source furnishes power to DC motors necessary for shutdown conditions.

During normal operation, the DC loads are powered from the battery chargers with the batteries floating on the system. In case of loss of normal power to the battery charger, the DC loads are automatically powered from the batteries.

The DC power distribution system is described in more detail in Bases for LCO 3.8.7, "Distribution Systems—Operating," and LCO 3.8.8, "Distribution Systems—Shutdown."

Each battery has adequate storage capacity to carry the required load continuously for approximately 4 hours (Ref. 11).

BASES

BACKGROUND (continued)

Each DC battery subsystem is separately housed in a ventilated room apart from its charger and distribution centers. Each subsystem is located in an area separated physically and electrically from the other subsystems to ensure that a single failure in one subsystem does not cause a failure in a redundant subsystem. There is no sharing between redundant Class 1E subsystems such as batteries, battery chargers, or distribution panels.

The batteries for DC electrical power subsystems are sized to produce required capacity at 80% of nameplate rating, corresponding to warranted capacity at end of life cycles and the 100% design demand. The minimum design voltage limit is 105/210 V.

Each battery charger of DC electrical power subsystem has ample power output capacity for the steady state operation of connected loads required during normal operation, while at the same time maintaining its battery bank fully charged. Each battery charger has sufficient capacity to restore the battery from the design minimum charge to its fully charged state within 24 hours while supplying normal steady state loads (Ref. 11).

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in the UFSAR, Chapter 6 (Ref. 4) and Chapter 15 (Ref. 5), assume that Engineered Safety Feature (ESF) systems are OPERABLE. The DC electrical power system provides normal and emergency DC electrical power for the EDGs, emergency auxiliaries, and control and switching during all MODES of operation. The OPERABILITY of the DC subsystems is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the unit. This includes maintaining sufficient DC sources OPERABLE during accident conditions in the event of:

a. An assumed loss of all offsite AC power or all onsite AC power; and

b. A worst case single failure.

The DC sources satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

BASES

LCO

The DC electrical power subsystems — with each DC subsystem consisting of two 130 VDC batteries in series, two battery chargers, and the corresponding control equipment and interconnecting cabling supplying power to the associated bus, are required to be OPERABLE to ensure the availability of the required power to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence (AOO) or a postulated DBA. Loss of any DC electrical power subsystem does not prevent the minimum safety function from being performed (Ref. 3).

APPLICABILITY

The DC electrical power sources are required to be OPERABLE in MODES 1, 2, and 3 to ensure safe unit operation and to ensure that:

a. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of AOOs or abnormal transients; and

b. Adequate core cooling is provided, and containment integrity and other vital functions are maintained in the event of a postulated DBA.

The DC electrical power requirements for MODES 4 and 5 are addressed in the Bases for LCO 3.8.5, "DC Sources — Shutdown."

ACTIONS

A.1 and B.1

Conditions A and B represent one division with a loss of ability to completely respond to an event, and a potential loss of ability to remain energized during normal operation. If one of the required DC electrical power subsystems is inoperable (e.g., inoperable battery, inoperable battery charger(s), or inoperable battery charger and associated inoperable battery), the remaining DC electrical power subsystems have the capacity to support a safe shutdown and to mitigate an accident condition. A subsequent worst case single failure could, however, result in the loss of minimum necessary DC electrical subsystems to mitigate a worst case accident. It is therefore imperative that the operator's attention focus on stabilizing the unit, minimizing the potential for complete loss of DC power to the affected

BASES

ACTIONS (continued)

division. The 4 hour Completion Time (Required Action A.1) for restoration of an inoperable battery charger allows time to replace the inoperable charger with an OPERABLE spare battery charger, if available. The four hour limit is reasonable based on the remaining capability of the battery to carry the loads for this period. The 2 hour limit for Required Action B.1 is consistent with the allowed time for an inoperable DC Distribution System division. The 2 hour Completion Time is based on Regulatory Guide 1.93 (Ref. 6) and reflects a reasonable time to assess unit status as a function of the inoperable DC electrical power subsystem and, if the DC electrical power subsystem is not restored to OPERABLE status, to prepare to effect an orderly and safe unit shutdown.

C.1

If the station service DC electrical power subsystem cannot be restored to OPERABLE status within the required Completion Time, the unit must be brought to a MODE in which overall plant risk is minimized. To achieve this status, the unit must be brought to at least MODE 3 within 12 hours.

Remaining in the Applicability of the LCO is acceptable because the plant risk in MODE 3 is similar to or lower than the risk in MODE 4 (Ref. 8) and because the time spent in MODE 3 to perform the necessary repairs to restore the system to OPERABLE status will be short. However, voluntary entry into MODE 4 may be made as it is also an acceptable low-risk state.

Required Action C.1 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 3. This Note prohibits the use of LCO 3.0.4.a to enter MODE 3 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 3, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

BASES

ACTIONS (continued)

The allowed Completion Time is reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.8.4.1

Verifying battery terminal voltage while on float charge for the batteries helps to ensure the effectiveness of the charging system and the ability of the batteries to perform their intended function. Float charge is the condition in which the charger is supplying the continuous charge required to overcome the internal losses of a battery (or battery cell) and maintain the battery (or a battery cell) in a fully charged state. The voltage requirements are based on the nominal design voltage of the battery and are consistent with the initial voltages assumed in the battery sizing calculations. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.8.4.2

Visual inspection to detect corrosion of the battery cells and connections, or measurement of the resistance of each inter-cell and terminal connection, provides an indication of physical damage or abnormal deterioration that could potentially degrade battery performance.

The connection resistance limits procedurally established for this SR are no more than 20% above the resistance as measured during installation and not above the ceiling value established by the manufacturer. This provides conservative measures to assure the Technical Specification limit is not exceeded.

For each inter-cell and terminal connection, the limit is 150 micro-ohm. The total resistance of each 130 VDC battery is also monitored. This resistance is the total aggregate measured resistance of the cell-to-cell and terminal connections of each 130 VDC battery. The limit for total connection resistance of each 130 VDC battery is 2700 micro-ohm.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.4.3

Visual inspection of the battery cells, cell plates, and battery racks provides an indication of physical damage or abnormal deterioration that could potentially degrade battery performance. The presence of physical damage or deterioration does not necessarily represent a failure of this SR, provided an evaluation determines that the physical damage or deterioration does not affect the OPERABILITY of the battery (its ability to perform its design function).

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.8.4.4 and SR 3.8.4.5

Visual inspection and resistance measurements of inter-cell and terminal connections provides an indication of physical damage or abnormal deterioration that could indicate degraded battery condition. The anti-corrosion material is used to help ensure good electrical connections and to reduce terminal deterioration. The visual inspection for corrosion is not intended to require removal of and inspection under each terminal connection.

The removal of visible corrosion is a preventive maintenance SR. The presence of visible corrosion does not necessarily represent a failure of this SR, provided visible corrosion is removed during performance of this Surveillance.

The connection resistance limits procedurally established for this SR are no more than 20% above the resistance as measured during installation, and not above the ceiling value established by the manufacturer. This provides conservative measures to assure the Technical Specification limit is not exceeded.

For each inter-cell and terminal connection, the limit is 150 micro-ohm. The total resistance of each 130 VDC battery is also monitored. This resistance is the total aggregate measured resistance of the cell-to-cell and terminal connections of each 130 VDC battery. The limit for total connection resistance of each 130 VDC battery is 2700 micro-ohm.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.8.4.6

Battery charger capability requirements are based on the design capacity of the chargers (Ref. 3). According to Regulatory Guide 1.32 (Ref. 9), the battery charger supply is required to be based on the largest combined demands of the various steady state loads and the charging capacity to restore the battery from the design minimum charge state to the fully charged state, irrespective of the status of the unit during these demand occurrences. The minimum required amperes and duration ensures that these requirements can be satisfied.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.8.4.7

A battery service test is a special test of the battery's capability, as found, to satisfy the design requirements (battery duty cycle) of the DC electrical power system. The discharge rate and test length corresponds to the design duty cycle requirements as specified in Reference 4.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

This SR is modified by a Note that allows the performance of a performance discharge test in lieu of a service test.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.4.8

A battery performance discharge test is a test of constant current capacity of a battery, normally done in the as found condition, after having been in service, to detect any change in the capacity determined by the acceptance test. The test is intended to determine overall battery degradation due to age and usage.

The battery performance discharge test is acceptable for satisfying SR 3.8.4.7 as noted in SR 3.8.4.7.

The acceptance criteria for this Surveillance is consistent with IEEE-450 (Ref. 7) and IEEE-485 (Ref. 10). These references recommend that the battery be replaced if its capacity is below 80% of the manufacturer's rating. A capacity of 80% shows that the battery rate of deterioration is increasing, even if there is ample capacity to meet the load requirements.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. If the battery shows degradation, or if the battery has reached 85% of its expected life and capacity is < 100% of the manufacturer's rating, the Surveillance Frequency is reduced to 12 months. However, if the battery shows no degradation but has reached 85% of its expected life, the Surveillance Frequency is only reduced to 24 months for batteries that retain capacity ≥ 100% of the manufacturer's rating. Degradation is indicated, according to IEEE-450 (Ref. 7), when the battery capacity drops by more than 10% relative to its capacity on the previous performance test or when it is 10% below the manufacturer's rating. All these Frequencies are consistent with the recommendations in IEEE-450 (Ref. 7).

This SR is modified by a Note. The reason for the Note is that performing the Surveillance would remove a required DC electrical power subsystem from service, perturb the electrical distribution system, and challenge safety systems. Credit may be taken for unplanned events that satisfy the Surveillance.

BASES

REFERENCES

1. 10 CFR 50, Appendix A, GDC 17.
2. Regulatory Guide 1.6.
3. IEEE Standard 308, 1978.
4. UFSAR, Chapter 6.
5. UFSAR, Chapter 15.
6. Regulatory Guide 1.93.
7. IEEE Standard 450.
8. NEDC-32988-A, Revision 2, Technical Justification to Support Risk- Informed Modification to Selected Required End States for BWR Plants, December 2002.
9. Regulatory Guide 1.32, February 1977.
10. IEEE Standard 485, 1983.
11. UFSAR, Section 8.3.2.

3.8 ELECTRICAL POWER SYSTEMS

3.8.1 AC Sources --Operating

LCO 3.8.1 The following AC electrical power sources shall be OPERABLE:

a. Two qualified circuits between the offsite transmission network and the onsite Class 1E AC Electrical Power Distribution System; and

b. Two emergency diesel generators (EDGs) per division.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

------------------------------------NOTE------------------------------------

LCO 3.0.4.b is not applicable to EDGs.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One EDG inoperable.	A.1	Perform SR 3.8.1.1 for OPERABLE offsite circuit(s).	1 hour AND Once per 8 hours thereafter
	AND		
	A.2	Declare required feature(s), supported by the inoperable EDG, inoperable when the redundant required feature(s) are inoperable.	4 hours from discovery of an inoperable EDG concurrent with inoperability of redundant required feature(s)
	AND		
	A.3	Verify the status of CTG 11-1.	Once per 8 hours
	AND		
			(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.4.1 Determine OPERABLE EDG(s) are not inoperable due to common cause failure.	24 hours
	<u>OR</u>	
	A.4.2 Perform SR 3.8.1.2 for OPERABLE EDG(s).	24 hours
	<u>AND</u>	
	A.5 Restore availability of CTG 11-1.	72 hours from discovery of Condition A concurrent with CTG 11-1 not available
	<u>AND</u>	
	A.6 Restore EDG to OPERABLE status.	14 days
B. Both EDGs in one division inoperable.	B.1 Perform SR 3.8.1.1 for OPERABLE offsite circuit(s).	1 hour <u>AND</u> Once per 8 hours thereafter
	<u>AND</u>	

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. (continued)	B.2	Declare required feature(s), supported by the inoperable EDGs, inoperable when the redundant required feature(s) are inoperable.	4 hours from discovery of the inoperable EDGs concurrent with inoperability of redundant required feature(s)
	AND		
	B.3.1	Determine OPERABLE EDG(s) are not inoperable due to common cause failure.	24 hours
	OR		
	B.3.2	Perform SR 3.8.1.2 for OPERABLE EDG(s).	24 hours
	AND		
	B.4	Restore one EDG in the division to OPERABLE status.	72 hours
C. One or both EDGs in both divisions inoperable.	C.1	Restore both EDGs in one division to OPERABLE status.	2 hours
D. One offsite circuit inoperable.	D.1	Perform SR 3.8.1.1 for OPERABLE offsite circuit.	1 hour AND Once per 8 hours thereafter
	AND		(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. (continued)	D.2	Declare required feature(s) with no offsite power available inoperable when the redundant required feature(s) are inoperable.	24 hours from discovery of no offsite power to one division concurrent with inoperability of redundant required feature(s)
	<u>AND</u>		
	D.3	Restore offsite circuit to OPERABLE status.	72 hours
E. Two offsite circuits inoperable.	E.1	Declare required feature(s) inoperable when the redundant required feature(s) are inoperable.	12 hours from discovery of Condition E concurrent with inoperability of redundant required feature(s)
	<u>AND</u>		
	E.2	Restore one offsite circuit to OPERABLE status.	24 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. One offsite circuit inoperable. AND One or both EDGs in one Division inoperable.	------------NOTE----------- Enter applicable Conditions and Required Actions of LCO 3.8.7, "Distribution Systems - Operating," when Condition F is entered with no AC power source to one or more 4160 V buses 64B, 64C, 65E or 65F. ---------------------------	
	F.1 Restore offsite circuit to OPERABLE status.	12 hours
	OR	
	F.2 Restore both EDGs in the Division to OPERABLE status.	12 hours
G. Required Action and Associated Completion Time of Condition A, B, C, D, E or F not met.	G.1 -------NOTE------- LCO 3.0.4.a is not applicable when entering MODE 3. ------------------	
	Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.1.1	Verify correct breaker alignment and indicated power availability for each offsite circuit.	In accordance with the Surveillance Frequency Control Program
SR 3.8.1.2	------NOTES------ 1. All EDG starts may be preceded by an engine prelube period and followed by a warmup period prior to loading. 2. A modified EDG start involving idling and gradual acceleration to synchronous speed may be used for this SR as recommended by the manufacturer. ------------	
	Verify each EDG starts and achieves steady state voltage $\geq$ 3950 V and $\leq$ 4580 V and frequency $\geq$ 58.8 Hz and $\leq$ 61.2 Hz.	In accordance with the Surveillance Frequency Control Program
SR 3.8.1.3	------NOTES------ 1. EDG loadings may include gradual loading as recommended by the manufacturer. 2. Momentary transients below the load limit do not invalidate this test. 3. This Surveillance shall be conducted on only one EDG at a time. ------------	
	Verify each EDG is synchronized and loaded and operates for $\geq$ 60 minutes at a load $\geq$ 2500 kW.	In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.4	Verify each day tank contains ≥ one hour supply of fuel oil.	In accordance with the Surveillance Frequency Control Program
SR 3.8.1.5	Check for and remove accumulated water from each day tank.	In accordance with the Surveillance Frequency Control Program
SR 3.8.1.6	Verify each fuel oil transfer system operates to automatically transfer fuel oil from storage tanks to the day tanks.	In accordance with the Surveillance Frequency Control Program
SR 3.8.1.7	------------------NOTE------------------ All EDG starts may be preceded by an engine prelube period and followed by a warmup period prior to loading. -- Verify each EDG starts from standby condition and achieves: a. In ≤ 10 seconds, voltage ≥ 3950 V and frequency ≥ 58.8 Hz; and b. Steady state voltage ≥ 3950 V and ≤ 4580 V and frequency ≥ 58.8 Hz and ≤ 61.2 Hz.	In accordance with the Surveillance Frequency Control Program
SR 3.8.1.8	Verify each EDG rejects a load greater than or equal to its associated single largest post-accident load, and following load rejection, the frequency is ≤ 66.75 Hz.	In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.9	Verify each EDG does not trip and voltage is maintained ≤ 5267 V during and following a load rejection of ≥ 2850 kW.	In accordance with the Surveillance Frequency Control Program
SR 3.8.1.10	------------------NOTE------------------ All EDG starts may be preceded by an engine prelube period. -- Verify on simulated loss of offsite power signal: a. De-energization of emergency buses; b. Load shedding from emergency buses; and c. EDG auto-starts and: 1. energizes permanently connected loads in ≤ 10 seconds, 2. energizes auto-connected shutdown loads through load sequencer, 3. maintains steady state voltage ≥ 3950 V and ≤ 4580 V, 4. maintains steady state frequency ≥ 58.8 Hz and ≤ 61.2 Hz, and 5. supplies permanently connected and auto-connected shutdown loads for ≥ 5 minutes.	In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.11	------------------NOTE------------------ All EDG starts may be preceded by an engine prelube period. -- Verify on an actual or simulated Emergency Core Cooling System (ECCS) initiation signal each EDG auto-starts and: a. In ≤ 10 seconds after auto-start and during tests, achieves voltage ≥ 3950 V and frequency ≥ 58.8 Hz; b. Achieves steady state voltage ≥ 3950 V and ≤ 4580 V, and frequency ≥ 58.8 Hz and ≤ 61.2 Hz; and c. Operates for ≥ 5 minutes.	In accordance with the Surveillance Frequency Control Program
SR 3.8.1.12	Verify each EDG's automatic trips are bypassed on an actual or simulated emergency start signal except: a. Engine overspeed; b. Generator differential current; c. Low lube oil pressure; d. Crankcase overpressure; and e. Failure to start.	In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.13	------------------NOTE------------------ Momentary transients outside the load range do not invalidate this test. --- Verify each EDG operates for ≥ 24 hours: a. For all but the final ≥ 2 hours loaded ≥ 2500 kW and ≤ 2600 kW; and b. For the final ≥ 2 hours of the test loaded ≥ 2800 kW and ≤ 2900 kW.	In accordance with the Surveillance Frequency Control Program
SR 3.8.1.14	------------------NOTES------------------ 1. This Surveillance shall be performed within 5 minutes of shutting down the EDG after the EDG has operated ≥ 2 hours loaded ≥ 2500 kW or until operating temperatures have stabilized. Momentary transients below the load limit do not invalidate this test. 2. All EDG starts may be preceded by an engine prelube period. --- Verify each EDG starts and achieves: a. In ≤ 10 seconds, voltage ≥ 3950 V and frequency ≥ 58.8 Hz; and b. Steady state voltage ≥ 3950 V and ≤ 4580 V and frequency ≥ 58.8 Hz and ≤ 61.2 Hz.	In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.15	Verify each EDG: a. Synchronizes with offsite power source while loaded with emergency loads upon a simulated restoration of offsite power; b. Transfers loads to offsite power source; and c. Returns to standby status.	In accordance with the Surveillance Frequency Control Program
SR 3.8.1.16	Verify interval between each sequenced load block is within ± 10% of design interval for each load sequencer timer.	In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.17	------------------NOTE------------------ All EDG starts may be preceded by an engine prelube period. -- Verify, on simulated loss of offsite power signal in conjunction with an actual or simulated ECCS initiation signal: a. De-energization of emergency buses; b. Load shedding from emergency buses; and c. EDG auto-starts and: 1. energizes permanently connected loads in ≤ 10 seconds, 2. energizes auto-connected emergency loads through load sequencer, 3. achieves steady state voltage ≥ 3950 V and ≤ 4580 V, 4. achieves steady state frequency ≥ 58.8 Hz and ≤ 61.2 Hz, and 5. supplies permanently connected and auto-connected emergency loads for ≥ 5 minutes.	In accordance with the Surveillance Frequency Control Program
SR 3.8.1.18	------------------NOTE------------------ All EDG starts may be preceded by an engine prelube period. -- Verify, when started simultaneously each EDG achieves, in ≤ 10 seconds, frequency ≥ 58.8 Hz.	In accordance with the Surveillance Frequency Control Program

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources — Operating

BASES

BACKGROUND

The unit Class 1E AC Electrical Power Distribution System AC sources consist of the offsite power sources, and the onsite standby power sources (emergency diesel generators (EDGs) 11, 12, 13, and 14). As required by 10 CFR 50, Appendix A, GDC 17 (Ref. 1), the design of the AC electrical power system provides independence and redundancy to ensure an available source of power to the Engineered Safety Feature (ESF) systems.

The Class 1E AC distribution system is divided into redundant load groups (Division I and Division II), so loss of any one group does not prevent the minimum safety functions from being performed. Each load group is connected to an offsite power supply and two EDGs. Additional capability exists for each load group to be connected to the alternate division's offsite power supply (referred to as the maintenance cross-tie).

Offsite power is supplied to the 120 kV and 345 kV switchyards from the transmission network by five transmission lines. From the 120 kV switchyard, an electrically and physically separated circuit provides AC power, through system service transformer 64, to 4.16 kV ESF buses 64B and 64C. From the 345 kV switchyard, an electrically and physically separated circuit provides AC power through system service transformer 65 to 4.16 kV buses 65E and 65F. A detailed description of the offsite power network and circuits to the onsite Class 1E ESF buses is found in the UFSAR, Sections 8.2 and 8.3 (Ref. 2).

An offsite circuit consists of all breakers, transformers, switches, interrupting devices, cabling, and controls required to transmit power from the offsite transmission network to the onsite Class 1E ESF bus or buses.

Transformers 64 and 65 are sized to accommodate the simultaneous starting of all ESF loads on receipt of an accident signal without the need for load sequencing.

BASES

BACKGROUND (continued)

The onsite standby power source for 4.16 kV ESF buses 64B, 64C, 65E, and 65F, consists of four EDGs; EDG 11, 12, 13, and 14 respectively. An EDG starts automatically on a loss of coolant accident (LOCA) signal (i.e., low reactor water level signal or high drywell pressure signal) or on an ESF bus degraded voltage or undervoltage signal. After the EDG has started, it automatically ties to its respective bus after offsite power is tripped as a consequence of ESF bus undervoltage or degraded voltage, independent of or coincident with a LOCA signal. The EDGs also start and operate in the standby mode without tying to the ESF bus on a LOCA signal alone. Following the trip of offsite power, load shed relays strip nonpermanent loads from the ESF bus. When the EDG is tied to the ESF bus, loads are then sequentially connected to its respective ESF bus by the automatic sequencer. The sequencing logic controls the permissive and starting signals to motor breakers to prevent overloading the EDG.

In the event of a loss of normal power, the ESF electrical loads are automatically connected to the EDGs in sufficient time to provide for safe reactor shutdown and to mitigate the consequences of a Design Basis Accident (DBA) such as a LOCA.

Certain required plant loads are returned to service in a predetermined sequence in order to prevent overloading of the EDGs in the process. Within approximately 55 seconds after the EDG breaker closure, all automatic and permanently connected loads needed to recover the unit or maintain it in a safe condition are returned to service (i.e., available to start according to designed start signals).

Ratings for the EDGs satisfy the requirements of Regulatory Guide 1.9 (Ref. 3). EDGs 11, 12, 13, and 14 have the following ratings:

a. 2850 kW — continuous;

b. 3135 kW—2 hour, short time;

c. 3100 kW — 2000 hours;

d. 3250 kW — 300 hours; and

e. 3500 kW — 30 minutes.

BASES

APPLICABLE SAFETY ANALYSES

The initial conditions of DBA and transient analyses in the UFSAR, Chapter 6 (Ref. 4) and Chapter 15 (Ref. 5), assume ESF systems are OPERABLE. The AC electrical power sources are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, Reactor Coolant System (RCS), and containment design limits are not exceeded. These limits are discussed in more detail in the Bases for Section 3.2, Power Distribution Limits; Section 3.4, Reactor Coolant System (RCS); and Section 3.6, Containment Systems.

The OPERABILITY of the AC electrical power sources is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the unit. This includes maintaining sufficient onsite or offsite AC sources OPERABLE during accident conditions in the event of:

a. An assumed loss of all offsite power or all onsite AC power sources; and

b. A worst case single failure.

AC sources satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Two qualified circuits between the offsite transmission network and the onsite Class 1E Distribution System and two separate and independent divisions of two EDGs (11 and 12; 13 and 14) ensure availability of the required power to shut down the reactor and maintain it in a safe shutdown condition after an anticipated operational occurrence (AOO) or a postulated DBA.

Qualified offsite circuits are those that are described in the UFSAR, and are part of the licensing basis for the unit.

Each offsite circuit must be capable of maintaining rated frequency and voltage, and accepting required loads during an accident, while connected to the ESF buses. Each offsite circuit is physically independent and consists of incoming breakers and disconnect to the respective system service 64 or 65 transformers, and the respective circuit path including feeder breakers to 4.16 kV ESF buses.

GDC-17, "Electric Power Systems" (Reference 1) requires that provisions be included to minimize the probability of losing electric power from any of the remaining supplies as a

BASES

LCO (continued)

result of, or coincident with the loss of power generated by the nuclear power unit. The 345 kV breaker alignment must be maintained such that this criteria continues to be met. The GDC-17 criteria are met when either BM or DF breaker is closed when the main generating unit is online. An example of not meeting the criteria is with both BM and DF breakers open, a main generator trip would open breakers CM and CF and cause a loss of the 345 kV preferred power source. Thus, an offsite circuit must be declared inoperable when the breaker alignment is such that a loss of the main generator could lead to a loss of the respective offsite circuit.

Each EDG must be capable of starting, accelerating to rated speed and voltage, and connecting to its respective ESF bus on detection of bus undervoltage. This sequence must be accomplished within 10 seconds. Each EDG must also be capable of accepting required loads within the assumed loading sequence intervals, and must continue to operate until offsite power can be restored to the ESF buses. These capabilities are required to be met from a variety of initial conditions, such as EDG in standby with the engine hot and EDG in standby with the engine at ambient condition. Additional EDG capabilities must be demonstrated to meet required Surveillances, e.g., capability of the EDG to revert to standby status upon restoration of offsite power.

Proper sequencing of loads, including tripping of nonessential loads, is a required function for EDG OPERABILITY.

The AC sources must be separate and independent (to the extent possible) of other AC sources as described in UFSAR Sections 8.2 and 8.3 (Ref. 2).

APPLICABILITY

The AC sources are required to be OPERABLE in MODES 1, 2, and 3 to ensure that:

a. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of AOOs or abnormal transients; and

b. Adequate core cooling is provided and containment OPERABILITY and other vital functions are maintained in the event of a postulated DBA.

The AC power requirements for MODES 4 and 5 are covered in LCO 3.8.2, "AC Sources-Shutdown."

BASES

ACTIONS

A Note prohibits the application of LCO 3.0.4.b to an inoperable EDG. There is an increased risk associated with entering a MODE or other specified condition in the Applicability with an inoperable EDG and the provisions of LCO 3.0.4.b, which allow entry into a MODE or other specified condition in the Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, should not be applied in this circumstance.

A.1

To ensure a highly reliable power source remains with one EDG inoperable, it is necessary to verify the availability of the OPERABLE offsite circuits on a more frequent basis. Since the Required Action only specifies "perform," a failure of SR 3.8.1.1 acceptance criteria does not result in a Required Action being not met. However, if a circuit fails to pass SR 3.8.1.1, it is inoperable. Upon offsite circuit inoperability, additional Conditions must then be entered.

A.2

Required Action A.2 is intended to provide assurance that a loss of offsite power, during the period that one EDG is inoperable, does not result in a complete loss of safety function of critical systems. These features are designed with redundant safety related divisions (i.e., single division systems are not included). Redundant required features failures consist of inoperable features associated with a division redundant to the division that has an inoperable EDG.

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." In this Required Action the Completion Time only begins on discovery that both:

a. An inoperable EDG exists; and

b. A required feature on the other division (Division 1 or 2) that is redundant to a feature supported by the inoperable EDG is inoperable.

BASES

ACTIONS (Continued)

If, at any time during the existence of this Condition (one EDG inoperable), a required feature subsequently becomes inoperable, this Completion Time begins to be tracked.

Discovering one required EDG inoperable coincident with one or more inoperable redundant required support or supported features, or both, that are associated with the OPERABLE EDGs results in starting the Completion Time for the Required Action. Four hours from the discovery of these events existing concurrently is acceptable because it minimizes risk while allowing time for restoration before subjecting the unit to transients associated with shutdown.

The remaining OPERABLE EDGs and offsite circuits are adequate to supply electrical power to the onsite Class 1E Distribution System. Thus, on a component basis, single failure protection for the required feature's function may have been lost; however, function has not been lost. The 4 hour Completion Time takes into account the component OPERABILITY of the redundant counterpart to the inoperable required feature. Additionally, the 4 hour Completion Time takes into account the capacity and capability of the remaining AC sources, reasonable time for repairs, and low probability of a DBA occurring during this period.

A.3

To minimize the impact of operation with an inoperable EDG, it is necessary to periodically ensure the availability of CTG 11-1. The verification of the status of CTG 11-1 is performed by an administrative check of breaker and line availability, and the CTG 11-1 ability to supply Division I loads. Since this Required Action only specifies "verify the status," even when CTG 11-1 is not available it does not result in this Required Actions being not met. However, upon discovery that CTG 11-1 is unavailable, the limitations of Required Action A.5 are imposed.

BASES

ACTIONS (continued)

A.4.1 and A.4.2

Required Action A.4.1 provides an allowance to avoid unnecessary testing of OPERABLE EDGs. If it can be determined that the cause of the inoperable EDG does not exist on the OPERABLE EDGs, SR 3.8.1.2 does not have to be performed. If the cause of inoperability exists on other EDG(s), they are declared inoperable upon discovery, and Condition B or C of LCO 3.8.1 may be entered. Once the failure is repaired, and the common cause failure no longer exists, Required Action A.4.1 is satisfied. If the cause of the initial inoperable EDG cannot be confirmed not to exist on the remaining EDG(s), performance of SR 3.8.1.2 suffices to provide assurance of continued OPERABILITY of those EDGs.

In the event the inoperable EDG is restored to OPERABLE status prior to completing either A.4.1 or A.4.2, the plant corrective action program will continue to evaluate the common cause possibility. This continued evaluation, however, is no longer under the 24 hour constraint imposed while in Condition A.

According to Generic Letter 84-15 (Ref. 7), 24 hours is a reasonable time to confirm that the OPERABLE EDGs are not affected by the same problem as the inoperable EDG.

A.5 and A.6

According to Regulatory Guide 1.93 (Ref. 6), operation may continue with no OPERABLE EDGs to one division for a period that should not exceed 72 hours. With one EDG in one division inoperable, the remaining OPERABLE EDGs and offsite circuits are adequate to supply electrical power to the onsite Class 1E Distribution System. Required Action A.5 imposes this 72 hour Completion Time from the discovery of the non-availability CTG 11-1. However, if CTG 11-1 is available to supply Division I loads (determined by administrative check of breaker, line availability, and CTG 11-1 status) Required Action A.5 would be met and Required Action A.6 would allow the restoration time of 14 days.

The 14 day Completion Time to restore all EDGs to OPERABLE status takes into account the capacity and capability of the remaining AC Sources, as well as the additional reliability afforded by the availability of CTG 11-1, and low probability of a DBA occurring during this period.

BASES

ACTIONS (continued)

B.1

To ensure a highly reliable power source remains with both EDGs in one division inoperable, it is necessary to verify the availability of the OPERABLE offsite circuits on a more frequent basis. Since the Required Action only specifies "perform," a failure of SR 3.8.1.1 acceptance criteria does not result in a Required Action being not met. However, if a circuit fails to pass SR 3.8.1.1, it is inoperable. Upon offsite circuit inoperability, additional Conditions must then be entered.

B.2

Required Action B.2 is intended to provide assurance that a loss of offsite power, during the period that both EDGs in one division are inoperable, does not result in a complete loss of safety function of critical systems. These features are designed with redundant safety related divisions (i.e., single division systems are not included). Redundant required features failures consist of inoperable features associated with a division redundant to the division that has an inoperable EDG.

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." In this Required Action the Completion Time only begins on discovery that both:

a. Both EDGs in one division are inoperable; and

b. A required feature on the other division (Division 1 or 2) that is redundant to a feature supported by the inoperable EDGs is inoperable.

If, at any time during the existence of this Condition (both EDGs in one division inoperable), a required feature subsequently becomes inoperable, this Completion Time begins to be tracked.

BASES

ACTIONS (continued)

Discovering both EDGs in one division inoperable coincident with one or more inoperable redundant required support or supported features, or both, that are associated with the OPERABLE EDGs results in starting the Completion Time for the Required Action. Four hours from the discovery of these events existing concurrently is acceptable because it minimizes risk while allowing time for restoration before subjecting the unit to transients associated with shutdown.

The remaining OPERABLE EDGs and offsite circuits are adequate to supply electrical power to the onsite Class 1E Distribution System. Thus, on a component basis, single failure protection for the required feature's function may have been lost; however, function has not been lost. The 4 hour Completion Time takes into account the component OPERABILITY of the redundant counterpart to the inoperable required feature. Additionally, the 4 hour Completion Time takes into account the capacity and capability of the remaining AC sources, reasonable time for repairs, and low probability of a DBA occurring during this period.

B.3.1 and B.3.2

Required Action B.3.1 provides an allowance to avoid unnecessary testing of OPERABLE EDGs. If it can be determined that the cause of the inoperable EDGs does not exist on the OPERABLE EDGs, SR 3.8.1.2 does not have to be performed. If the cause of inoperability exists on other EDGs, they are declared inoperable upon discovery, and Condition C of LCO 3.8.1 may be entered. Once the failure is repaired, and the common cause failure no longer exists, Required Action B.3.1 is satisfied. If the cause of the initial inoperable EDGs cannot be confirmed not to exist on the remaining EDGs, performance of SR 3.8.1.2 suffices to provide assurance of continued OPERABILITY of those EDGs.

In the event the inoperable EDGs are restored to OPERABLE status prior to completing either B.3.1 or B.3.2, the plant corrective action program will continue to evaluate the common cause possibility. This continued evaluation, however, is no longer under the 24 hour constraint imposed while in Condition B.

According to Generic Letter 84-15 (Ref. 7), 24 hours is a reasonable time to confirm that the OPERABLE EDGs are not affected by the same problem as the inoperable EDG.

BASES

ACTIONS (continued)

B.4

According to Regulatory Guide 1.93 (Ref. 6), operation may continue with no OPERABLE EDGs to one division for a period that should not exceed 72 hours. With both EDGs in one division inoperable, the remaining OPERABLE EDGs and offsite circuits are adequate to supply electrical power to the onsite Class 1E Distribution System. Required Action B.4 imposes this 72 hour Completion Time.

The 72 hour Completion Time to restore one EDG in the division to OPERABLE status takes into account the capacity and capability of the remaining AC sources, reasonable time for repairs, and low probability of a DBA occurring during this period.

C.1

With one or both EDGs on both divisions inoperable, there may be no remaining standby AC source. Thus, with an assumed loss of offsite electrical power, insufficient standby AC sources are available to power the minimum required ESF functions. Since the offsite electrical power system is the only source of AC power for a significant percentage of ESF equipment at this level of degradation, the risk associated with continued operation for a very short time could be less than that associated with an immediate controlled shutdown. (The immediate shutdown could cause grid instability, which could result in a total loss of AC power.) Since any inadvertent unit generator trip could also result in a total loss of offsite AC power, however, the time allowed for continued operation is severely restricted. The intent here is to avoid the risk associated with an immediate controlled shutdown and to minimize the risk associated with this level of degradation.

According to Regulatory Guide 1.93 (Ref. 6), with both divisions with EDGs inoperable, operation may continue for a period that should not exceed 2 hours.

BASES

ACTIONS (continued)

D.1

To ensure a highly reliable power source remains with one offsite circuit inoperable, it is necessary to verify the availability of the remaining required offsite circuit on a more frequent basis. Since the Required Action only specifies "perform," a failure of SR 3.8.1.1 acceptance criteria does not result in a Required Action not met. However, if a second required circuit fails SR 3.8.1.1, the second offsite circuit is inoperable and Condition E for two offsite circuits inoperable, is entered.

D.2

Required Action D.2, which only applies if the division cannot be powered from an offsite source, is intended to provide assurance that an event with a coincident single failure of the associated EDG does not result in a complete loss of safety function of critical systems. These features are designed with redundant safety related divisions (i.e., single division systems are not included). Redundant required features failures consist of inoperable features associated with a division redundant to the division that has no offsite power.

The Completion Time for Required Action D.2 is intended to allow time for the operator to evaluate and repair any discovered inoperabilities. This Completion Time also allows an exception to the normal "time zero" for beginning the allowed outage time "clock." In this Required Action, the Completion Time only begins on discovery that both:

a. The division has no offsite power supplying its loads and

b. A required feature on the other division is inoperable.

If, at any time during the existence of this Condition (one offsite circuit inoperable) a required feature subsequently becomes inoperable, this Completion Time would begin to be tracked.

BASES

ACTIONS (continued)

Discovering no offsite power to one 4160 V ESF bus of the onsite Class 1E Power Distribution System coincident with one or more inoperable required support or supported features, or both, that are associated with any other ESF bus that has offsite power, results in starting the Completion Times for the Required Action. Twenty-four hours is acceptable because it minimizes risk while allowing time for restoration before the unit is subjected to transients associated with shutdown.

The remaining OPERABLE offsite circuit and EDGs are adequate to supply electrical power to the onsite Class 1E Distribution System. Thus, on a component basis, single failure protection may have been lost for the required feature's function; however, function is not lost. The 24 hour Completion Time takes into account the component OPERABILITY of the redundant counterpart to the inoperable required feature. Additionally, the 24 hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period.

D.3

According to Regulatory Guide 1.93 (Ref. 6), operation may continue in Condition D for a period that should not exceed 72 hours. With one offsite circuit inoperable, the reliability of the offsite system is degraded and the potential for a loss of offsite power is increased, with attendant potential for a challenge to the plant safety systems. In this condition, however, the remaining OPERABLE offsite circuit and EDGs are adequate to supply electrical power to the onsite Class 1E Distribution System.

The 72 hour Completion Time takes into account the capacity and capability of the remaining AC sources, reasonable time for repairs, and the low probability of a DBA occurring during this period.

BASES

ACTIONS (continued)

E.1 and E.2

Required Action E.1 addresses actions to be taken in the event of inoperability of redundant required features concurrent with inoperability of two offsite circuits. Required Action E.1 reduces the vulnerability to a loss of function. The Completion Time for taking these actions is reduced to 12 hours from that allowed with one division without offsite power (Required Action D.2). The rationale for the reduction to 12 hours is that Regulatory Guide 1.93 (Ref. 6) allows a Completion Time of 24 hours for two required offsite circuits inoperable, based upon the assumption that two complete safety divisions are OPERABLE. When a concurrent redundant required feature failure exists, this assumption is not the case, and a shorter Completion Time of 12 hours is appropriate. These features are designed with redundant safety related divisions, (i.e., single division systems are not included in the list). Redundant required features failures consist of any of these features that are inoperable because any inoperability is on a division redundant to a division with inoperable offsite circuits.

The Completion Time for Required Action E.1 is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." In this Required Action, the Completion Time only begins on discovery that both:

a. All required offsite circuits are inoperable and

b. A required feature is inoperable.

If, at any time during the existence of this Condition (two offsite circuits inoperable), a required feature subsequently becomes inoperable, this Completion Time begins to be tracked.

BASES

ACTIONS (continued)

According to Regulatory Guide 1.93 (Ref. 6), operation may continue in Condition E for a period that should not exceed 24 hours. This level of degradation means that the offsite electrical power system does not have the capability to effect a safe shutdown and to mitigate the effects of an accident; however, the onsite AC sources have not been degraded. This level of degradation generally corresponds to a total loss of the immediately accessible offsite power sources.

Because of the normally high availability of the offsite sources, this level of degradation may appear to be more severe than other combinations of two AC sources inoperable that involve one or more EDGs inoperable. However, two factors tend to decrease the severity of this degradation level:

a. The configuration of the redundant AC electrical power system that remains available is not susceptible to a single bus or switching failure and

b. The time required to detect and restore an unavailable offsite power source is generally much less than that required to detect and restore an unavailable onsite AC source.

With both of the required offsite circuits inoperable, sufficient onsite AC sources are available to maintain the unit in a safe shutdown condition in the event of a DBA or transient. In fact, a simultaneous loss of offsite AC sources, a LOCA, and a worst case single failure were postulated as a part of the design basis in the safety analysis. Thus, the 24 hour Completion Time provides a period of time to effect restoration of one of the offsite circuits commensurate with the importance of maintaining an AC electrical power system capable of meeting its design criteria.

According to Regulatory Guide 1.93 (Ref. 6), with the available offsite AC sources two less than required by the LCO, operation may continue for 24 hours. If two offsite sources are restored within 24 hours, unrestricted operation may continue. If only one offsite source is restored within 24 hours, power operation continues in accordance with Condition D.

BASES

ACTIONS (continued)

F.1 and F.2

Pursuant to LCO 3.0.6, the Distribution System ACTIONS would not be entered even if all AC sources to it were inoperable, resulting in de-energization. Therefore, the Required Actions of Condition F are modified by a Note to indicate that when Condition F is entered with no AC source to any ESF bus, ACTIONS for LCO 3.8.7, "Distribution Systems - Operating," must be immediately entered. This allows Condition F to provide requirements for the loss of the offsite circuit and one EDG without regard to whether a division is de-energized. LCO 3.8.7 provides the appropriate restrictions for a de-energized division.

According to Regulatory Guide 1.93 (Ref. 6), operation may continue in Condition F for a period that should not exceed 12 hours. In Condition F, individual redundancy is lost in both the offsite electrical power system and the onsite AC electrical power system. Since power system redundancy is provided by two diverse sources of power, however, the reliability of the power systems in this Condition may appear higher than that in Condition E (loss of both required offsite circuits). This difference in reliability is offset by the susceptibility of this power system configuration to a single bus or switching failure. The 12 hour Completion Time takes into account the capacity and capability of the remaining AC sources, reasonable time for repairs, and the low probability of a DBA occurring during this period.

G.1

If the inoperable AC electrical power sources cannot be restored to OPERABLE status within the associated Completion Time, the unit must be brought to a MODE in which overall plant risk is minimized. To achieve this status, the unit must be brought to at least MODE 3 within 12 hours.

Remaining in the Applicability of the LCO is acceptable because the plant risk in MODE 3 is similar to or lower than the risk in MODE 4 (Ref. 8) and because the time spent in MODE 3 to perform the necessary repairs to restore the system to OPERABLE status will be short. However, voluntary entry into MODE 4 may be made as it is also an acceptable low-risk state.

BASES

ACTIONS (continued)

Required Action G.1 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 3. This Note prohibits the use of LCO 3.0.4.a to enter MODE 3 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 3, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

The allowed Completion Time is reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

The AC sources are designed to permit inspection and testing of all important areas and features, especially those that have a standby function, in accordance with 10 CFR 50, GDC 18 (Ref. 9). Periodic component tests are supplemented by extensive functional tests during refueling outages (under simulated accident conditions). The SRs for demonstrating the OPERABILITY of the EDGs are based on the recommendations of Regulatory Guide 1.9 (Ref. 3), Regulatory Guide 1.108 (Ref. 10), and Regulatory Guide 1.137 (Ref. 11), as addressed in the UFSAR.

Where the SRs discussed herein specify voltage and frequency tolerances, the following summary is applicable. The minimum steady state output voltage of 3950 V corresponds to the most limiting voltage needed to supply Division I buses under degraded voltage with LOCA conditions.

BASES

SURVEILLANCE REQUIREMENTS (continued)

This value is also bounding for Division II and ensures that adequate voltage is available to the equipment supported by Division I and II of the EDGs. The specified maximum steady state output voltage of 4580 V is equal to the maximum operating voltage specified for 4000 V motors. It ensures that for a lightly loaded distribution system, the voltage at the terminals of 4000 V motors is no more than the maximum rated operating voltages. The specified minimum and maximum frequencies of the EDG are 58.8 Hz and 61.2 Hz, respectively. These values are equal to ± 2% of the 60 Hz nominal frequency and are derived from the recommendations found in Regulatory Guide 1.9 (Ref. 3).

SR 3.8.1.1

This SR ensures proper circuit continuity for the offsite AC electrical power supply to the onsite distribution network and availability of offsite AC electrical power. The breaker alignment verifies that each breaker is in its correct position to ensure that distribution buses and loads are connected to their preferred power source and that appropriate independence of offsite circuits is maintained. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.8.1.2 and SR 3.8.1.7

These SRs help to ensure the availability of the standby electrical power supply to mitigate DBAs and transients and maintain the unit in a safe shutdown condition.

To minimize the mechanical stress and wear on moving parts that do not get lubricated when the engine is not running, these SRs have been modified by a Note (Note 1 for SR 3.8.1.2 and the Note for SR 3.8.1.7) to indicate that all EDG starts for these Surveillances may be preceded by an engine prelube period and followed by a warmup prior to loading.

BASES

SURVEILLANCE REQUIREMENTS (continued)

For the purposes of SR 3.8.1.2 testing, the EDGs are started anywhere from standby to hot conditions by using one of the following signals:

- Manual,
- Simulated loss-of-offsite power by itself,
- Simulated loss-of-offsite power in conjunction with an ESF actuation test signal, or
- An ESF actuation test signal by itself.

In order to reduce stress and wear on diesel engines, the EDG manufacturer recommends a modified start in which the starting speed of EDGs is limited, warmup is limited to this lower speed, and the EDGs are gradually accelerated to synchronous speed prior to loading. These start procedures are the intent of Note 2, which is only allowed to satisfy SR 3.8.1.2 but are not applicable when performing SR 3.8.1.7.

SR 3.8.1.7 requires that the EDG starts from standby conditions and achieves required voltage and frequency within 10 seconds. Standby conditions for an EDG mean that the diesel engine coolant and oil are being continuously circulated and temperature is being maintained consistent with manufacturer recommendations. The 10 second start requirement supports the assumptions in the design basis LOCA analysis of UFSAR, Section 6.3 (Ref. 13). The 10 second start requirement is not applicable to SR 3.8.1.2. Since SR 3.8.1.7 does require a 10 second start, it is more restrictive than SR 3.8.1.2, and it may be performed in lieu of SR 3.8.1.2. In addition to the SR requirements, the time for the EDG to reach steady state operation, unless the modified EDG start method is employed, is periodically monitored and the trend evaluated to identify degradation of governor and voltage regulator performance.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.3

This Surveillance provides assurance that the EDGs are capable of synchronizing and accepting greater than or equal to the equivalent of the maximum expected accident loads without the risk of overloading the EDG. The EDG is tested at approximately 90% of its continuous load rating, which provides margin to excessive EDG loading, while demonstrating the EDG capability to carry loads near the maximum expected accident loads. A minimum run time of 60 minutes is required to stabilize engine temperatures, while minimizing the time that the EDG is connected to the offsite source.

Although no power factor requirements are established by this SR, the EDG is normally operated at a power factor between 0.8 lagging and 1.0. The 0.8 value is the design rating of the machine, while 1.0 is an operational limitation to ensure circulating currents are minimized. Routine overloading may result in more frequent teardown inspections in accordance with vendor recommendations in order to maintain EDG OPERABILITY.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

Note 1 modifies this Surveillance to indicate that diesel engine runs for this Surveillance may include gradual loading, as recommended by the manufacturer, so that mechanical stress and wear on the diesel engine are minimized.

BASES

SURVEILLANCE REQUIREMENTS (continued)

Note 2 modifies this Surveillance by stating that momentary transients (e.g., because of changing bus loads) do not invalidate this test. Similarly, momentary power factor transients outside the normal range do not invalidate the test.

Note 3 indicates that this Surveillance should be conducted on only one EDG at a time in order to avoid common cause failures that might result from offsite circuit or grid perturbations.

SR 3.8.1.4

This SR provides verification that there is an adequate inventory of fuel oil in the day tank to support the EDG operation for a minimum of one hour at full load. The volume of fuel oil equivalent to one hour supply is 210 gallons.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.8.1.5

Microbiological fouling is a major cause of fuel oil degradation. There are numerous bacteria that can grow in fuel oil and cause fouling, but all must have a water environment in order to survive. Periodic removal of water from the fuel oil day tanks eliminates the necessary environment for bacterial survival. This is the most effective means of controlling microbiological fouling. In addition, it eliminates the potential for water entrainment in the fuel oil during EDG operation. Water may come from any of several sources, including condensation, ground water, rain water, contaminated fuel oil, and breakdown of the fuel oil by bacteria. Frequent checking for and removal of accumulated water minimizes fouling and provides data regarding the watertight integrity of the fuel oil system. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.6

This Surveillance demonstrates that each required fuel oil transfer pump operates and transfers fuel oil from its associated storage tank to its associated day tank. It is required to support continuous operation of standby power sources. This Surveillance provides assurance that the fuel oil transfer pump is OPERABLE, the fuel oil piping system is intact, the fuel delivery piping is not obstructed, and the controls and control systems for automatic fuel transfer systems are OPERABLE.

The design of fuel transfer systems is such that pumps operate automatically in order to maintain an adequate volume of fuel oil in the day tank during or following EDG testing. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.8.1.7

See SR 3.8.1.2.

SR 3.8.1.8

Each EDG is provided with an engine overspeed trip to prevent damage to the engine. Recovery from the transient caused by the loss of a large load could cause diesel engine overspeed, which, if excessive, might result in a trip of the engine. This Surveillance demonstrates the EDG load response characteristics and capability to reject the largest single load while maintaining a specified margin to the overspeed trip. The largest single load for each EDG is a residual heat removal pump (1684 kW). This Surveillance may be accomplished by:

a. Tripping the EDG output breaker with the EDG carrying greater than or equal to its associated single largest post-accident load while paralleled to offsite power, or while solely supplying the bus; or

b. Tripping its associated single largest post-accident load with the EDG solely supplying the bus.

BASES

SURVEILLANCE REQUIREMENTS (continued)

As required by IEEE-308 (Ref. 15), the load rejection test is acceptable if the increase in diesel speed does not exceed 75% of the difference between synchronous speed and the overspeed trip setpoint, or 15% above synchronous speed, whichever is lower. This represents 66.75 Hz, equivalent to 75% of the difference between nominal speed and the overspeed trip setpoint.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.8.1.9

This Surveillance demonstrates the EDG capability to reject a full load without overspeed tripping or exceeding the predetermined voltage limits. The EDG full load rejection may occur because of a system fault or inadvertent breaker tripping. This Surveillance ensures proper engine generator load response under the simulated test conditions. This test simulates the loss of the total connected load that the EDG experiences following a full load rejection and verifies that the EDG does not trip upon loss of the load. These acceptance criteria provide EDG damage protection. While the EDG is not expected to experience this transient during an event, and continues to be available, this response ensures that the EDG is not degraded for future application, including reconnection to the bus if the trip initiator can be corrected or isolated.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.10

As required by Regulatory Guide 1.108 (Ref. 10), paragraph 2.a.(1), this Surveillance demonstrates the as designed operation of the standby power sources during loss of the offsite source. This test verifies all actions encountered from the loss of offsite power, including shedding of the nonessential loads and energization of the emergency buses and respective loads from the EDG, including automatic start of the EDG cooling water pump. It further demonstrates the capability of the EDG to automatically achieve the required voltage and frequency within the specified time.

The EDG auto-start time of 10 seconds is derived from requirements of the accident analysis for responding to a design basis large break LOCA. The Surveillance should be continued for a minimum of 5 minutes in order to demonstrate that all starting transients have decayed and stability has been achieved.

The requirement to verify the connection and power supply of permanent and auto-connected loads is intended to satisfactorily show the relationship of these loads to the EDG loading logic. In certain circumstances, many of these loads cannot actually be connected or loaded without undue hardship or potential for undesired operation. For instance, Emergency Core Cooling Systems (ECCS) injection valves are not desired to be stroked open, or systems are not capable of being operated at full flow, or RHR systems performing a decay heat removal function are not desired to be realigned to the ECCS mode of operation. In lieu of actual demonstration of the connection and loading of these loads, testing that adequately shows the capability of the EDG system to perform these functions is acceptable. This testing may include any series of sequential, overlapping, or total steps so that the entire connection and loading sequence is verified.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

This SR is modified by a Note allowing EDG starts to be preceded by an engine prelube period. The reason for the Note is to minimize wear and tear on the EDGs during testing.

SR 3.8.1.11

This Surveillance demonstrates that the EDG (including its associated cooling water pump) automatically starts and achieves the required minimum voltage and frequency within the specified time (10 seconds) from the design basis actuation signal (LOCA signal) and operates for ≥ 5 minutes. The 5 minute period provides sufficient time to demonstrate stability.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

This SR is modified by a Note allowing EDG starts to be preceded by an engine prelube period. The reason for the Note is to minimize wear and tear on the EDGs during testing.

SR 3.8.1.12

This Surveillance demonstrates that EDG non-critical protective functions (e.g., high jacket water temperature) are bypassed on an actual or simulated emergency start (LOCA or loss of offsite power) signal. The non-critical trips are bypassed during DBAs and provide an alarm on an abnormal engine condition. This alarm provides the operator with sufficient time to react appropriately. The EDG availability to mitigate the DBA is more critical than protecting the engine against minor problems that are not immediately detrimental to emergency operation of the EDG.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.8.1.13

Regulatory Guide 1.108 (Ref. 10), paragraph 2.a.(3), requires demonstration that the EDGs can start and run continuously at full load capability for an interval of not less than 24 hours—22 hours of which is at a load equivalent to the continuous rating of the EDG, and 2 hours of which is at a load equivalent to 110% of the continuous duty rating of the EDG. Fermi-2 has taken an exception to this requirement and performs the 22 hour run at approximately 90% of the continuous rating (2500 kW-2600 kW), and performs the 2 hour run at approximately the continuous rating (2800 kW-2900 kW). The EDG starts for this Surveillance can be performed either from standby or hot conditions. The provisions for prelube and warmup, discussed in SR 3.8.1.2, and for gradual loading, discussed in SR 3.8.1.3, are applicable to this SR.

Although no power factor requirements are established by this SR, the EDG is normally operated at a power factor between 0.8 lagging and 1.0. The 0.8 value is the design rating of the machine, while the 1.0 is an operational limitation to ensure circulating currents are minimized. A load band is provided to avoid routine overloading of the EDG. Routine overloading may result in more frequent teardown inspections in accordance with vendor recommendations in order to maintain EDG OPERABILITY.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

This Surveillance has been modified by a Note. The Note states that momentary transients due to changing bus loads do not invalidate this test.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.14

This Surveillance demonstrates that the diesel engine can restart from a hot condition, such as subsequent to shutdown from normal Surveillances, and achieve the minimum required voltage and frequency within 10 seconds and maintain a steady state voltage and frequency range. The 10 second time is derived from the requirements of the accident analysis to respond to a design basis large break LOCA. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

This SR is modified by two Notes. Note 1 ensures that the test is performed with the diesel sufficiently hot. The requirement that the diesel has operated for at least 2 hours near full load conditions prior to performance of this Surveillance is based on manufacturer recommendations for achieving hot conditions. Routine overloads may result in more frequent teardown inspections in accordance with vendor recommendations in order to maintain EDG OPERABILITY. Momentary transients due to changing bus loads do not invalidate this test. Note 2 allows all EDG starts to be preceded by an engine prelube period to minimize wear and tear on the diesel during testing.

SR 3.8.1.15

As required by Regulatory Guide 1.108 (Ref. 10), paragraph 2.a.(6), this Surveillance ensures that the manual synchronization and load transfer from the EDG to the offsite source can be made and that the EDG can be returned to standby status when offsite power is restored. It also ensures that the auto-start logic is reset to allow the EDG to restart and reload if a subsequent loss of offsite power occurs. The EDG is considered to be in standby status when the EDG is shutdown with the output breaker open, the load sequence timers are reset, and is able to restart and reload on a subsequent bus under voltage.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.16

Under accident conditions with loss of offsite power loads are sequentially connected to the bus by the automatic load sequencer. The sequencing logic controls the permissive and starting signals to motor breakers to prevent overloading of the EDGs due to high motor starting currents. The 10% load sequence time interval tolerance ensures that sufficient time exists for the EDG to restore frequency and voltage prior to applying the next load and that safety analysis assumptions regarding ESF equipment time delays are not violated. Reference 2 provides a summary of the automatic loading of ESF buses.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.8.1.17

In the event of a DBA coincident with a loss of offsite power, the EDGs are required to supply the necessary power to ESF systems so that the fuel, RCS, and containment design limits are not exceeded.

This Surveillance demonstrates EDG operation, as discussed in the Bases for SR 3.8.1.10, during a loss of offsite power actuation test signal in conjunction with an ECCS initiation signal. In lieu of actual demonstration of connection and loading of loads, testing that adequately shows the capability of the EDG system to perform these functions is acceptable. This testing may include any series of sequential, overlapping, or total steps so that the entire connection and loading sequence is verified.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

This SR is modified by a Note allowing EDG starts to be preceded by an engine prelube period. The reason for the Note is to minimize wear and tear on the EDGs during testing.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.18

This Surveillance demonstrates that the EDG starting independence has not been compromised. Also, this Surveillance demonstrates that each engine can achieve proper speed within the specified time when the EDGs are started simultaneously.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

This SR is modified by a Note allowing EDG starts to be preceded by an engine prelube period. The reason for the Note is to minimize wear on the EDG during testing.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 17.
2. UFSAR, Sections 8.2 and 8.3.
3. Regulatory Guide 1.9.
4. UFSAR, Chapter 6.
5. UFSAR, Chapter 15.
6. Regulatory Guide 1.93.
7. Generic Letter 84-15.
8. NEDC-32988-A, Revision 2, Technical Justification to Support Risk- Informed Modification to Selected Required End States for BWR Plants, December 2002.
9. 10 CFR 50, Appendix A, GDC 18.
10. Regulatory Guide 1.108.
11. Regulatory Guide 1.93.
12. Deleted.
13. UFSAR, Section 6.3.
14. ASME Boiler and Pressure Vessel Code, Section XI.
15. IEEE Standard 308.

3.3 INSTRUMENTATION

3.3.2.1 Control Rod Block Instrumentation

LCO 3.3.2.1 The control rod block instrumentation for each Function in Table 3.3.2.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.2.1-1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One rod block monitor (RBM) channel inoperable.	A.1 Restore RBM channel to OPERABLE status.	24 hours
B. Required Action and associated Completion Time of Condition A not met. OR Two RBM channels inoperable.	B.1 Place one RBM channel in trip.	1 hour
C. Rod worth minimizer (RWM) inoperable during reactor startup.	C.1 Suspend control rod movement except by scram. OR	Immediately (continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. (continued)	C.2.1.1 Verify ≥ 12 rods withdrawn.	Immediately
	OR	
	C.2.1.2 Verify by administrative methods that startup with RWM inoperable has not been performed in the current calendar year.	Immediately
	AND	
	C.2.2 Verify movement of control rods is in compliance with the prescribed withdrawal sequence by a second licensed operator or other qualified member of the technical staff.	During control rod movement
D. RWM inoperable during reactor shutdown.	D.1 Verify movement of control rods is in accordance with the prescribed withdrawal sequence by a second licensed operator or other qualified member of the technical staff.	During control rod movement

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
E. One or more Reactor Mode Switch-Shutdown Position channels inoperable.	E.1	Suspend control rod withdrawal.	Immediately
	AND		
	E.2	Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately

SURVEILLANCE REQUIREMENTS

------------------------------------NOTES------------------------------------

1. Refer to Table 3.3.2.1-1 to determine which SRs apply for each Control Rod Block Function.

2. When an RBM channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains control rod block capability.

SURVEILLANCE		FREQUENCY
SR 3.3.2.1.1	-----------------NOTE----------------- Not required to be performed until 1 hour after any control rod is withdrawn at ≤ 10% RTP in MODE 2. ---------------------------------------	
	Perform CHANNEL FUNCTIONAL TEST.	In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.3.2.1.2	------------------NOTE------------------ Not required to be performed until 1 hour after THERMAL POWER is ≤ 10% RTP in MODE 1. --- Perform CHANNEL FUNCTIONAL TEST.	In accordance with the Surveillance Frequency Control Program
SR 3.3.2.1.3	Perform CHANNEL FUNCTIONAL TEST.	In accordance with the Surveillance Frequency Control Program
SR 3.3.2.1.4	------------------NOTE------------------ Not required to be performed until 1 hour after reactor mode switch is in the shutdown position. --- Perform CHANNEL FUNCTIONAL TEST.	In accordance with the Surveillance Frequency Control Program
SR 3.3.2.1.5	Verify the RBM is not bypassed when THERMAL POWER is ≥ 30% RTP.	In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.2.1.6	-----------------NOTE----------------- Neutron detectors are excluded. ------------------------------------- Perform CHANNEL CALIBRATION.	In accordance with the Surveillance Frequency Control Program
SR 3.3.2.1.7	Verify control rod sequences input to the RWM are in conformance with the prescribed withdrawal sequence.	Prior to declaring RWM OPERABLE following loading of sequence into RWM

Table 3.3.2.1-1 (page 1 of 1)
Control Rod Block Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Rod Block Monitor				
a. Upscale	(a)	2	SR 3.3.2.1.3 SR 3.3.2.1.5 SR 3.3.2.1.6	As specified in the COLR
b. Inop	(a)	2	SR 3.3.2.1.3	NA
c. Downscale	(a)	2	SR 3.3.2.1.3 SR 3.3.2.1.6	As specified in the COLR
2. Rod Worth Minimizer	1[(b)], 2[(b)]	1	SR 3.3.2.1.1 SR 3.3.2.1.2 SR 3.3.2.1.7	NA
3. Reactor Mode Switch - Shutdown Position	(c)	2	SR 3.3.2.1.4	NA

(a) THERMAL POWER ≥ 30% RTP.

(b) With THERMAL POWER ≤ 10% RTP.

(c) Reactor mode switch in the shutdown position.

Table 8	MINIMUM STEAM COOLING PRESSURE (MSCP)
No. of open SRVs	MSCP (PSIG)
5 or more	230
4	290
3	392
2	595

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 Recirculation Loops Operating

LCO 3.4.1 Two recirculation loops with matched recirculation loop jet pump flows shall be in operation;

OR

One recirculation loop may be in operation provided the following limits are applied when the associated LCO is applicable:

1. LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," single loop operation limits specified in the COLR;
2. LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," single loop operation limits specified in the COLR;
3. LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," Function 2.b (Average Power Range Monitors Simulated Thermal Power–Upscale) Allowable Value of Table 3.3.1.1-1 is reset for single loop operation, when in MODE 1; and
4. THERMAL POWER is ≤ 66.1% RTP.

--------------------------NOTE--------------------------

Application of the required limitations for single loop operation may be delayed for up to 4 hours after transition from two recirculation loop operations to single recirculation loop operation.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Recirculation jet pump loop flow mismatch not within limits.	A.1	Declare recirculation loop with lower flow: "not in operation."	2 hours
B. No recirculation loops operating.	B.1	Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.1.1	-------------------NOTE------------------- Not required to be performed until 24 hours after both recirculation loops are in operation. -- Verify recirculation loop jet pump flow mismatch with both recirculation loops in operation is: a. $\leq$ 10% of rated core flow when operating at < 70% of rated core flow; and b. $\leq$ 5% of rated core flow when operating at $\geq$ 70% of rated core flow.	In accordance with the Surveillance Frequency Control Program

OFFGAS LOG RADIATION MONITORS CONVERSION FACTORS FOR TECHNICAL SPECIFICATIONS 3.7.5

SAMPLE INFORMATION

Offgas Radioactivity Rate..	1.20E-01 mCi/sec
Offgas System Flow Rate..	12.4 scfm
Offgas Log Rad Monitor D11-K601A Reading...........................	12.8 mr/hr
Offgas Log Rad Monitor D11-K601B Reading...........................	14.4 mr/hr

CONVERSION FACTORS

RAD MONITOR D11-K601A	RAD MONITOR D11-K601B
2.703E-03 mCi/sec / mr/hr x cfm	**2.403E-03** mCi/sec / mr/hr x cfm

These conversion factors are used to obtain the Technical Specification offgas release rate in mCi/sec from the offgas log radiation monitors by applying the following equation.

mCi/sec = (conversion factor) x (rad monitor reading) x (offgas flow rate)*

*Note: Offgas flow is taken from N62R808 O/G Outlet Flow Recorder Pen Labeled Charcoal Unit 1-6.

Performed By: ______________ Date: Today

Chemist/Designee: ______________ Date: Today

Noted By SM: ______________ Date: Today

Forward this page to operations after receiving Chemistry Lab Supervisor/Designee approval.

FORM 78.000.09 Att 3 P1/1 IP:I DTC: TGCHEM File: 1327.01