Examination Outline Cross-reference: RO-1	Level	RO		SRO
Rev. 0 Date: 12/16/2016	Tier	1		
	Group	1		
	K/A	295001.AA1.01		
Level of Difficulty: 4	Importance Rating	3.5		

Partial or Complete Loss of Forced Core Flow Circulation: Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION: Recirculation system

Question # 1

CGS is operating in Mode 1. Reactor power 100%.

- Reactor Recirculation (RRC) Pump B, RRC-P-1B, tripped.
- The crew has entered ABN-RRC-LOSS, Loss of Reactor Recirculation flow.
- RRC A Drive Flow (RRC-FI-M/A/R676A) – 43,500 gpm
- RRC Jet Pump Loop A Flow (MS-FI-611A) – 56 Mlb/hr
- RRC Pump A frequency (RRC-HZM-R670A) – 57 hz
- Control rods are inserted to the 100% rod line.

What actions should the crew take to stabilize the plant?

The crew should verify the Loop “A” Auto/Manual controller, RRC-M/A-676A, is in MANUAL, then…

A. raise RRC Jet Pump Loop A flow to GE 57.5 Mlb/hr.

B. Immediately attempt to restart RRC-P-1B.

C. Lower RRC Loop A Drive flow to LT 41,725 gpm.

D. raise RRC Loop A Drive flow by raising RRC Pump A frequency to 60 hz.

Answer: C

K/A Match:
The K/A discusses operating the recirculation system during a full or partial loss of flow. This question asks the student to understand how recirculation flow is adjusted after losing one recirculation pump.

SRO Only:
N/A.

Explanation:

ABN-RRC-LOSS step 4.3.4 directs reducing RRC Loop A drive flow to LT 41,725GPM.

A. Incorrect. This distractor is plausible since actions to restore boiling boundary to GE 4.0 is referenced in step 4.3.13 of ABN-RRC-LOSS. The distractor is also plausible because the maximum jet pump flow allowed in single look operations is 57.5Mlb/hr. The distractor is incorrect because RRC Loop Flow should be lowered, not raised.

B. Incorrect. Plausible because starting RRC-P-1B restore the plant to its previous normal lineup, but is incorrect because an RRC pump should not be restarted until it is understood why the pump tripped and appropriate procedures are followed to control reactor power.

C. Correct. ABN-RRC-LOSS step 4.3.4 directs reducing RRC Loop A drive flow to LT 41,725GPM.

D. Incorrect. This distractor is plausible because 60Hz is the maximum frequency that the ASD system and step 4.3.5 states that RRC flow should be maximized without exceeding 57.5Mlb/hr Jet Pump Flow or 41,725GPM Drive Flow. The distractor is incorrect, because RRC flow should be lowered, not raised, to keep drive flow less than 41,725GPM.

Technical Reference(s)		Attached w/ Revision # See Comments / Reference
ABN-RRC-LOSS, Loss of Reactor Recirculation Flow		

Proposed references to be provided during examination: None

Learning Objective: RO-1484: Respond to a loss of RRC-P-1A or RRC-P-1B

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 41.7
55.43 ____

Comments / Reference: ABN-RRC-LOSS, step 4.3.4	Revision: Major 013 Minor 001

Number: ABN-RRC-LOSS	Use Category: CONTINUOUS	Major Rev: 013 Minor Rev: 001 Page: 14 of 32
Title: Loss of Reactor Recirculation Flow		

4.3.4 IF RRC Loop A Drive Flow is GT 41,725 gpm,
THEN **ADJUST** RRC Loop A Drive Flow to LT 41,725 gpm as follows:
{P-113021}

a. IF controlling from the Main Control Room,
THEN **ADJUST** flow at H13-P602. ____

b. IF controlling from the ASD Building,
THEN **ADJUST** the following Local Speed Adjust potentiometers (Local Control and Diagnostic Panel). N/A any not adjusted.

- RRC-IMD-ASD1A/1 ____
- RRC-IMD-ASD1A/2 ____

CAUTION

Due to potential for Jet Pump damage in single loop operation (GE RICSIL 092), Each Jet Pump Flow is limited to LE 57.5 Mlb/hr in single loop operation. In the next step DO NOT reduce the Jet Pump Loop Flow. The Jet Pump Loop Flow will be reduced later in this section when directed. {AR-230496}

Comments / Reference: ABN-RRC-LOSS, step 4.3.13	Revision: Major 013 Minor 001

4.3.13 IF operating in the AIA,
AND the Boiling Boundary is LT 4.0 feet,
THEN **PERFORM** the following:

a. **INITIATE** action to restore the Boiling Boundary to GT 4.0 feet by adjusting control rods per the Fast Shutdown Sequence or per the SNE recommendations,
OR **EXIT** the AIA within 4 hours by reducing CTP, as directed by the CRS. _____

b. **CONTINUOUSLY MONITOR** for indications of a core instability until the Boiling Boundary is GT 4.0 feet or the AIA has been exited. _____

Comments / Reference: ABN-RRC-LOSS, step 4.3.5	Revision: Major 013 Minor 001

CAUTION

Due to potential for Jet Pump damage in single loop operation (GE RICSIL 092), Each Jet Pump Flow is limited to LE 57.5 Mlb/hr in single loop operation. In the next step DO NOT reduce the Jet Pump Loop Flow. The Jet Pump Loop Flow will be reduced later in this section when directed. {AR-230496}

4.3.5 IF the Core Thermal Power is GT 25%,
AND RRC Loop A Drive Flow is between 4173 gpm and 33,000 gpm,
THEN **RAISE** the Jet Pump Loop A Flow to the maximum value allowed, but not GT 57.5 Mlb/hr as follows:

a. IF controlling from the Main Control Room,
THEN **ADJUST** flow at H13-P602. _____

b. IF controlling from the ASD Building,
THEN **ADJUST** the following Local Speed Adjust potentiometers (Local Control and Diagnostic Panel). N/A any not adjusted.

- RRC-IMD-ASD1A/1 _____
- RRC-IMD-ASD1A/2 _____

Comments / Reference: ABN-RRC-LOSS, Notes prior to step 4.3.1	Revision: Major 013 Minor 001

4.3 Loss of RRC-P-1B

CAUTION

If operating on a rod line GT 70%, it is possible to enter Region A following the removal of RRC-P-1B from service. Operation in Region A has an elevated risk of a core instability event. Operation in Region A is an accepted risk. Minimize operation in Region A by reducing the rod line as soon as practical to reduce the potential of a core instability event. {P-77714}

NOTE: This section provides proper actions to be taken in the event of an automatic tripping of RRC-P-1B or when RRC-P-1B is manually tripped per procedural direction.

NOTE: To avoid Region A, the 100% rod line requires 53 Hz for the operating RRC pump, while the 90% rod line requires 46 Hz for the operating RRC pump.

NOTE: The limit for Single Loop RRC Drive flow is LT 41,725 gpm.

NOTE: RRC A Drive Flow indication in gpm can be obtained from any of the following: RRC-FI-M/A/R676A, computer point B037GPM, B129GPM, B131GPM, B038GPM.

NOTE: RRC Loop A Flow indication in Mlb/hr can be obtained from either of the following: MS-FI-611A or TDAS pt. X034.

Examination Outline Cross-reference:	Level	RO		SRO
Rev. 0 Date: 12/16/2016	Tier	1		
	Group	1		
	K/A	295003.AA2.01		
Level of Difficulty: 3	Importance Rating	3.4		

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

Question # 2

CGS is operating in Mode 1, 100% core thermal power.

A plant transient occurs.

Current conditions:

- Reactor power approximately 65%.
- Reactor Recirculation (RRC) pumps running at 30 hz.
- HPCS Service Water pump, HPCS-P-2, running.

These conditions are caused by a loss of power to…

A. SM-1.

B. SM-2.

C. SM-3.

D. SM-4.

Answer: B

K/A Match:

The candidate must analyze conditions in the stem and determine the bus loss (loss of power) that caused the conditions.

SRO Only:

N/A

Explanation:

A. Incorrect. Plausible since a loss of SM-1 will cause a loss of Condensate Pump 1A and Condensate Booster Pump 2A, which will cause a trip of Reactor Feedwater Pump 1B on low suction pressure. (See ABN-ELEC-SM1/SM7, Attachment 7.1). The loss of a feed pump will cause RPV level to lower. When RPV level reaches less than 31.5" (Level 4) with one feed pump offline, Reactor Recirculation (RRC) pumps will run back to 30 hz to prevent a low RPV level scram. The distractor is incorrect because E-SM-4 is normally powered from SM-2, a loss of SM-1 will not cause a start of 4-DG3 and the HPCS Service Water pump will not be running.

B. Correct. A loss of power to SM-2 will cause a loss of Condensate Pump 1B and Condensate Booster Pump 2B, which will cause a trip of Reactor Feedwater Pump 1B on low suction pressure. (See ABN-ELEC-SM1/SM7, Attachment 7.1). The loss of a feed pump will cause RPV level to lower. When RPV level reaches less than 31.5" (Level 4) with one feed pump offline, Reactor Recirculation (RRC) pumps will run back to 30 hz to prevent a low RPV level scram. Additionally, since SM-2 normally powers SM-4, a loss of SM-2 will cause the HPCS emergency diesel generator, 4-DG3, to start. The HPCS Service Water pump, HPCS-P-2, will start to support emergency diesel generator operations.

C. Incorrect. Plausible since a loss of SM-3 will cause a loss of Condensate Pump 1C and Condensate Booster Pump 2C, which will cause a trip of Reactor Feedwater Pump 1B on low suction pressure. (See ABN-ELEC-SM3/SM8, Attachment 7.1). The loss of a feed pump will cause RPV level to lower. When RPV level reaches less than 31.5" (Level 4) with one feed pump offline, Reactor Recirculation (RRC) pumps will run back to 30 hz to prevent a low RPV level scram. However, since E-SM-4 is normally powered from SM-2, a loss of SM-3 will not cause a start of 4-DG3 and the HPCS Service Water pump will not be running.

D. Incorrect. Plausible since tripping CB-2/4 will cause the HPCS diesel generator, 4-3DG to start. This will start the HPCS Service Water pump. However, a loss of power to SM-4 will not affect the condensate and feed system. Subsequently, reactor power will remain at 100% and RRC pumps will be operating at their normal frequency for full power.

Technical Reference(s)		
ABN-ELEC-SM3/SM8, SM-3, SM-8, SM-85, SM-82, SL-81, SL-83 & SL-31 Distribution System Failures		Attached w/ Revision # See Comments / Reference
ABN-ELEC-SM1/SM7, SM-1, SM-7, SM-75, SM-72, SL-71, SL-73 & SL-11 Distribution System Failures		
ABN-ELEC-SM2/SM4, SM-2, SM-4 and SL-21 Distribution System Failures		

Proposed references to be provided during examination: None

Learning Objective: 6809- Given a loss of SM-2, identify those automatic actions that may have occurred.

Question Source: Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 41.10
55.43 ______

Comments / Reference: ABN-ELEC-SM1/SM7, Attachment 7.1	Revision: Major 018, Minor 001

Number: ABN-ELEC-SM1/SM7	Use Category: CONTINUOUS	Major Rev: 018 Minor Rev: 001 Page: 25 of 29
Title: SM-1, SM-7, SM-75, SM-72, SL-71, SL-73 & SL-11 Distribution System Failures		

AUTOMATIC ACTIONS

7.1.1 Loss of E-SM-1

a. IF E-SM-7 was energized from E-SM-1, THEN Emergency Diesel Generator 1 starts.

b. The following supply breakers **OPEN**:

- E-CB-1/7
- E-CB-7/1
- E-CB-S1
- E-CB-N1/1
- E-CB-1/11

c. If the Backup Transformer is available, then E-CB-B/7 closes after a time delay.

d. IF the Backup Transformer is not available, THEN DG 1 **CLOSES** in on E-SM-7 after a time delay.

e. Loss of E-SM-1 will cause a loss of COND-P-1A (Condensate Pump 1A) and COND-P-2A (Condensate Booster Pump 2A).

f. A loss of one Reactor Feedwater Pump on low suction pressure with a Reactor Level 4 will initiate a RRC Runback to 30 Hz.

Comments / Reference: ABN-ELEC-SM2/SM4, sections 7.2 and 7.4	Revision: Major 006, Minor 003

Number: ABN-ELEC-SM2/SM4	Use Category: CONTINUOUS	Major Rev: 006 Minor Rev: 003 Page: 11 of 12
Title: SM-2, SM-4 and SL-21 Distribution System Failures		

AUTOMATIC ACTIONS

7.2 Loss of E-SM-2

7.2.1 The HPCS Emergency Diesel Generator **STARTS** and **SUPPLIES** SM-4.

7.2.2 The following supply breakers should be **OPEN**:

- E-CB-2/4
- E-CB-4/2
- E-CB-2/21
- E-CB-S2
- E-CB-N1/2

7.2.3 HPCS-P-2, HPCS Service Water Pump **STARTS**.

7.2.4 Loss of E-SM-2 will cause a loss of COND-P-1B (Condensate Pump 1B) and COND-P-2B (Condensate Booster Pump 2B).

7.2.5 A loss of one Reactor Feedwater Pump on low suction pressure with a Reactor Level 4 will initiate a RRC Runback to 30 Hz.

7.4 Loss of E-SM-4

7.4.1 The following supply breakers **OPEN**:

- E-CB 2/4
- E-CB 4/2

7.4.2 HPCS Emergency Diesel Generator **STARTS** and **SUPPLIES** E-SM-4.

7.4.3 HPCS-P-2 (HPCS Service Water Pump) **STARTS** when E-SM-4 is re-energized.

Comments / Reference: ABN-ELEC-SM3/SM8, section 7.1.1	Revision: Major 018

Number: ABN-ELEC-SM3/SM8	Use Category: CONTINUOUS	Major Rev: 018 Minor Rev: N/A Page: 23 of 27
Title: SM-3, SM-8, SM-85, SM-82, SL-81, SL-83 & SL-31 Distribution System Failures		

AUTOMATIC ACTIONS

7.1.1 Loss of E-SM-3

a. IF E-SM-8 was energized from E-SM-3,
THEN Emergency Diesel Generator 2 starts.

b. The following supply breakers **OPEN**:

- E-CB-3/8
- E-CB-8/3
- E-CB-S3
- E-CB-N1/3
- E-CB-3/31

c. IF the Backup Transformer is available,
THEN E-CB-B/8 closes after a time delay

d. IF the Backup Transformer is not available,
THEN DG 2 closes in on E-SM-8 after a time delay.

e. Loss of E-SM-3 will cause a loss of COND-P-1C (Condensate Pump C) and COND-P-2C (Condensate Booster Pump 2C).

f. A loss of one Reactor Feedwater Pump on low suction pressure with a Reactor Level 4 will initiate a RRC Runback to 30 Hz.

Examination Outline Cross-reference:	Level	RO		SRO
Rev. 0 Date: 12/16/2016	Tier	1		
	Group	1		
	K/A	295004.2.4.46		
Level of Difficulty: 3	Importance Rating	4.2		

Partial or Complete Loss of D.C. Power: Ability to verify that the alarms are consistent with the plant conditions.

Question # 3

CGS is operating in Mode 1.

A fault causes multiple annunciators to alarm, including:

- DEH TROUBLE (H13-P820.B1 – 2-5)
- CRD PUMPS ABNORMAL OPERATION (H13-P603.A7 – 4-6)
- RCIC TURBINE TRIP (H13-P601.A4 – 1-5)
- REACTOR FEEDWATER CONTROL SYSTEM TROUBLE (H13-P603.A8 – 1-7)

These annunciators will alarm with a loss of voltage on…

A. DP-S1/1.

B. DP-S1/2.

C. DP-S2/1.

D. DP-S1/7.

Answer: A

K/A Match:
The question requires an understanding of systems/components lost with a loss of DC power and the subsequent annunciators that will alarm.

SRO Only:
N/A

Explanation:

A. Correct. A loss of voltage on DP-S1/1 (Div. 1 125 vdc) will cause a DEH TROUBLE annunciator due to a loss of one DEH power supply. CRD pump 1A indication and control are lost causing the CRD PUMPS ABNORMAL OPERATION annunciator alarm. The REACTOR FEEDWATER CONTROL SYSTEM TROUBLE alarm is received since RPV Narrow Range Level C fails downscale with a loss of Div. 1 125 vdc. Indication and control is lost for the majority of the RCIC controls at H13-P60, causing the RCIC TURBINE TRIP annunciator to alarm.

B. Incorrect. Plausible since a loss of Division 2 125 vdc (DP-S1/2) will cause the DEH TROUBLE annunciator due to a loss of one DEH power supply. Additionally, CRD pump 1B indication and control are lost. This should cause the CRD PUMPS ABNORMAL OPERATION annunciator to alarm. Incorrect because a loss of Division 2 125 vdc causes a loss of annunciator power to H13-P601, H13-P602 and H13-P603 except “ANNUNCIATOR 125 VDC LOSS” H13-P603.A7-1.1, which alarms.

C. Incorrect. Plausible since a loss of 250VDC will cause a loss of indication and control for RCIC components and RFP Turbine Emergency Oil Pumps. A RCIC DIV 1 OUT OF SERVICE annunciator will occur. Incorrect because a DEH trouble annunciator would not be received.

D. Incorrect. Plausible because a loss of DP-S1/7 causes a loss of control panel RFW-DT-1A. It is also called the BOP 125 VDC panel which implies it would cause BOP related annunciators such as DEH TROUBLE. Incorrect because the RCIC turbine trip annunciator will not alarm.

Technical Reference(s)	
ABN-ELEC-125VDC, Plant BOP, DIV 1,2 & 3 125 VDC Distribution System Failures	Attached w/ Revision # See Comments / Reference
ABN-ELEC-250VDC, Plant 250 VDC Distribution System Failures	

Proposed references to be provided during examination: None

Learning Objective: 7652 – Predict the effects(s) a failure of 125VDC bus S1-1 will have on: (l) CR Annunciators

Question Source: Bank # ___
Modified Bank # ___ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis __X__

10 CFR Part 55 Content: 55.41 41.10
55.43 ______

Comments / Reference: ABN-ELEC-125VDC	Revision: 14

Number: ABN-ELEC-125VDC	Use Category: CONTINUOUS	Major Rev: 014 Minor Rev: N/A Page: 12 of 35
Title: Plant BOP, DIV 1,2 & 3 125 VDC Distribution System Failures		

4.2 125 VDC Division 1

NOTE: The overall effect on Plant due to a loss of E-DP-S1/1 from 100% power is that the Plant continues at 100% with the operation of Div 1 equipment severely limited.

NOTE: A alphabetical list of major loads supplied by 125 VDC Div 1 distribution system and their status is provided in Attachment 7.2 and a complete list is found in SOP-ELEC-DC-LU.

NOTE: Either 125 VDC charger E-C1-1A or E-C1-1B is capable of supplying the Div 1 125 VDC bus E-DP-S1/1 with the 125 VDC battery B1-1 disconnected from the bus.

NOTE: IF DC Power is completely lost, THEN loss of indication and the ability to operate the following equipment is expected:

- H13-P601 SRV Relief Operation Solenoid)
- H13-P601 Div 1 ADS (A Solenoid)
- P628 SRV Div 1 ADS Operation (A Solenoid)
- On line inverter IN-3A or IN-3B (Will auto transfer to AC source)
- RCIC Flow Controller causing RCIC trip if operating
- DG-1 DC powered pumps
- RCIC (Div 1) motor operated valves
- DG 1 Control Panels and Swgr Control Power
- Remote Shutdown & Alt. Remote Shutdown Room DC loads
- Div 1 remotely operated circuit bkr position indication and closing/tripping power
- RHR-P-2A and LPCS-P-1 (Will continue to run if previously operating)
- SW-P-1A
- CRD-P-1A

NOTE: A loss of power to E-PP-7AA results in a loss of RCC to the Drywell due to a BOP outboard isolation from a loss of RC-1.

NOTE: A loss of power to E-PP-7AA causes an RC-1 half trip. Consider protecting the RC-2 logic train and operating equipment. Refer to PPM 1.3.83 and OI-41. {AR-238691}

NOTE: H13-P800.C1-8.3, 125 VDC IN-3A/3B TROUBLE alarming may be an indication of E-IN-3A or E-IN-3B failing to transfer to its alternate AC source, E-PP-7A, after a loss of E-DP-S1/1.

Number: ABN-ELEC-125VDC	Use Category: CONTINUOUS	Major Rev: 014 Minor Rev: N/A Page: 30 of 35
Title: Plant BOP, DIV 1,2 & 3 125 VDC Distribution System Failures		

SYSTEM/COMPONENT STATUS ON LOSS OF E-DP-S1/1

NOTE: A complete list of equipment supplied by 125 VDC Div I (E-DP-S1/1) distribution system is provided in Attachment 7.3 of this PPM and in SOP-ELEC-DC-LU.

NOTE: The alphabetical listing of affected systems/components that follows provides an overview of the effect on the following equipment if DC power to E-DP-S1/1 is completely lost.

- Alternate Remote Shutdown panel, will lose control and Indication for SRV's, RHR A and SW Loop A DC loads.
- Annunciation - Loss of annunciation on H13-P672, loss of audible alarm function for Board S and all rear control room panels with the exception of "ANNUNCIATOR 125 VDC LOSS" P672-A9-3-1 which alarms.
- ATWS/ARI, Division I valves will lose control and indication and the auto logic will be lost.
- CRD, cannot be started but would remain running if previously started. Lose control and indication for CRD-P-1A.
- DG-1, will lose control and indication power and cannot be started nor can DG-1 be emergency tripped from the control room. If DG-1 was previously started it will continue running, but at full load (due to mechanical governor setting) and supplying voltage but unable to control it. DG-1 will also lose power to the DC powered pumps.
- FWLC, no effect of RPV level, however Narrow Range "C" RPV level indicator fails downscale. Also a Channel "C" Level 8 trip would be received following restoration of power (which could be reset).
- LPCS-P-1, cannot be started but would remain running if previously started. Loss of control and indication of LPCS pump.
- RCC, loss of control and indication power for RCC-P-1A and lose of indication and control for RCC-V-6 (will auto close on power restoration).
- RCIC, will trip if running. Indication and control is lost for the majority of the RCIC controls at H13-P601. RCIC operation would be limited to local operation via manual reset and manual positioning of the throttle valve from RB 421 with no available indication of operating parameters. RCIC DC motor operated valves will also lose power so any operation of these valves would also require local manual operation.
- RSD Panel, loss of control and Indication of RCIC.

Comments / Reference: ABN-ELEC-250VDC	Revision: Major 004 Minor 001

Number: ABN-ELEC-250VDC	Use Category: CONTINUOUS	Major Rev: 004 Minor Rev: 001 Page: 4 of 9
Title: Plant 250 VDC Distribution System Failures		

4.0 SUBSEQUENT OPERATOR ACTIONS

NOTE: If operating from the Remote Shutdown Panel, the following valves continue to have valve position indication but cannot be repositioned:

- RHR-V-8 (Shutdown Cooling Isolation)
- RHR-V-23 (Reactor Head Spray Isolation)
- RCIC-V-1 (RCIC Turbine Trip and Throttle Valve)
- RCIC-V-13 (RCIC Pump Discharge)
- RCIC-V-19 (RCIC Minimum Flow Bypass Valve)
- RCIC-V-22 (Test Return to CSTs)
- RCIC-V-45 (Steam Admittance Valve to RCIC Turbine)
- RCIC-V-69 (Vacuum Pump Discharge Valve)

NOTE: If DC power is completely lost to DP-S2-1 a loss of indication and the ability to operate the following equipment is expected. (All valves fail “as-is”)

E-MC-S2/1A

- RHR-V-8 (Shutdown Cooling Isolation)
- RHR-V-23 (Reactor Head Spray Isolation)
- RWCU-V-4 (Outboard Suction from RPV)
- RCIC-V-1 (RCIC Turbine Trip and Throttle Valve)
- RCIC-V-13 (RCIC Pump Discharge)
- RCIC-V-19 (RCIC Minimum Flow Bypass Valve)
- RCIC-V-22 (Test Return to CSTs)
- RCIC-V-59 (Test Return to CSTs)
- RCIC-V-69 (Vacuum Pump Discharge Valve)
- RCIC-V-45 (Steam Admittance Valve to RCIC Turbine)
- RCIC-P-2 (RCIC Vacuum Pump)
- RCIC-P-4 (RCIC Condensate Pump)

E-MC-S2/1B

- TO-P-EOP (Main Turbine Emergency Oil Pump)
- SO-P-ASBU (Air Side Seal Oil Backup Pump)
- RFT-P-EOP/1A (RFP 1A Turbine Emergency Oil Pump)
- RFT-P-EOP/1B (RFP 1B Turbine Emergency Oil Pump)

E-DP-S2/1

- DC Supply to IN-1

Examination Outline Cross-reference:	Level	RO		SRO
Rev. 1 Date: 12/16/2016	Tier	1		
	Group	1		
	K/A	295005.AK1.03		
Level of Difficulty: 2	Importance Rating	3.5		

Main Turbine Generator Trip: Knowledge of the operational implications of the following concepts as they apply to MAIN TURBINE GENERATOR TRIP : Pressure effects on reactor level

Question # 4

CGS is operating in Mode 1. Reactor power is 100%.

- A loss of lube oil causes a main turbine trip

How does actual RPV level initially respond to this event?

Actual RPV level will initially…

A. rise due to loss of recirculation pumps.

B. lower due to the collapse of voids in the core.

C. rise due to the rapid reduction in steam flow.

D. lower due to initiation of RPV level setpoint setdown.

Answer: B

K/A Match:
This question requires candidates to identify RPV level response due to the pressure transient caused by a main turbine trip.

SRO Only:
N/A

Explanation:

A. Incorrect. Plausible since recirculation pumps will trip on a main turbine trip from greater than 30% reactor power and this will tend to raise RPV level. However, this effect is overridden by the rapid collapse of voids in the core which will initially cause RPV level to lower.

B. Correct. A main turbine trip will cause a large RPV pressure spike. A rapid collapse of voids in the core will cause RPV level to initially lower.

C. Incorrect. Plausible since steam flow will rapidly lower on a main turbine trip while feed flow will gradually lower as the level control system compensates for reduced steam flow. This effect would cause RPV level to rise initially while feed flow is greater than steam flow. However, the rapid collapse of voids in the core due to RPV pressure rise will override this effect and RPV level will initially lower.

D. Incorrect. Plausible since the RPV level setpoint setback feature is in service on a RPS initiated scram, which will occur when the main turbine trips greater than 100% reactor power. However, this feature determines where RPV level will be controlled steady state. The initial RPV level reduction occurs as core voids collapse as RPV pressure rapidly rises.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
CGS Final Safety Analysis Report (FSAR)	
T.S. Bases, 3.3.4.1	

Proposed references to be provided during examination: None

Learning Objective: 11647 - Explain the reasons for the following responses as they apply to Main Turbine trip: d. Generator Trip

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

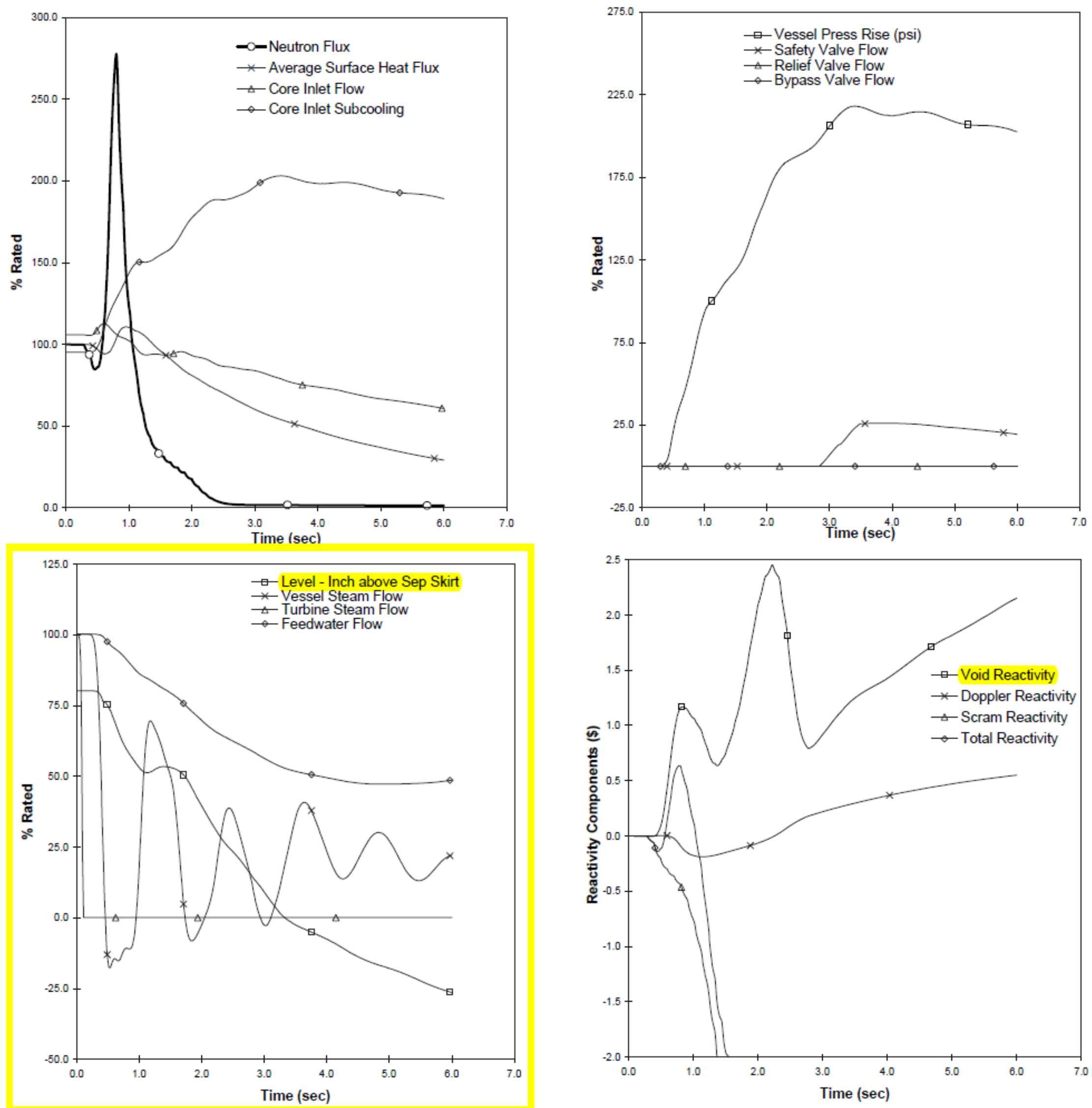
Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ____

10 CFR Part 55 Content: 55.41 41.5
55.43 ____

Comments / Reference: FSAR	Revision: Major 63 Minor 008

Amendment 60
December 2009



Columbia Generating Station Final Safety Analysis Report	Turbine Trip with Bypass Failure
	Draw. No. 020361.71 | Rev. | Figure 15.2-4

Form No. 960690
LDCN-08-035

Table 15.2-5

Sequence of Events for Figure 15.2-4

Turbine Trip with Bypass Failure
at 100% Power/106% Core Flow

Time (sec)	Event
0	Turbine trip initiates closure of main stop (throttle) valves.
0	Turbine bypass valves fail to operate.
0.01	Main turbine stop valves reach 90% open position and initiate reactor scram trip.
0.10	Turbine stop valves closed.
0.20	Recirculation pump motor circuit breakers open causing decrease in core flow to natural circulation.
(a)	Group 1 relief valves actuated.
(a)	Group 2 relief valves actuated.
2.83	Group 3 relief valves actuated.
3.18	Group 4 relief valves actuated.

[a] Not used - out of service for this analysis.

Comments / Reference: T.S. Bases, 3.3.4.1	Revision: 92

B 3.3 INSTRUMENTATION

B 3.3.4.1 End of Cycle Recirculation Pump Trip (EOC-RPT) Instrumentation

BASES

BACKGROUND

The EOC-RPT instrumentation initiates a recirculation pump trip (RPT) to reduce the peak reactor pressure and power resulting from turbine trip or generator load rejection transients to provide additional margin to the core thermal MCPR Safety Limit (SL).

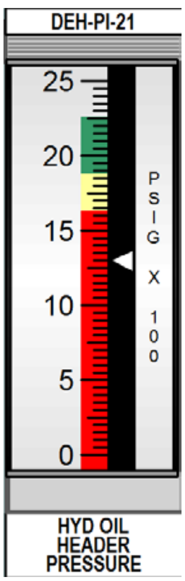
The need for the additional negative reactivity in excess of that normally inserted on a scram reflects end of cycle reactivity considerations. Flux shapes at the end of cycle are such that the control rods may not be able to ensure that thermal limits are maintained by inserting sufficient negative reactivity during the first few feet of rod travel upon a scram caused by Turbine Governor Valve (TGV) Fast Closure, Trip Oil Pressure - Low, or Turbine Throttle Valve (TTV) - Closure. The physical phenomenon involved is that the void reactivity feedback due to a pressurization transient can add positive reactivity at a faster rate than the control rods can add negative reactivity.

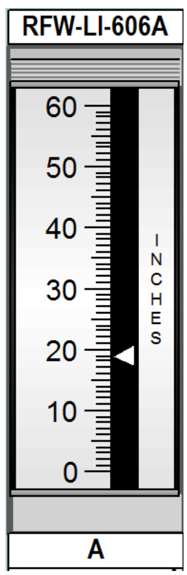
Examination Outline Cross-reference:	Level	RO		SRO
Rev. 0 Date: 12/16/2016	Tier	1		
	Group	1		
	K/A	295006.AK2.04		
Level of Difficulty: 2	Importance Rating	3.6		

SCRAM: Knowledge of the interrelations between SCRAM and the following: Turbine trip logic

Question # 5

The reactor is operating at 35%.
You see the indications pictured below for DEH-PI-21(Hydraulic Oil Header Pressure), and RFW-LI-606A (RPV Level).





Which of the following occurs?

A. Turbine Trip, Reactor SCRAM

B. Turbine Trip, Reactor does not SCRAM

C. No Turbine Trip, Reactor SCRAM

D. No Turbine Trip, Reactor does not SCRAM

Answer: A

K/A Match:
A turbine trip above 30% will result in a reactor SCRAM. A turbine trip below 30% will not result in a reactor SCRAM. The question determines whether the student knows the power value at which a SCRAM will occur due to a turbine trip and whether they know a turbine trip can cause a SCRAM.

SRO Only:
N/A

Explanation:
The candidate must analyze the indication for DEH pressure and recognize it is below the turbine trip value of 1600 psig. They must then determine whether or not a SCRAM occurs based on reactor power exceeding 30%. The correct answer is A.

A. Correct. DEH pressure is below 1600psig resulting in a turbine trip. Because reactor power is above 30%, a reactor scram also occurs.

B. Incorrect. Plausible since a turbine trip occurs, but the distractor is incorrect since a SCRAM will also occur.

C. Incorrect. Plausible because RPV level is lower than normal and approaching the SCRAM setpoint of 13 inches. However, a turbine trip will occur.

D. Incorrect. Plausible because RPV level is low, but has not reached the SCRAM setpoint of 13”. However, a turbine trip will occur and therefore, the reactor will scram.

Technical Reference(s)		Attached w/ Revision # See Comments / Reference
SD000129 r12	Main Turbine System Description	
4.603.A8 5-4	Annunciator Response Procedure	

Proposed references to be provided during examination: None

Learning Objective: 5566 – List all parameters and setpoints that will cause a turbine trip

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam ____

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 41.7
55.43 ____

Comments / Reference: SD000129	Revision: 12

COLUMBIA SYSTEMS September 2014
MT SD000129, r12 mr0

4. DEH System leaks are addressed in ABN-DEH-LEAK and require a timely response to control and isolate the leak for the following reasons:

   - Allowing DEH pump discharge pressure to decrease to LT 1600 psig will result in the trip of the Main Turbine and loss of Bypass Valve pressure control capability. NLO-12600c NLO-12596
   - Allowing DEH-TK-2 level to decrease to LT 7.6" will result in the trip and lockout of DEH-P-1A and 1B, the trip of DEH-EHC-1 and the

Comments / Reference: 4.603.A8	Revision: 36

Number: 4.603.A8	Use Category: CONTINUOUS	Major Rev: 036 Minor Rev: N/A Page: 49 of 65
Title: 603.A8 Annunciator Panel Alarms		

5-4 TURBINE GOVERNOR VALVE/THROTTLE VALVE TRIP BYPASS

5-4 WINDOW	SOURCE	AUTOMATIC ACTIONS
TURBINE GOV VLV/ THROTTLE VLV TRIP BYPASS	MS-PS-3B (RPS-RLY-K9B) OR MS-PS-3D (RPS-RLY-K9D) (1st Stage Pressure LE 142.3 PSIG, 30% equivalent)	• RPS-RLY-K9B bypasses the trip logic for RPT3A and RPT4B. RPS-RLY-K9D bypasses the trip logic for RPT4A and RPT3B. • RPS-RLY-K9B bypasses the turbine trip logic for RPS channel B1. RPS-RLY-K9D bypasses the turbine trip logic for RPS channel B2.

NOTE: Provides indication that Main Turbine first stage pressure is LE 30% equivalent Reactor Power.

1. IF Reactor Power is GT 30%,
   THEN **REFER** to Technical Specifications 3.3.1.1 and 3.3.4.1.

CAUTION

Bypass valves open with Reactor Power GT 30% may cause the Reactor Scram

Examination Outline Cross-reference:	Level	RO		SRO
Rev. 0 Date: 12/16/2016	Tier	1		
	Group	1		
	K/A	295016.AA1.06		
Level of Difficulty: 2	Importance Rating	4.0		

Control Room Abandonment: Ability to operate and/or monitor the following as they apply to CONTROL ROOM ABANDONMENT: Reactor water level

Question # 6

Which of the following describes the RPV level indications available at the RSD panel and ARSD panel?

At the RSD panel, ___①___ RPV level indication(s) is/are available. At the ARSD panel, ___②___ RPV level indication(s) is/are available.

A. ① Wide Range only
② Wide Range only

B. ① Wide Range only
② Wide Range and Fuel Zone

C. ① Wide Range and Fuel Zone
② Wide Range only

D. ① Wide Range and Fuel Zone
② Wide Range and Fuel Zone

Answer: A

K/A Match:

The question evaluates the candidates understanding of what indications are available to monitor reactor water level during control room abandonment which would impact their ability to correctly monitor RPV level during an accident when control room abandonment is required.

SRO Only:

N/A.

Explanation:

Only wide range indication is available at the RSD and ARSD panels.

A. Correct. Only wide range indication is available at the RSD and ARSD panels.

B. Incorrect. Plausible because fuel zone indication is an actual RPV level indication available in the control room and is designed for accident conditions and mitigating severe transients. Plausible because fuel zone indication would be desirable if a LOCA occurred while control room abandonment is necessary. Incorrect because fuel zone indication is not available outside the control room.

C. Incorrect. Plausible because fuel zone indication is an actual RPV level indication available in the control room and is designed for accident conditions and mitigating severe transients. Plausible because fuel zone indication would be desirable if a LOCA occurred while control room abandonment is necessary. Incorrect because fuel zone indication is not available outside the control room.

D. Incorrect. Plausible because fuel zone indication is an actual RPV level indication available in the control room and is designed for accident conditions and mitigating severe transients. Plausible because fuel zone indication would be desirable if a LOCA occurred while control room abandonment is necessary. Incorrect because fuel zone indication is not available outside the control room.

Technical Reference(s)		Attached w/ Revision # See Comments / Reference
SD000210, RSD/ARSD System Text (pages 19,20)		

Proposed references to be provided during examination: None

Learning Objective: RO-1057 Perform actions for a control room evacuation
5582 List calibration conditions and nominal ranges for each of the five ranges of level instruments

Question Source: Bank # LO01281
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 41.7
55.43

Comments / Reference: SD000126 r13 mr1	Revision:

COLUMBIA SYSTEMS November 2014
RSD SD000210, R10 MR0

VI. INSTRUMENTATION AND ALARMS

A. Control Room Annunciators

NOTE: See associated Annunciator Procedures for current setpoints.

1. P602.A5-1.6 RSD OR ALT RSD CONTR/TRANSFER SWITCH ACTIVATED
2. P602.A5-1.8 RHR-V-9 POWER ENABLED

B. RSP and ARS SRV position indication

RSP and ARS SRV position indicators identify the status of power to the operating solenoid; ie., solenoid deenergized is green and solenoid energized is red. The use of indirect position indication may not identify an open SRV or the failure of an SRV to open on demand.

C. Remote Shutdown Panel C61-P001, Instrumentation

RCIC-SI-1	Turbine speed	0 to 6 (X 1000) RPM
RCIC-FI-1R/1	RCIC flow	0 to 700 gpm
MS-PI-2	RPV pressure	0 to 15 (X 100) psig
MS-LI-10	RPV level	-150 to +60 inches
RHR-FI-5	RHR B Flow	0 to 1,000 (X 10)GPM

COLUMBIA SYSTEMS November 2014
RSD SD000210, R10 MR0

G. Alternate Remote Shutdown PNL E-CP-ARS

1.	CMS-TI-44AR	Supp Pool Air Temp	50 to 400°F
2.	MS-TI-41AR	Supp Pool Wtr Temp	50 to 400°F
3.	CMS-LI-1AR	Supp Pool Wtr Level	-25 to +25 inches
4.	RHR-FI-4AR	RHR Loop A Flow	0 to 1,000 (X 10) gpm
5.	MS-LI-10AR	RPV Level	-150 to +60 inches
6.	MS-PI-11AR	RPV Pressure	0 to 15 (X 100) psig
7.	SW-PI-32AR	SW-P1A Discharge Pressure	0 to 300 psig

Examination Outline Cross-reference:	Level	RO		SRO
Rev. 0 Date: 12/16/2016	Tier	1		
	Group	1		
	K/A	295018.AK3.01		
Level of Difficulty: 2	Importance Rating	2.9		

Partial or Complete Loss of Component Cooling Water: Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER : Isolation of non-essential heat loads

Question # 7

RCC-V-6, Reactor/Radwaste Building Isolation Valve, closes on low RCC system flow to provide sufficient cooling flow to the…

A. Reactor Recirculation (RRC) pump seals.

B. Fuel Pool Cooling (FPC) Heat Exchangers.

C. Control Rod Drive (CRD) Pump Bearing and Oil Coolers.

D. Reactor Water Cleanup (RWCU) Non-Regenerative Heat Exchangers.

Answer: A

K/A Match:
Question determines if the candidate understands the reason RCC-V-6 closes on low flow. It determines if they know which component is protected when the isolation occurs.

SRO Only:
N/A

Explanation:

On low flow in the RCC system for 10 seconds, RCC-V-6 will close isolating the reactor building and radwaste buildings from the system. The only components with flow remaining will be RRC pumps and drywell air cooling units.

A. Correct.

B. Incorrect. Plausible because Fuel Pool Cooling Heat Exchangers are cooled by RCC, but incorrect because RCC-V-6 does not close to provide additional flow to this component.

C. Incorrect. Plausible because CRD pumps are cooled by RCC, but incorrect because RCC-V-6 does not close to provide additional flow to this component.

D. Incorrect. Plausible because RWCU Heat Exchangers are cooled by RCC, but incorrect because RCC-V-6 does not close to provide additional flow to this component.

Technical Reference(s)		Attached w/ Revision # See Comments / Reference
SD000196, RCC System Text		

Proposed references to be provided during examination: None

Learning Objective: 5705 – State the purpose of the following components: (b) RCC-V-6

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ____

10 CFR Part 55 Content: 55.41 41.5
55.43 ____

Comments / Reference: SD000196	Revision: r14 mr1

discharge valves RCC-V-4A (B, C) as necessary. The discharge valves are normally full-open, but can be throttled if system pressure is too high. Lifting of the EDR-HX-2 relief valve (RCC-RV-44 @ 150 psig) is an indication of unacceptably high system pressure.

B. Abnormal

1. Loss of RCC (ABN-RCC) LO-7668 NLO-12207 NLO-12210a

   a) Reactor/Radwaste Building isolation valve RCC-V-6 automatically isolates the RCC cooling supply to all loads except those inside the Primary Containment, during a sustained RCC low-flow condition of GT 10 seconds (LT two RCC pumps running as sensed by breaker position).

   b) Automatic closure of RCC-V-6 isolates RCC cooling to the Reactor Bldg & Radwaste Bldg. requiring increased surveillance of the following Reactor Bldg loads: LO-11739b LO-11739c

      - Control Rod Drive (CRD) pump seals and oil cooler
      - Reactor Water Cleanup (RWCU) pump motor cavity temperature
      - RWCU Nonregenerative heat exchangers

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Examination Outline Cross-reference:	Level	RO		SRO
Rev. 0 Date: 12/16/2016	Tier	1		
	Group	1		
	K/A	295019.AA2.01		
Level of Difficulty: 3	Importance Rating	3.5		

Partial or Complete Loss of Instrument Air: Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR : Instrument air system pressure

Question # 8

Given the following:

- A leak has developed on the control air header.
- Control air header pressure is 92 psig and down slow.

As pressure continues to drop, select the next automatic action that will occur in the control and service air system.

A. Standby control air compressor STARTS.

B. Control/service air crosstie valve, SA-PCV-2, CLOSES.

C. Service air compressor, SA-C-1, STARTS.

D. Desiccant dryer bypass valve, CAS-PCV-1, OPENS.

Answer: B

K/A Match:
Question determines if the candidate understands automatic actions in the CAS system related to system pressure.

SRO Only:
N/A

Explanation:

Both standby air compressors start at 100psig. SA-PCV-2 closes at 80psig. CAS-PCV-1 opens at 75psig. Therefore the next action is for CAS-PCV-1 to close.

A. Incorrect. because the standby Control Air Compressors would have already started at 100psig. Plausible because standby air compressors start on lowering pressure.

B. Correct.

C. Incorrect. The service air compressor is normally running and does not have auto start functionality. Plausible because the service air compressor

D. Incorrect. CAS-PCV-1 closes at 75 psig and is not the **next** automatic action. Plausible because the valve has automatic actions on low pressure.

Technical Reference(s)		Attached w/ Revision # See Comments / Reference
ABN-CAS,Control Air System Failure		

Proposed references to be provided during examination: None

Learning Objective: 5878 – List the expected automatic Control Air system response due to a leak in the Control Air system.

Question Source: Bank # LO00435
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam 1996

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 41.10
55.43 ______

Comments / Reference: ABN-CAS	Revision: 9

Number: ABN-CAS	Use Category: CONTINUOUS	Major Rev: 009 Minor Rev: N/A Page: 3 of 27
Title: CONTROL AIR SYSTEM FAILURE		

1.0 ENTRY CONDITIONS

Loss of Control or Service Air header pressure, or inability of the CAS compressors to maintain pressure GT their setpoints.

2.0 AUTOMATIC ACTIONS

2.1 Both Standby Air Compressors start at 100 psig.

2.2 Service Air Header isolates at 80 psig (SA-PCV-2 **CLOSES**).

2.3 Control Air Desiccant Dryer bypasses at 75 psig (CAS-PCV-1 **OPENS**).

3.0 IMMEDIATE OPERATOR ACTIONS

None

Examination Outline Cross-reference:	Level	RO		SRO
Rev. 0 Date: 12/16/2016	Tier	1		
	Group	1		
	K/A	295021.2.4.11		
Level of Difficulty: 3	Importance Rating	4.0		

Loss of Shutdown Cooling: Knowledge of abnormal condition procedures.

Question # 9

CGS is in Mode 4.

- RHR Loop A is in Shutdown Cooling mode
- RHR Loop B is in Suppression Pool Cooling mode
- OSP-ELEC-M701, DG1 Monthly Operability Test is in progress

The following indications are observed:

- RHR A PUMP TRIP annunciator is alarming
- RHR-V-8, RHR-V-9 and RHR-V-6A, SDC Suction Isolation Valves, are OPEN
- RHR-V-48A, HX Shell Side Bypass, is throttled partially OPEN
- RHR-V-53A, SDC Return Valve, is OPEN
- RPV Pressure indicates 2 psig

What procedure should be entered?

A. ABN-ELEC-SM1/SM7

B. ABN-RHR-SDC-LOSS

C. ABN-RHR-SDC-ALT

D. ABN-RHR-SDC-PRESS

Answer: B

K/A Match:
The question determines whether a candidate can determine that shutdown cooling has been lost based on plant indications and select the correct procedure based on those conditions.

SRO Only:
N/A

Explanation:

SDC is lost because of RHR-P-1A tripping. The correct procedure to enter is ABN-RHR-SDC-LOSS

A. Incorrect. There is no loss of voltage condition stated in the stem. Plausible because an EDG surveillance is in progress and SM7 powers RHR A.

B. Correct. RHR-P-1A is tripped. Entry conditions are met for ABN-RHR-SDC-LOSS.

C. Incorrect. ABN-RHR-SDC-ALT is used when a normal SDC lineup is not available. This is not the case. Plausible because ABN-RHR-SDC-ALT has actions to correct a loss of SDC.

D. Incorrect. ABN-RHR-SDC-PRESS is used when there is an unexpected RISE in SDC suction pressure. Those conditions are not listed in the stem. ABN-RHR-SDC-PRESS is plausible because it relates to SDC system performance.

Technical Reference(s)	
ABN-RHR-SDC-LOSS, : Loss of Shutdown Cooling	Attached w/ Revision # See Comments / Reference
ABN-RHR-SDC-ALT, Residual Heat Removal Alternate Shutdown Cooling	
ABN-RHR-SDC-PRESS, : Leakage Into RHR SDC Suction Line	

Proposed references to be provided during examination: None

Learning Objective: RO-1302 – Respond to a loss of shutdown cooling

Question Source: Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 41.10
55.43 ______

Comments / Reference: ABN-RHR-SDC-LOSS	Revision: 6

1.0 ENTRY CONDITIONS

Loss of an in-service RHR shutdown cooling (SDC) loop.

2.0 AUTOMATIC ACTIONS

2.1 RHR valves (RHR-V-8, 9, 23, 53A, 53B) close on the following signals:

2.1.1 High RHR room area ΔT (GT 55°F)

2.1.2 High reactor steam dome pressure (125 psig)

2.1.3 Low reactor vessel water level (LT 13")

2.1.4 RHR system excess flow (125% of two loop flow)

2.1.5 High RHR room temp (GT 130 - 150°F)

2.2 IF RHR-V-8(9) starts to close,
THEN RHR-P-2A AND RHR-P-2B trip due to no suction flow path

3.0 IMMEDIATE OPERATOR ACTIONS

None

Comments / Reference: ABN-RHR-SDC-ALT	Revision: 13

Number: ABN-RHR-SDC-ALT	Use Category: CONTINUOUS	Major Rev: 013 Minor Rev: N/A Page: 3 of 54
Title: Residual Heat Removal Alternate Shutdown Cooling		

1.0 ENTRY CONDITIONS

Any of the following:

- RPV pressure is LT 135 psig, and normal Shutdown Cooling alignment is not available
- Normal Shutdown Cooling suction valves are inoperable
- RHR SDC is required to be restarted, but RPV pressure is GT 30 psig
- CRS/Shift Manager determine Alternate Shutdown Cooling operation is required during high heat load conditions

2.0 AUTOMATIC ACTIONS

Comments / Reference: ABN-RHR-SDC-PRESS	Revision: 4

Number: ABN-RHR-SDC-PRESS	Use Category: CONTINUOUS	Major Rev: 004 Minor Rev: N/A Page: 3 of 8
Title: Leakage Into RHR SDC Suction Line		

1.0 ENTRY CONDITIONS

Unexplained increase in RHR SDC suction pressure.

2.0 AUTOMATIC ACTIONS

Examination Outline Cross-reference:	Level	RO		SRO
Rev. 0 Date: 12/16/2016	Tier	1		
	Group	1		
	K/A	295023.AK1.03		
Level of Difficulty: 3	Importance Rating	3.7		

Refueling Accidents: Knowledge of the operational implications of the following concepts as they apply to REFUELING ACCIDENTS : Inadvertent criticality

Question # 10

CGS is in Mode 5. Fuel Shuffling is in progress.

Initial source range counts:

- SRM-A: 75 cps, steady
- SRM-B: 105 cps, steady
- SRM-C: 125 cps, steady
- SRM-D: 93 cps, steady

As a fuel bundle is moved into the core, the control room operator notes the following source range response:

- SRM-A: 152 cps, steady
- SRM-B: 188 cps, up slow
- SRM-C: 305 cps, up slow
- SRM-D: 156 cps, steady

What actions should be taken?

Immediately stop fuel bundle insertion. Then…

A. withdraw the fuel bundle from the core and evaluate the cause of the source range level increase.

B. bundle insertion may be continued slowly with close observation of subcritical multiplication behavior.

C. perform a subcritical check and obtain permission from the Station Nuclear Engineer prior to moving the fuel bundle.

D. validate proper source range response and obtain peer check prior to moving the fuel bundle.

Answer: A

K/A Match:
Question determines if candidates are able to identify that inadvertent criticality has occurred and what actions must be taken based on this.

SRO Only:
N/A

Explanation:

PPM 6.3.2, Fuel Shuffling and/or Offloading and Reloading states that if SRM count rate exceeds 300 counts/second, immediately stop bundle insertion and withdraw the assembly from the core.

A. Correct.

B. Incorrect. Plausible because if counts double, bundle insertion may continue following observation of subcritical multiplication behavior. Incorrect because 300 counts have been exceeded on SRM indications.

C. Incorrect. Plausible because PPM 6.3.2 directs a subcritical check be performed at various points during the core loading process. Incorrect because 300 counts have been exceeded on SRM indications.

D. Incorrect. Plausible because source range response is continually monitored during fuel loading operations. Also plausible because peer checks are used for reactivity manipulations. Incorrect because 300 counts have been exceeded on SRM indications.

Technical Reference(s)		Attached w/ Revision # See Comments / Reference
PPM 6.3.2, Fuel Shuffling and/or Offloading and Reloading		

Proposed references to be provided during examination: None

Learning Objective: 7700 – State the indications used to identify criticality during fuel loading

Question Source: Bank # LX00330
Modified Bank # X (Note changes or attach parent)
New

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 41.8
55.43

Comments / Reference: PPM 6.3.2 Section 6.2.1	Revision: 23 mr 2

CAUTION

SRM count rates should not be allowed to exceed 300 counts per second during bundle insertion.

b. During each bundle insertion into the core, the SRMs should be monitored by the Control Room Operator in direct communication with the refueling bridge to ascertain that criticality will not be attained.

NOTE: Inserting a bundle near an SRM may have a strong effect on the SRM count rate. An increase in SRM count rate may occur when any bundle is inserted within two face adjacent locations from an SRM. If any other bundle insertion causes a substantial SRM count rate increase, as described below, it is considered unexpected.

c. If an unexpected doubling in average SRM count rate occurs, or two doublings of any single SRM count rate occurs, then stop the insertion of the bundle. Bundle insertion may then be continued slowly with close observation of subcritical multiplication behavior. Once the bundle is inserted, notify the Station Nuclear Engineer, they will determine whether a substantial reduction in margin to criticality has occurred. Resumption of the shuffle shall require the concurrence of the Reactivity Manager.

1) If any SRM count rate exceeds 300 counts/second, immediately stop bundle insertion and withdraw the assembly from the core. Contact the Station Nuclear Engineer for evaluation. Shuffling will not continue until the cause is determined. Resumption of the shuffle shall require the concurrence of Plant Management

Comments / Reference: PPM 6.3.2 Section 6.2.4 and 6.4.1	Revision: 23 mr 2

c. The Core Alt Supervisor or designee should document all moves on the NCTL.

d. Verification of each step should be documented by the verifier's/designee's initials. {P-104550}

6.2.3 When fuel shuffling has been completed, perform a full core verification per PPM 6.3.5.

6.2.4 After full core verification, perform a subcritical check per PPM 6.3.3. {P-104550}

6.3 Full Core Offload Procedure

6.3.1 Initiate FLEX actions in accordance with SOP-FLEX-FULL CORE OFFLOAD.

6.3.2 Perform the steps, in order listed on the Nuclear Components Transfer List (NCTL).

6.3.3 The Core Alt Supervisor should verify the identity of each fuel assembly by location

loading pattern the count rate will be the average of the coupled SRMs.)

e. After the initial four bundles are loaded around each SRM, if the 1/M plots indicate that the next bundle insertion will result in a critical condition, the loading procedure should be suspended at that point and a subcritical check should be performed to demonstrate that adequate margin to criticality exists. {P-104550}

Examination Outline Cross-reference:	Level	RO		SRO
Rev. 1 Date: 12/16/2016	Tier	1		
	Group	1		
	K/A	295024.EK2.08		
Level of Difficulty: 3	Importance Rating	4.0		

High Drywell Pressure: Knowledge of the interrelations between HIGH DRYWELL PRESSURE and the following: ADS

Question # 11

CGS is operating in Mode 1.

- A large LOCA into primary containment occurred.
- Wetwell pressure cannot be restored and maintained less than PSP.
- Emergency depressurization is required.

Why is it preferable to open ADS SRVs with the current plant conditions?

When compared to other SRVs, ADS SRVs are preferred since…

A. pressure will be lowered more rapidly due to their larger capacity.

B. the instantaneous trip channel results in faster initiation.

C. they can be opened with a loss of containment instrument air.

D. they are designed to operate at higher primary containment temperatures.

Answer: D

K/A Match:
The K/A asks for understanding of the relationship between high drywell pressure and SRVs. The correct answer is based on the fact that ADS SRVs are designed to operate in more adverse conditions (higher drywell pressure and temperature) than other SRVs.

SRO Only:
N/A

Explanation:

Per PPM 5.0.10, ADS SRVs are preferred because of their qualifications.

A. Incorrect. ADS SRVs due not have a higher capacity than other SRVs. Plausible because ADS SRVs are designed differently (logic, qualifications, etc.) than other SRVs.

B. Incorrect. The instantaneous trip channel does NOT result in faster initiation. Plausible because an instantaneous trip channel exists in the system.

C. Incorrect. Plausible since all SRVs will operate upon a loss of containment instrument air. Therefore, this is not a difference between ADS SRVs and non-ADS SRVs.

D. Correct. ADS SRVs are designed to maintain operability during adverse combinations of loadings and forces resulting from a Loss of Coolant Accident (LOCA) so that the core can be effectively cooled.

Technical Reference(s)		Attached w/ Revision # See Comments / Reference
PPM 5.0.10, Flowchart Training Manual		
CGS System Description, Vol. 7, Chap. 5, Automatic Depressurization.		

Proposed references to be provided during examination: None

Learning Objective: 11874 - Describe the physical connection and/or cause-and-effect relationship between the Automatic Depressurization System and the following: Drywell pressure

Question Source: Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 41.7
55.43 ______

Comments / Reference: PPM 5.0.10	Revision: 21 mr 1

Number: 5.0.10	Use Category: INFORMATION	Major Rev: 021 Minor Rev: 001 Page: 196 of 320
Title: Flowchart Training Manual		

d. Step P-4:

1) If Wetwell water level is at or below 17 ft, then insufficient water is in the Wetwell to adequately cover the SRV discharge device. Steam discharged through the SRVs could pass directly into the Wetwell airspace, and cause direct pressurization of the Wetwell airspace. Since the extent of this pressurization cannot be predicted and may exceed the pressure capability of the primary containment, SRV operation is prohibited and alternate methods of depressurization must be attempted.

e. Step P-5:

1) Depressurization of the RPV is most easily and rapidly performed by opening SRVs; thus instructions for operation of these valves are specified first, in preference to steps directing the use of other depressurization systems and mechanisms. Of the SRVs, those dedicated to the ADS function are generally the most reliable because of their qualifications, pneumatic supply, the design and operation of initiation circuitry, and the availability of control power. Additionally, the relative location of their quenchers distributes the heat load around the Wetwell.

2) Concurrent opening of all ADS valves is within analyzed plant design limits. Other steps in the EOP flowcharts provide instructions for maintaining sufficient Wetwell heat capacity to accommodate simultaneous opening of all ADS valves at any RPV pressure.

3) The specific method to be used for opening ADS valves is purposely not stated in this step. Depending on event-specific considerations, one

Comments / Reference: CGS System Description, Automatic Depressurization, section II, Design Bases	Revision: 12

COLUMBIA SYSTEMS MAY 2016
ADS SD000186, r12

I. PURPOSE

The Automatic Depressurization System (ADS) is an emergency system designed to relieve steam pressure in the main steam lines and reactor vessel to allow the Low Pressure ECCS systems to inject. LO-5067

In the event of a small break in the Reactor Coolant Pressure Boundary (RCPB) concurrent with a failure of the High Pressure Core Spray (HPCS) system to adequately cool the reactor core, ADS will automatically open seven safety relief valves (Fig. 1) allowing the RHR system and the Low Pressure Core Spray (LPCS) System to flood the core.

II. DESIGN BASES

ADS is designed to:

1. Provide automatic depressurization of the RCPB in the event of small breaks in the RPV Pressure Boundary so that the LPCI and the LPCS systems can operate to flood the reactor vessel to protect the fuel barrier from excessive temperature.
2. Permit verification of system operability.
3. Withstand and maintain operability during adverse combinations of loadings and forces resulting from a Loss of Coolant Accident (LOCA) so that the core can be effectively cooled.
4. Be capable of operation regardless of the availability of offsite power supplies and the normal power generating system of the plant.

The ADS SRV 42 gallon accumulator will provide for at least one ADS actuation with drywell at the maximum design pressure during a Design Basis Accident and the reactor at 0 psig.

Examination Outline Cross-reference:	Level	RO		SRO
Rev. 0 Date: 12/16/2016	Tier			
	Group			
	K/A	295025.EK3.02		
Level of Difficulty: 3	Importance Rating	3.9		

<u>High Reactor Pressure</u>: Knowledge of the reasons for the following responses as they apply to HIGH REACTOR PRESSURE : Recirculation pump trip

Question # 12

CGS is operating in Mode 1.

- A transient caused reactor pressure to rise to 1153 psig.
- Both Reactor Recirculation (RRC) Pumps have tripped.

Which of the following describes the reason for the trip of the RRC pumps?

Tripping the RRC pumps…

A. increases core inlet subcooling which reduces Reactor power.

B. adds additional negative reactivity by increasing the voiding in the core.

C. moves the boiling boundary up the fuel channel which adds negative reactivity.

D. overcomes the power increase caused by the moderator temperature increase due to the rising RPV pressure.

Answer: B

K/A Match:
Question direct asks the reason that a high pressure trip of RRC pumps is needed.

SRO Only:
N/A

Explanation:

The basis for technical specification 3.3.4.2 states that the high pressure RRC pump trip is to add negative reactivity due to increase in steam voiding as core flow decreases.

A. Incorrect. Tripping RRC pumps does not increase core inlet subcooling. Plausible because starting RRC pumps does increase core inlet subcooling.

B. Correct.

C. Incorrect. Tripping RRC pumps does not raise the boiling boundary. Plausible because starting RRC pumps or raising RRC flow does increase the boiling boundary.

D. Incorrect. The reason for tripping RRC pumps is NOT

Technical Reference(s)		Attached w/ Revision # See Comments / Reference
TS B3.3.4.2,ATWS without SCRAM RRC Pump Trip		

Proposed references to be provided during examination: None

Learning Objective: 5022 – Describe the physical and or caused and effect relationship between the RRC system and the following: (a)Core Flow (b) Reactor Power (c) NBI

Question Source: Bank # LO001750
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam 2009, question #12

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 41.5
55.43 ______

Comments / Reference: B 3.3.4.2	Revision: 91

B 3.3 INSTRUMENTATION

B 3.3.4.2 Anticipated Transient Without Scram Recirculation Pump Trip (ATWS-RPT) Instrumentation

BASES

BACKGROUND

The ATWS-RPT System initiates a recirculation pump trip, adding negative reactivity, following events in which a scram does not, but should occur, to lessen the effects of an ATWS event. Tripping the recirculation pumps adds negative reactivity from the increase in steam voiding in the core area as core flow decreases. When Reactor Vessel Water Level - Low Low, Level 2 or Reactor Vessel Steam Dome Pressure - High setpoint is reached, the recirculation pump motor breakers trip.

The ATWS-RPT System (Ref. 1) includes sensors, relays, bypass capability, circuit breakers, and switches that are necessary to cause initiation of a recirculation pump trip. The channels include electronic equipment (e.g., trip relays) that compares measured input signals with pre-established setpoints. When the setpoint is exceeded, the channel outputs an ATWS-RPT signal to the trip logic.

The ATWS-RPT consists of two independent trip systems, with two channels of Reactor Vessel Steam Dome Pressure - High and two channels of Reactor Vessel Water Level - Low Low, Level 2, in each trip system. Each ATWS-RPT trip system is a two-out-of-two logic for each

Examination Outline Cross-reference:	Level	RO		SRO
Rev. 0 Date: 12/16/2016	Tier	1		
	Group	1		
	K/A	295026.EA1.01		
Level of Difficulty: 3	Importance Rating	4.1		

Suppression Pool High Water Temperature: Ability to operate and/or monitor the following as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE: Suppression pool cooling

Question # 13

Columbia is operating in Mode 1.

- A LOCA occurred 15 minutes ago.
- RPV water level is -140 inches, up slow.
- Wetwell temperature is 95°F, up slow.
- RHR Loops B and C are injecting to recover RPV level.

Control room operators are aligning RHR A for Suppression Pool cooling due to high Wetwell temperature.

- RHR-V-42A (LPCI injection valve) is OPEN
- RHR-V-4A (Suppression Pool suction valve) is OPEN
- RHR-V-24A (Suppression Pool cooling valve) is CLOSED
- RHR-V-6A (Shutdown Cooling suction valve) is CLOSED
- RHR-V-48A (RHR Heat Exchanger bypass valve) is OPEN
- RHR-P-2A is RUNNING
- SW-P-1A is RUNNING

What actions are required?

A. Throttle open RHR-V-24A, throttle open RHR-V-48A as needed.

B. Close RHR-V-42A, throttle open RHR-V-24A, close RHR-V-48A.

C. Open RHR-V-24A, open RHR-V-6A, close RHR-V-4A.

D. Close RHR-V-42A, open RHR-V-24A, throttle open RHR-V-48A as needed.

Answer: B

K/A Match:

The question requires the candidate to MONITOR the system by analyzing the current status and then OPERATE the system to change the status in response to High Suppression Pool Temperature.

SRO Only:

N/A

Explanation:

The correct answer is B. The system is in LPCI Injection Mode due to actuation at -129" and RHR-V-42A being open. To lineup for suppression pool cooling, RHR-V-42A must be closed manually prior to RHR-V-24A is opened. RHR-V-48A is closed to maximize cooling in EOP situations.

A. Incorrect. RHR-V-24A will not open unless RHR-V-42A is closed. Plausible because answer would be correct if automatic actuation had not occurred and it was not an emergency situation.

B. Correct Answer

C. Incorrect. RHR-V-24A will not open unless RHR-V-42A is closed. Plausible because IF RHR-V-42A were closed in the stem, this would result in a working lineup.

D. Incorrect. RHR-V-48A should not be throttle in this situation, but fully closed. Plausible because RHR-V48A is allowed to be throttled when not in an emergency situation.

Technical Reference(s)		Attached w/ Revision # See Comments / Reference
SOP-RHR-SPC-QC, Placing RHR in Suppression Pool Cooling QC		

Proposed references to be provided during examination: None

Learning Objective: RO-0218 – Initiate RHR system in suppression pool cooling.

Question Source: Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 41.7
55.43 ______

Comments / Reference: SOP-RHR-SPC-QC	Revision: 5

Number: SOP-RHR-SPC-QC	Use Category: CONTINUOUS	Major Rev: 005 Minor Rev: N/A Page: 4 of 5
Title: Placing RHR Loop A(B) in Suppression Pool Cooling - Quick Card		

2.0 PROCEDURE

2.1 Placing RHR A(B) in Suppression Pool Cooling During EOP's

2.1.1 **VERIFY** RHR-P-2A(B) running. ____

2.1.2 **VERIFY** SW-P-1A(B) running. ____

NOTE: RHR-V-48A(B) may be closed concurrently while opening RHR-V-24A(B).

2H 2.1.3 **THROTTLE OPEN** RHR-V-24A(B) to between 4500 and 7000 gpm. ____

2H 2.1.4 **CLOSE** RHR-V-48A(B). ____

CAUTION

Operation of multiple ECCS pumps following a LOCA, with one RHR heat exchanger not operable, may exceed the maximum calculated temperature for NPSH. If either RHR heat exchanger is inoperable following a LOCA, then minimize operation of ECCS pumps not required for Adequate Core Cooling or Containment Integrity. {P-255468}

2.1.5 IF operating per the EOPs,
THEN **MAXIMIZE** cooling flow. ____

2.1.6 IF NOT operating per the EOPs,
THEN **THROTTLE** RHR-V-48A(B) to maintain temperature between 55-90°F. ____

Examination Outline Cross-reference:	Level	RO		SRO
Rev. 0 Date: 12/16/2016	Tier	1		
	Group	1		
	K/A	295028.2.4.6		
Level of Difficulty: 3	Importance Rating	3.7		

High Drywell Temperature: Knowledge of EOP mitigation strategies.

Question # 14

Given the following parameters:

- HPCS and CRD are injecting into RPV.
- RPV Pressure is 550 psig and down slow.
- RPV Level is –158 inches and steady.
- Drywell Temperature is 285 degrees F and up slow.
- Drywell Pressure is 8 psig and up slow.
- Drywell conditions are within the Drywell Spray Initiation Limit (DSIL).
- Wetwell Pressure is 1.5 psig and up slow.
- Wetwell level is 29 feet and steady.

Which of the following should be prioritized as the next action to be taken?

A. Initiate Drywell sprays

B. Raise WW level with RHR

C. Initiate Wetwell Sprays

D. Open 7 ADS SRVs

Answer: A

K/A Match:

Question determines if candidate knows major EOP mitigation strategies based on high drywell pressure.

SRO Only:

N/A

Explanation:

Per PPM 5.2.1, when drywell temperature cannot be maintained below 135 degrees, drywell sprays should be initiated provided WW level is below 51ft and drywell temp is below DSIL.

A. Correct.

B. Incorrect. Plausible because WW level needs to be raised. However, WW level is raised using HPCS during EOPs per PPM 5.5.23

C. Incorrect. WW pressure is too low for wetwell spray. Plausible because WW spray is directed when drywell pressure is above 1.68psig (PPM 5.2.1 step P-2 to P-5)

D. Incorrect. Emergency depressurization limits have not been met. Plausible based on low RPV level and accident conditions.

Technical Reference(s)		Attached w/ Revision # See Comments / Reference
PPM 5.2.1, Primary Containment Control		

Proposed references to be provided during examination: None

Learning Objective: RO-0635 – Maintain Drywell temperature below 330 degrees F

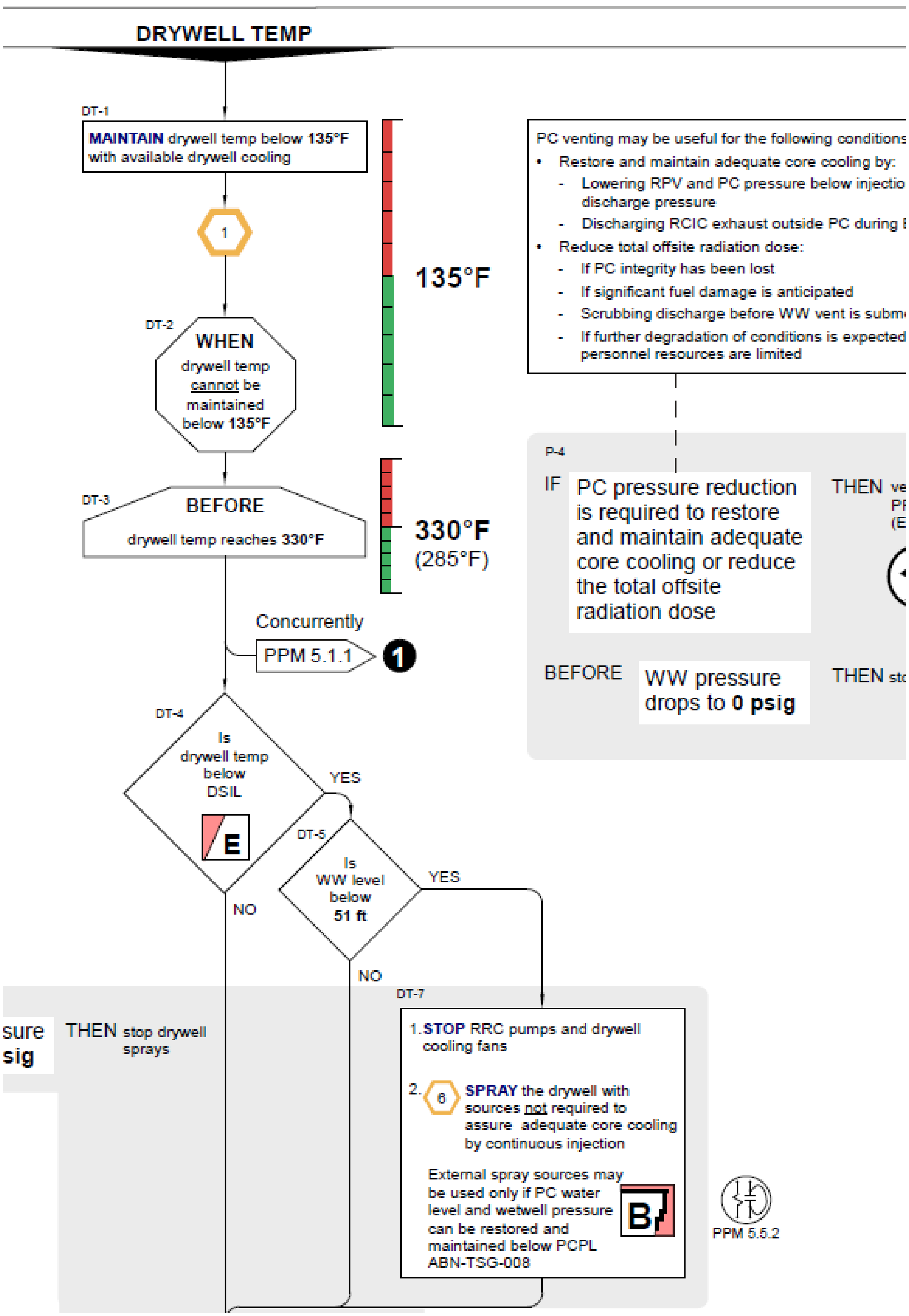
Question Source: Bank # LO01890
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam 2009, question #14

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 41.10
55.43 ______

Comments / Reference: PPM 5.2.1	Revision:



Examination Outline Cross-reference:	Level	RO		SRO
Rev. 0 Date: 12/16/2016	Tier	1		
	Group	1		
	K/A	295030.EK1.01		
Level of Difficulty: 3	Importance Rating	3.8		

Low Suppression Pool Water Level: Knowledge of the operational implications of the following concepts as they apply to LOW SUPPRESSION POOL WATER LEVEL: Steam condensation

Question # 15

Refer to the following conditions:

- All control rods are fully inserted
- Drywell Pressure is 12 psig and rising
- RPV Pressure is 550 psig and stable
- RPV Water Level is -50 inches
- RHR A PUMP ROOM HIGH WATER LEVEL annunciator is locked in
- Wetwell Level is 18’ 9” and lowering
- Wetwell Pressure is 8 psig and rising
- HPCS PUMP MOTOR OVERCURRENT annunciator is locked in

What action should be taken:

A. Depressurize the RPV. Maintain less than a 100°F/HR cooldown rate.

B. Stop and Prevent ECCS pumps per Table 18, Vortex and NPSH Limits.

C. Emergency Depressurize using 7 ADS SRVs.

D. Spray the Wetwell using RHR A.

Answer: C

K/A Match:
The candidate must recognize from the plant conditions in the stem that WW level is below 19'2", the point at which inadequate steam condensation becomes a concern. They must then select the correct action to take (operational implications) based on that condition.

SRO Only:
N/A

Explanation:
With WW level below 19'2", Emergency Depressurization is required per PPM 5.2.1 step L-6.

A. Incorrect. Emergency depressurization is required per PPM 5.2.1 step L-6. Plausible because with the reactor shutdown and no boron injection required, the RPV is cooled down per PPM 5.1.2 step P-6 and P-8.

B. Incorrect. Wetwell level is still too high to direct stopping and preventing of ECCS pumps. Plausible because table 18 lists WW levels in which ECCS pump operation is no longer allowed.

C. Correct. PPM 5.2.1 step L-6 directs emergency depressurization.

D. Incorrect. Wetwell sprays should not be directed using RHR A because room water levels are high. Plausible because wetwell spray could be directed per step P-6 of PPM 5.2.1

Technical Reference(s)		Attached w/ Revision # See Comments / Reference
PPM 5.2.1, Primary Containment Control, step L-6		

Proposed references to be provided during examination: None

Learning Objective: 13567 – Given a copy of EOPs and an event, describe the basis for each variable and figure used to execute EOP strategies without error.

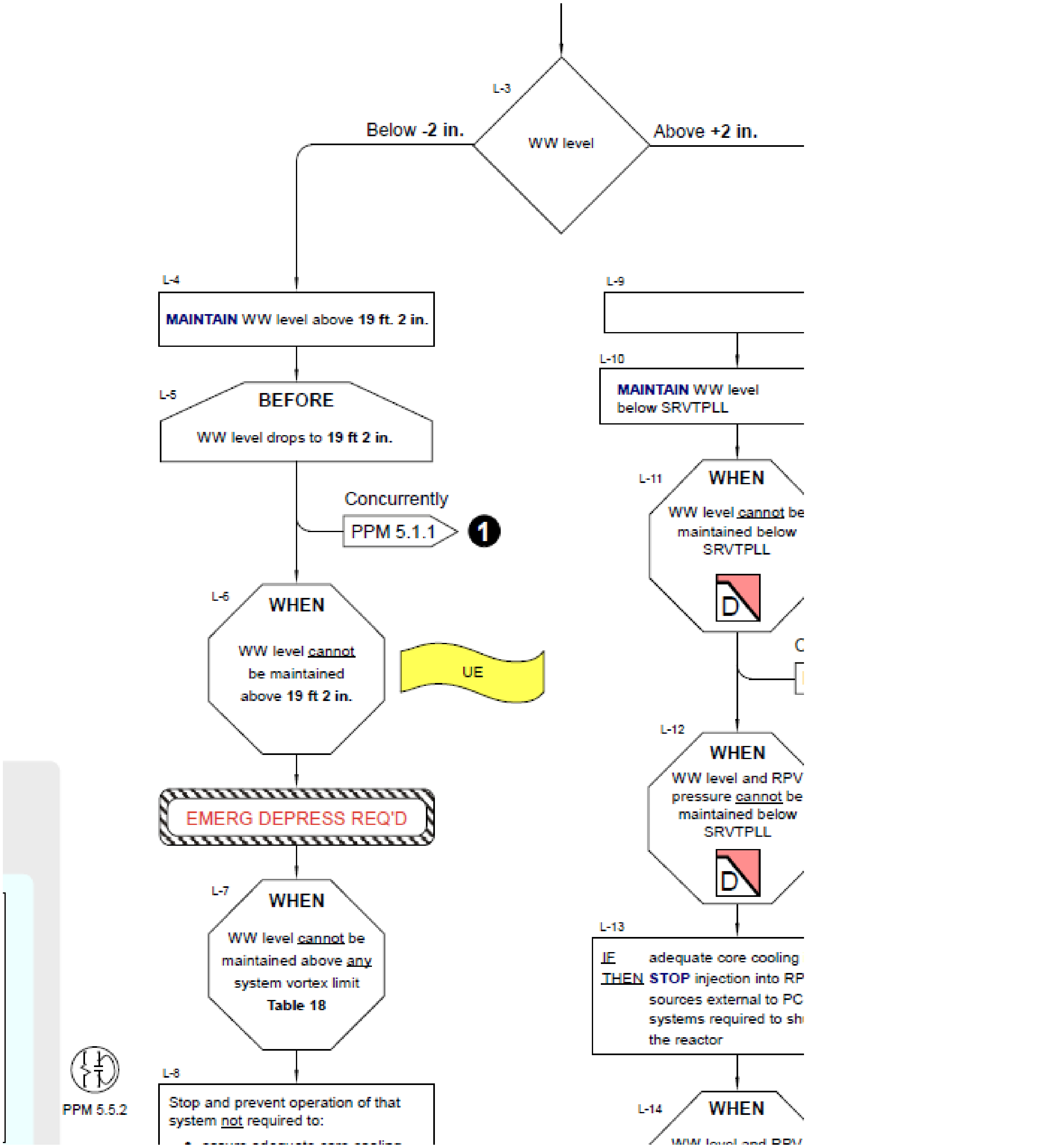
Question Source:
Bank # ___
Modified Bank # ___ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level:
Memory or Fundamental Knowledge X
Comprehension or Analysis ___

10 CFR Part 55 Content:
55.41 41.8 to 41.10
55.43 ___

Comments / Reference: PPM 5.2.1	Revision: 23



Examination Outline Cross-reference:	Level	RO		SRO
Rev. 0 Date: 12/16/2016	Tier	1		
	Group	1		
	K/A	295031.EA1.05		
Level of Difficulty: 3	Importance Rating	4.3		

Reactor Low Water Level: Ability to operate and/or monitor the following as they apply to REACTOR LOW WATER LEVEL : Reactor core isolation system

Question # 16

CGS is operating in Mode 1.

- A loss of all feedwater causes a reactor scram.
- RPV level lowered to -65 inches prior to restoring feed.

Current plant conditions:

- Actions of PPM 3.3.1, Reactor Scram are complete through scram reset.
- The crew is controlling RPV level +13 inches to +54 inches.
- Drywell pressure is 0.8 psig, up slow.
- The CRS has directed resetting NS4 logic per ABN-FAZ.

What actions should the crew take to reset containment isolation logic?

Prior to depressing the Isolation Logic A&B and C&D pushbuttons on H13-P601…

A. vent the Drywell per SOP-CN-CONT-VENT.

B. return both SGT trains to standby.

C. restore power to busses load shed from SM-7 and SM-8.

D. place the RCC pumps in pull-to-lock.

Answer: D

K/A Match:
Question asks how to operate the reactor core isolation system (called NS4 system at Columbia) following an actuation on low RPV level (A signal).

SRO Only:
N/A

Explanation:

Per ABN-FAZ step 4.2.19, RCC pumps should be place in PTL prior to resetting the signal in step 4.2.20.

A. Incorrect. Venting the drywell is not required for an “A” signal. Plausible because venting the drywell is required for an “F” signal (high drywell pressure).

B. Incorrect. Returning both SGT trains to standby is not required prior to reset. Plausible because SGT trains start on an “A” signal.

C. Incorrect. Restoring load shed busses is not required PRIOR to reset. Plausible because any FAZ signal causes non-critical loads to be shed.

D. Correct answer.

Technical Reference(s)		Attached w/ Revision # See Comments / Reference
ABN-FAZ, FAZ		

Proposed references to be provided during examination: None

Learning Objective: 15764 – With the procedures available, discuss all contingencies associated with the subsequent operator actions of ABN-FAZ

Question Source:
Bank # ____
Modified Bank # LO03416 (Note changes or attach parent)
New ____

Question History: Last NRC Exam None

Question Cognitive Level:
Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content:
55.41 41.7
55.43 ____

Comments / Reference: ABN-FAZ	Revision: 17 mr 5

Number: ABN-FAZ	Use Category: CONTINUOUS	Major Rev: 017 Minor Rev: 005 Page: 7 of 34
Title: FAZ		

4.2.19 **PLACE** the control switches for the following RCC pumps in **PTL**:

- RCC-P-1A ____
- RCC-P-1B ____
- RCC-P-1C ____

CAUTION

The resetting of any containment isolation logic and subsequent opening of any containment isolation valve prior to ensuring that fuel failure has not occurred may result in radiological release to the environment.

NOTE: The Emergency Director and/or CRS/Shift Manager's permission is to be obtained to reset any containment logic following any containment isolation, to prevent inadvertent opening of isolation valves, or to open containment isolation valves.

4.2.20 WHEN RPV level and Drywell pressure have stabilized,
AND the F and A signals no longer exist,
THEN **OBTAIN** the Emergency Director and/or CRS/Shift Manager's permission to reset NS4 logic and restore isolated systems, ____
AND **DEPRESS** the Isolation Logic A&B and C&D pushbuttons on H13-P601. ____

Comments / Reference: Original Question LO03416	Revision: N/A

A valid "A" signal has been received due to a loss of all feedwater. RPV level has been returned to a band of +13" to +54".

What is the required order for the following steps from ABN-FAZ?

(1) Reset the NS4 logic

(2) Place the RCC pumps in pull-to-lock

(3) Restore power to busses load shed from SM-7 and SM-8

(4) Start an RCC pump

(5) Open the RCC primary containment valves

A. (1)-(2)-(3)-(4)-(5)

B. (1)-(3)-(2)-(4)-(5)

C. (1)-(5)-(2)-(3)-(4)

D. (2)-(1)-(3)-(5)-(4)

Answer: D

Examination Outline Cross-reference:	Level	RO		SRO
Rev. 0 Date: 12/16/2016	Tier	1		
	Group	1		
	K/A	295037.EK3.05		
Level of Difficulty: 3	Importance Rating	3.2		

SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown: Knowledge of the reasons for the following responses as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKOWN: Cold shutdown boron weight.

Question # 17

CGS is operating in Mode 1.

- A transient causes an ATWS.
- Reactor power is 24%.
- The MSIVs are open.
- SLC-P-1A and 1B are in operation and injecting normally.
- Reactor level is -70", controlled with Feed and Condensate.

Which of the following is correct concerning these conditions?

Cooldown…

A. is not permitted until Cold Shutdown Boron Weight has been injected because core reactivity response for a partially borated core is unpredictable.

B. is not permitted because additional heat load will be imposed on the primary containment that could lead to containment failure.

C. is permitted to start when Hot Shutdown Boron Weight has been injected because core reactivity response for a partially borated core is unpredictable.

D. is permitted as long as it is secured if the core returns to power.

Answer: A

K/A Match:
Question directly asks the REASON why cooldown is not allowed in an ATWS situation when cold shutdown boron weight has not been reached.

SRO Only:
N/A

Explanation:

Per PPM 5.0.10, cooldown is not permitted in a failure to scram condition due to unpredictable reactivity response.

A. Correct.

B. Incorrect. The concern is not related to drywell parameters. Plausible because cooling down the reactor with SRVs raises containment pressure.

C. Incorrect. The requirement is related to COLD Shutdown Boron Weight, not HOT Shutdown Boron Weight. Plausible because Hot Shutdown Boron Weight exists and applies to BIIT (Boron Injection Initiation Temperature).

D. Incorrect. Cooldown is not allowed. Plausible because stopping cooldown would stop the insertion of positive reactivity.

Technical Reference(s)		Attached w/ Revision # See Comments / Reference
PPM 5.0.10, Flowchart Training Manual		
PPM 5.2.1, RPV Control - ATWS		

Proposed references to be provided during examination: None

Learning Objective: 8184 – Given a list, identify the statement that describes plant response to conducting PPM 3.2.1, “Plant Shutdown”, during an ATWS if Cold Shutdown Weight of boron has not been injected.

Question Source: Bank # LO01280
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam None.

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 41.5
55.43 ______

Comments / Reference: 5.0.10	Revision: 21 mr 1

Number: 5.0.10	Use Category: INFORMATION	Major Rev: 021
Title: Flowchart Training Manual		Minor Rev: 001 Page: 69 of 320

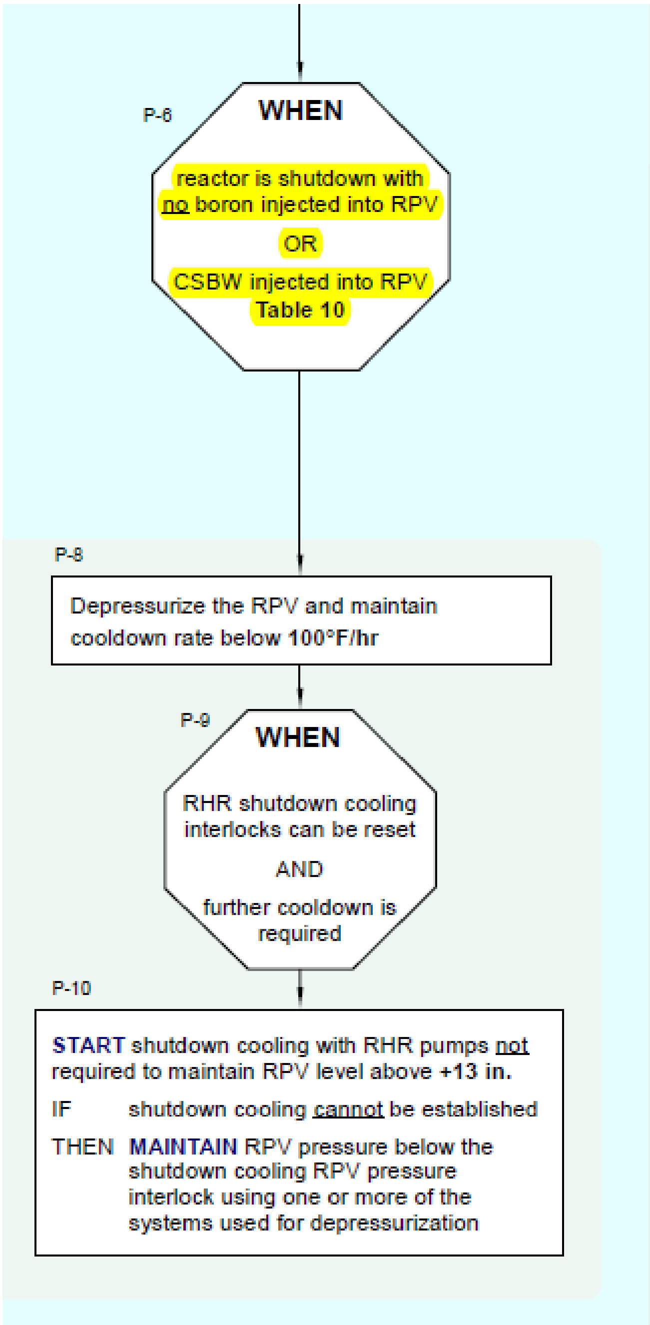
7.1.3 In practice, the weight of elemental boron cannot be measured directly because it is used in a solution of sodium pentaborate or weights of borax and boric acid. If boron injection is initiated with the SLC system and initial tank volume is 4500 gal, the CSBW of boron is injected when SLC tank level drops to 2995 gallons. This corresponds to a volume of 1505 gallons of sodium pentaborate solution at the Technical Specification minimum required boron concentration. If the SLC tank level is below 1505 gallons when the SLC system is initiated, the SLC tank will need to be refilled in order to inject the CSBW.

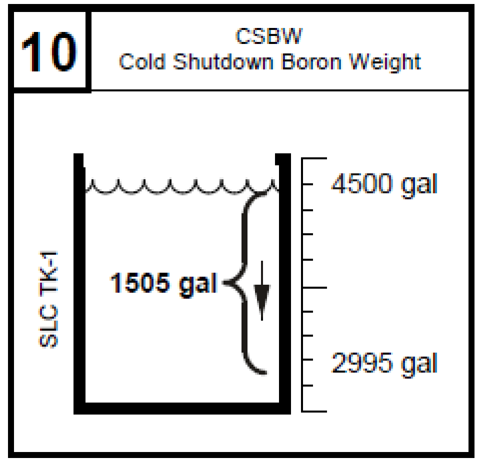
7.1.4 If any amount of boron less than the CSBW has been injected into the RPV, cooldown is not permitted in failure-to-scram conditions unless it can be determined that control rod insertion alone assures the reactor will remain shutdown under all conditions. The core reactivity response from cooldown in a partially borated core is unpredictable and subsequent steps may not prescribe the correct actions for such conditions if criticality were to occur.

7.1.5 If no boron has been injected into the RPV, depressurization and cooldown in failure-to-scram conditions may proceed as long as control rod insertion is sufficient to shut down the reactor. Such action is permitted even though the existing margin to criticality is small. A return to criticality under these conditions is acceptable because termination of the cooldown will stop the reactor power increase.

7.1.6 The CSBW, (Table 10), is utilized in the RPV pressure control flowpath of PPM 5.1.2 and in PPM 5.1.5.

Comments / Reference: PPM 5.1.2, RPV Control, ATWS	Revision: 24





Examination Outline Cross-reference:	Level	RO		SRO
Rev. 0 Date: 12/16/2016	Tier	1		
	Group	1		
	K/A	295038.EK2.07		
Level of Difficulty: 3	Importance Rating	3.5		

High Off-Site Release Rate: Knowledge of the interrelations between HIGH OFF-SITE RELEASE RATE and the following: Control room ventilation

Question # 18

CGS is operating in Mode 1

- A High Reactor Building HVAC Exhaust Plenum Radiation Level isolation signal (‘Z’ signal) is generated.

How will the Control Room HVAC system respond?

The Control Room HVAC system will shift to…

A. PRESSURIZATION MODE with the Remote Air Intakes isolated.

B. RECIRCULATION MODE with the Normal Air Intake isolated.

C. PRESSURIZATION MODE with the Normal Air Intake isolated.

D. RECIRCULATION MODE with the Remote Air Intakes isolated.

Answer: C

K/A Match:
The question asks how the Control Room Ventilation System will respond on a high exhaust plenum radiation level (Z signal).

SRO Only:
N/A

Explanation:
On a “Z” signal, the Control Room Ventilation air intake valves isolate and suction is shifted to the remote air intakes to PRESSURIZE the control room.

A. Incorrect. The remote air intakes would not be isolated. Plausible because the remote air intakes are not normally used.

B. Incorrect. The system would be in PRESSURIZATION mode. Plausible because the normal air intakes are isolated and because RECIRCULATION implies no outside air enters the control room.

C. Correct.

D. Incorrect. The remote air intakes are not isolated. Plausible because the remote air intakes are not normally used and because RECIRCULATION implies that no outside air enters the control room.

Technical Reference(s)		Attached w/ Revision # See Comments / Reference
SD000201, CR-HVAC System Description		

Proposed references to be provided during examination: None

Learning Objective: 5225 – State the automatic features associated with the following Control Room HVAC components: (f) Normal Supply Isolation, (g) Remote Intake Valves

Question Source: Bank # ______
Modified Bank # LO02947 (Note changes or attach parent)
New ______

Question History: Last NRC Exam None

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 41.7
55.43 ______

Comments / Reference: SD000201, CR-HVAC System Description	Revision: 15 mr1

COLUMBIA SYSTEMS December 2015
CR-HVAC SD000201, r15 mr1

2. Isolation Signal

The outside air supply dampers to the control room air-handling units WMA-AD-51A1/B1 and the normal control room air intake valves WOA-V-51C/52C auto isolate after FAZ signal. The CR Emergency Filtration system starts due to a FAZ signal and takes suction from the remote ground level intakes to pressurize the control room. The toilet/kitchen exhaust fan WEA-FN-51 is also tripped. NLO-12432b LO-7649

Reset of the FAZ signal CRHVAC isolation is accomplished by depressing the CR Emergency Filter Sys. Div. 1 FAZ Reset pushbuttons WMA-RMS-FAZ-3AXY/BXY located on the RC-1/2 panels in the control room. These pushbuttons restore Main Control Room Ventilation lineup to normal.

WMA-AD-51A1/B1 have a design leak rates of 0.5% of rated flow. Because inlet pressure of the emergency filter fan is lower than the inlet pressure of the normal supply fan, negative pressure develops in the inlet of the bypass damper when the emergency unit operates. This prevents

Comments / Reference: Original Bank Question, LO02947	Revision: N/A

An offsite release is in progress when the following alarms are received:

REMOTE INTAKE DIV 1 RAD HI-HI
REMOTE INTAKE DIV 2 RAD HI-HI

Considering only these alarms, how will the Control Room HVAC system respond?

A. WOA-V-51C and WOA-V-52C (Normal Outside Air Intake Isolation Valves) close
WMA-AD-51A1 and WMA-AD-51B1 (Outside Air Supply Dampers) close
WMA-FN-54A and WMA-FN-54B (Emergency Fltr Supply Fans) start
WEA-FN-51 (Kitchen Exhaust Fan) stops

B. WOA-V-51A and WOA-V-51B (Remote Air Intakes) open
WOA-V-51C and WOA-V-52C (Normal Outside Air Intake Isolation Valves) close
WMA-FN-54A and WMA-FN-54B (Emergency Fltr Supply Fans) start

C. WMA-AD-51A1 and WMA-AD-51B1 (Outside Air Supply Dampers) close
WEA-FN-51 (Kitchen Exhaust Fan) stops
WEA-AD-51 (Outlet Damper) closes

D. WMA-AD-51A1 and WMA-AD-51B1 (Outside Air Supply Dampers) close
WMA-FN-54A and WMA-FN-54B (Emergency Fltr Supply Fans) start
WEA-FN-51 (Kitchen Exhaust Fan) stops
WEA-AD-51 (Outlet Damper) closes

Answer: C

Examination Outline Cross-reference:	Level	RO		SRO
Rev. 0 Date: 12/16/2016	Tier	1		
	Group	1		
	K/A	600000.AA2.14		
Level of Difficulty: 3	Importance Rating	3.0		

Plant Fire On Site: Ability to determine and interpret the following as they apply to PLANT FIRE ON SITE: Equipment that will be affected by fire suppression activities in each zone

Question # 19

CGS was operating in Mode 1 when a fire in the reactor building was reported. The crew entered ABN-FIRE, Fire. Immediate actions have been completed and the crew is performing step 4.1.4 of subsequent operator actions.

> NOTE: An "Appendix R Fire" is defined as a fire or explosion in a Fire Protection (FP) vital area where safe shutdown system parameters indicate degraded performance or there is visible damage to vital plant equipment or cabling. If uncertain if the fire is Appendix R, then treat the fire as an Appendix R fire. Refer to Attachment 13.2 for Fire Areas identified as FP Vital Areas.

4.14 IF the fire is an Appendix R Fire,
OR becomes an Appendix R Fire,
THEN PERFORM Section 5.0 concurrently with Section 6.0, and Section 8.0. _____

The fire brigade leader has reported the following:

- The fire is out and a re-flash watch is stationed.
- There is extensive smoke in Room 410. Specific equipment damage cannot be ascertained at this time.

Using the references provided, what actions are REQUIRED for post-fire safe shutdown (PFSS)?

If a reactor shutdown is required, …

A. use Division 2 PFSS systems. Start RHR-P-2A within 30 minutes.

B. use Division 1 PFSS systems. Verify RCIC-V-22 OR RCIC-V-59 is open.

C. use Division 2 PFSS systems. Open door R408 to provide passive cooling to Room 410.

D. use Division 1 PFSS systems. Close RCC-V-130, FPC HX RCC Outlet, from the control room.

Answer: D

K/A Match:

Question asks candidates to use plant procedures and maps to determine equipment impacted by the fire in certain fire protection zones and actions required based on this.

SRO Only:

N/A

Explanation:

Students must analyze the reactor 522 SE Quadrant Map to determine that the equipment in room 410 contains “vital” systems and should be classified as “Appendix R”. They must then correlate this with another section of the procedure to determine what actions to take for area R-18, which is to close RCC-V-130.

A. Incorrect. Actions are from the wrong area. Plausible because fire areas other than R-18 use Division 2 systems and have actions as specified in this distractor.

B. Incorrect. Actions are from the wrong area. Plausible because fire areas other than R-18 use Division 1 systems and have actions as specified in this distractor.

C. Incorrect. Actions are from the wrong area. Plausible because fire areas other than R-18 use Division 2 systems and have actions as specified in this distractor.

D. Correct.

Technical Reference(s)		Attached w/ Revision # See Comments / Reference
ABN-FIRE, Fire		
PFB-RB-522, Reactor 522 SE Quadrant Map		

Proposed references to be provided during examination: PFP-RB-522, Reactor 522 SE Quadrant Map
ABN-FIRE pages 39-42, 56-58

Learning Objective: 15765 – With procedures available, discuss all contingencies associated with the subsequent operator actions of ABN-FIRE.

Question Source: Bank # _____
Modified Bank # _____ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 ______________
55.43 ______________

Comments / Reference: ABN-FIRE	Revision: 36 mr 1

Number: ABN-FIRE	Use Category: CONTINUOUS	Major Rev: 036 Minor Rev: 001 Page: 39 of 67
Title: Fire		

APPENDIX R FIRE AREA OPERATOR MANUAL ACTIONS

Fire Area R-1

Location: Reactor Building General Area all levels (excluding Containment and ECCS Pump Rooms)

Protected Level Inst: MS-LR/PR-623B

Required Actions:

★ • **USE** Division 2 Post Fire Safe Shutdown systems. _____

CAUTION

Fire in the RB 471 east could result in RB door locking, and preventing access.

• **PERFORM** the following: {AR-12926}

a. **NOTIFY** CAS/SAS of intent to open E-CB-DP/SS/IN4/A/4 _____

b. **OBTAIN** key 157 from the Control Room key locker. _____

c. **UNLOCK** and **OPEN** grated door C-245 (RW 467 vestibule to the Division 1 battery room) _____

d. **OPEN** E-CB-DP/SS/IN4/A/4 _____

CAUTION

Loss of keep fill pressure may cause a water hammer during pump start. The following actions should be taken from the Control Room.

★ • **START** RHR-P-2A within 30 minutes (Control Room). {12.1}, {P-80442} _____

★ • **START** RHR-P-2B within 30 minutes (Control Room). {12.1}, {P-80442} _____

NOTE: Fire in this area may affect RHR-FIS-10B, which could affect operation of RHR-FCV-64B.

★ • **VERIFY** RHR-FCV-64B opens upon the start of RHR-P-2B (Control Room) {AR-42457} _____

Attachment 13.1, Appendix R Fire Area Operator Manual Actions

Number: ABN-FIRE	Use Category: CONTINUOUS	Major Rev: 036 Minor Rev: 001 Page: 40 of 67
Title: Fire		

- IF RHR-FI-603B is erratic or inoperable, THEN **USE** RHR-P-2B ammeter (RHR-AM-P/2B) to verify RHR flow exists (H13-P601). _____

CAUTION

Radiological monitoring is suggested prior to entry. The use of SCBA may be required.

★ • **VERIFY CLOSE** RWCU-V-32 within 3 hours, Manual Blowdown Isolation (RB 501 SW). _____

NOTE: Door wedges are in portable lantern boxes.

★ • **OPEN** the door R408 to MCC 8B/8BA within 3 hours to provide passive cooling to the room (RB 522 SE). _____

★ • **CLOSE** RCC-V-50 within 3.5 hours of the start of the fire (RCC Surge Tank Isolation) (RB 572 NE). {12.1} _____

NOTE: The Shift Managers extension 2441 is PFSS credited and should be used as the primary control room contact number, the CRS extension is to be used as the backup.

★ **MONITOR** the Spent Fuel Pool temperature and level locally within 4 hours. Infrared temperature monitoring device is available in the tool crib. **REFER** to ABN-FPC-LOSS. {P-80442} _____

Attachment 13.1, Appendix R Fire Area Operator Manual Actions

Number: ABN-FIRE	Use Category: CONTINUOUS	Major Rev: 036 Minor Rev: 001 Page: 41 of 67
Title: Fire		

Fire Area R-4

Location: RHR-P-2B Pump Room, RHR B Pipe Chase from RB 470 to RB 548, RB 548 South Valve Room, RHR-HX-B room RB-548 and 572.

Protected Inst: MS-LR/PR-623A

Required Actions:

★ • **USE** Division 1 Post Fire Safe Shutdown systems. _____

★ • **CLOSE** RCC-V-130 from the Control Room (FPC HX RCC Outlet) (H13-P626) within 1 hour of the start of the fire. This action is required since a fire may cause SW-V-187B to open and cause service water to be cross-tied to RCC. Resulting flooding and a loss of LPCS-P-2 (RHR A/LPCS keep fill pump). {12.1} _____

Fire Area R-6

Location: RCIC Pump Room, RB 422 and 444

Protected Level Inst: MS-LR/PR-623A

Required Actions:

★ • **USE** Division 1 Post Fire Safe Shutdown systems. _____

• **DECLARE** RCIC Inoperable and **NOTIFY** Security _____

Fire Area R-7

Location: RHR-P-2C Pump Room, RB 422 and 444

Protected Inst: MS-LR/PR-623A

Required Actions:

★ • **USE** Division 2 Post Fire Safe Shutdown systems. _____

> **CAUTION**
>
> Loss of keep fill pressure may cause a water hammer during pump start. The following actions should be taken from the Control Room.

★ • **START** RHR-P-2B within 30 minutes. {12.1},{P-80442} _____

Attachment 13.1, Appendix R Fire Area Operator Manual Actions

Number: ABN-FIRE	Use Category: CONTINUOUS	Major Rev: 036 Minor Rev: 001 Page: 42 of 67
Title: Fire		

Fire Area R-8

Location: LPCS Pump Room, RB 422 and 444

Protected Level Inst: MS-LR/PR-623B

Required Actions:

★ • **USE** Division 2 Post Fire Safe Shutdown systems. ______

CAUTION

Loss of keep fill pressure may cause a water hammer during pump start. The following actions should be taken from the Control Room.

★ • **START** RHR-P-2A within 30 minutes (Control Room). {12.1}, {P-80442} ______

Fire Area R-18

Location: DIV 2 MCC Room, RB 522

Protected Level Inst: MS-LR/PR-623A

Required Actions:

★ • **USE** Division 1 Post Fire Safe Shutdown systems. ______

CAUTION

Loss of keep fill pressure may cause a water hammer during pump start. The following actions should be taken from the Control Room.

★ • **START** RHR-P-2B within 30 minutes. {12.1}, {P-80442} ______

★ • **CLOSE** RCC-V-130 from the Control Room (FPC HX RCC Outlet) (H13-P626) within 1 hour of the start of the fire. This action is required since a fire may cause SW-V187B to open and cause service water to be cross-tied to RCC. Resulting flooding and a loss of LPCS-P-2 (RHR A/LPCS keep fill pump). {12.1} ______

Number: ABN-FIRE	Use Category: CONTINUOUS	Major Rev: 036 Minor Rev: 001 Page: 56 of 67
Title: Fire		

FIRE AREAS

FIRE AREA	PSFF DIV LOST	FIRE LOCATION	FP VITAL AREA
DG-1	1	DG 441-455 HPCS Generator Room	Yes
DG-2	1	DG 441-455 Diesel Generator 1 Room	Yes
DG-3	2	DG 441-455 Diesel Generator 2 Room	Yes
DG-4	1	DG 441 Diesel Generator 1 Diesel Oil Tank Pump Room	Yes
DG-5	2	DG 441 Diesel Generator 2 Diesel Oil Tank Pump Room	Yes
DG-6	N	DG 441 HPCS Diesel Oil Tank Pump Room	No
DG-7	N	DG 441 HPCS Diesel Day Tank Room	No
DG-8	1	DG 441 DG-1 Diesel Oil Day Tank Room	Yes
DG-9	2	DG 441 DG-2 Diesel Oil Day Tank Room	Yes
DG-10	N	DG 455 Deluge Valve Equipment Room	No
R-1	1	Reactor Building 471-606 General Equipment Area including 422 NW Stairwell, 441 Train Bay, and NE Vestibule 471-606. (Does not include instrument rack fire areas)	Yes
R-2	U	Primary Containment (Inerted during operation)	No
R-3	N	RB 422 HPCS Pump Room	No
R-4	2	RB 422 RHR-B Pump Room, 470-548 Pipe Chase, 471 SW Valve Room, 492 & 563 Pipe Tunnels, 548 South Valve Room, 548 & 572 Heat Exchanger Equipment Rooms and 572 Hydrogen Recombiner Room	Yes
R-5	1	RB 422 RHR-A Pump Room, 470-548 Chase, 492 & 563 Pipe Tunnels (west half) and 548 Heat Exchanger Equipment Room	Yes
R-6	2	RB 422 RCIC Pump Room	Yes
R-7	1	RB 422 RHR-C Pump Room	Yes
R-8	1	RB 422 LPCS Pump Room	Yes
R-9	N	RB 422-623 SW Stair A6	No
R-10	N	RB 422-623 SW Elevator 2	No
R-11	N	RB 422-623 NE Stair A5	No

Attachment 13.2, Fire Areas

Number: ABN-FIRE	Use Category: CONTINUOUS	Major Rev: 036 Minor Rev: 001 Page: 57 of 67
Title: Fire		

FIRE AREA	PSFF DIV LOST	FIRE LOCATION	FP VITAL AREA
R-12	N	RB 441-623 NE Elevator 1	No
R-15	N	RB 422 Lobby Outside of Stair A5 & Elevator 1	No
R-18	2	Division 2 MCC Room	Yes
R-21	2	RB 522 South Valve Room	Yes
M-9	2	RB 471 Instrument Rack E-IR-H22/P009 Enclosure	Yes
M-21	2	RB 501 Instrument Rack E-IR-H22/P021 Enclosure	Yes
M-27	2	RB 522 Instrument Rack E-IR-H22/P027 Enclosure	Yes
RC-1	N	RW 437-507 General Equipment Areas (excluding below rooms)	No
RC-1 (Vital Portion)	2	RW 437 Rms C-106 & C-108	Yes
RC-2	1	RW 484 Cable Spreading Room	Yes
RC-3	1	RW 467-525 Cable Chase	Yes
RC-4	1	RW 467 Division 1 Electrical Equipment Rooms (Battery Charger Room No. 1 and RPS Room No. 1)	Yes
RC-5	1	RW 467 Battery Room 1	Yes
RC-6	2	RW 467 Battery Room 2	Yes
RC-7	2	RW 467 Division 2 Electrical Equipment Rooms (Battery Charger Room No. 2 and RPS Room No. 2)	Yes
RC-8	2	RW 467 Switchgear Room 2	Yes
RC-9	2	RW 467 Remote Shutdown Room	Yes
RC-10	U	RW 510 Main Control Room	Yes
RC-11	1	RW 525 Unit A Air Condition Room Division 2 PFSS Feeders	Yes
RC-12	2	RW 525 Unit B Air Conditioning Room	Yes
RC-13	2	RW 525 Communications Room, Emergency Chiller Area and HVAC Chase	Yes
RC-14	1	RW 467 Switchgear Room 1	Yes
RC-19	2	RW 467 Vital Island Corridor C-205	Yes
RC-20	1	RW 467 Pipe Chase and 487 PASS Area	Yes

Attachment 13.2, Fire Areas

Number: ABN-FIRE	Use Category: CONTINUOUS	Major Rev: 036 Minor Rev: 001 Page: 58 of 67
Title: Fire		

FIRE AREA	PSFF DIV LOST	FIRE LOCATION	FP VITAL AREA
SW-1	1	Standby Service Water Pump House 1A	Yes
SW-2	2	Standby Service Water Pump House 1B	Yes
TG-1	N	Turbine Generator Building General Areas 441-501 (including E-W corridors adjacent RW & RB)	No
Zone TG-2	N	TB 441 East Turbine Oil Storage Room	No
TG-3	N	TB 441-518 East Stair A1	No
TG-4	N	TB 441-518 East Elevator 3	No
Zone TG-5	N	TB 441 Auxiliary Boiler Room	No
TG-6	N	TB 441-501 North Stair A3	No
Zone TG-7	N	TB 441 Hydrogen Seal Oil Room	No
TG-8	N	TB 441-501 NW Stair A4	No
Zone TG-9	N	TB 471 Turbine Oil Reservoir Room	No
Zone TG-10	N	TB 441 West Transformer Vault	No
Zone TG-11	N	TB 441 East Transformer Vault (by Aux. Boiler)	No
Zone TG-12	2	441 Corridors between RW & RB, DG & RB (includes Rooms C121, D104 and D113)	Yes

NOTE: Plant areas not listed do not contain post-fire safe shutdown equipment or cables and can be considered non-FP vital.

Division Code Legend

DIV 1 Fire area contains Div. 1 post-fire safe shutdown equipment or cables which may be lost. Use Division 2 PFSS systems for a fire in this area.

DIV 2 Fire area Contains Div. 2 post-fire safe shutdown equipment or cables which may be lost. Use Division 1 PFSS systems for a fire in this area.

DIV N Fire area does not contain equipment or cables for either division of post-fire safe shutdown equipment.

DIV U This code indicates that this fire area has been uniquely analyzed for post fire safe shutdown.

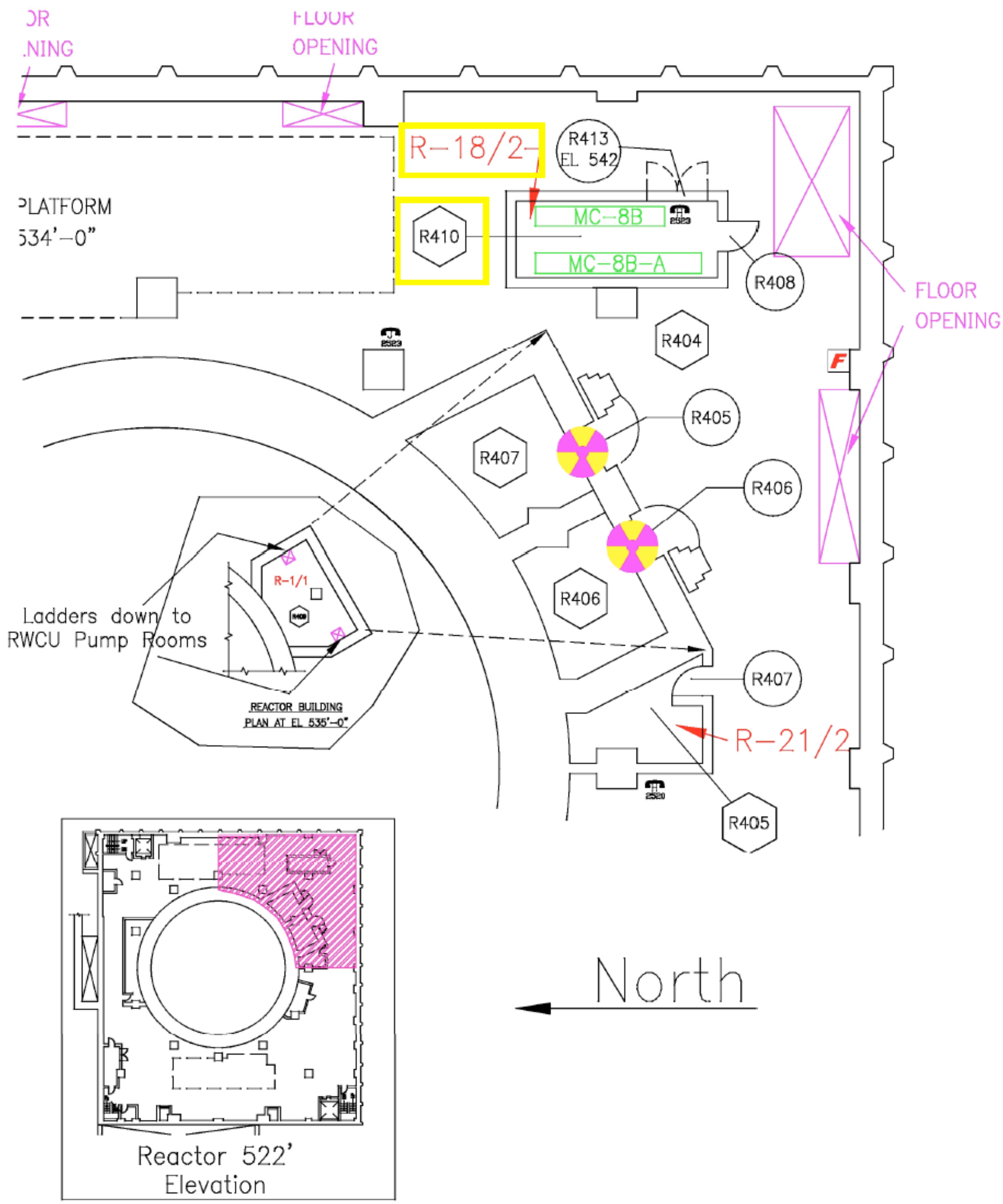
END

Attachment 13.2, Fire Areas

Comments / Reference: PFP-RB-522	Revision: 5

Number: PFP-RB-522	Use Category: N/A	Major Rev: 005 Minor Rev: N/A Page: 10 of 14
Title: REACTOR 522	Fire Area: R-1, 18, 21, M-27	

Location: SE QUADRANT	Fire Area: R-1/1, 18/2, 21/2	Page 1 of 2



Number: PFP-RB-522	Use Category: N/A	Major Rev: 005 Minor Rev: N/A Page: 11 of 14
Title: REACTOR 522	Fire Area: R-1, 18, 21, M-27	

Location: SE QUADRANT	Fire Area: R-1/1, 18/2, 21/2	Page 2 of 2

Areas covered by this Pre-Fire Plan:

- General Floor Area R404
- Pipe Space R405
- RWCU Pump Room 1B R406
- RWCU Pump Room 1A R407
- Access Area above RWCU Pump Rooms R409
- MCC Room R410

Post Fire Safe Shutdown Equipment or Associated Cabling:

Fire Area R-1/1, Room R404

SW-42-8BA10C	Motor Starter for SW-M-V/187B (NEMA 1)	N7/3.9

Fire Area R-21/2, Room R405;

RHR-V-42B	LPCI Isolation	N1/5.7

Fire Area R-18/2, Room R410;

E-MC-8B	Motor Control Center 8B	N/3.8
E-MC-8BA	Motor Control Center 8BA+	N/3.8

Special Entry Considerations:
RWCU Pump Rooms are Locked HIGH HIGH Radiation Areas. Contact HP for keys to access.

Fire Detection:
Ionization detectors.
Manual Pull Station Alarms

Suppression Systems and Isolation Location:
N/A

Electrical Disconnects or Special Shutoffs:
None

Exposure or Fire Extension Routes:
Smoke could extend throughout general floor area and up or down through open hatches to elevations. Because of the small amount of combustible materials, fire extension is improbable from this fire area.

Radiological Considerations:
Potential To Be High. Consult HP Tech for current conditions.

Combustibles or Gasses:
Cables, Hydraulic fluid, Cabinets of protective clothing

Fire Loading:
LOW

Emergency Ventilation:
Building Exhaust System

Special Hazards:
None

Number: PFP-RB-522	Use Category: N/A	Major Rev: 005 Minor Rev: N/A
Title: REACTOR 522	Fire Area: R-1, 18, 21, M-27	Page: 14 of 14

Location: FIRE FIGHTING EQUIPMENT LEGEND	Fire Area: None	Page 1 of 1

Containment Fire Barrier (Unique)

Essential Fire Barrier (3 Hr Rated)

Non–Essential Fire Barrier (3 Hr Rated)

Non–Essential Fire Barrier (2 Hr Rated)

Gaseous Suppression System

Dry Chemical Suppression System

Deluge Suppression System

Wet Pipe Sprinkler System

Pre Action Suppression system

Wheeled Dry Chem Extinguisher

Suppression System Iso Valve

Yard Hydrant

Hose Station

ABC PFX

Halon PFX

Foam/Water PFX

Foam Cart

Fire Protection Control Panel

CO2 Hose Reel

FLC Flammable Liquid Cabinet

F Manual Pull Station

FS Manual Supp Activation Pull Station

H 1234 Phonejack & number, WITH handset

J 5678 Phonejack & number, NO handset

Locked High or Very High Radiation Area

R320 Door Number

R320 Room Number

Area of Concern identifier on Key Plan

Fire Area designator explanation:

Example, using R–1/1

R–1 is the actual Fire Area

The number after the / is the Division of PFSS Components that are not protected from the effects of fire.

1 = Div 1, 2 = Div 2, # = none

U = Both Divs where alt SD is utilized

Legend for Pre Fire Plans

Examination Outline Cross-reference:	Level	RO		SRO
Rev. 0 Date: 12/16/2016	Tier	1		
	Group	1		
	K/A	700000.2.4.4		
Level of Difficulty: 3	Importance Rating	4.5		

Generator Voltage and Electric Grid Disturbances: Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures.

Question # 20

CGS is operating in Mode 1 when the following alarms are received:

- XFMR TR-S TROUBLE, P800.C4.2-7
- OSCILLOGRAPH STARTED, P800.C4.4-3

The crew observes the following Startup Transformer (E-TR-S) input phase voltages:

- ΦA: 235 kv
- ΦB: 10 kv
- ΦC: 235 kv

What actions should the crew take?

The crew should enter…

A. ABN-ELEC-GRID, Degraded Off Site Power Grid.

B. ABN-ELEC-LOOP, Loss of All Off-Site Electrical Power.

C. ABN-ELEC-SM2/SM4, SM-2, SM-4 and SL-21 Distribution System Failures.

D. ABN-ELEC-SM1/SM7, SM-1, SM-7, SM-75, SM-72, SL-71, SL-73 & SL-11 Distribution System Failures.

Answer: A

K/A Match:

Students must analyze phase voltages to determine an abnormal condition exists and then determine which procedure to enter based on those conditions. In this case, the indications are related to a grid disturbance/malfunction.

SRO Only:

N/A

Explanation:

The conditions presented in the stem match entry conditions in ABN-ELEC-GRID which state, "A loss of one phase on E-TR-S as determined by the following: Low voltage on one or more phases"

A. Correct Answer

B. Incorrect. Offsite power is still available through other sources so ABN-ELEC-LOOP is not applicable. Plausible because TR-S is an offsite source.

C. Incorrect. Power to SM-2 and SM-4 are from TR-N (Mode 1) and have not lost power. Plausible because TR-S powers SM-2 and SM-4 when the plant is shutdown.

D. Incorrect. Power to SM-1 and SM-7 are from TR-N (Mode 1) and have not lost power. Plausible because TR-S powers SM-1 and SM-7.

Technical Reference(s)		Attached w/ Revision # See Comments / Reference
ABN-ELEC-GRID, Degraded Off Site Power Grid		

Proposed references to be provided during examination: None

Learning Objective: 12153 – Given plant annunciation and indications, evaluate conditions for entry into ABN-ELEC-GRID.

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 41.10

55.43

Comments / Reference: ABN-ELEC-GRID	Revision: 7 mr 3

Number: ABN-ELEC-GRID	Use Category: CONTINUOUS	Major Rev: 007 Minor Rev: 003 Page: 3 of 16
Title: Degraded Off Site Power Grid		

1.0 ENTRY CONDITIONS

1.1 The BPA dispatcher notifies Columbia Generating Station Control Room of any of the following:

- The grid analysis program has projected a degraded reliability of the 500 kV system. }
- The grid analysis program has projected a degraded reliability of the 230 kV or 115 kV power source to CGS, such that the post trip voltage for TR-S or TR-B will be LT the allowable values. {P-226926}
- Current 500 kV system condition is degraded, and cannot be immediately restored.
- Current TR-S voltage is LE 235.0 kV, and cannot be immediately restored to Technical Specification limits. {AR-298300
- Current TR-B voltage is LE 115.0 kV, and cannot be immediately restored to Technical Specification limits. {AR-298300
- The grid analysis program is inoperable, or not available. {P-229616}

1.2 Unanticipated loss of E-TR-S or E-TR-B.

1.3 A loss of one phase on E-TR-S as determined by the following:{AR-OE-257993}{AR-259724}

- H13-800.C4-2.7 (TRANSFORMER TR-S TROUBLE) due to loss of one phase.
- Low voltage on one or more phases
- H13-800.C4,4-3 OSCILLOGRAPH STARTED (alarm point J15, 230KV Undervoltage (any phase))
- Plant loads tripping on overcurrent
- Plant loads unable to be started

1.4 A loss of one phase on E-TR-B as determined by the following: {AR-263740}

Examination Outline Cross-reference:	Level	RO		SRO
Rev. 0 Date: 12/16/2016	Tier	1		
	Group	2		
	K/A	295002.AA2.04		
Level of Difficulty: 2	Importance Rating	2.8		

Loss of Main Condenser Vacuum: Ability to determine and/or interpret the following as they apply to LOSS OF MAIN CONDENSER VACUUM : Offgas system flow

Question # 21

The plant is operating at 21% power. A large condenser air leak causes main condenser backpressure to rise from 2" Hg to 18" Hg.

Offgas flow is now ______(1)_____than its original value because ______(2)______.

A. (1) higher
(2) more non-condensable gasses are being removed from the condenser

B. (1) higher
(2) second stage air ejectors automatically come online as backpressure rises

C. (1) lower
(2) the turbine tripped

D. (1) lower
(2) MSIVs have closed

Answer: A

K/A Match:
The candidate must understand the implications of high main condenser backpressure including the expected offgas flow change.

SRO Only:
N/A

Explanation:

Offgas flow is expected to rise with rising condenser backpressure due to the removal of additional non-condensable gasses.

A. Correct.

B. Incorrect. Second stage air ejectors do not automatically come online with increasing backpressure. Plausible because the steam supply to the second stage air ejectors changes automatically based on main steam pressure.

C. Incorrect. The turbine does not trip when reactor power is below 30%. Plausible because the turbine would trip if power were above 30% which would impact offgas flow and condenser vacuum.

D. Incorrect. The MSIVs do not close until 21.6" Hg backpressure. Plausible because closing MSIVs would remove driving steam from the air ejectors.

Technical Reference(s)		Attached w/ Revision # See Comments / Reference
ABN-BACKPRESSURE, Loss of Main Condenser Backpressure		

Proposed references to be provided during examination: None

Learning Objective: 5111 Discuss the interrelationships between the Air Removal System and the following: (d) Off-Gas System

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 41.10
55.43

Comments / Reference: ABN-BACKPRESSURE	Revision: 5 mr 1

Number: ABN-BACKPRESSURE	Use Category: CONTINUOUS	Major Rev: 005 Minor Rev: 001 Page: 3 of 14
Title: Loss of Main Condenser Backpressure		

1.0 ENTRY CONDITIONS

1.1 Unexpected rise of Main Condenser backpressure.

1.2 High SJAE drain tank temperature (GT 141 degrees)

2.0 AUTOMATIC ACTIONS

2.1 Main Turbine trip

- For turbine loads LT 560 MWe, 5.5" Hg back pressure.
- For Turbine loads 560-835 MWe, linear rise from 5.5-8.0" Hg back pressure.
- For Turbine loads GE 835 MWe, 8.0" Hg back pressure.

2.2 21.6" Hg back pressure - Main Steam Isolation Valve closure

2.3 22.9" Hg back pressure - Turbine bypass valve closure

2.4 29.9" Hg back pressure - Reactor Feedwater Drive Turbine trip

4.18 Offgas flow provides indication of air in-leakage to the Main Condenser or associated systems that could cause a loss of back pressure condition. The normal offgas flow would be that recorded during recent shifts. This flow may vary from cycle to cycle based on system integrity.

4.19 This step checks the plant for possible in-leakage paths to the Main Condenser. Identified leaks should be stopped or minimized to recover back pressure. The components listed are those that could be leaking. The normal Offgas flow would be that recorded during recent shifts and may vary between cycles.

Examination Outline Cross-reference:	Level	RO		SRO
Rev. 0 Date: 12/16/2016	Tier	1		
	Group	2		
	K/A	295009.AK2.01		
Level of Difficulty: 2	Importance Rating	3.9		

Low Reactor Water Level: Knowledge of the interrelations between LOW REACTOR WATER LEVEL and the following: Reactor water level indication.

Question # 22

CGS is operating in Mode 1 when an automatic reactor scram is initiated.

Current plant conditions:

- Narrow Range RPV level: 0 inches
- Wide Range RPV level: -149 inches and stable
- Compensated Fuel Zone RPV level: -129 inches and stable
- Upset Range RPV level: 0 inches
- Drywell Temperature: 175° F up slow
- Reactor Pressure 480 psig down slow

Which of the following is correct concerning these indications?

The RO should report RPV water level as…

A. 0 inches.

B. -129 inches.

C. -149 inches.

D. cannot be determined.

Answer: B

K/A Match:

The question presents a situation where a low RPV level exists. Candidates must know the requirements for when different indications are considered correct which is a direct understanding of the capabilities of the various RPV level instruments.

SRO Only:

N/A

Explanation:

Per PPM 5.2.1 caution 1, Wide Range RPV level is not usable below -147”. Upset level is not usable below +15” when Drywell temperature is above 161F. Therefore the correct answer is to use Fuel Zone RPV level.

A. Incorrect. Upset Range and Narrow Range indications should not be used based on PPM 5.2.1. Plausible because they are actual RPV level indications and could be used if the situation was different.

B. Correct.

C. Incorrect. Wide Range indication should not be used because it is below -147” per PPM 5.2.1. Plausible because WR indication would be preferred if level was higher.

D. Incorrect. Level can be determined. Plausible because several level indications are not available based on current conditions.

Technical Reference(s)		Attached w/ Revision # See Comments / Reference
PPM 5.2.1, Primary Containment Control		

Proposed references to be provided during examination: None

Learning Objective: 11774 – Describe the operational implications of the following concepts as they apply to the Nuclear Boiler Instrumentation System: (a) Vessel level measurement

Question Source:
Bank # LO01558
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam N/A

Question Cognitive Level:
Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 41.7
55.43

Comments / Reference: PPM 5.2.1	Revision: 23

To prevent taking action based on erroneous RPV level indication, an RPV level instrument may not be used to determine RPV level if any of the following conditions exist for that instrument:

a. Drywell temp is at or above RPV Saturation Temp. and erroneous / erratic indication is observed.

b. The instrument is identified as unusable per ABN-HELB, ABN-FIRE or ABN-INSTRUMENTATION

c. For any of the instruments in the following table, the instrument reads below the Minimum Usable Level:

Instrument	Range (in.)	Drywell Temp Range (°F)	Minimum Usable Level (in.)
Wide Range	-150 to +60	100 - 550	-147
Fuel Zone Range	-310 to -110	100 - 550	-310
Narrow Range	0 to +60	100 - 550	+5
Upset Range	0 to +180	100 - 160 161 - 200 201 - 250 251 - 300 301 - 350 351 - 400 above 400	+0 +15 +36 +59 +86 +117 +151
Shutdown Flooding Range	0 to +400	100 - 150 151 - 200 201 - 250 251 - 300 301 - 350 351 - 400 above 400	+20 +33 +49 +67 +87 +112 +205

Examination Outline Cross-reference:	Level	RO		SRO
Rev. 0 Date: 12/16/2016	Tier	1		
	Group	2		
	K/A	295012.AK3.01		
Level of Difficulty: 3	Importance Rating	3.5		

High Drywell Temperature: Knowledge of the reasons for the following responses as they apply to HIGH DRYWELL TEMPERATURE : Increased drywell cooling

Question # 23

CGS is operating in Mode 1

- A reactor scram has been initiated due to a steam leak in the drywell.
- Drywell pressure is 1.5 psig, up slow
- Drywell temperature is 137°F, up slow

The crew has entered PPM 5.2.1, Primary Containment Control, and is performing the following step:

DT-1

MAINTAIN drywell temp below **135°F** with available drywell cooling

Which of the following describes the reason for this direction?

A. This action assures that the normal method of temperature control is attempted in advance of more complex actions.

B. This action assumes normal cooling is not functional and to use whatever cooling is “available” under the given plant conditions.

C. Other means to control temperature such as containment spray are not available until a LOCA signal has been received.

D. This direction is given as an initial action since drywell cooling equipment will load shed if conditions degrade, resulting in a LOCA signal.

Answer: A

K/A Match:
Question asks the reason drywell cooling is MAXIMIZED in step DT-1 to maintain drywell temperature below normal values of 135F. If drywell fans were not running or cooling was not available it would be expected that operators start the fans to maximize cooling.

SRO Only:
N/A

Explanation:

Per PPM 5.0.10, the step assures normal methods are used prior to more complex methods that might have long term consequences.

A. Correct.

B. Incorrect. Plausible because step DT-1 is not clear and simply says, “available drywell cooling”.

C. Incorrect. Plausible because an F or A signal will isolate RCC from containment and other methods of cooling become necessary.

D. Incorrect. Plausible because an F or A signal will isolate RCC from containment and other methods of cooling become necessary.

Technical Reference(s)		Attached w/ Revision # See Comments / Reference
PPM 5.0.10, Flowchart Training Manual		
PPM 5.2.1, Primary Containment Control		

Proposed references to be provided during examination: None

Learning Objective: 8312 – Given a list, identify the statement that describes the purpose of using drywell cooling as the first method of attempting to control drywell temperature.

Question Source: Bank # LO00105
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam 1999, question #65

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 41.5
55.43 ______

Comments / Reference: PPM 5.0.10	Revision: 21 mr 1

Number: 5.0.10	Use Category: INFORMATION	Major Rev: 021 Minor Rev: 001 Page: 267 of 320
Title: Flowchart Training Manual		

8.8.5 Drywell Temperature

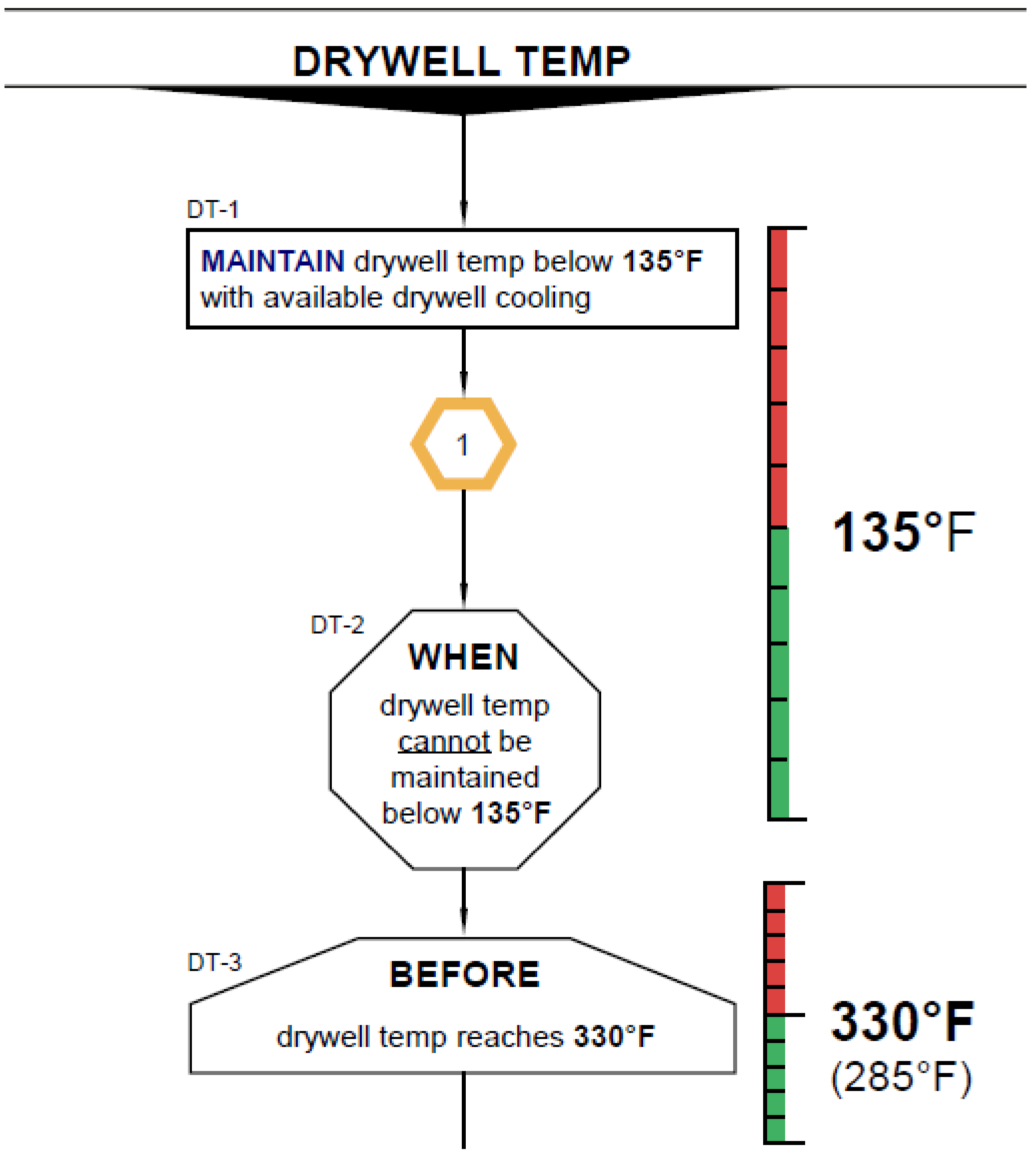
a. Step DT-1:

1) The initial action taken to control drywell temperature employs the same method typically used during normal plant operations: monitoring its status and placing available drywell cooling in operation as required to maintain drywell temperature below the LCO value. Step DT-1 thus provides a smooth transition from general plant procedures to emergency operating procedures, and assures that the normal method of drywell temperature control is attempted in advance of initiating more complex actions to terminate increasing drywell temperature.

2) As long as average drywell temperature remains below normal operating limits, no further action is required in this flowpath other than continuing to monitor and control average drywell temperature using available drywell cooling.

3) The applicability of Caution #1 highlights the direct effect that high drywell temperature has on RPV water level indication.

Comments / Reference: PPM 5.2.1, Primary Containment Control	Revision: 23



Examination Outline Cross-reference:	Level	RO		SRO
Rev. 0 Date: 12/16/2016	Tier	1		
	Group	2		
	K/A	295015.AA1.02		
Level of Difficulty: 3	Importance Rating	4.0		

Incomplete SCRAM: Ability to operate and/or monitor the following as they apply to INCOMPLETE SCRAM : RPS

Question # 24

CGS is operating in Mode 1.

While preparing to shift RPS to Alternate power following the trip of RPS-MG-1, the following indications are observed at H13-P610:

- The white indicating light located above the MG Set Transfer Switch for GENERATOR A FEED is extinguished.
- The white indicating lights for ALTERNATE FEED and for GENERATOR B FEED are both illuminated.
- The MG Set Transfer Switch is selected to NORM.

Which of the following correctly describes the expected response for the given MG Set Transfer Switch operations?

Placing the MG Set Transfer Switch in...

A. ALT B would not affect any of the white indicating lights but will cause a full reactor scram.

B. ALT A would not affect any of the white indicating lights and RPS Bus A would remain de-energized.

C. ALT A would cause the white GENERATOR A FEED light to illuminate and re-energize RPS Bus A from the alternate source.

D. ALT B would cause the white GENERATOR A FEED light to illuminate and re-energize RPS Bus A from RPS-MG-2.

Answer: A

K/A Match:
A loss of a RPS MG set causes a half scram. The question challenges the candidate's ability to restore power via the alternate source and correctly interpret indications.

SRO Only:
N/A

Explanation:

Indicating lights do not match or reflect switch position. The indicating lights are the status of power available to the RPS 3 power sources. The operator should place the switch in the position where power is NOT available to line up the alternate source to that bus. The lights will not change based on this switch manipulation.

A. Correct.

B. Incorrect. Plausible because switching to ALT A is the desired action. Incorrect because RPS Bus A would be energized.

C. Incorrect. Plausible because switching to ALT A is the desired action. Incorrect because the white GENERATOR A FEED light will NOT come on.

D. Incorrect. Plausible because the white GENERATOR B FEED light is on. Incorrect because the white GENERATOR A FEED light will NOT come on.

Technical Reference(s)		Attached w/ Revision # See Comments / Reference
SD000161, CGS System Description, Volume 6, Chapter 8, Reactor Protection System		

Proposed references to be provided during examination: None

Learning Objective: 5961 – Describe the electrical alignment of RPS when the MG SET TRANSFER switch (on control room panel H13-P610) is in: NORM/ALT A/ALT B

Question Source: Bank # LR00171
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 41.7

55.43 ______________

Comments / Reference: SD000161	Revision: 17 mr1

4. MG SET TRANSFER SWITCH LO-7678 LO-7680 LO-11681i

a) The MG SET TRANSFER SWITCH (see Figure 4A) located on control room back panel H13-P610, connects one RPS bus to either the normal or alternate power source.

b) The MG SET TRANSFER SWITCH is a break-before-make switch that prevents paralleling the two different RPS power supplies and allows only one RPS bus to be energized from the alternate supply.

c) Operating the MG SET TRANSFER SWITCH causes a momentary loss of power to the selected RPS bus resulting in the trips and actuations normally associated with a loss of power to the RPS bus.

d) Three "power available" lights above the switch indicate the availability of the three RPS power supplies.

e) These lights are on when the power source is energized and the associated Electric Power Monitoring Assembly (EPA) breakers are closed.

Examination Outline Cross-reference:	Level	RO		SRO
Rev. 0 Date: 12/16/2016	Tier	1		
	Group	2		
	K/A	295029.EK1.01		
Level of Difficulty: 3	Importance Rating	3.4		

High Suppression Pool Water Level: Knowledge of the operational implications of the following concepts as they apply to HIGH SUPPRESSION POOL WATER LEVEL : Containment integrity

Question # 25

Following a major plant transient, primary containment water level has risen to 555'.

Which of the following is correct?

If adequate core cooling is assured, injection from sources external to the primary containment should be stopped in order to prevent...

A. loss of the ability to determine off-site radiation release rates release.

B. loss of primary containment integrity and possible substantial radioactivity release.

C. failure of low pressure Emergency Core Cooling System (ECCS) suction piping due to the static head of water.

D. overloading Emergency Core Cooling System (ECCS) pump motors due to the additional flow that would result from the higher suction head.

Answer: B

K/A Match:
Question asks if candidates understand the consequences and operational implications if primary containment water level rises to 555'. The correct answer is related to containment integrity.

SRO Only:
N/A

Explanation:

Per PPM 5.0.10, the reason for PCPL limits is primary containment integrity.

A. Incorrect. Plausible because the bottom of the drywell vent is at 552.2ft elevation. Incorrect because loss of ability to determine off site release rates is not the primary concern.

B. Correct.

C. Incorrect. Plausible because ECCS pumps draw suction from the suppression pool. Incorrect because failure of piping is not the primary concern.

D. Incorrect. Plausible because ECCS pumps draw suction from the suppression pool. Incorrect because ECCS pump failure is not the primary concern.

Technical Reference(s)		Attached w/ Revision # See Comments / Reference
PPM 5.0.10, Flowchart Training Manual		

Proposed references to be provided during examination: None

Learning Objective: 13567 – Given a copy of Emergency Operating Procedures (EOPs) and an event, describe the basis for each variable and figure used to execute the strategies of the EOPs without error.

Question Source: Bank # LR00600
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam N/A

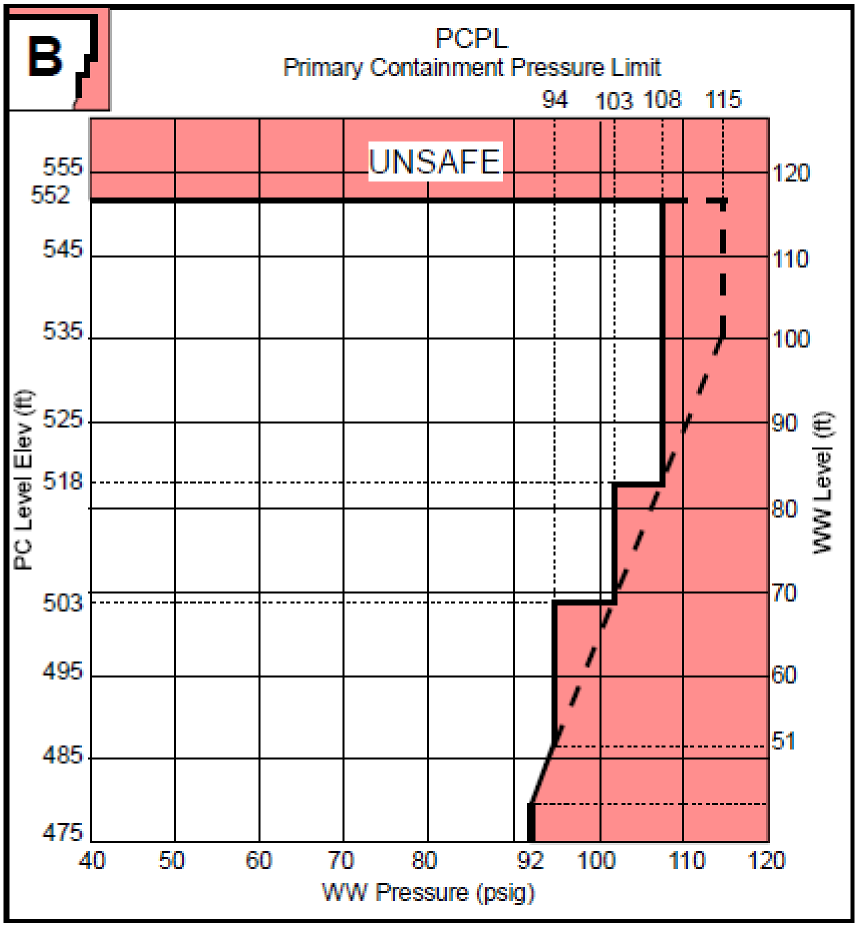
Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 41.8
55.43

Comments / Reference: PPM 5.0.10	Revision: 21 mr 1

Number: 5.0.10	Use Category: INFORMATION	Major Rev: 021 Minor Rev: 001 Page: 85 of 320
Title: Flowchart Training Manual		

7.13 Primary Containment Pressure Limit



7.13.1 The Primary Containment Pressure Limit (PCPL) is the pressure capability of the limiting component in the primary containment. If primary containment water level is less than 535 ft, the limit is based on SRV operability. Above 535 ft, the limit is base on the pressure capability of the girder joint of the primary containment. The PCPL is used to preclude failure of the containment and resultant loss of systems required to maintain adequate core cooling.

Examination Outline Cross-reference:	Level	RO		SRO
Rev. 0 Date: 12/16/2016	Tier	1		
	Group	2		
	K/A	295034.2.2.40		
Level of Difficulty: 3	Importance Rating	3.4		

Secondary Containment Ventilation High Radiation: Ability to apply Technical Specifications for a system.

Question # 26

The following describes different operating conditions of the reactor plant:

(1) Mode 1
(2) Mode 2
(3) Mode 3
(4) Mode 4
(5) Mode 5
(6) Operations with a potential for draining the reactor vessel (OPDRVs)

Which of the following describes **all** the conditions where the “Reactor Building Vent Exhaust Plenum Radiation – High” function of Secondary Containment Isolation Instrumentation is required by technical specifications?

A. (1), (2), (3)

B. (4), (5), (6)

C. (1), (2), (3), (6)

D. (3), (4), (5), (6)

Answer: C

K/A Match:
Question discriminates candidates ability to correct apply technical specification applicability requirements related to secondary containment isolation ability.

SRO Only:
N/A

Explanation:

Per given reference, student must analyze table 3.3.6.2.1 to determine that C is the correct answer.

A. Incorrect. Plausible because it logically follows other distractor format.

B. Incorrect. Plausible because it would be correct if note (a) did not apply.

C. Correct.

D. Incorrect. Plausible because manual initiation is not always required in technical specifications for systems and if candidate does not correctly read table 3.3.6.2.1(a).

Technical Reference(s)		Attached w/ Revision # See Comments / Reference
T.S. 3.3.6.2, Secondary Containment Isolation Instrumentation LCO		

Proposed references to be provided during examination: None

Learning Objective: 5606 – Given a copy of Technical Specifications, locate and apply all of the T.S. that are related to the NS4 system.

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ____

10 CFR Part 55 Content: 55.41 41.10
55.43 ____

Comments / Reference: Tech Spec 3.3.6.2	Revision: 237

Table 3.3.6.2-1 (page 1 of 1)
Secondary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Reactor Vessel Water Level - Low Low, Level 2	1, 2, 3, (a)	2(c)	SR 3.3.6.2.2 SR 3.3.6.2.3 SR 3.3.6.2.4	≥ -58 inches
2. Drywell Pressure - High	1, 2, 3	2(c)	SR 3.3.6.2.2 SR 3.3.6.2.3 SR 3.3.6.2.4	≤ 1.88 psig
3. Reactor Building Vent Exhaust Plenum Radiation - High	1, 2, 3, (a)	2	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.3 SR 3.3.6.2.4	≤ 16.0 mR/hr
4. Manual Initiation	1, 2, 3, (a)	4	SR 3.3.6.2.4	NA

(a) During operations with a potential for draining the reactor vessel.

(b) Deleted

(c) Also required to initiate the associated LOCA Time Delay Relay Function pursuant to LCO 3.3.5.1.

Examination Outline Cross-reference:	Level	RO		SRO
Rev. 0 Date: 12/16/2016	Tier	1		
	Group	2		
	K/A	295036.EK1.01		
Level of Difficulty: 3	Importance Rating	2.9		

Secondary Containment High Sump/Area Water Level: Knowledge of the operational implications of the following concepts as they apply to SECONDARY CONTAINMENT HIGH SUMP/AREA WATER LEVEL : Radiation releases

Question # 27

CGS is operating in Mode 1.

- Reactor Building EDR Sump Pump, EDR-P-5A is running to pump down reactor building sumps.

A steam leak occurs in the RCIC room.

- Reactor Building Ventilation exhaust radiation level is 15 mr/hr, up slow.

What is the condition of the Equipment Drain (EDR) system?

EDR-SUMP-R5 Pump Discharge Isolation valves, EDR-V-394/395, are…

A. open and EDR-P-5A is running.

B. closed and EDR-P-5A is secured.

C. closed and EDR-P-5A is running.

D. open and EDR-P-5A is secured.

Answer: B

K/A Match:
This question examines the candidate's understanding of the response of the Plant Drain system to mitigate a uncontrolled radiation release during a high radiation level/high sump level condition.

SRO Only:
N/A

Explanation:

A. Incorrect. Plausible since the sump pump would be running with a high sump level if the sump pump discharge valves were open. However, the sump pump discharge valves close when reactor building ventilation exhaust radiation levels are ≥ 13 mr/hr ('Z' signal), and the sump pump will not start regardless of sump level.

B. Correct. EDR-SUMP-R5 Pump Discharge Isolation valves, EDR-V-394/395, will close on a 'Z' signal (reactor building ventilation exhaust radiation level ≥ 13 mr/hr). With the discharge isolation valves closed, the sump pump will not run in discharge mode or recirculation mode.

C. Incorrect. Plausible since EDR-SUMP-R5 Pump Discharge Isolation valves, EDR-V-394/395, will close on a 'Z' signal (reactor building ventilation exhaust radiation level ≥ 13 mr/hr), and the sump pump would be running with a high sump level if the sump pump discharge valves were open. However, the sump pump discharge valves close when reactor building ventilation exhaust radiation levels are ≥ 13 mr/hr ('Z' signal), and the sump pump will not start regardless of sump level.

D. Incorrect. Plausible since the equipment drain system primary containment isolation valves automatically close only on high drywell pressure ('A' signal) and RPV low Level 2 ('F' signal). Both of these signals are not present. However, EDR-SUMP-R5 Pump Discharge Isolation valves, EDR-V-394/395, will close on a 'Z' signal (reactor building ventilation exhaust radiation level ≥ 13 mr/hr). With the discharge isolation valves closed, the sump pump will not run in discharge mode or recirculation mode.

Technical Reference(s)	
SD000130, CGS System Description, Volume 9, Chapter 7, Plant Drains	Attached w/ Revision # See Comments / Reference
PPM 4.602.A13, 602.A13 Annunciator Panel Alarms	

Proposed references to be provided during examination: None

Learning Objective: 5333 – List the isolation signals and setpoints for the following valves: (e) EDR-V-394 (f) EDR-V-395

Question Source:
Bank #
Modified Bank # (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level:
Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 41.8
55.43

Comments / Reference: SD000130, section V.B.3.1, EDR sump pump starting logic	Revision: Major 12

COLUMBIA SYSTEMS
PLANT DRAINS

November 2014
SD000130, r12 mr0

3. Reactor Building EDR Sump Pump EDR-P-5A/B
4-position switch, PTL/STOP/AUTO/START (Spring return to AUTO)

NOTE: Local transfer switch (EDR-RMS-P5A/B) selects which pump is connected to the MCC. The other pump is available as an installed spare.

PTL - Pump out of service.

STOP - Pump stop

AUTO - 1. Pump starts if the following conditions exist: LO-5331

- Sump level > Low level setpoint **AND**
- EDR-V-394 and EDR-V-395 Open **AND**
- Either EDR-V-21 Open **OR** Sump temperature GT Low temperature setpoint **AND**
- Either High Sump Temperature (GT 140°F) **OR** sump level GT high level setpoint

Comments / Reference: SD000130, section V.C.4, EDR sump discharge isolation valve closing logic	Revision: Major 12

COLUMBIA SYSTEMS
PLANT DRAINS

November 2014
SD000130, r12 mr0

2. COND/CRD Floor Drain Sump – FDR-Sump-R3 Inlet FDR-V-608 3-position switch, CLOSE/AUTO/OPEN (maintained) LO-9328 LO-5332c

   CLOSE - Valve closes.

   AUTO - Valve automatically closes if there is a high level in Sump R3 (HPCS Pump Room).

   OPEN - Valve opens.

3. LPCS Floor Drains Sump – FDR-Sump-R4 Inlet FDR-V-609 3-position switch, CLOSE/AUTO/OPEN (maintained) LO-5332a

   CLOSE - Valve closes.

   AUTO - Valve automatically closes if there is a high level in Sump R4 (C RHR Pump Room).

   OPEN - Valve Opens

4. EDR-SUMP-R5 Pump Discharge Isolation EDR-V-394_ 3-position switch, CLOSE/AUTO/OPEN (Spring return to AUTO) LO-5333

   CLOSE - Valve closes.

   AUTO - Valve automatically closes for any of the following isolation signals

   "F" - High Drywell Pressure (1.68 psig)

   "A" - RPV Level 2 (-50″)

   "Z" - RB Exhaust Plenum High Radiation (13 mrem/hr)

   OPEN - Valve opens if no FAZ signal exists

5. EDR-SUMP-R5 Sump Pump Discharge Isolation EDR-V-395

   The controls for this valve are identical to EDR-V-394

Comments / Reference: Ref. B, Annunciator 3-1, REACTOR BUILDING EQUIPMENT SUMP HIGH LEVEL, response.	Revision: Major 024

Number: 4.602.A13	Use Category: CONTINUOUS	Major Rev: 024 Minor Rev: N/A Page: 33 of 64
Title: 602.A13 Annunciator Panel Alarms		

3-1 REACTOR BUILDING EQUIPMENT SUMP HIGH LEVEL

3-1 WINDOW	SOURCE	AUTOMATIC ACTIONS
REACTOR BLDG EQUIP SUMP R5 LEVEL HI-HI	EDR-LS-14B (420'9")	None

CAUTION

There may be very little time from the receipt of the alarm, until the sump overflows. Actions to restore level should be performed expeditiously. {P-206442}

1. **VERIFY** one of the following operating:
   - EDR-P-5A
   - EDR-P-5B
2. **VERIFY** the following **OPEN**:
   - EDR-V-394 (EDR discharge to Radwaste) (H13-P632)
   - EDR-V-395 (EDR discharge to Radwaste) (H13-P632)

Comments / Reference: Ref. B, Annunciator 4-1, REACTOR BUILDING EQUIPMENT SUMP TEMPERATURE HIGH, response.	Revision: Major 024

Number: 4.602.A13	Use Category: CONTINUOUS	Major Rev: 024 Minor Rev: N/A Page: 37 of 64
Title: 602.A13 Annunciator Panel Alarms		

4-1 REACTOR BUILDING EQUIPMENT SUMP TEMPERATURE HIGH

4-1 WINDOW	SOURCE	AUTOMATIC ACTIONS
REACTOR BLDG EQUIP SUMP TEMP HIGH	EDR-TIS-601 (GE140°F)	1. Starts EDR-P-5A. 2. Closes Pump Discharge Valve (EDR-V-21) and opens Heat Exchanger Recirc Valve (EDR-V-22).

NOTE: If sump R-5 hi level exists, EDR-V-21 will remain open, and EDR-V-22 will remain closed, regardless of sump temperature.

1. **MONITOR** EDR-TIS-601 (sump R5 temperature) (H13-P602).
2. **VERIFY** EDR-P-5A operating (R5 Sump pump).

Examination Outline Cross-reference:	Level	RO		SRO
Rev. 0 Date: 12/16/2016	Tier	2		
	Group	1		
	K/A	203000.K5.02		
Level of Difficulty: 2	Importance Rating	3.5		

RHR/LPCI: Injection Mode: Knowledge of the operational implications of the following concepts as they apply to RHR/LPCI: INJECTION MODE: Core cooling methods

Question # 28

Select the valve that has an automatic open signal for the first ten minutes after an emergency core cooling system (ECCS) initiation signal.

A. RHR-V-42A (injection valve).

B. RHR-V-48A (heat exchanger bypass).

C. RHR-V-53A (shutdown cooling return isolation).

D. RHR-V-24A (suppression pool cooling return).

Answer: B

K/A Match:

Question asks for candidate knowledge of ECCS interlocks related to core cooling (HTXR bypass valve).

SRO Only:

Type an explanation as to how this question clearly meets the NRC guidance for SRO only questions. If this cannot be done without a lengthy explanation the question may be RO level.

Explanation:

RHR-V-48A is fully open for the first 10 minutes following an ECCS initiation signal.

A. Incorrect. RHR-V-42A does not have a 10 minute timer. Plausible because it is an RHR valve and opens on ECCS initiation signal.

B. Correct.

C. Incorrect. RHR-V-53A does not receive an open signal. Plausible because it is an RHR valve related to cooling.

D. Incorrect. RHR-V-24A does not receive an open signal. Plausible because it is an RHR valve related to cooling.

Technical Reference(s)		Attached w/ Revision # See Comments / Reference
SD000198, CGS System Description, Volume 7, Chapter 4, Residual Heat Removal		

Proposed references to be provided during examination: None

Learning Objective: 11801 – Describe the function, purpose and design features of the following RHR components: (p) RHR-V-48A

Question Source: Bank # LO00834
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 41.5
55.43

Comments / Reference: SD000198	Revision: 16 mr 0

COLUMBIA SYSTEMS
RHR

May 2016
SD000198, r16 mr0

30. RHR-V-48A HX Shell Side Bypass LO-11801p

    3-position switch CLOSE/AUTO/OPEN (spring return to AUTO, throttleable)

    CLOSE The valve closes if the switch is held in this position. Will remain closed when switch is released, if 10 minutes have elapsed since an initiation signal occurred.

    AUTO The valve opens if closed at the time of LPCI initiation and will auto open if attempted to be closed within ten minutes after the initiation.

    OPEN The valve opens if the switch is held in this position.

31. RHR-V-48B HX Shell Side Bypass

    3-position switch CLOSE/AUTO/OPEN (spring return to AUTO, throttle) LO-11801p

    CLOSE The valve closes if the switch is held in this position. Will remain closed when switch is released, if 10 minutes have elapsed since an initiation signal occurred.

    AUTO The valve opens if closed at the time of LPCI initiation and will auto open if attempted to be closed within ten minutes after the initiation.

    OPEN The valve opens if the switch is held in this position.

32. RHR-V-49, RHR Discharge to Radwaste Inboard LO-11801n

Examination Outline Cross-reference:	Level	RO	X	SRO
Rev. 1 Date: 12/12/2016	Tier	2		
	Group	1		
	K/A	203000.A1.03		
Level of Difficulty: 3	Importance Rating	3.8		

RHR/LPCI: Injection Mode: Ability to predict and/or monitor changes in parameters associated with operating the RHR/LPCI: INJECTION MODE controls including: System flow.

Question # 29

CGS is operating in Mode1.

A LOCA has occurred, requiring the crew to scram the reactor and enter the EOPs.

Current plant conditions:

- RPV level is – 135 inches, up slow.
- RPV pressure is 350 psig, down slow.
- Drywell pressure increased to 1.60 psig. Current pressure is 1.55 psig, down slow.
- RHR-P-2B failed to automatically start. All other automatic actions were satisfactorily completed.

What is the status of RHR Loop B if the operator ARMS and DEPRESSES RHR-RMS-S60 (RHR B and C MANUAL INITIATION pushbutton)?

RHR-P-2B is _(1)__ , RHR-V-64B (RHR Loop B Minimum Flow Bypass valve) is _(2)__ and RHR Loop B is _(3)__ into the RPV.

A. (1) not running
(2) open
(3) not injecting

B. (1) not running
(2) closed
(3) not injecting

C. (1) running
(2) closed
(3) injecting

D. (1) running
(2) open
(3) not injecting

Answer: D

K/A Match:
This question requires the candidate to demonstrate knowledge of how the LPCI system responds when manually started after a LOCA.

SRO Only:
N/A

Explanation:

A. Incorrect. Plausible since bus SM-8 is powered from TR-S after a reactor scram/main turbine trip, and RHR-P-2B will not start without a valid “F” or “A” signal. However, with level at -135 inches, a valid “A” signal is present and RHR-P-2B will start, and RHR-V-64B will not open unless the RHR-P-2B breaker is closed.

B. Incorrect. Plausible since bus SM-8 is powered from TR-S after a reactor scram/main turbine trip, and RHR-P-2B will not start without a valid “F” or “A” signal. However, with level at -135 inches, a valid “A” signal is present and RHR-P-2B will start.

C. Incorrect. Plausible since RHR-P-2B will start. However, since RPV pressure is greater than the pump shutoff head of 220 psig, there will be no injection into the RPV.

D. Correct. With a valid “F” initiation signal, RHR-P-2B will start. Since RPV pressure is greater than the pump shutoff head of 220 psig, there will be no injection into the RPV and the minimum flow bypass valve will be open.

Technical Reference(s)		Attached w/ Revision # See Comments / Reference
SOP-RHR-INJECTION, RHR RPV Injection		

Proposed references to be provided during examination: None

Learning Objective: 5779 – Describe the expected system response for any routine lineup, when the initiation logic for the LPCI mode of the RHR system is satisfied.

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 41.5
55.43 ____

Comments / Reference: SOP-RHRA-INJECTION | Revision: Major 003 Minor 002

Number: SOP-RHR-INJECTION	Use Category: CONTINUOUS	Major Rev: 003 Minor Rev: 002 Page: 9 of 22
Title: RHR RPV Injection		

5.2 RHR Loop B & C Automatic/Manual Initiation Logic Start

NOTE: This section is EOP related.

NOTE: If H13-P601.A4-3.1(H13-P601.A2-6.5), RHR B(C) PUMP DISCH PRESS HIGH/LOW is in alarm, RHR-P-2B(C) may be started for EOP related activities.

NOTE: If SM-8 is energized from TR-S, the RHR B and C MANUAL INITIATION pushbutton function will not start RHR-P-2B(C), unless there is an associated valid F or A signal. If needed, manually initiate RHR System B(C).

5.2.1 IF performing a Manual Initiation of RHR-SYS-B and RHR-SYS-C, THEN **ARM** and **DEPRESS** RHR-RMS-S60 (RHR B and C MANUAL INITIATION pushbutton). _____

5.2.2 **VERIFY** DG-2 running. _____

5.2.3 **VERIFY** SW-P-1B running. _____

CAUTION

To minimize cavitation and increased pump hydraulic loads/vibrations, minimize operating with RHR-FCV-64B(C) (Minimum Flow) as its only discharge path. {C-9448}

5.2.4 **VERIFY** RHR-P-2B running. _____

5.2.5 **VERIFY** RHR-FCV-64B opens during low flow conditions (approximately 800 gpm) (Minimum Flow Bypass). {P-103296} {C-9448} _____

5.2.6 **VERIFY** RHR-P-2C running. _____

5.2.7 **VERIFY** RHR-FCV-64C opens during low flow conditions (approximately 800 gpm) (Minimum Flow Bypass). {P-103296} {C-9448} _____

5.2.8 WHEN RPV pressure decreases to LT 470 psig, THEN **VERIFY** RHR-V-42B opens (LPCI Isolation). _____

5.2.9 WHEN RPV pressure decreases to LT 470 psig, THEN **VERIFY** RHR-V-42C opens (LPCI Isolation). _____

5.2.10 WHEN RPV pressure decreases to approximately 220 psig, THEN **VERIFY** RHR B injection into the RPV. _____

Examination Outline Cross-reference:	Level	RO		SRO
Rev. 0 Date: 12/16/2016	Tier	2		
	Group	1		
	K/A	205000.K6.01		
Level of Difficulty: 3	Importance Rating	3.3		

Shutdown Cooling System (RHR Shutdown Cooling Mode): Knowledge of the effect that a loss or malfunction of the following will have on the SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE) : A.C. electrical power

Question # 30

CGS is in Mode 4.

- RHR Loop A is in Shutdown Cooling mode.
- DG-2 is inoperable and unavailable.
- RRC pumps are secured.
- A Loss of Offsite Power (LOOP) occurs.
- Breaker CB-B8, TR-B to SM-8 supply breaker, failed to close.

With no operator action, what is the status of the following SDC components 1 minute after the loss of TR-S?

(1) RHR-V-8, RHR Shutdown Cooling Suction Outboard Isolation
(2) RHR-V-9, RHR Shutdown Cooling Suction Inboard Isolation
(3) RHR-P-2A, RHR Pump

A. (1) Closed
(2) Closed
(3) Secured

B. (1) Open
(2) Open
(3) Running

C. (1) Closed
(2) Open
(3) Secured

D. (1) Open
(2) Closed
(3) Running

Answer: C

K/A Match:
This question requires the candidate to demonstrate knowledge of the effect of a loss of an AC bus on the shutdown cooling lineup.

SRO Only:
N/A

Explanation:

A. Incorrect. Plausible since on a loss of offsite power where both SM-7 and SM-8 are restored, this would be the expected lineup. However, since SM-8 is de-energized, RHR-V-9 will not close even though it has a signal to do so.

B. Incorrect. Plausible since RHR-P-2A would be running following a loss of offsite power with a LOCA signal. However, with a LOOP only, RHR-P-2A would be secured.

C. Correct. Following a LOOP, RHR-P-2A will be secured. Since both RPS buses are lost, an isolation signal is generated for all NS^4 inboard and outboard valves. However, since SM-8 is de-energized, RHR-V-9 will not close.

D. Incorrect. Plausible since RHR-P-2A would be running following a loss of offsite power with a LOCA signal. Valve positions are plausible if it is believed that loss of RPS B only generates a NS^4 inboard isolation signal and RPS A was not lost. However, both RPS A and B are lost generating an isolation signal for both inboard and outboard valves. RHR-P-2A is secured following a LOOP only.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
SD000182, CGS System Description, Vol. 1 Chap. 2, AC Distribution	
SD000161, CGS System Description, Vol. 6 Chap. 8, Reactor Protection System	

Proposed references to be provided during examination: None

Learning Objective: 5781 - List the interlocks and trips associated with the following RHR system components: a. RHR pumps, d. RHR-V-8 & RHR-V-9

Question Source:
- Bank #
- Modified Bank # (Note changes or attach parent)
- New X

Question History: Last NRC Exam N/A

Question Cognitive Level:
- Memory or Fundamental Knowledge
- Comprehension or Analysis X

10 CFR Part 55 Content:
- 55.41 41.7
- 55.43

Comments / Reference: AC Distribution System Description	Revision: 19

9. **SM-7 Undervoltage**

Immediately	Start DG1 and close 7-DG1 (normally closed)
	Trip 7-1 tie bkr
	Trip MC-7C and MC-7E
	Trip LPCS, RHR-2A, SW-1A, CRD-1A pumps
	Trip REA-1A and ROA-1A fans
5.5 sec later	Close B-7
7-10 sec later	Close DG1-7 if UV still exists

10. **SM-8 Undervoltage**

Immediately	Start DG2 and close 8-DG2 (normally closed)
3 sec later	Trip 8-3 tie bkr
	Trip MC-8C and MC-8E
	Trip RHR 2B & 2C, SW-1B and CRD-1B pumps
	Trip REA-1B and ROA-1B fans
5.5 sec later	Close B-8
7-10 sec later	Close DG2-8 if UV still exists

Loss of AC power; MC-7A or MC-8A will result in loss of RPS bus A or RPS bus B resulting in a half-scram (no rod motion) and a partial NS4 isolation. Loss of both MC-7A and MC-8A will result in a full scram signal wih insertion of all control rods and a partial NS4 isolation. The alternate source of power from MC-6B is manually transferred in the control room at panel P-610. See the RPS and NS4 chapters for more detail on the effects of loss of RPS power.

F. AC distribution loads

7. Loss of Offsite Power (LOOP) (ABN-ELEC-LOOP)

LO-11827
NLO-12322a

A loss of 230 KV Startup transformer TR-S when Normal transformers TR-N1 and TR-N2 are in service will not result in the loss of any station electrical loads. A loss of TR-S while it is

supplying station loads will result in electrical separation between critical and non-critical buses SM-1, SM-2, SM-3 and SM-7, SM-4 and SM-8 respectively. DG3 will assume critical bus SM-4 loads and Backup Transformer, TR-B will assume critical bus SM-7 and 8 loads following the separation of buses.

Comments / Reference: RPS System Description	Revision: Major 17 Minor 001

d) On a scram signal, the inboard vent and drain valves close followed by the outboard vent and drain valves.

e) When the scram signal is reset, the outboard vent and drain valves open followed by the inboard vent valve.

(1) The inboard drain valve opens after a two minute delay following the reset of the scram signal.

(2) An RPS time-delay relay delays energization of solenoid pilot valve SPV-186 delaying the opening of the inboard drain valve, CRD-V-11.

H. RPS Power

LO-5950
LO-7684a
LO-7684b
LO-11681e
LO-11681f
LO-11681g
LO-12528
LO-12530a
LO-12530b
LO-12530c
LO-12530d
LO-12526a
LO-12526d

1. The RPS busses (see Figure 4) consist of two independent, redundant, normally energized, 120 VAC buses.

2. Normal Power

a) Power from critical busses SM-7 and SM-8 is reduced from 4160V to 480V and directed to motor control center MC-7A and MC-8A respectively.

b) Power from MC-7A and MC-8A drives the motor end of its respective RPS motor-generator (RPS-MG-A & RPS-MG-B).

c) The 120V single-phase output of RPS-MG-A & RPS-MG-B is connected to the RPS bus "A" and "B" respectively.

(2) Each MG set is equipped with an 880 pound flywheel to provide a constant voltage and frequency output during power interruptions to MC-7A or MC-8A for up to three seconds in duration.

2. Loss of RPS (ABN-RPS)

LO-7683b
LO-7684
LO-12530

RPS is designed to fail such that a total loss of the RPS results in a reactor scram. Simultaneous activation of all reactor protection trip annunciators is indicative of an RPS failure; these annunciators activate on annunciator block A7 (RPS channel A) or A8 (RPS channel B) as appropriate. Other indications are APRMs A, C, E or B, D, F readings downscale, rod groups 1, 2, 3, & 4 power available lights (white) at control room panels H13-P603, H13-P609, and H13-P611 extinquished and RPS Bus A, B, or Alternate Supply power available light at control room panel H13-P610 extinquished. A loss of a single RPS bus results in a half-scram, as well as other abnormal operating conditions subsequently described. The following is a list of the more important automatic actions that occur on a loss of RPS bus A or B:

LO-7683c
LO-7683f
LO-7683e

- RPS A
  - Half scram System A.
  - NS[4] Outboard isolation of Groups 2, 5, 6 and 7
  - RWCU-V-4 closes and RWCU pumps trip
  - AR-EX-1A trips due to MSL radiation monitor power loss
  - AR-P-1A trips due to MSL radiation monitor power loss
- RPS B
  - Half scram System B
  - NS[4] Inboard and Outboard Isolation of Groups 2, 5, 6, and 7
  - RWCU-V-1 & RWCU-V-4 close and RWCU pumps trip
  - AR-EX-1B trips due to MSL radiation monitor power loss
  - AR-P-1B trips due to MSL radiation monitor power loss
  - CRD-V-11 Scram Discharge Drain closes (results in a closed indication on control room panel H13-P603 for the SDV Drains).

Examination Outline Cross-reference:	Level	RO		SRO
Rev. 0 Date: 12/16/2016	Tier	2		
	Group	1		
	K/A	209001.A1.01		
Level of Difficulty: 3	Importance Rating	3.4		

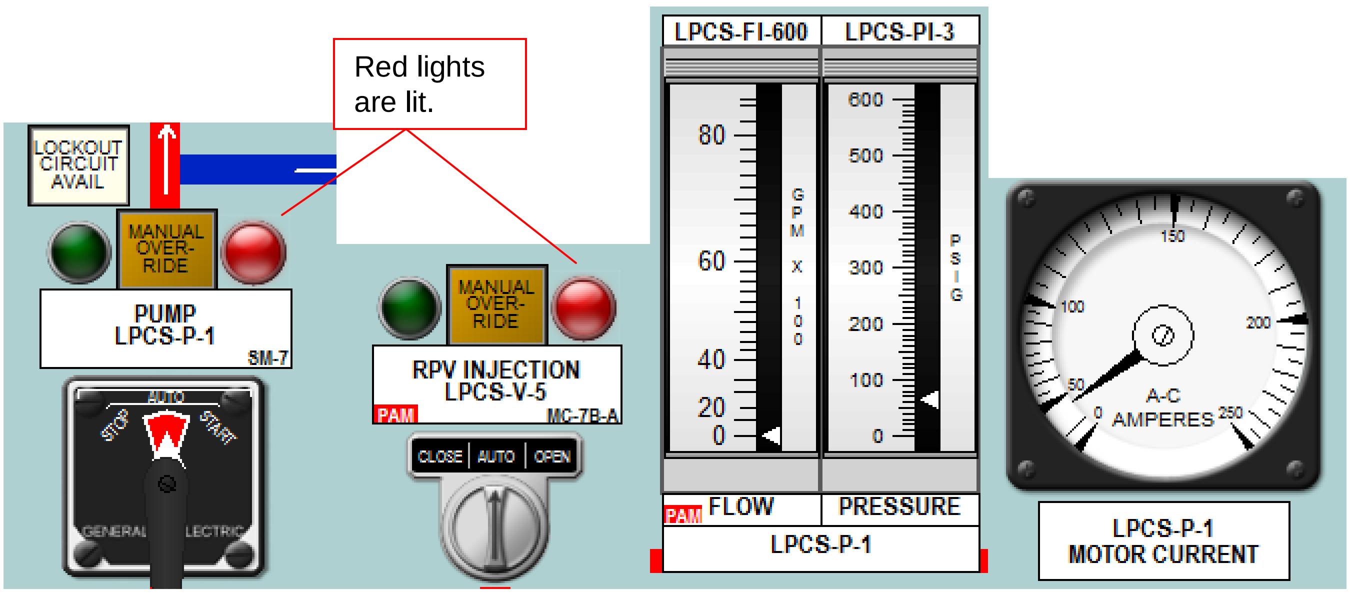
Low Pressure Core Spray System: Ability to predict and/or monitor changes in parameters associated with operating the LOW PRESSURE CORE SPRAY SYSTEM controls including: Core spray flow

Question # 31

Given the following:

- LPCS-P-1 automatically initiated on low RPV water level.
- LPCS flow is observed to be 7200gpm.

3 minutes later, RPV pressure is 240 psig and the following is observed:



Which of the following describes the reason for these indications?

A. RPV Pressure is above LPCS-P-1 shutoff head.

B. LPCS-P-1 shaft seizure.

C. LPCS-P-1 shaft shear

D. LPCS-P-1 supply breaker opened on overcurrent.

Answer: C

K/A Match:
Candidate is using LPCS indications to monitor a change in the system status and describe the cause of that change.

SRO Only:
N/A

Explanation:
The red indicating light for LPCS-P-1 means the breaker is still closed and the motor is running. Current is abnormally low and both flow and discharge pressure have lowered to zero. This indicates a sheared shaft.

A. Incorrect. RPV pressure of 350psig is below the shutoff head for LPCS-P-1. Plausible because LPCS flow would lower to zero if below shutoff head and the red indicating light for LPCS-P-1 would still be on as it is in the pictures.

B. Incorrect. Shaft seizure would result in a breaker trip and a green light on LPCS-P-1. Plausible because a shaft seizure would lower LPCS pump flow and discharge pressure.

C. Correct.

D. Incorrect. Supply breaker opening would result in a green light on LPCS-P-1. Plausible because the supply breaker opening would result in

Technical Reference(s)		Attached w/ Revision # See Comments / Reference
GFES BC05Sr4_Motors		

Proposed references to be provided during examination: None

Learning Objective: 11586 Describe the function, purpose and design features of the following Low Pressure Core Spray System components: b. LPCS pump LPCS-P-1

Question Source: Bank # ___
Modified Bank # ___ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge ___
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 41.5
55.43 ___

Comments / Reference: GFES BC05Sr4_Motors	Revision: May 2011

BC05Sr4_Motors May 2011

PUMP STARTS

Due to the high currents during pump motor starts, successive start attempts and the number of pump motor starts should be minimized to prevent excessive heat generation. Resistive losses in the copper coils and eddy currents within the rotor core produce a significant amount of heat, which breaks down insulation and reduces motor life expectancy. Insulation breakdown affects any type of rotating machine (both motors and generators). Insulation breakdown can result in short circuits and grounds, motor or generator trips, blown fuses and degraded resistance readings.

- Instantaneous increase in motor winding temperatures
- Possible breaker trip

The second problem is a sheared rotor. A sheared rotor permits the operation or spinning of the shaft without turning the pump impeller. Essentially, the motor and pump have been disconnected. Indicators of a sheared rotor are as follows:

- No load running current
- Decrease in pump discharge pressure
- Decrease in system flow rate

Another mechanical problem that can affect motors (and generators) is overheating of the

Examination Outline Cross-reference:	Level	RO		SRO
Rev. 0 Date: 12/16/2016	Tier	2		
	Group	1		
	K/A	209002.K4.07		
Level of Difficulty: 2	Importance Rating	3.5		

High Pressure Core Spray System (HPCS): Knowledge of HIGH PRESSURE CORE SPRAY SYSTEM (HPCS) design feature(s) and/or interlocks which provide for the following: Override of reactor water level interlock

Question # 32

Select the statements below that describe how the HPCS Injection valve (HPCS-V-4) will respond when the HPCS RPV INJ VALVE INTERLOCK OVERRIDE keylock switch (HPCS-RMS-S25) is placed in the "OVERRIDE" position.

HPCS-V-4 ___①___ be throttled and ___②___ open on a HPCS initiation signal.

A. ① cannot
② will not

B. ① can
② will not

C. ① cannot
② will

D. ① can
② will

Answer: B

K/A Match:
Questions asks how overriding the HPCS Injection Valve low RPV water level interlock will impact its behavior.

SRO Only:
N/A

Explanation:

Overriding the switch allows HPCS-V-4 to be throttled. The valve will no longer open on low RPV water level signals.

A. Incorrect. HPCS-V-4 can be throttled once the switch is in the OVERRIDE positon.

B. Correct.

C. Incorrect. HPCS-V-4 can be throttled once the switch is in the OVERRIDE positon.

D. Incorrect. HPCS-V-4 will no longer open on low RPV level.

Technical Reference(s)		Attached w/ Revision # See Comments / Reference
SD000174, CGS System Description, Volume 7, Chapter 2, High Pressure Core Spray		

Proposed references to be provided during examination: None

Learning Objective: 7661 – State the function of the HPCS RPV Injection Valve Interlock Override key switch

Question Source: Bank # LO02256
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 41.7
55.43 ______

Comments / Reference: SD000174	Revision: r13 mr0

closed.

OPEN — Associated valve moves in open direction. Valve position is indicated on the vertical portion of P601 on a 0-100% OPEN meter indication.

5. HPCS Injection Valve (HPCS-V-4) Interlock Override (HPCS-RMS-S25) LO-7661

2-position maintained keylock switch NORMAL/OVERRIDE at P625.

NORMAL — All injection valve normal functions occur.

OVERRIDE — The injection valve open and closed seal-in contacts are bypassed. The valve must be throttled open and closed. The injection valve will not open on initiation signal but will auto close on Level-8. This allows HPCS-V-4 to be throttled for better level control when directed by the EOPs.

6. HPCS RPV High Water Level Override (HPCS-RMS-S26) LO-7662 LO-11728

2-position maintained keylock switch NORMAL/OVERRIDE at P625

NORMAL — All injection valve (HPCS-V-4) normal functions occur.

Examination Outline Cross-reference:	Level	RO		SRO
Rev. 0 Date: 12/16/2016	Tier	2		
	Group	1		
	K/A	211000.A4.08		
Level of Difficulty: 2	Importance Rating	4.2		

Standby Liquid Control System: Ability to manually operate and/or monitor in the control room: System initiation

Question # 33

While placing the control switches for SLC-P-1A and SLC-P-1B to OPERATE, SLC-V-1A (Storage tank outlet valve) fails to open (all other components functioned as designed).

Which of the following correctly describes the SLC system response to this condition?

A. SLC-P-1A and SLC-P-1B will start.

B. SLC-P-1A and SLC-P-1B will not start.

C. SLC-P-1A will not start, SLC-P-1B will start.

D. SLC-P-1A will start and trip, SLC-P-1B will start.

Answer: A

K/A Match:

The question determines if a candidate is able to understand system initiation response given indications available in the control room.

SRO Only:

N/A

Explanation:

SLC pumps will start provided ONE of the tank outlet valves SLC-V-1A or SLC-V-1B is open after the switch for SLC-P-1A and SLC-P-1B are placed in the OPERATE POSTION. The suction to the pumps is connected downstream of the two suction valves.

A. Correct Answer.

B. Incorrect because SLC-P-1B WILL start. Plausible because it is common for pumps to have start interlocks based on their specific suction valve.

C. Incorrect because SLC-P-1A will NOT trip. Plausible because pumps often have low suction pressure trips that would trip the pump if the suction valve were closed.

D. Incorrect because both pumps WILL start. Plausible because some systems have multiple suction valves in series before the pumps that must be open before the system starts.

Technical Reference(s)		Attached w/ Revision # See Comments / Reference
SD000172, CGS System Description, Volume 5, Chapter 3, Standby Liquid Control		

Proposed references to be provided during examination: None

Learning Objective: 5925 – Describe the expected response to placing the SLC SYSTEM A or B keylock switch in the OPERATE position.

Question Source: Bank # LR00922

Modified Bank # ______ (Note changes or attach parent)

New ______

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 41.7

55.43 ______

Comments / Reference: SD000172	Revision: 13

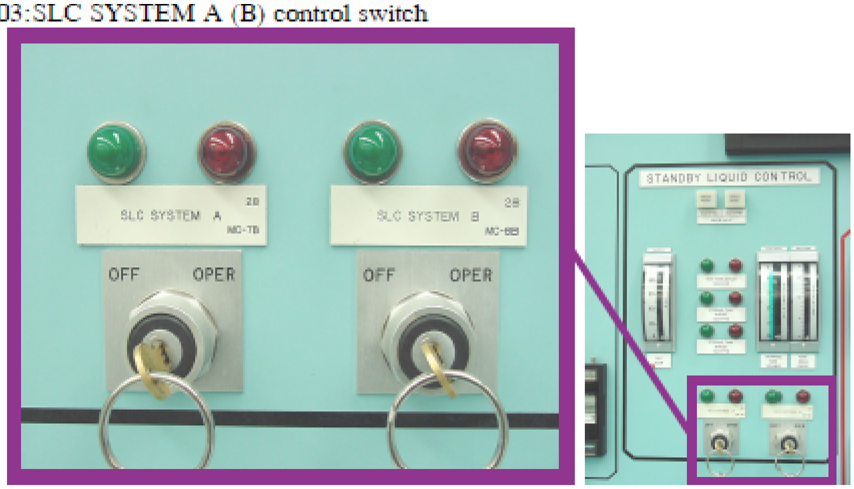
COLUMBIA SYSTEMS
STANDBY LIQUID CONTROL

October 2014
SD000172, r13 mr0

V. CONTROL THEORY AND INTERLOCKS

A. Control Room Controls

P603:SLC SYSTEM A (B) control switch



Two Position, OFF/OPER, Keylock Switch, Maintained Positions.

OFF - Pump stops. LO-5925

OPER - Either system control switch causes all of the following:

- Both SLC-V-4A & 4B, explosive valves, detonate.
- Both SLC-V-1A & 1B, SLC storage tank outlet valves open, **if** SLC-V-31, test tank outlet valve is fully closed (normally a locked closed valve).
- The associated SLC pump starts when suction path verified by either SLC-V-1A or 1B, SLC storage tank outlet valves is open fully, or SLC-V-31, test tank outlet to pump suction is fully open.
- RWCU-V-4, Outboard RWCU system Containment isolation valve closes

NOTE: Plant procedures require both pumps to be initiated when required for operation

SLC Pump SLC-P-1A & 1B trips following start from P603 are as follows:

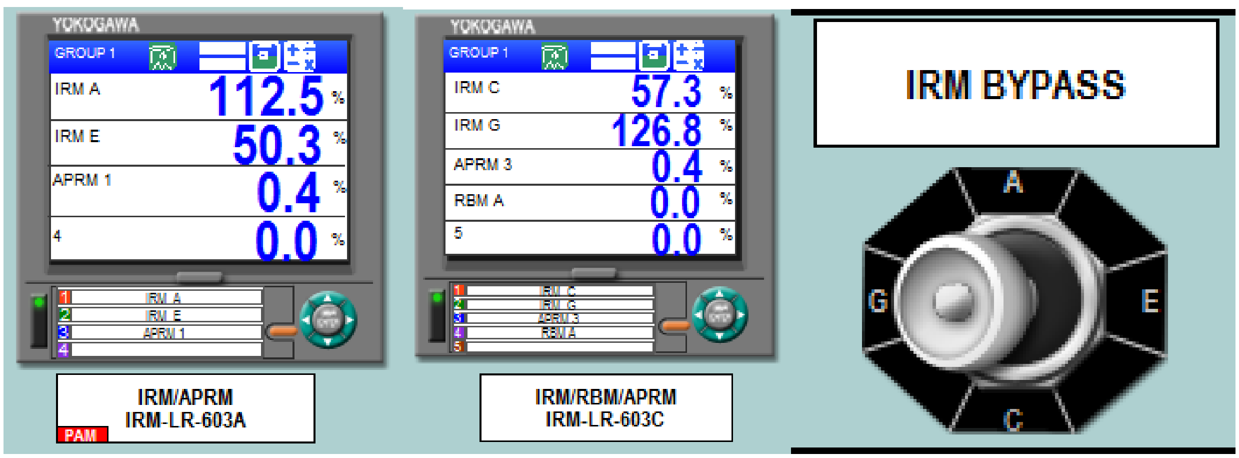
Examination Outline Cross-reference:	Level	RO		SRO
Rev. 0 Date: 12/16/2016	Tier	2		
	Group	1		
	K/A	212000.A3.01		
Level of Difficulty: 3	Importance Rating	4.4		

Reactor Protection System: Ability to monitor automatic operations of the REACTOR PROTECTION SYSTEM including: Reactor Power

Question # 34

CGS is operating in Mode 2.

- The following indications are observed:



What is the status of RPS A?

A. IRM ACEG UPSCL TRIP OR INOP Annunciator, Rod Block.

B. IRM MONITOR UPSCALE Annunciator, NO Rod Block

C. NO Rod Block, ½ SCRAM

D. Rod Block, NO ½ SCRAM

Answer: D

K/A Match:
Student must interpret/monitor system indications for IRM and determine whether automatic RPS actions have occurred. Indications are based on reactor power and IRM readings.

SRO Only:
N/A

Explanation:
Any IRM channel above 120% will cause a ½ SCRAM.
Any IRM channel above 108% will cause a ROD BLOCK.
Rod Blocks, ½ SCRAMs, and Annunciators are disabled if an IRM channel is bypassed for that channel only.

Because IRM Channel G is bypassed, there is no ½ SCRAM. IRM A, however, has a rod block signal.

A. Incorrect because ONLY a rod block occurs. IRM ACEG Upscale Trip or Inop comes in at 120%. Plausible because channel G is at 126.8%.

B. Incorrect because a rod block condition exists. Plausible because IRM Monitor Upscale annunciator comes in at 108%.

C. Incorrect because no ½ SCRAM occurs. Plausible because channel G is at 126.8%.

D. Correct Answer.

Distractors are plausible based on possible combinations of switch positions and proper interpretation of readings.

Technical Reference(s)		Attached w/ Revision # See Comments / Reference
SD000138, CGS System Description, Volume 6, Chapter 2, Intermediate Range Monitor		
PPM 4.603.A7, Annunciator Response Procedure		

Proposed references to be provided during examination: None

Learning Objective: 11794 – Describe the physical connection and/or cause-and-effect relationship between the Intermediate Range Monitoring System and the following: (a) RPS

Question Source: Bank # ___
Modified Bank # ___ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge ___
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 41.7
55.43

Comments / Reference: 4.603.A7	Revision: 51

Number: 4.603.A7	Use Category: CONTINUOUS	Major Rev: 051 Minor Rev: N/A Page: 11 of 71
Title: 603.A7 Annunciator Panel Alarms		

1-5 INTERMEDIATE RANGE MONITORING SYSTEM A, C, E OR G
UPSCALE TRIP OR INOPERABLE

1-5 WINDOW	SOURCE	AUTOMATIC ACTIONS
IRM A,C,E,G UPSCL TRIP OR INOP	Intermediate range channels A,C,E or G (K90) Setpoints Upscale - GE 120/125 Inoperable - • Detector high voltage low • Any module unplugged • IRM mode switch not in OPERATE • Loss of -24V power RPS-RLY-K12A, K12C, K12E, K12G	Reactor Half Scram if Mode Switch is not in RUN.

1. IF a Full Scram has occurred,
THEN **REFER** to PPM 3.3.1, Reactor Scram.

Comments / Reference: SD000138	Revision: 10 mr 1

B. Local Controls

NONE

C. TRIPS/INTERLOCKS

Setpoint	Result	Setpoint	
5/125	Rod Block	5/125	LO-5459 LO-11798d
108/125	Rod Block	108/125	
120/125	Scram	120/125	
Detector Not full in	Rod Block	Detector Not full in	
SOOO MU HVL NVL	Rod Block Scram (IRM INOP)	SOOO MU HVL NVL	LO-5449a,b,c,d LO-11798b

SOOO - Switch out of Operate; ***MU*** - Module Unplugged; ***HVL*** - High Voltage Power Supply Low; ***NVL*** - Negative Voltage Power Supply Lost

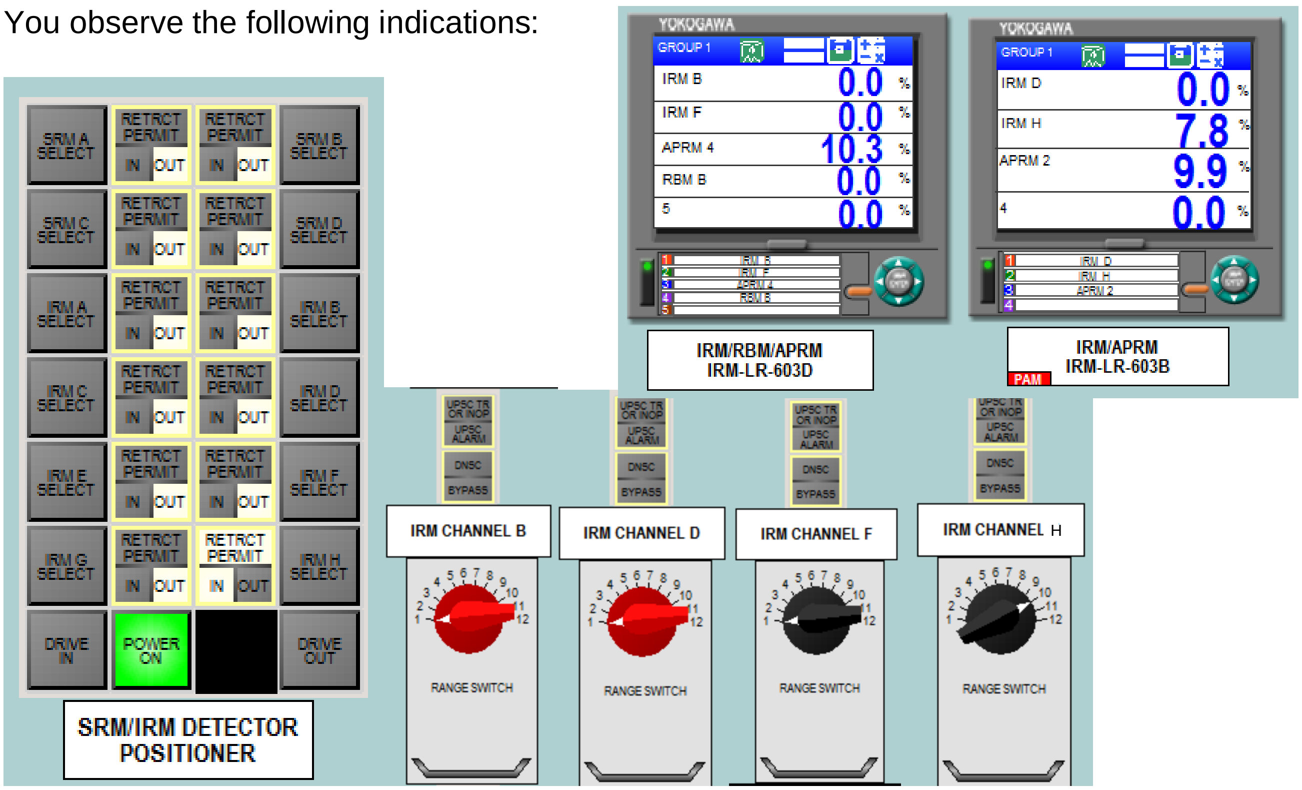
Examination Outline Cross-reference:	Level	RO		SRO
Rev. 0 Date: 12/16/2016	Tier	2		
	Group	1		
	K/A	215003.2.1.31		
Level of Difficulty: 3	Importance Rating	4.6		

Intermediate Range Monitor (IRM) System: Ability to locate control room switches, controls, and indications, and to determine that they correctly reflect the desired plant lineup.

Question # 35

A cold reactor startup is in progress per PPM 3.1.2, Reactor Startup.

You observe the following indications:



Which of the following is correct regarding system lineup?

A. IRM H needs to be withdrawn from the core.

B. IRM B, D, and F range switches need to be raised.

C. IRM/APRM overlap requirements are not met on IRM H.

D. IRM B, D, and F need to be inserted for IRM/APRM overlap verification.

Answer: A	

K/A Match:
The K/A requires the ability to determine if IRM system are in the correct lineup for plant conditions. This question requires candidates to understand the required IRM lineup during a reactor startup, to determine that the current IRM lineup is not in the required lineup, and to identify the problem.

SRO Only:
N/A

Explanation:

Following APRM/IRM overlap at 5% power, all IRMs are inserted per PPM 3.1.2 step Q42. APRM power indicates close to 10% and IRM H is still withdrawn.

A. Correct.

B. Incorrect. IRM range switches should be on Range 1 per step Q42 of PPM 3.1.2. Plausible since during a reactor startup, the IRM range switches must be raised as power increases to keep power on scale.

C. Incorrect. IRM/APRM overlap requirements would be met and have already been verified. Plausible because inserting IRM detectors occurs following IRM/APRM overlap verification.

D. Incorrect. Lowering range switch 10 could cause

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
PPM 3.1.2, Reactor Startup, Attachment 7.3, Startup Flowchart	

Proposed references to be provided during examination: None

Learning Objective: 11796 – Describe the operational implications of the following concepts as they apply to the IRM system: (a) Detector operation.

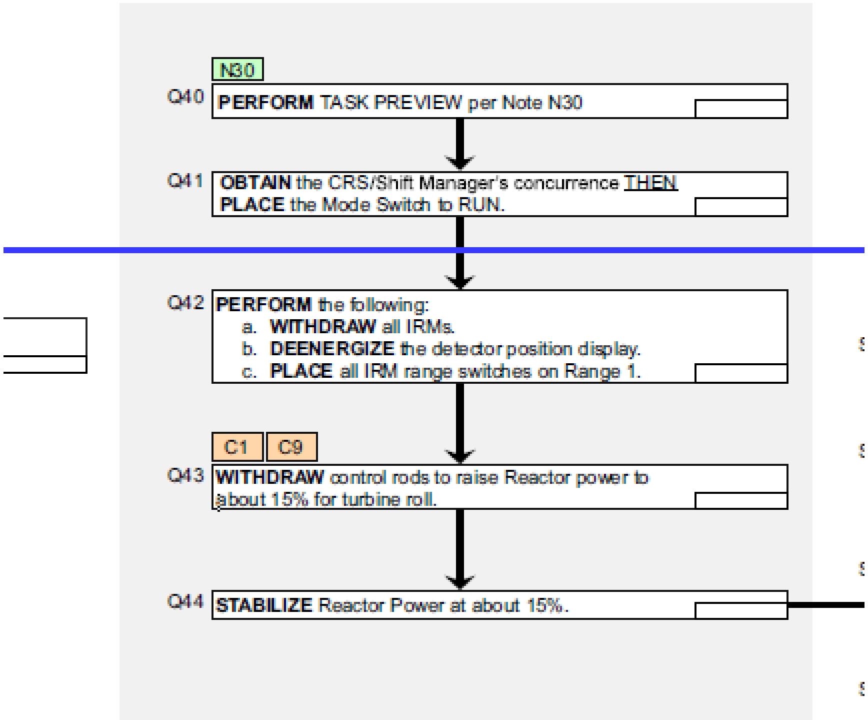
Question Source: Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

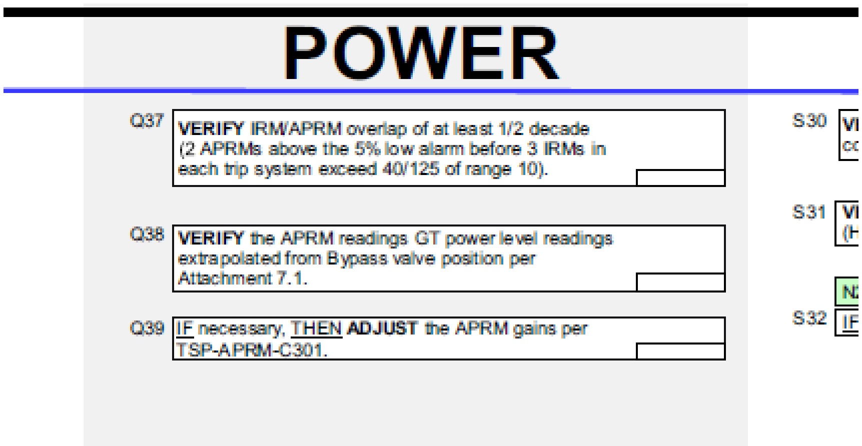
Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 41.10
55.43 ______

Comments / Reference:	Revision:
(Ref. B) PPM 3.1.2, Reactor Startup, Attachment 7.3, Startup Flowchart, steps Q40 to Q44	Major Rev. 081 Minor Rev. 001



Comments / Reference:	Revision:
(Ref. B) PPM 3.1.2, Reactor Startup, Attachment 7.3, Startup Flowchart, steps Q37 to Q39	Major Rev. 081 Minor Rev. 001



Examination Outline Cross-reference:	Level	RO		SRO
Rev. 0 Date: 12/16/2016	Tier	2		
	Group	1		
	K/A	215003.A2.06		
Level of Difficulty: 3	Importance Rating	3.0		

Intermediate Range Monitor (IRM) System: Ability to (a) predict the impacts of the following on the INTERMEDIATE RANGE MONITOR (IRM) SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Faulty range switch

Question # 36

CGS is operating in Mode 2.

- A plant startup is in progress with power increasing.
- Power indicates 37/125 with IRM G on range 5.
- All other IRM channels are on range 5 and indicate between 20/125 and 30/125.
- The range switch for IRM G fails from range 5 to range 4.
- IRM G range can no longer be manually changed.

Which of the following is **next** action that needs to be taken?

A. Shutdown the reactor per PPM 3.2.1, Normal Reactor Shutdown.

B. Reset the ½ SCRAM using SOP-RPS-OPS.

C. Enter PPM 3.3.1, Reactor SCRAM.

D. Bypass IRM G per ARP 4.603.A7.

Answer: D

K/A Match:
The candidate must determine the impact to the system based on the failure such as whether or not a SCRAM will occur. They must also know the correct action to take per procedures.

SRO Only:
N/A

Explanation:
The failure will cause a rod block and half scram on RPS A. Startup can continue if the IRM channel is bypassed.

A. Incorrect. Shutting down is not necessary. Plausible because ½ SCRAM occurs and the IRM system is not functioning as designed.

B. Incorrect. The stem asked for the NEXT action to be taken. Resetting the ½ SCRAM will not work until the IRM channel is bypassed. Plausible because a ½ SCRAM occurs.

C. Incorrect. An automatic SCRAM does not occur and a manual SCRAM is not required. Plausible because a ½ SCRAM occurs.

D. Correct.

Technical Reference(s)		Attached w/ Revision # See Comments / Reference
PPM 4.603.A7, Annunciator Response Procedure		

Proposed references to be provided during examination: None

Learning Objective: 11798 – Predict the impacts of the following on the Intermediate Range Monitoring System: (f) Faulty Range Switch

Question Source: Bank #
Modified Bank # LO01128 (Note changes or attach parent)
New

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 41.5
55.43

Comments / Reference: PPM 4.603.A7	Revision: 51

Number: 4.603.A7	Use Category: CONTINUOUS	Major Rev: 051 Minor Rev: N/A Page: 11 of 71
Title: 603.A7 Annunciator Panel Alarms		

1-5 INTERMEDIATE RANGE MONITORING SYSTEM A, C, E OR G
UPSCALE TRIP OR INOPERABLE

1-5 WINDOW	SOURCE	AUTOMATIC ACTIONS
IRM A,C,E,G UPSCL TRIP OR INOP	Intermediate range channels A,C,E or G (K90) <u>Setpoints</u> <u>Upscale</u> - GE 120/125 <u>Inoperable</u> - • Detector high voltage low • Any module unplugged • IRM mode switch not in OPERATE • Loss of -24V power RPS-RLY-K12A, K12C, K12E, K12G	Reactor Half Scram if Mode Switch is not in RUN.

1. <u>IF</u> a Full Scram has occurred,
   <u>THEN</u> **REFER** to PPM 3.3.1, Reactor Scram.
2. **DETERMINE** which IRM channel(s) are upscale or inoperable.
3. **ADJUST** Range upscale IRM(s) up to bring on scale.
4. <u>IF</u> CRS directs,
   <u>THEN</u> **BYPASS** the inoperable IRM channel.
5. **RESET** the Half Scram.
6. <u>IF</u> instrument operability is in doubt,
   <u>THEN</u> **REFER** to the following:

Comments / Reference: LO01128, Original Question	Revision: N/A

A plant startup is in progress with power at 0.2% and increasing. Maintenance activities has caused RPS B to have a ½ scram in on it. A range switch has failed causing IRM G to remain on Range 4. The failure goes unnoticed. All other plant systems operate as designed.

Which of the following is correct for these conditions?

A. Enter PPM 3.3.1 Reactor Scram

B. The startup continues after the rod block is reset.

C. The power increase stops when the rod block is received.

D. Enter PPM 5.1.2 RPV Control ATWS.

Answer: A

Examination Outline Cross-reference:	Level	RO		SRO
Rev. 0 Date: 12/16/2016	Tier	2		
	Group	1		
	K/A	215004.K1.01		
Level of Difficulty: 3	Importance Rating	3.6		

Source Range Monitor (SRM) System: Knowledge of the physical connections and/or cause-effect relationships between SOURCE RANGE MONITOR (SRM) SYSTEM and the following: Reactor protection system

Question # 37

Given the following:

- The reactor is in Mode 4.
- Reactor Protection System (RPS) shorting links have been removed as part of an approved special test.
- As the crew is withdrawing control rods for the special test, the following source range indications are observed:
  - SRM A: $6.0x10^4$ cps
  - SRM B: $1.3x10^4$ cps
  - SRM C: $2.1x10^5$ cps
  - SRM D: $1.6x10^4$ cps

What is the condition of RPS channels ‘A’ and ‘B’ and the rod withdrawal block?

A. RPS A: not tripped
RPS B: not tripped
ROD BLOCK: not present

B. RPS A: tripped
RPS B: not tripped
ROD BLOCK: not present

C. RPS A: not tripped
RPS B: tripped
ROD BLOCK: present

D. RPS A: tripped
RPS B: tripped
ROD BLOCK: present

Answer: D

K/A Match:
Question determines if candidate understands when an RPS scram trip would occur based on signals from the SRM system.

SRO Only:
N/A

Explanation:

With the shorting links removed, SRM scram trip capability is enabled. Also, with shorting links removed, logic is non-coincidental (i.e. one SRM channel will cause both RPS channels to trip). With a channel above $2.0x10^5$ counts, RPS trips and a Rod block will occur.

A. Incorrect. RPS A and B trips will occur.

B. Incorrect. A trip will occur on BOTH RPS A and B.

C. Incorrect. A trip will occur on BOTH RPS A and B.

D. Correct.

Technical Reference(s)		Attached w/ Revision # See Comments / Reference
SD000161, CGS System Description, Volume 6, Chapter 8, Reactor Protection System		
SD000132, CGS System Description, Volume 6, Chapter 1, Source Range Monitor		

Proposed references to be provided during examination: None

Learning Objective: 11997 – Describe the function, purpose, and design features of the following major SRM system components: (j) trip circuits

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 41.2
55.43 ____

Comments / Reference: SD000132	Revision: 12

COLUMBIA SYSTEMS JUNE 2010
SRM SD000132, r12 mr0

3. Scram Trip LO-11999a

a) The SRMs have the ability to provide an upscale trip reactor scram. LO-12001a

b) Used during initial fuel loading and during dynamic shut down margin determination. Columbia Generating Station normally determines shutdown margin analytically thereby making the activation of this scram trip unnecessary. But this scram trip is enabled after core reload during sub-critical checks. LO-12002b LO-12004a

c) The activation of this scram trip capability requires the removal of shorting links in the RPS circuitry. The removal of the shorting links not only activates the SRM scram trip capability, but it also makes the IRM and APRM scram trips non-coincidental. This circuitry is covered in detail in the RPS System Description LO-12005d LO-5943

d) Upscale Trip at 2×10^5 cps.

4. Rod Withdrawal Blocks LO-11999b

a) Upscale alarm 10^5 cps LO-12002a

b) INOP:

- Switch out of Operate
- Module Unplugged
- High Voltage Low
- Negative 15 VDC Voltage Lost LO-12005d

Comments / Reference: SD000161	Revision: 17 mr 1
2. The reset circuit is disabled for ten seconds following any full scram signal to ensure that all rods have adequate time to fully insert. F. Shorting Links 1. The RPS trip logic can be configured to provide a scram from any single Source Range Monitor (SRM) or Intermediate Range Monitor (IRM). This logic condition is called non-coincident.	LO-7677 LO-7684j

Examination Outline Cross-reference:	Level	RO		SRO
Rev. 0 Date: 12/16/2016	Tier	2		
	Group	1		
	K/A	215005.K2.02		
Level of Difficulty: 3	Importance Rating	2.6		

Average Power Range Monitor/Local Power Range Monitor System: Knowledge of electrical power supplies to the following: APRM channels

Question # 38

What is the power supply to Average Power Range Monitor (APRM) channel "3"?

A. Critical Instrument Power Inverter IN-3.

B. 125 VDC Distribution Panel DP-S1-1A.

C. 24 VDC Distribution Panel DP-SO-A.

D. RPS A or RPS B

Answer: D

K/A Match:
Question specifically asks the power supply to APRMs.

SRO Only:
N/A

Explanation:

APRMs are powered by both RPS A and B via an auctioneered system.

A. Incorrect. Plausible because IN-3 is called the "Critical Instrument Power Inverter"

B. Incorrect. Plausible because DP-S-1A powers safety related DC components.

C. Incorrect. Plausible because DP-SO-A powers safety related DC components.

D. Correct.

Technical Reference(s)		Attached w/ Revision # See Comments / Reference
SD0001819, CGS System Description, Vol. 6 Chap. 10, Power Range Neutron Monitor		

Proposed references to be provided during examination: None

Learning Objective: 13709 – Identify the normal, alternate, and or emergency power supplies for major PRNM system components.

Question Source: Bank # LO03310
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 41.7
55.43 ______

Comments / Reference: SD001819	Revision: 2

6. Each APRM Chassis:

   a) APRM/OPRM channel – is powered from one of the Quad Low Voltage Power supplies (from either RPS Division A or B).

Examination Outline Cross-reference:	Level	RO		SRO
Rev. 0 Date: 12/16/2016	Tier	2		
	Group	1		
	K/A	217000.K3.04		
Level of Difficulty: 3	Importance Rating	3.6		

Reactor Core Isolation Cooling System (RCIC): Knowledge of the effect that a loss or malfunction of the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) will have on following: Adequate core cooling

Question # 39

Current plant conditions:

- A reactor scram was initiated due to a LOCA
- Reactor Core Isolation Cooling (RCIC) is the only system available for RPV injection.
- RCIC is injecting into the RPV.
- RPV pressure is 850 psig.

Emergency Depressurization (ED) has been initiated.

When should emergency depressurization be terminated?

ED should be terminated when RPV pressure is…

A. approximately 10 psig below DW pressure to minimize radiological release.

B. within 40 psig of WW pressure to minimize energy addition to the WW.

C. LT 125 psig to allow transition to shutdown cooling using the RHR system.

D. between 175 psig and 300 psig to maintain adequate core cooling with RCIC.

Answer: D

K/A Match:
This question requires that the candidate demonstrate the understanding that maintaining adequate core cooling by maintaining RCIC injection, when RCIC is the only RPV injection source, during a situation that requires emergency depressurization.

SRO Only:
N/A

Explanation:

A. Incorrect. Plausible because RPV pressure below drywell pressure would minimize release. Incorrect because this is not the reason ED is terminated.

B. Incorrect. Plausible because WW pressure can be a concern. Incorrect because this is not the reason ED is terminated.

C. Incorrect. Plausible because 125 psig is the highest pressure in which SDC can be placed in service. Incorrect because this is not the reason ED is terminated.

D. Correct. As stated in Ref. B, section 4.2.4, when RCIC is the only injection source and emergency depressurization is required, the depressurization is stopped when RPV pressure is between 175 psig and 300 psig. This is accomplished to maintain RCIC flow and restore RPV level to greater than TOP of Active Fuel (TAF), which is -161 inches. If RCIC flow is not maintained, loss of adequate core cooling will occur (see Ref. A, 4.2.1 for definition of adequate core cooling).

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
PPM 5.0.10, Flowchart Training Manual	
OI-15, EOP and EAL Clarifications	

Proposed references to be provided during examination: None

Learning Objective: 11229 – Analyze plant conditions and determine the bases for prioritizing emergency procedure implementation during emergency operations.

Question Source: Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 41.7
55.43 ______

Comments / Reference: PPM 5.0.10, section 4.2.1, definition of Adequate Core Cooling	Revision: Major 021 Minor 001

Number: 5.0.10	Use Category: INFORMATION	Major Rev: 021 Minor Rev: 001 Page: 17 of 320
Title: Flowchart Training Manual		

4.2 Definitions

4.2.1 The following words, acronyms and abbreviations are given specific definitions in order to ensure a correct and consistent interpretation of EOP flowchart terminology. A standard English language dictionary should be consulted for the definition of words and abbreviations not included in this list.

- ADD
  1) To combine; to place into.
- ADEQUATE CORE COOLING
  1) Heat removal from the reactor sufficient to prevent rupturing the fuel clad. Four viable mechanisms of adequate core cooling exist; in order of preference they are:
     - Core Submergence (-161 in.)
     - Steam cooling with injection of makeup water to the RPV (-186 in.)
     - Steam cooling without injection of makeup water to the RPV (-198 in.)
     - Spray cooling with HPCS or LPCS injecting at equal to or greater than 6000 gpm with RPV water level at or above 2/3 core height (-210 in.)

Comments / Reference: PPM 5.0.10, section 8.4.1, Purpose of emergency depressurization	Revision: Major 021 Minor 001

Number: 5.0.10	Use Category: INFORMATION	Major Rev: 021 Minor Rev: 001 Page: 189 of 320
Title: Flowchart Training Manual		

8.4 PPM 5.1.3, Emergency RPV Depressurization

The actions specified in PPM 5.1.3, Emergency RPV Depressurization, rapidly depressurize the RPV when RPV injection is available under non-failure-to-scram conditions.

8.4.1 Purpose

a. PPM 5.1.3 depressurizes the RPV as rapidly as possible within plant design limits and maintains it in a depressurized state. The steps of PPM 5.1.3, Emergency RPV Depressurization, may be required to:

- Establish or maintain adequate core cooling.
- Stop or minimize discharge of reactor coolant from unisolable primary system breaks.
- Reduce the energy within the RPV before reaching plant conditions for which the pressure suppression system may not be able to safely accommodate an SRV opening or condense steam discharged through the downcomers.
- Minimize radioactivity release from the RPV to the primary containment, secondary containment, and areas beyond the primary and secondary containments.

b. The RPV is considered “depressurized” when the RPV to suppression chamber differential pressure is less than the Decay Heat Removal Pressure (DHRP). The DHRP is used as the basis for the depressurized state since, below this differential, the rate of energy addition to the primary containment will be within the capacity of the primary containment vent path. Refer to Section 7 for a discussion of the DHRP.

Comments / Reference: PPM 5.0.10, section 8.4.2, Overview of emergency depressurization strategy	Revision: Major 021 Minor 001

- Primary containment pressure
- Wetwell water level (two places)

- PPM 5.3.1, Secondary Containment Control

- PPM 5.4.1, Radioactivity Release Control

b. The above requirements do not transfer control of RPV pressure to this flowchart, they only identify the need for depressurizing the RPV. Actual transfer of RPV pressure control to this procedure is accomplished by an override in the RPV pressure control flowpath of PPM 5.1.1.

c. If the main condenser is available, the fourth IF/THEN in PPM 5.1.1 Override P-1 permits rapid RPV depressurization through the main turbine bypass valves in anticipation of an emergency depressurization requirement.

d. Since PPM 5.1.3 directly affects the control of RPV pressure, entry to this flowchart is made from PPM 5.1.1 only. Therefore, when entry to PPM 5.1.3 is required, RPV pressure control actions specified in PPM 5.1.1 are superseded by override P-1 which directs entry to this flowchart. This hierarchical structure for transferring operator actions precludes having concurrently effective and conflicting instructions for controlling RPV pressure.

e. If a sufficient Wetwell water level exists, the actions in PPM 5.1.3 direct the operator to open the ADS number of SRVs. If the ADS number of SRVs cannot be opened, alternate RPV depressurization methods are specified. When the shutdown cooling interlock clears, Shutdown Cooling is placed in service, if necessary, and the plant is cooled to cold shutdown conditions (<200°F). PPM 5.1.3 remains in effect as the RPV pressure control procedure until all EOPs are exited or it is determined that RPV water level cannot be determined.

f. It is generally desirable to fully depressurize the RPV and cool down the plant to cold shutdown conditions. However, while RPV pressure reductions will tend to increase flow from motor-driven injection sources, full depressurization may result in loss of steam-driven injection systems. Emergency RPV depressurization must therefore be coordinated with core cooling strategies.

g. Full depressurization and cooldown is appropriate only if adequate core cooling will not be sacrificed as a result. Loss of adequate core cooling would compound the plant challenges requiring emergency depressurization and increase any resulting radioactivity release. Core cooling is thus prioritized over other EOP objectives. If, at any time during RPV depressurization, it is anticipated that continued pressure reduction will result in loss of injection flow required for adequate core cooling, the depressurization is terminated. Pressure should then be controlled as low as practicable but above the minimum value at which the required injection flow can be sustained.

Comments / Reference: PPM 5.0.10, section 8.4.3.g.1, Definition of Decay Heat Removal Pressure (DHRP)	Revision: Major 021 Minor 001

Number: 5.0.10	Use Category: INFORMATION	Major Rev: 021 Minor Rev: 001 Page: 197 of 320
Title: Flowchart Training Manual		

f. Step P-6:

1) Only two open SRVs are needed to depressurize the RPV when decay heat is generating steam. A minimum of seven SRVs, however, is specified in this step to ensure the RPV will depressurize at a sufficiently rapid rate without relying on less desirable, alternate means of depressurization. Seven SRVs is the Minimum Number of SRVs Required for Emergency Depressurization (MNSRED). Refer to Section 7 for discussion of the MNSRED.

2) The phrase “can be opened” is used in this decision since the second IF/THEN in Override P-1 may require that SRVs be closed before the RPV is fully depressurized. Use of Table 13 RPV depressurization methods in Step P-8 would then be inappropriate.

g. Step P-7:

1) If this step is reached with fewer than seven SRVs open, alternate depressurization methods are required, but only if the RPV has remained at pressure. If the RPV has already depressurized due to prior RPV pressure control actions or a break in the reactor coolant system, the alternate depressurization methods may not be needed. Alternate RPV depressurization methods should be avoided if possible because their use may be labor intensive, require defeating safety system interlocks, and possibly lead to an offsite radioactivity release.

2) The RPV pressure of 40 psig above Wetwell pressure is the Decay Heat Removal Pressure (DHRP). The DHRP is used to define the depressurized condition since, below this differential, the rate of energy addition to the primary containment will be within the capacity of the primary containment vent path. Refer to Section 7 for discussion of the DHRP.

Comments / Reference: OI-15, section 4.2.4, RPV Pressure Control	Revision: Major 025

Number: OI-15	Use Category: INFORMATION	Major Rev: 025 Minor Rev: N/A Page: 9 of 31
Title: EOP and EAL Clarifications		

- When in the EOP flowcharts and the low ECCS water leg pump pressure annunciator is received, or when a loss of keep fill pressure occurs, the HPCS/LPCS/RHR pumps may be started if needed for EOP related activities. If time permits since the loss of water leg pump pressure, the operator should attempt to restore keep fill pressure through fill and vent per procedure. If while restoring keep fill pressure plant conditions approach a limiting condition for adequate core cooling or maintenance of containment integrity, the ECCS pumps are started regardless.

4.2.4 RPV Pressure Control

- If an ADS actuation has occurred and the SRVs are open it is not expected or necessary to stop the depressurization from continuing, unless specifically directed by EOPs.

- The operator has the latitude to control RPV cooldown at any rate from that resulting from natural decay heat reduction up to the Technical Specification cooldown rate of 100 °F/hr.

- When initiating flow through an ADS valve from H13-P601, the individual SRV control switches should be used. If the "C" solenoid is not available and emergency depressurization is necessary, the use of the ADS logic initiation arm-and-depress pushbuttons is acceptable. However, if pressure control is ongoing from the ADS valve switches on H13-P628/P631, then use of the back panel switches for emergency RPV depressurization is preferred.

- For overrides that allow termination of depressurization, examples are 5.1.1 P-4, or 5.1.3 P-1, the preferred pressure band is 175 psig to 300 psig and this assumes RCIC is the only source of injection. This ensures RCIC stays above 2100 RPM, but also minimizes heat input to the suppression pool.

- Table 4, It is expected that SRVs will be operated from H13-P601 using the individual SRV control switches. If the normal pneumatic supply is unavailable, the H13-P601 control switches will be placed in AUTO and SRV control will continue from the back panel (H13-P628/P631) using the ADS supply. If the normal pneumatic supply is restored, the back panel control switches will be returned to AUTO and SRV control resumed from H13-P601. If the ADS pneumatic supply is depleted such that only one nitrogen bottle remains in each division, the continuous nitrogen supply is declared unavailable. Loss of continuous nitrogen supply shall mean that all backup means of pneumatic supply to the SRVs have been lost. If the bottles in the Diesel Hallway are available and valved in, the "Loss of continuous nitrogen supply" occurs when the truck bay bottles have been sequenced through.

Examination Outline Cross-reference:	Level	RO		SRO
Rev. 0 Date: 12/16/2016	Tier	2		
	Group	1		
	K/A	217000.A2.14		
Level of Difficulty: 3	Importance Rating	3.3		

Reactor Core Isolation Cooling System (RCIC): Ability to (a) predict the impacts of the following on the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Rupture disc failure: Exhaust-Diaphragm

Question # 40

Given the following:

- A reactor scram was initiated due to a LOCA.
- RCIC is in operation and maintaining RPV level GT -161 inches.
- The inboard RCIC Turbine Exhaust rupture disc has ruptured.

What is the impact on the RCIC system?

The RCIC-P-1 turbine is…

A. in operation. The RCIC turbine must be tripped.

B. tripped. The RCIC system should be restored to service.

C. in operation. The RCIC system may remain in service.

D. tripped. The RCIC turbine must remain tripped.

Answer: D

K/A Match:
The question asks how a rupture disc failure (exhaust) will impact the RCIC system and what the next actions should be.

SRO Only:
N/A

Explanation:

Four pressure switches (RCIC-PS-12A, B, C, and D) cause a high pressure trip of the RCIC turbine. The actuation causes a RCIC isolation signal which will cause the turbine to trip and the system to be automatically isolated. The system cannot be used if this were to happen.

A. Incorrect. An automatic RCIC isolation occurs. Plausible because there are two rupture discs. A failure of the upstream disc will not result in steam release due to the outboard rupture disc. It is in place because it is exposed to cyclic stresses when the pump is started and stopped.

B. Incorrect. The system cannot be restored to service until the disc is repaired. There is no procedural guidance to bypass the high exhaust pressure isolation. Plausible because the RCIC pump will trip.

C. Incorrect. An automatic RCIC isolation occurs. Plausible because there are two rupture discs. A failure of the upstream disc will not result in steam release. It is in place because it is exposed to cyclic stresses when the pump is started and stopped.

D. Correct.

Technical Reference(s)		Attached w/ Revision # See Comments / Reference
SD000180, CGS System Description, Volume 7, Chapter 6, Reactor Core Isolation Cooling		

Proposed references to be provided during examination: None

Learning Objective: 11671 – From memory, draw a simplified diagram of the RCIC system showing all major components and flow paths. (g) rupture discs

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 41.5
55.43

Comments / Reference: SD000180	Revision: 16 mr 1

U. Turbine Exhaust Line Overpressure Protection (Rupture Discs)

The RCIC turbine casing and exhaust piping are protected against overpressure by a dual rupture disc arrangement. Each rupture disc is designed to rupture at 150 psig and is supported to prevent reversal by vacuum. The rupture disc assemblies are located upstream of the exhaust check valve and are subject to pressure cycles associated with turbine operation. The rupture discs are designed to pass 75,000-lbm/hr and limit exhaust pressure to 165 psig when the exhaust line is isolated with the governor valve full open. The downstream rupture disc prevents the unnecessary release of steam in the event of a premature failure of the upstream disc from accumulated cyclic stresses. The space between the rupture discs is instrumented with pressure switches and vented through a 1/8" orifice to the steam tunnel side of the downstream disc. The exhaust line travels through the main steam tunnel and ends in the Turbine Building 471' Overhead area.

Comments / Reference: SD000180	Revision: r16 mr1

6. Turbine exhaust rupture diaphragm pressure

Four pressure switches (RCIC-PS-12A, B, C and D) sense the pressure between the two rupture diaphragms. Division 1 uses signals from two switches while Division 2 uses the other two. If only one pressure switch is actuated in either logic, the respective division RCIC high turbine exhaust diaphragm pressure annunciator will be actuated in the control room. If both pressure switches in either logic circuit are actuated simultaneously, a RCIC isolation signal will be initiated.

Examination Outline Cross-reference:	Level	RO		SRO
Rev. 0 Date: 12/16/2016	Tier	2		
	Group	1		
	K/A	218000.K4.03		
Level of Difficulty: 3	Importance Rating	3.8		

Automatic Depressurization System: Knowledge of AUTOMATIC DEPRESSURIZATION SYSTEM design feature(s) and/or interlocks which provide for the following: ADS logic control

Question # 41

Given the following:

- A loss of coolant accident (LOCA) has occurred concurrent with a loss of offsite power.
- Both diesel generator #1 (DG1) and diesel generator #2 (DG2) have tripped and cannot be re-started.
- Vessel level is -120" and lowering slowly.
- Reactor core isolation cooling (RCIC) has tripped on high exhaust pressure.
- High pressure core spray (HPCS) is operating as expected and injecting into the vessel.

Which of the following describes the response of the automatic depressurization system (ADS) when vessel level reaches Level 1?

A. ADS timers will start and time out. ADS will then initiate.

B. ADS timers will not start. ADS will not initiate.

C. ADS timers will start and time out. ADS will not initiate.

D. ADS timers will already be timed-out when Level 1 is reached. ADS will initiate immediately upon reaching Level 1.

Answer: C

K/A Match:
Question provides plant conditions and discriminates whether a candidate knows the ADS logic behavior based on those conditions.

SRO Only:
N/A

Explanation:

With DG1 and DG2 out of service, no low pressure ECCS pumps are available so ADS initiation will not occur. The ADS timer, however, will still start when RPV level reaches level 1 (-129").

A. Incorrect. ADS will not initiation. Plausible because if low pressure ECCS pumps were available, ADS would initiate.

B. Incorrect. ADS timers WILL start and timeout. Plausible because ADS does not initiate when no low pressure ECCS pumps are running.

C. Correct.

D. Incorrect. ADS timers start when Level 1 is reached, not before. Plausible because ADS senses when water level is below BOTH level 3 and level 1 and the stem states that RPV level is below level 3.

Technical Reference(s)		Attached w/ Revision # See Comments / Reference
SD000186, CGS System Description, Volume 7, Chapter 5, Automatic Depressurization		

Proposed references to be provided during examination: None

Learning Objective: 5070 – State the interlocks (conditions) that must be satisfied prior to automatic or manual initiation of ADS.

Question Source: Bank # LR00641
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 41.7
55.43 ______

Comments / Reference: SD000186	Revision: 12

3. Time Delay trip channel (Fig. 9)

The time delay trip is assigned to logic channel "A" for Division 1 ADS and "B" for Division 2. The time delay trip channels monitor input from reactor vessel level 3 and level 1, low pressure ECCS pumps, and status of the 105 second delay timer. When reactor level has been below level 3 and level 1 for 105 seconds, and low pressure ECCS pump pressure is sufficient to indicate that at least one of two divisionally related low pressure pumps is running, the channel is tripped. The 105 second timer provides the HPCS system enough time to restore RPV water level above level 1.

Examination Outline Cross-reference:	Level	RO		SRO
Rev. 0 Date: 12/16/2016	Tier	2		
	Group	1		
	K/A	223002.K6.04		
Level of Difficulty: 4	Importance Rating	3.3		

Primary Containment Isolation System/Nuclear Steam Supply Shut-Off: Knowledge of the effect that a loss or malfunction of the following will have on the PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF: Nuclear boiler instrumentation

Question # 42

CGS is operating in Mode 1.

- MS-LT-61A, WR RPV Level Transmitter, failed low.
- P601.A12.2-4, NS4 GROUP 1 ISOLATION RPV LEVEL LOW (-129"), is LIT.

Subsequently, a LOCA occurred. The crew scrammed the reactor and entered PPM 3.3.1, Reactor Scram and PPM 5.1.1, RPV Control. Current plant conditions:

- MSIV ISOL SYS A/B LOW RPV LVL / HI STM TUNNEL TEMP BYPASS switches are NOT in “BYPASS”
- Wide Range RPV Level indication:
  - MS-LIS-200A: Downscale
  - MS-LIS-200B: -115 inches
  - MS-LIS-200C: -132 inches
  - MS-LIS-200D: -125 inches

What is the condition of the MSIV inboard and outboard isolation valves?

MSIV Inboard Isolation valves, MS-V-22A-D, are __(1)__ and MSIV Outboard Isolation valves, MS-V-28A-D, are __(2)__ .

A. (1) OPEN
(2) OPEN

B. (1) CLOSED
(2) OPEN

C. (1) OPEN
(2) CLOSED

D. (1) CLOSED
(2) CLOSED

Answer: A

K/A Match:
This question evaluates the candidates knowledge of the effect on the NS4 system when one RPV level transmitter fails and an additional level transmitter actuates a NS4 isolation signal.

SRO Only:
N/A

Explanation:
Low RPV level on MS-LIS-200A/B/C/D results in a group 1 isolation (MSIVs). Group 1 isolation has a one out of two taken twice logic meaning that one transmitter in the A/C channels must be low and one transmitter in the B/D channels must be low for an actuation to occur. In this case, BOTH A and C channels are below the trip setpoint, but neither B or D are below the setpoint, so no actuation would occur.

A. Correct.

B. Incorrect. All MSIVs would still be open. Plausible because different NS4 logic schemes exist for different group isolation signals.

C. Incorrect. All MSIVs would still be open. Plausible because different NS4 logic schemes exist for different group isolation signals.

D. Incorrect. All MSIVs would still be open. Plausible because different NS4 logic schemes exist for different group isolation signals.

Technical Reference(s)	
SD000173, CGS System Description, Volume 8, Chapter 5, Nuclear Steam Supply Shutoff System (NS4)	Attached w/ Revision # See Comments / Reference
SD000126, CGS System Description, Volume 4, Chapter 2, Nuclear Boiler Instrumentation (NBI)	
PPM 4.601.A12, Annunciator Panel Alarms, Annunciator 2-4 Response	

Proposed references to be provided during examination: None

Learning Objective: 5596 – Describe the isolation logic used by the NS4 system for MSIV isolation and Group 3 and 4.

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 41.7
55.43 ____

Comments / Reference: SD000173	Revision: 14 mr2

IV. COMPONENT DESCRIPTION

LO-5596

A. Single Point Vulnerability

Single Point Vulnerability (SPV) refers to a component whose failure is determined to cause one or more of the following conditions:

- Unit Trip
- 10% or more reduction in RX Power
- Configuration that requires a Unit Shutdown

The following system components have been determined to be SPV's:

- None Identified

B. Isolation Logic

The NS4 Isolation logic is comprised of trip systems "A" and "B" and logic channels "A", "B", "C" and "D". The isolation logic varies with the different NS4 groups as the following paragraphs describe.

1. Group 1, MSIV logic is the only group that uses a one-out-of-two-taken-twice logic (similar to RPS). See Figure 1. The MSIVs do not close on a single trip system actuation. All eight MSIVs will close only when both trip systems are actuated. Trip system "A" consists of logic channels "A" and "C" and trip system "B" consists of logic channels "B" and "D". For example; there are four level switches that cause a -129" isolation of MSIVs. One level switch in channel "A" or "C" must trip, along with one level switch in channel "B" or "D" to cause all eight MSIVs to close. No MSIVs will close if only one level switch, in any channel, is actuated.

   There is no combination of isolation signals that could cause only the inboard or only the outboard MSIVs to close. When a valid isolation signal is received, all MSIVs will close.

2. Group 1 MSL drains and Group 2 thru 7 use a two-out-of-two-taken-once logic (with the exception of portions of group 6 and 7). See Figures 1

Comments / Reference: SD000126	Revision: 13 mr 1

D. Reactor Level Indications

RANGE	PANEL	INDICATION	TRANSMITTER
Narrow 0 to 60"	P603	RFW-LI-606A,B,C	RFW-DPT-4A,B,C
		RFW-LR-608 (pt. 2)	RFW-DPT-4A or B
	P602	RFW-LI-606D	RFW-DPT-4C
Wide -150" to 60"	P601	MS-LR/PR-623A	MS-LT-26A
		MS-LR/PR-623B	MS-LT-26D
	P603	MS-LI-604	MS-LT-26C
	RSP	MS-LI-10AR	
	P609/611	MS-LIS-200A,B,C,D	MS-LT-61A-D
Upset 0 to 180"	P603	RFW-LR-608 (point 1)	RFW-DPT-17
Shutdown 0 to 400"	P602	MS-LI-605	MS-LT-27
Fuel Zone -310" to -110"	P601	MS-LR-615 (comp)	MS-LT-44A
		MS-LI-612 (comp)	MS-LT-44B
		MS-LI-610 (uncomp)	MS-LT-44B

Comments / Reference: P601.A12.2-4 Annunciator Response	Revision: Major 026 Minor 001

Number: 4.601.A12	Use Category: CONTINUOUS	Major Rev: 026 Minor Rev: 001 Page: 17 of 34
Title: 601.A12 Annunciator Panel Alarms		

2-4 NS4 GROUP 1 ISOLATION RPV LEVEL LOW (-129")

2-4 WINDOW	SOURCE	AUTOMATIC ACTIONS
NS4 GROUP 1 ISOLATION RPV LEVEL LOW (-129")	RPS-RLY-K38A (-129" RPV Level) (MS-LIS-200A) OR RPS-RLY-K38B (-129" RPV Level) (MS-LIS-200B)	Single switch actuation • Half Isolation Both switches actuating closes: • MS-V-19 • MS-V-67A through D • MS-V-22A through D • MS-V-28A through D

Examination Outline Cross-reference:	Level	RO		SRO
Rev. 0 Date: 12/16/2016	Tier	2		
	Group	1		
	K/A	239002.K5.05		
Level of Difficulty: 3	Importance Rating	2.6		

Relief/Safety Valves: Knowledge of the operational implications of the following concepts as they apply to RELIEF/SAFETY VALVES : Discharge line quencher operation

Question # 43

Which of the following describes the reason for Emergency Depressurization with a high Suppression Pool Level?

A. SRV discharge would result in exceeding code allowable stresses which could cause a failure of the SRV tailpipes.

B. A large LOCA would exceed the Heat Capacity Temperature Limit resulting in the failure of the wetwell/drywell interface.

C. A large LOCA would result in exceeding the SRV Tail Pipe Level limit and exceed the Primary Containment Pressure Limit.

D. SRV discharge would cause excessive containment pressure on the drywell floor and exceed the Primary Containment Pressure Limit.

Answer: A

K/A Match:
Question determines if candidate understands the reason SRVs are opened on high WW level which directly ties to the design and capabilities of the quenchers.

SRO Only:
N/A

Explanation:

SRVs are opened to Emergency Depressurize the reactor prior to wetwell level being too high (SRVTPLL) and challenging the quenchers/tailpipes.

A. Correct.

B. Incorrect. The reason/concern related to high wetwell level is NOT exceeding HCTL. Plausible since HCTL is related to wetwell temperature.

C. Incorrect. This condition does not challenge PCPL. Plausible since distractor includes SRVTPLL.

D. Incorrect. Pressure on the drywell floor is not the concern. Plausible since the wetwell performs the quenching function to minimize the impact on primary containment.

Technical Reference(s)		Attached w/ Revision # See Comments / Reference
PPM 5.0.10, Flowchart Training Manual		

Proposed references to be provided during examination: None

Learning Objective: 8387 – Given a list, identify the statement that describes the reason for emergency depressurizing the RPV if wetwell level and reactor pressure cannot be restored and maintained below SRVTPLL.

Question Source: Bank # LO00144
Modified Bank # ________ (Note changes or attach parent)
New ________

Question History: Last NRC Exam N/A

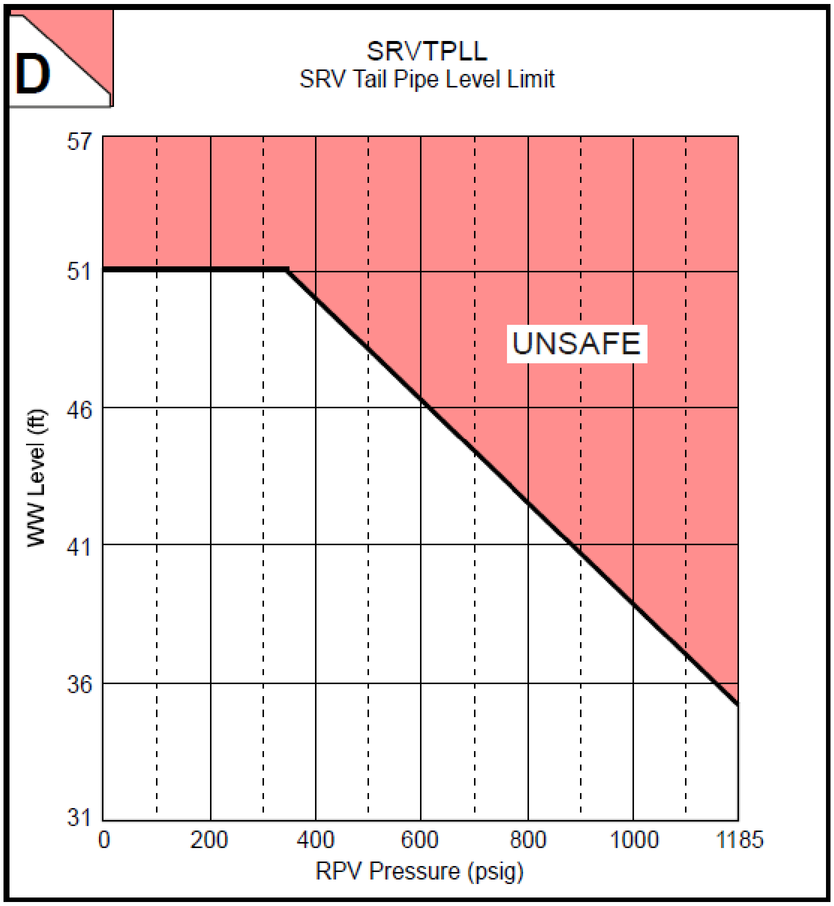
Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ________

10 CFR Part 55 Content: 55.41 41.5
55.43 ________

Comments / Reference: 5.0.10	Revision: 21 mr 1

Title: Flowchart Training Manual	Page: 87 of 320

7.14 SRV Tail Pipe Level Limit



7.14.1 The SRV Tail Pipe Level Limit (SRVTPLL) is the highest Wetwell 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. This water level is a function of RPV pressure, and the Limit is utilized to preclude SRV system damage and containment failure.

Examination Outline Cross-reference:	Level	RO		SRO
Rev. 0 Date: 12/16/2016	Tier	2		
	Group	1		
	K/A	259002.A1.04		
Level of Difficulty: 3	Importance Rating	3.6		

Reactor Water Level Control System: Ability to predict and/or monitor changes in parameters associated with operating the REACTOR WATER LEVEL CONTROL SYSTEM controls including: Reactor water level control controller indications

Question # 44

RFW-P-1B is being placed in service as the second Reactor Feed Pump per SOP-RFT-START. Feedwater Level Control is aligned as follows:

- RFW-P-1A is in AUTO.
- RFW-P-1B is in MDEM with RFW-V-102B (Pump Discharge Valve) open. RFW-P-1B discharge pressure is at shutoff head.
- RPV level is being maintained by RFW-LIC-600 (RPV Master Level Controller) in AUTO

If CRO1 depresses the UP arrow for RFW-P-1B on RFT-COMP-1, RFW-P-1B speed will…

A. rise and RFW-P-1A speed will remain the same.

B. rise and RFW-P-1A speed will rise.

C. rise and RFW-P-1A speed will lower.

D. remain the same and RFW-P-1A speed will remain the same.

Answer: C

K/A Match:
Question determines if candidate can predict changes in the feedwater level control system controllers following a feedwater pump speed adjustment.

SRO Only:
N/A

Explanation:

A. Incorrect: See A. If RFW-V-102B were not open, this distractor would be correct. RFW-V-102B is opened when RFW-P-1B discharge pressure is within 20 to 30 psi of RFW-P-1A discharge pressure. As a result, once speed is raised, RFW-P-1B will feed the reactor and raising the speed of one pump will cause the speed of the other pump to lower.

B. Incorrect: See A. The UP arrows on RFT-COMP-1 control feed turbine speeds individually. The response in this distractor would be expected if the INC button on RFW-LIC-600 was depressed with both pumps in AUTO.

C. Correct: With the conditions provided in the stem, depressing the UP arrow for RFW-P-1B will cause the B feed turbine speed to rise. This will result in an increase in feed flow, which the FWLC system will detect and respond to by lowering the speed of RFW-P-1A.

D. Incorrect: See A. The UP arrow will not raise the speed of a feed turbine if the turbine has not been reset following a trip. The speed of the other feed turbine would also remain unchanged in that situation.

Technical Reference(s)		Attached w/ Revision # See Comments / Reference
SD000157, CGS System Description, Volume 4, Chapter 3, Feedwater Level Control		

Proposed references to be provided during examination: None

Learning Objective: 5394 – Describe the function of each of the following controls and how they relate to each other: (a)Turbine Speed Controllers

Question Source: Bank # LO02781
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 41.5
55.43 ______

Comments / Reference: SD000157	Revision: 16

G. Feedwater Level Components Controlled by the Invensys Triconex (Trident Model) system

1. Turbine A or B Speed Controller On either RFT-COMP-1 or RFT-COMP-2 (H13-P840 Figures 4 and 4a) LO-5394c LO-11701e

Each speed controller is a passive device, which works in conjunction with the Invensys Triconex (Trident Model) system to control the speed to the RFW turbine. The controller has three modes of operation: Automatic (Auto), Manual Demand Request (MDEM) and Manual Direct Valve Positioning (MDVP).

In Automatic (A) mode, the feedwater flow demand from the PLC (H13-P612) is converted to a speed demand, compared to actual speed measured at the RFW turbine to generate a governor valve position signal and sent to the local control panels RFT-CP-2A and 2B (discussed later). In Auto, the speed of the RFW turbine can be biased or changed in relation to the speed of the other RFW turbine by pressing one of the Raise or Lower PBs on the controller. The amount of bias is displayed on the vertical bias bar on the controller while in Automatic mode, and under the bars in digital format. The selected controller will shift to Automatic mode when the controller's Auto touch pad button is pressed provided:

a) A Valid demand signal is being received from the PLC within pumping speed range of the governor.

b) The associated feed turbine is reset (not tripped)

c) The Master Controller output and current manual speed demand deviation is within +/- 25 RPM

d) The system sees a valid speed signal.

In Manual Demand Request (MDEM) mode, the controller generates a manual speed demand by pressing the up arrow or down arrow, compares it to actual speed measured at the RFW turbine to generate a governor valve position signal, which is sent to the Trident Invensys Triconex and passed on to the electric motor positioner. The amount of deviation between the PLC speed demand (Master Controller output) and current manual speed demand is displayed on the selected controller's vertical Deviation bar, and under the bars in digital format, while in Manual Demand Request (MDEM) mode.

In MDEM mode, the gain and reset rate circuits of the controller will slow down the controller response at lower demand signals to make control smoother

Examination Outline Cross-reference:	Level	RO		SRO
Rev. 0 Date: 12/16/2016	Tier	2		
	Group	1		
	K/A	261000.A2.03		
Level of Difficulty: 3	Importance Rating	2.9		

Standby Gas Treatment System: Ability to (a) predict the impacts of the following on the STANDBY GAS TREATMENT SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: High train temperature

Question # 45

CGS is operating in Mode 1

- A LOCA occurred.
- Both Standby Gas Treatment (SGT) trains automatically started during the event.
- SGT System 'A' has been placed in standby, SGT System 'B' is running.
- The following annunciators have just alarmed:

  H13-P811.K2-2.4, CHARCOAL FLTR B-1 OUTLET TEMP HIGH
  H13-P811.K2-2.5, CHARCOAL FLTR B-2 OUTLET TEMP HIGH

What actions should the crew take to address these alarms per ABN-SGT-TEMP/RAD, Standby Gas Treatment Charcoal High Temperature/Radiation?

The crew should …

A. start SGT System 'A', place SGT System 'B' in standby, and initiate emergency deluge on SGT System 'B'.

B. start SGT System 'A', place SGT System 'B' in recirculation, and visually check for fire on SGT System 'B'.

C. ensure SGT Systems 'A' and 'B' are in standby and initiate emergency deluge on SGT System 'B'.

D. Maintain SGT System 'A' in standby, place SGT System 'B' in recirculation, and visually check for fire on SGT System 'B'.

Answer: B

K/A Match:
Question determines if candidates know the correct actions to take in the event of high temperature condition in the SGT system.

SRO Only:
N/A

Explanation:

Per ABN-SGT-TEMP/RAD the major actions/strategy that should be taken is to place the running (high temp) train in recirculation and to start the standby train. ABN-SGT-TEMP/RAD describes that high temperatures can occur from decay of particulates following a LOCA and that operating the associated SGT fan will cool the charcoal bed. This is the reason that the effected train should remain in recirculation and not secured.

A. Incorrect. SGT Train B should be placed in recirculation, not standby. Plausible because all other actions in the distractor are appropriate and shutting down a train that is experiencing high temperatures would be appropriate in other plant systems and conditions.

B. Correct.

C. Incorrect. SGT Train B should be placed in recirculation, not standby. Plausible because securing systems that might be impacted by high temperatures is a correct action in other plant systems and conditions. Also, initiating the emergency deluge system would reduce temperatures, but is only allowed in a fire situation.

D. Incorrect. SGT Train A should be placed in service. Plausible because other portions of the distractor are correct.

Technical Reference(s)		Attached w/ Revision # See Comments / Reference
ABN-SGT-TEMP/RAD, Standby Gas Treatment Charcoal High Temperature/Radiation		

Proposed references to be provided during examination: None

Learning Objective: 15809 – With procedures available, discuss all contingencies associated with the subsequent operator actions of ABN-SGT-TEMP/RAD

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 41.5
55.43 ______

Comments / Reference: ABN-SGT-TEMP/RAD	Revision: 3

Number: ABN-SGT-TEMP/RAD	Use Category: CONTINUOUS	Major Rev: 003 Minor Rev: N/A Page: 4 of 8
Title: Standby Gas Treatment Charcoal High Temperature/Radiation		

4.0 SUBSEQUENT OPERATOR ACTIONS

WARNING

The following steps require an Equipment Operator and a Health Physics technician present locally to monitor for smoke and airborne contaminants.

4.1 **START** the unaffected SGT unit per SOP-SGT-START. ____

NOTE: Step 4.2 provides instruction if a high temperature or high radiation in the charcoal beds exists. Step 4.3 provides instruction if a fire in the charcoal beds exists.

4.2 **START** and **RECIRCULATE** the SGT train that has the charcoal high temperature or high radiation as follows: {P-206014}

Comments / Reference: ABN-SGT-TEMP/RAD	Revision: 3

4.2 SGT charcoal may have a high temperature due to a faulted strip heater, an external temperature source, or from radioactive decay of particulates following a LOCA. SGT charcoal may have a high radiation due to radioactive decay of particulates following a LOCA. Operating the associated SGT fan will cool the charcoal bed.

4.3 SGT charcoal can only reach ignition temperature from an external source. SGT charcoal will not reach ignition temperatures from radioactive decay of particulates following a LOCA.

Examination Outline Cross-reference:	Level	RO		SRO
Rev. 0 Date: 12/16/2016	Tier	2		
	Group	1		
	K/A	262001.A3.01		
Level of Difficulty: 3	Importance Rating	3.1		

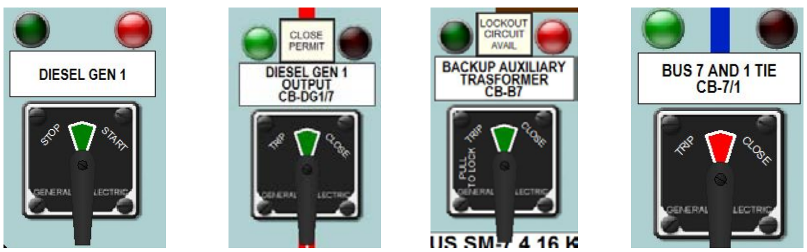
A.C. Electrical Distribution: Ability to monitor automatic operations of the A.C. ELECTRICAL DISTRIBUTION including: Breaker tripping

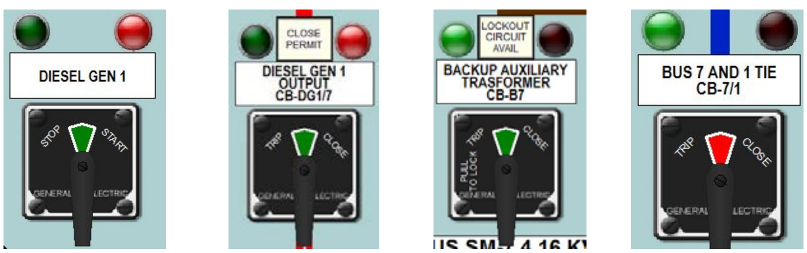
Question # 46

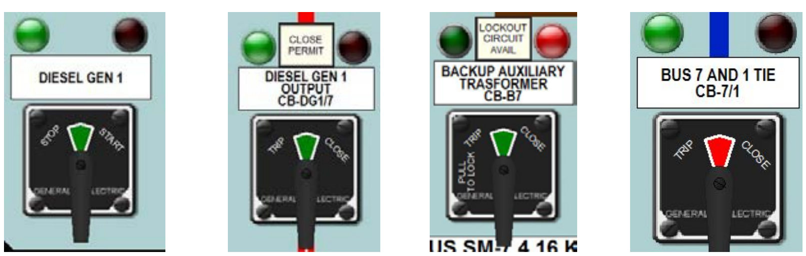
CGS is operating in Mode 1.

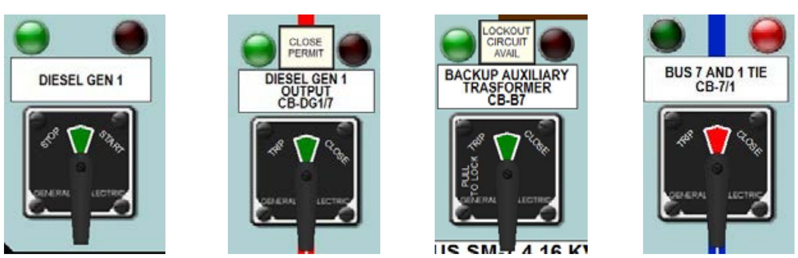
- A switchyard fault caused the concurrent loss of the main transformer and the startup transformer.

Which of the following is the expected lineup for bus SM-7 supply breakers?

A. 

B. 

C. 

D. 

Answer: A

K/A Match:
Question determines if candidates can select the correct status of the electrical distribution system following a trip and loss of the main and startup transformer.

SRO Only:
N/A

Explanation:
Bus SM-7 will experience an undervoltage condition which will cause an automatic transfer of power to the backup transformer (TR-B). In addition, the EDG will start, but will not connect to the bus due to the availability of TR-B.

A. Correct.

B. Incorrect. CB-DG1/7 would not close. Plausible because if TR-B was, unavailable this distractor would be correct.

C. Incorrect. DG 1 should start. Plausible because CB-B7 will close and supply power to SM-7.

D. Incorrect. DG1 should start. Plausible because this is the normal lineup for SM-7 and candidate must understand that SM-7 is powered by the startup transformer through CB-7/1.

Technical Reference(s)		Attached w/ Revision # See Comments / Reference
SD000182, CGS System Description, Volume 1, Chapter 2, AC Distribution		

Proposed references to be provided during examination: None

Learning Objective: 5051 – Explain the design features and/or system interlocks or response which provide for the following: (e) SM-7(8) response to undervoltage, including load shedding.

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam ____

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 41.7
55.43 ____

Comments / Reference: SD000182	Revision: r19 mr0

COLUMBIA SYSTEMS
AC DISTRIBUTION

October 2015
SD000182, r19 mr0

4. SM7 and SM8 Undervoltage Response
   Primary or Secondary UV (after an 8 sec TD) causes the following actions to occur in sequence. Note that on Secondary UV, the tripping of breaker 7-1(8-3) at 8 sec. begins the sequence. LO-11824

   a) DG1(2) starts (immediately after loss of bus voltage)
   b) Breaker 7-1 (8-3) trips (@ 3 sec. on Pri or 8 sec. on Sec UV)
   c) SM-7(8) load breakers open (@ 3 sec. after loss of bus voltage)
   d) Backup Transformer breaker B-7(B-8) closes if TR-B is available (@ 5.5 sec.)
   e) Diesel Generator breaker DG1-7(2-8) closes if B-7(B-8) is open. (@7-10 sec.)

Examination Outline Cross-reference:	Level	RO		SRO
Rev. 0 Date: 12/16/2016	Tier	2		
	Group	1		
	K/A	262002.A4.01		
Level of Difficulty: 3	Importance Rating	2.8		

Uninterruptable Power Supply (A.C./D.C.): Ability to manually operate and/or monitor in the control room: Transfer from alternative source to preferred source

Question # 47

CGS is operating in Mode 1.

- Efforts are underway to transfer IN-1 from the Alternate AC Source through the Maintenance Bypass Switch to the UPS Inverter through the Static Switch.

Concerning the transfer, which of the following is correct?

Power is being transferred to the UPS Inverter from…

A. MC-7F. It will be a make-before-break transfer.

B. MC-7A. It will be a make-before-break transfer.

C. MC-7F. It will be a break-before-make transfer.

D. MC-7A. It will be a break-before-make transfer.

Answer: A

K/A Match:
Question directly asks the expected conditions/response when transferring IN-1 from the maintenance source back to the normal source.

SRO Only:
N/A

Explanation:
The maintenance source is powered by MC-7F. The maintenance source goes through the static switch which is a break-before-make connection.

A. Correct.

B. Incorrect. MC-7A is not the power source.

C. Incorrect. It is not a break-before-make transfer.

D. Incorrect. MC-7A is not the power source.

Distractors are plausible because MC-7A is the power supply to the BYPASS source which is also a break-before-make transfer.

Technical Reference(s)		Attached w/ Revision # See Comments / Reference
SD000194, CGS System Description, Volume 1, Chapter 3, Uninterruptible Power Supply System		

Proposed references to be provided during examination: None

Learning Objective: 5896 – List the power supplies to each inverter: (a) E-IN-1
5891 – State the purpose and various functions of the following with respect to E-IN-1: (c) static switch, bypass switch

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ____

10 CFR Part 55 Content: 55.41 41.7
55.43 ____

Comments / Reference: SD000194	Revision: 13 mr0

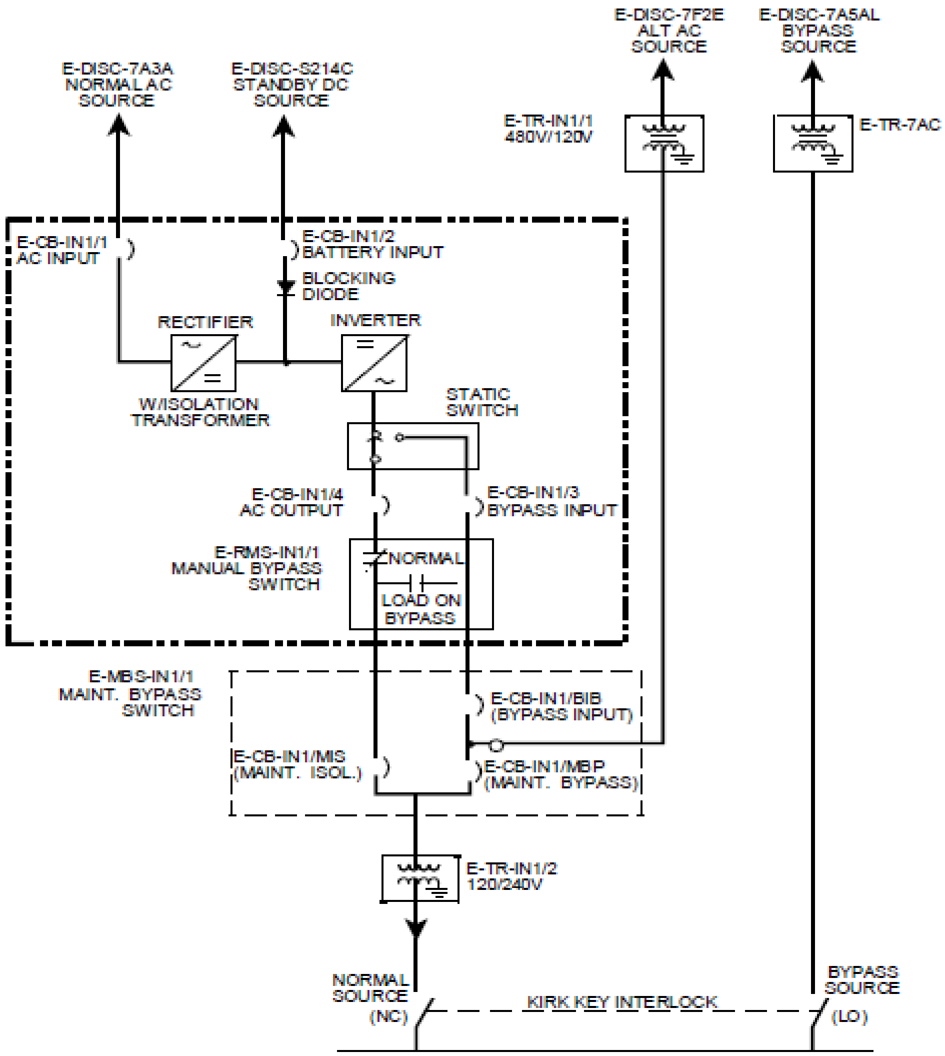


FIGURE 3: E-IN-1 ONE LINE SCHEMATIC

Comments / Reference: SD000194	Revision: 13 mr 0

2. The Kirk Key Bypass Source for E-PP-US is fed from critical E-MC-7A (cubicle 5AL) to the transformer E-TR-7AC at 480 VAC and stepped down to 120/240 VAC single phase (1Ø). This power supply terminates at the normally locked open incoming breaker at the bottom of E-PP-US. Use of the Bypass Source may require the plant to be off line for the "dead bus" transfer to or from this power source.

3. All branch circuits out of the E-PP-US in control room are fused for protection of the circuit and the UPS system.

CAUTION: DO NOT perform a Kirk-Key transfer while Refueling is in Progress, because the transfer will affect the refueling interlocks. CAUTION 5.3 SOP-ELEC-IN1-OPS

C. E-IN-1 (Figures 2-8)

1. The purpose of E-IN-1 is to supply uninterruptible, reliable power to Panel E-PP-US which in turn supplies critical plant loads such as the Rod Drive Control System, DEH, and the Process Computer. See section IX. POWER SUPPLIES for a more detailed listing.

   a) The inverter receives DC current from either the rectifier-charger by direct connection or the batteries via the Battery Input Circuit Breaker (E-CB-IN1/2). The inverter is a pulse-width-modulated type (PWM) employing Insulated Gate Bipolar Transistors (IGBTs), which provide continuous and uninterruptible AC power while operating from any DC source within its input operating range. LO-5896a NLO-12571c NLO-12568a,b

   b) The Normal source of power to the inverter is supplied from the 480 VAC MC-7A (cubicle 3A) to the Rectifier Charger. The Standby DC source of power to the inverter is supplied from the 250 VDC Bus S2-1 (cubicle 4C).

   c) The Alternate AC source is fed from Critical MC-7F (cubicle 2E) into transformer E-TR-IN1/T1, at 480 VAC, 3Ø, 60 Hz. where it is transformed down to 120 V, 1Ø, 60 Hz. and then fed to the Maintenance Bypass Switch E-CB-IN1/MBP. At the Maintenance Bypass Switch Alternate AC power is then normally fed to the Bypass Input circuit breaker (E-CB-IN1/3) and Static Switch via the Bypass Input Breaker (E-CB-IN1/BIB). The Alternate AC power from the Static Switch through circuit breaker E-CB-IN1/4 is then fed to E-PP-US via the Maintenance Isolation Breaker (E-CB-IN1/MIS). The Alternate AC source provides a stable power supply that can be automatically switched on line in event of inverter failure (see Static Switch Transfer Criteria below).

Examination Outline Cross-reference:	Level	RO		SRO
Rev. 0 Date: 12/16/2016	Tier	2		
	Group	1		
	K/A	263000.2.2.22		
Level of Difficulty: 3	Importance Rating	4.0		

D.C. Electrical Distribution: Knowledge of limiting conditions for operations and safety limits.

Question # 48

CGS is operating in Mode 1.

What is the minimum equipment required to be operable in the Division 1 125 vdc distribution system?

The corresponding control equipment and interconnecting cabling supplying power to DC bus S1-1,…

A. and battery B1-1 are required to be operable.

B. and one battery charger (C1-1A OR C1-1B) are required to be operable.

C. one battery charger (C1-1A OR C1-1B) and battery B1-1 are required to be operable.

D. both battery chargers (C1-1A AND C1-1B) and battery B1-1 are required to be operable.

Answer: C

K/A Match:

This question requires the candidate to demonstrate knowledge of LCO 3.8.4, DC Sources – Operating, by identifying the equipment necessary to make a DC distribution system operable. This information is contained in the TS bases.

SRO Only:

N/A

Explanation:

A. Incorrect. Plausible since one battery charger is sized to carry normal and post-accident DC loads without the assistance of the battery (see Ref. A, section IV.F.1). However per LCO 3.8.4, DC Sources – Operating, TS bases, one battery charger and one battery must be operable to make the DC subsystem operable. This LCO is applicable in Mode 3 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.

B. Incorrect. Plausible since one battery is capable of supplying power for at least two hours to its respective safe shutdown loads without the use of the associated battery charger. (see Ref. A, section IV.F.1). However per LCO 3.8.4, DC Sources – Operating, TS bases, one battery charger and one battery must be operable to make the DC subsystem operable. This LCO is applicable in Mode 3 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.

C. Correct. Per LCO 3.8.4, DC Sources – Operating, TS bases, one battery charger and one battery must be operable to make the DC subsystem operable. This LCO is applicable in Mode 3 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.

D. Incorrect. Plausible since for each division, one charger will be on line while the other will normally be available for service. Removing one battery charger from service would require an entry in the plant inoperability log. Per LCO 3.8.4, DC Sources – Operating, TS bases, ONLY one battery charger and one battery must be operable to make the DC subsystem operable. Although both battery chargers operable meets the requirements of LCO 3.8.4, the question asks for the minimum equipment required to meet technical specifications.

Technical Reference(s)		Attached w/ Revision # See Comments / Reference
A	SD000188, CGS System Description, Volume 1, Chapter 1, DC Distribution	
B	CGS Technical Specifications, LCO 3.8.4, DC Sources, Operating	
C	CGS Technical Specifications Bases, LCO 3.8.4, DC Sources Operating	

Proposed references to be provided during examination: None

Learning Objective:

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 41.5
55.43

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Comments / Reference: Ref. A, section IV.F.1, Battery Chargers and Battery data.	Revision: Major 10

COLUMBIA SYSTEMS
DC DISTRIBUTION

September 2014
SD000188, r10 mr0

3. The DC distribution panels at the TMU Pumphouse have provisions for a locally mounted ground detection relay, lights, and test pushbuttons.

F. Battery and Charger Data

1. Division 1 and 2 125 VDC (Figure 1, 14 and 14a) NLO-12390b

a. Each **Division 1 and 2 battery (B1-1, B1-2)** has 58 cells with a capacity of 1190 amp-hours and is capable of supplying power for at least two hours to their respective safe shutdown loads without the use of the associated battery charger.

b. Each **Division 1 and 2 charger (C1-1A, C1-1B, C1-2A, C1-2B)** is sized to carry normal and postaccident steady dc loads, while simultaneously recharging the battery from its discharged state (1.81-V per cell) to its fully charged state (charging current has stabilized) within 24 hr. The battery chargers receive 480-V ac input power from their respective Divisions 1 and 2 480-V MC-7A and 8A. For each division, one charger will be on line while the other will normally be available for service if the on-line charger fails or has to be removed from service. The chargers are rated 200 amps dc output. The old 125 V dc charger E-C1-1, (formally Division 1) is disconnected and spared in place for use as temporary power. LO-11840

Comments / Reference: Ref. B, T.S. 3.8.4, DC Sources – Operating, LCO and Applicability	Revision: Amendment 237

DC Sources - Operating
3.8.4

3.8 ELECTRICAL POWER SYSTEMS

3.8.4 DC Sources - Operating

LCO 3.8.4 The Division 1, Division 2, and Division 3 DC electrical power subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

Comments / Reference: Ref. C, T.S. Bases for LCO 3.8.4, DC Sources - Operating	Revision: 91

BASES

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in the FSAR, Chapter 6 (Ref. 7) and Chapter 15 (Ref. 8), assume that ESF systems are OPERABLE. The DC electrical power system provides normal and emergency DC electrical power for the DGs, 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 DC sources OPERABLE during accident conditions in the event of:

a. An assumed loss of all offsite AC power or of all onsite AC power; and

b. A worst case single failure.

The DC sources satisfy Criterion 3 of Reference 9.

LCO

The DC electrical power subsystems, each subsystem consisting of one battery, one battery charger, and the corresponding control equipment and interconnecting cabling supplying power to the associated bus within the divisions, 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. 4).

Examination Outline Cross-reference:	Level	RO		SRO
Rev. 0 Date: 12/16/2016	Tier	2		
	Group	1		
	K/A	263000.K5.01		
Level of Difficulty: 3	Importance Rating	2.6		

D.C. Electrical Distribution: Knowledge of the operational implications of the following concepts as they apply to D.C. ELECTRICAL DISTRIBUTION: Hydrogen generation during battery charging

Question # 49

CGS is operating in Mode 1. An equalizing battery charge is in progress on 250 vdc battery B2-1 when the following conditions are noted:

- Annunciator 826.P1.3-4, BATTERY ROOM FAN 53A DIFFERENTIAL PRESSURE LOW, is in alarm
- Battery Room No. 1 Exhaust Fan, WEA-FN-53A, is OFF.
- Field operators report that the breaker for WEA-FN-53A is tripped and will not reset.
- Battery Room No. 1 temperature is 82°F and steady.

What actions should the crew take?

The crew should secure the equalizing charge on battery B2-1 and…

A. install temporary ventilation in Battery Room No. 1 in accordance with ABN-HVAC, HVAC Trouble, to reduce hydrogen buildup.

B. remove battery B2-1 from service in accordance with SOP-ELEC-250VDC-SHUTDOWN, 250 vdc System Shutdown, to reduce hydrogen buildup.

C. install temporary ventilation in Battery Room No. 1 in accordance with ABN-HVAC, HVAC Trouble, to prevent battery electrolyte temperature from affecting battery performance.

D. remove battery B2-1 from service in accordance with SOP-ELEC-250VDC-SHUTDOWN, 250 vdc System Shutdown, to prevent battery electrolyte temperature from affecting battery performance.

Answer: A

K/A Match:
This questions requires the candidates to demonstrate understanding of the method used to prevent hydrogen buildup during a battery charge and the operational requirements if ventilation is lost.

SRO Only:
N/A

Explanation:

Per ABN-HVAC, if WMA-FN-53A is lost, portable fans are installed to direct ventilation to battery room 1.

A. Correct.

B. Incorrect. The battery should not be removed from service. Plausible because removing the battery from service would stop hydrogen generation.

C. Incorrect. Battery electrolyte temperature is not the concern. Plausible because temporary ventilation should be installed in battery room 1.

D. Incorrect. The battery should not be removed from service. Plausible because battery electrolyte temperatures are related to battery operation.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
CGS System Description Volume1, Chapter 1, DC Distribution	
SOP-ELEC-250V-OPS, 250 vdc System Operations	
ABN-HVAC, HVAC Trouble	

Proposed references to be provided during examination: None

Learning Objective: 15777 – With the procedures available, discuss all contingencies associated with the subsequent operator actions of ABN-HVAC.

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 41.5
55.43 ____

Comments / Reference:SOP-ELEC-250V-OPS	Revision: 2

Number: SOP-ELEC-250V-OPS	Use Category: CONTINUOUS	Major Rev: 002 Minor Rev: N/A Page: 3 of 9
Title: 250 VDC SYSTEM OPERATIONS		

1.0 PURPOSE

Provide instructions for operation of the 250 VDC Electrical System.

2.0 REFERENCES

2.1 245 VDC Found in E-DP-S2/1 During Live-Dead-Live Checks {AR 197253}

2.2 EWD-50E-0017

2.3 E503, Sh 6, Auxiliary One Line Diagram

2.4 E505-2, DC One Line Diagram

2.5 Technical Specification 3.8.2.1, DC Sources – Operating

2.6 Technical Specification 3.8.2.2, DC Sources – SHUTDOWN

2.7 SOP-ELEC-250V-START, 250 VDC System Start

2.8 SOP-ELEC-250V-SHUTDOWN, 250 VDC System Shutdown

2.9 SOP-ELEC-DC-LU, DC Electrical Distribution System Breaker Lineup

3.0 PREREQUISITES

3.1 **VERIFY** 250 VDC Distribution System Power Supply Checklist has been completed per SOP-ELEC-DC-LU, or as directed by CRS/Shift Manager. _____

3.2 **VERIFY** Battery Room HVAC in service. _____

Comments / Reference: ABN-HVAC	Revision: 13

- C237 (Batt Rm 1 Access from RPS Rm1) _____

NOTE: The following step installs auxiliary fans, which will promote cooling of vital island (These fans are staged underneath the stairs in Rm C214).

4.5.8 **INSTALL** portable fans to maximize cooling as follows:

- **INSTALL** a fan in doorway C221 with flow directed into room (Battery Charger Rm 1) (Rm C216) _____
- **INSTALL** a fan in doorway C218 with flow directed into room (Battery Rm 1) (Rm C239/C210) _____

4.5.9 IF required

Examination Outline Cross-reference:	Level	RO		SRO
Rev. 0 Date: 12/16/2016	Tier	2		
	Group	1		
	K/A	264000.K1.05		
Level of Difficulty: 2	Importance Rating	3.2		

Emergency Generators (Diesel/Jet): Knowledge of the physical connections and/or cause-effect relationships between EMERGENCY GENERATORS (DIESEL/JET) and the following: Emergency generator fuel oil supply system

Question # 50

How is diesel fuel delivered to the Emergency Diesel Generator DG-1 and DG-2 fuel injectors during diesel generator operation?

A. An engine-driven main supply pump with a DC-powered backup pump.

B. An AC-powered main supply pump with a DC-powered backup pump.

C. A DC-powered main supply pump with an engine-driven backup pump.

D. An engine-driven main supply pump with an AC-powered backup pump.

Answer: A

K/A Match:
The question determines whether a candidate understands what type of pump supplies fuel to the engine and how it is powered.

SRO Only:
N/A

Explanation:
Per SD000200, the fuel is supplied by an engine driven pump during normal operation with a DC powered backup pump available as a backup.

A. Correct.

B. Incorrect. The primary supply pump is NOT AC powered.

C. Incorrect. The primary supply pump is NOT DC powered.

D. Incorrect. The backup pump is NOT AC powered.

Distractors are plausible because the Fuel Oil Transfer Pumps are AC powered and distractors are variations of the possible combinations.

Technical Reference(s)		Attached w/ Revision # See Comments / Reference
SD000200, CGS System Description, Volume 7, Chapter 8, Diesel Generator		

Proposed references to be provided during examination: None

Learning Objective: 12403 – Explain the function and operation of the following components, including any interlocks or automatic features associated with them: (f) Fuel Oil Storage and Transfer System

Question Source: Bank # EO00573
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 41.2
55.43 ______

Comments / Reference: SD000200 (pages 12 and 13	Revision: 12 mr 1

4. Fuel Oil Pumps

The fuel oil supply from the day tanks to each diesel engine being served consists of two mutually redundant systems. Either system is capable of supplying fuel oil to the engine fuel header. Each system contains a fuel supply line, strainer, fuel oil pump, duplex filter, pressure gage, and relief and check valves. A fuel oil filter and strainer system is provided in each fuel line to eliminate passage of particles five microns or greater in size to the engine injectors. Separate fuel return lines from the relief valves to the day tanks are provided for each system on diesel generators 1A and 1B. The HPCS diesel utilizes a common return line to the day tank.

One of the fuel supply pumps is mechanically driven by the engine and is normally used during engine operation. The other supply pump is driven by a 125 volt DC motor and is used to fill the fuel oil system and fuel header prior to initial operation and after maintenance has been performed on system piping and components. The motor driven pump also automatically starts when the engine starts and delivers a slightly lower discharge pressure than the engine driven pump and is, therefore, available as a backup in the event the engine driven pump system fails.

Examination Outline Cross-reference:	Level	RO		SRO
Rev. 0 Date: 12/16/2016	Tier	2		
	Group	1		
	K/A	300000.K2.01		
Level of Difficulty: 3	Importance Rating	2.8		

Instrument Air System (IAS): Knowledge of electrical power supplies to the following: Instrument air compressor

Question # 51

What is the power supply to air compressor CAS-C-1B?

A. MC-7A

B. MC-8A

C. MC-6C

D. MC-6B

Answer: B

K/A Match:
The candidate must recall the power supply to the CAS compressors.

SRO Only:
N/A

Explanation:
CAS-C-1B is powered by MC-8A

A. Incorrect. CAS-C-1B is powered by MC-8A. Plausible because CAS-C-1A is powered by MC-7A.

B. Correct.

C. Incorrect. CAS-C-1B is powered by MC-8A. Plausible because service air compressor SA-C-1 is powered by MC-6C.

D. Incorrect. CAS-C-1B is powered by MC-8A. Plausible because service air dryer SA-DY-1 is powered by MC-6B.

Technical Reference(s)		Attached w/ Revision # See Comments / Reference
SD000205, CGS System Description, Volume 2, Chapter 11, Control and Service Air Cas System Description		

Proposed references to be provided during examination: None

Learning Objective: 5881 – Given a list of various plant systems, describe their interrelationship with the Control Air and Service Air systems. (a) AC Power Distribution

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 41.7
55.43

Comments / Reference: SD000205	Revision: 11 mr 1

IX. POWER SUPPLIES

Air Compressor 1A (CAS-C-1A)	MC-7A (critical)
Air Compressor 1B (CAS-C-1B)	MC-8A (critical)
Air Compressor 1C (CAS-C-1C)	MC-2P
Air Dryer (CAS-DY-3A & 4A)	PP-7AZ (critical)
Air Dryer (CAS-DY-3B & 4B)	PP-8AZ (critical)
Service Air Compressor (SA-C-1)	MC-6C
Service Air Dryer (SA-DY-1)	MC-6B
Service Air Bleed Air Blower	MC-6B
SA-PCV-2 Solenoid Pilot Valve	PP-1B-A
Vac Breaker Testing Air Supply CAS-V-453	PP-7A-A
Cooling Jacket Water Pump (CJW-P-1A)	PP-7E-A
Cooling Jacket Water Pump (CJW-P-1B)	MC-8A (critical)

Examination Outline Cross-reference:	Level	RO		SRO
Rev. 0 Date: 12/16/2016	Tier	2		
	Group	1		
	K/A	300000.K6.13		
Level of Difficulty: 3	Importance Rating	2.5		

Instrument Air System (IAS): Knowledge of the effect that a loss or malfunction of the following will have on the INSTRUMENT AIR SYSTEM: Valves

Question # 52

CGS is operating in Mode 1.

- During a valve lineup, field personnel erroneously operated the CN to CIA crosstie isolation valve, CIA-V-728, in the close direction.
- The error was detected prior to fully closing the valve and CIA-V-728 was restored to the full open position.
- The pressure transient on the Containment Instrument Air (CIA) header is denoted in the following table:

Time (minutes)	Pressure (psig)
0	185
1	150
2	100
3	90
4	145
5	185

What is the current status of the CIA system?

Inboard MSIVs are __(1)__ , ADS Header Supply valves, CIA-V-39A & B __(2)__ .

A. (1) closed
(2) remained open

B. (1) closed
(2) closed and reopened

C. (1) open
(2) remained open

D. (1) open
(2) closed and reopened

Answer: D

K/A Match:
This question requires the candidates to understand the effects on the instrument air system when the nitrogen supply valve is shut.

SRO Only:
N/A

Explanation:

A. Incorrect. Plausible since MSIVs will close when CIA pressure drops to 50-80 psig. However, CIA pressure only reached 90 psig. Additionally, ADS Header Supply valves, CIA-V-39A & B, will close if CIA pressure drops below 160 psig for 3 minutes, and then reopen when CIA header pressure is > 160 psig.

B. Incorrect. Plausible since the ADS Header Supply valves, CIA-V-39A & B, will close if CIA pressure drops below 160 psig for 3 minutes, and then reopen when CIA header pressure is > 160 psig. However, MSIVs will close when CIA pressure drops to 50-80 psig. CIA pressure only reached 90 psig and the MSIVs will remain open.

C. Incorrect. Plausible since MSIVs will remain open if CIA pressure remains above 50-80 psig. However, ADS Header Supply valves, CIA-V-39A & B, will close if CIA pressure drops below 160 psig for 3 minutes, and then reopen when CIA header pressure is > 160 psig.

D. Correct. MSIVs will remain open if CIA pressure remains above 50-80 psig. ADS Header Supply valves, CIA-V-39A & B, will close if CIA pressure drops below 160 psig for 3 minutes, and then reopen when CIA header pressure is > 160 psig.

Technical Reference(s)		Attached w/ Revision # See Comments / Reference
SD000156, CGS System Description, Vol. 8, Chap. 4, Containment Instrument Air (CIA)		
ABN-CIA, Containment Instrument Air System Failure		

Proposed references to be provided during examination: None

Learning Objective: 11755 – Describe the function, purpose and design features of the following Containment Instrument Air System: (e) Nitrogen Bank Bottles.

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 41.7
55.43 ____

Comments / Reference: CGS System Description, CIA	Revision: 11

COLUMBIA SYSTEMS
CONTAINMENT INSTRUMENT AIR

November 2014
SD000156, r11 mr0

V. CONTROL THEORY AND INTERLOCKS

A. Control Room Controls

1. Control Room Solenoid actuated air operated valves CIA-V-39A [39B] (H13-P840 [H13-P820]), 3 position spring return to AUTO:

CLOSE	Valve Closes
AUTO	Valve closes LE 160 psig (CIA-PS-39A [B]) for 3 minutes. Re-opens when GT 160 psig.
OPEN	Valve Opens

- Indication of CIA-V-39A (B) not open

Should pressure on the non-safety related header upstream of CIA-V-39A [B] be restored to GT 160 psig, the valves will automatically reopen providing a supply source to the safety related header. The initiation signal to the programmers will clear, however any bottles previously aligned by the programmer stay aligned until the programmer is manually reset.

Abnormal

1. Loss of CIA (ABN-CIA) NLO-12379

   A decrease in pressure of the CIA Main Header Pressure as indicated on CIA–PI-20 or locally on CIA–PI-4 (RB 522) indicates a leak in the system is apparent. This indication is detrimental to system in the following ways:

   a) Low ADS Header Pressure

      Low ADS header pressure could be caused by:

      - Mispositioning and/or auto-closure of CN to CIA crosstie isolation valve CIA-V-728 or auto-closure of CIA-PCV-39A (B).
      - Improper adjustment of CN to CIA regulator CN-PCV-10, set point 186 psig (located outside by CN tank) or low Containment Nitrogen Inerting System tank level. If tank level is too low there is an insufficient source of liquid nitrogen to maintain pressure.
      - Improper operation of backup nitrogen bottle station programmers, if they did not function to restore pressure as designed to place adequate bottles in-service.
      - Exhausted backup bottles, by providing insufficient supply
      - CIA-V-5A (B) lifts or sticks open

   b) Low MSIV/SRV Header Pressure NLO-12380a

      Low MSIV/SRV header pressure may be expected for short periods of time during high system (CN) flow rates. If a low-pressure condition might cause MSIV closure, personnel can prevent their closure by transferring header supply from CN to CAS. Normal header pressure is slightly greater than 100 psig. MSIV closure occurs at approximately 80 psig nitrogen header pressure due to four-way pilot valve shifting.

Comments / Reference: ABN-CIA	Revision: Major 006 Minor 004

Number: ABN-CIA	Use Category: CONTINUOUS	Major Rev: 006 Minor Rev: 004 Page: 3 of 8
Title: Containment Instrument Air System Failure		

1.0 ENTRY CONDITIONS

Inability of the CIA system to maintain designed pressure (180-190 psig).

2.0 AUTOMATIC ACTIONS

2.1 CIA-V-39A(B) closes on low header pressure of 160 psig (following a 3 minute time delay).

2.2 CIA-PROG-1A(B) begins sequencing on backup nitrogen bottles if two of the following three conditions are satisfied:

- ADS Header A(B) LT 160 psig.
- ADS Header A(B) LT 156 psig.
- CIA-V-39A(B) closed.

Number: ABN-CIA	Use Category: CONTINUOUS	Major Rev: 006 Minor Rev: 004 Page: 4 of 8
Title: Containment Instrument Air System Failure		

4.0 SUBSEQUENT OPERATOR ACTIONS

CAUTION

INBD MSIV closure will occur at approximately 50-80 psig.

Examination Outline Cross-reference:	Level	RO		SRO
Rev. 0 Date: 12/16/2016	Tier	2		
	Group	1		
	K/A	400000.K3.01		
Level of Difficulty: 2	Importance Rating	2.9		

Component Cooling Water System (CCWS): Knowledge of the effect that a loss or malfunction of the CCWS will have on the following: Loads cooled by CCWS

Question # 53

Columbia is operating in MODE 1.

- RCC-P-1C, Reactor Closed Cooling Water Pump 'C', is tagged out for maintenance.
- RCC-P-1B, Reactor Closed Cooling Water Pump 'B', breaker trips due to a fault.

How will the plant respond?

(1) RCC-V-6, Radwaste/Rx Bldg Supply Valve, will...
(2) RWCU Nonregenerative Heat Exchanger Outlet Temperature will...

A. (1) remain open.
(2) rise.

B. (1) shut.
(2) rise.

C. (1) remain open.
(2) remain unchanged.

D. (1) shut.
(2) remain unchanged.

Answer: B

K/A Match:
Question presents a malfunction in the RCC (CCWS) system and determines if candidate can properly evaluate the expected response on loads in the RCC system.

SRO Only:
N/A

Explanation:
With only one RCC pump running, RCC-V-6 automatically shuts to prioritize providing cooling to drywell loads. RWCU Non-Regenerative HTXR is not a drywell load so temperatures are expected to rise.

A. Incorrect. RCC-V-6 will not remain open with two RCC pumps secured. Plausible because RWCU Non-Regenerative Heat Exchanger Outlet Temperatures will rise.

B. Correct.

C. Incorrect. RCC-V-6 will not remain open and RWCU Non-Regenerative Heat Exchanger Outlet Temperature will rise. Plausible based on understanding of which loads are isolated by RCC-V-6.

D. Incorrect. RWCU Non-Regenerative Heat Exchanger Outlet temperature will rise. Plausible based on understanding of which loads are isolated by RCC-V-6.

Technical Reference(s)		Attached w/ Revision # See Comments / Reference
SD000196, CGS System Description, Volume 3, Chapter 1, Reactor Closed Cooling Water		

Proposed references to be provided during examination: None

Learning Objective: 5705 – State the purpose of the following components: (c) Reactor/Radwaste Building Isolation Valve RCC-V-6

Question Source: Bank # LO03470
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 41.7
55.43 ______

Comments / Reference: SD000196	Revision: 14 mr 1

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RCC SD000196, r14 mr1

b) RADWASTE/RX BLDG SUPPLY RCC-V-6 Three-position switch: CLOSE, NOR, OPEN, Spring return to NOR NLO-12203h

CLOSE - the valve closes

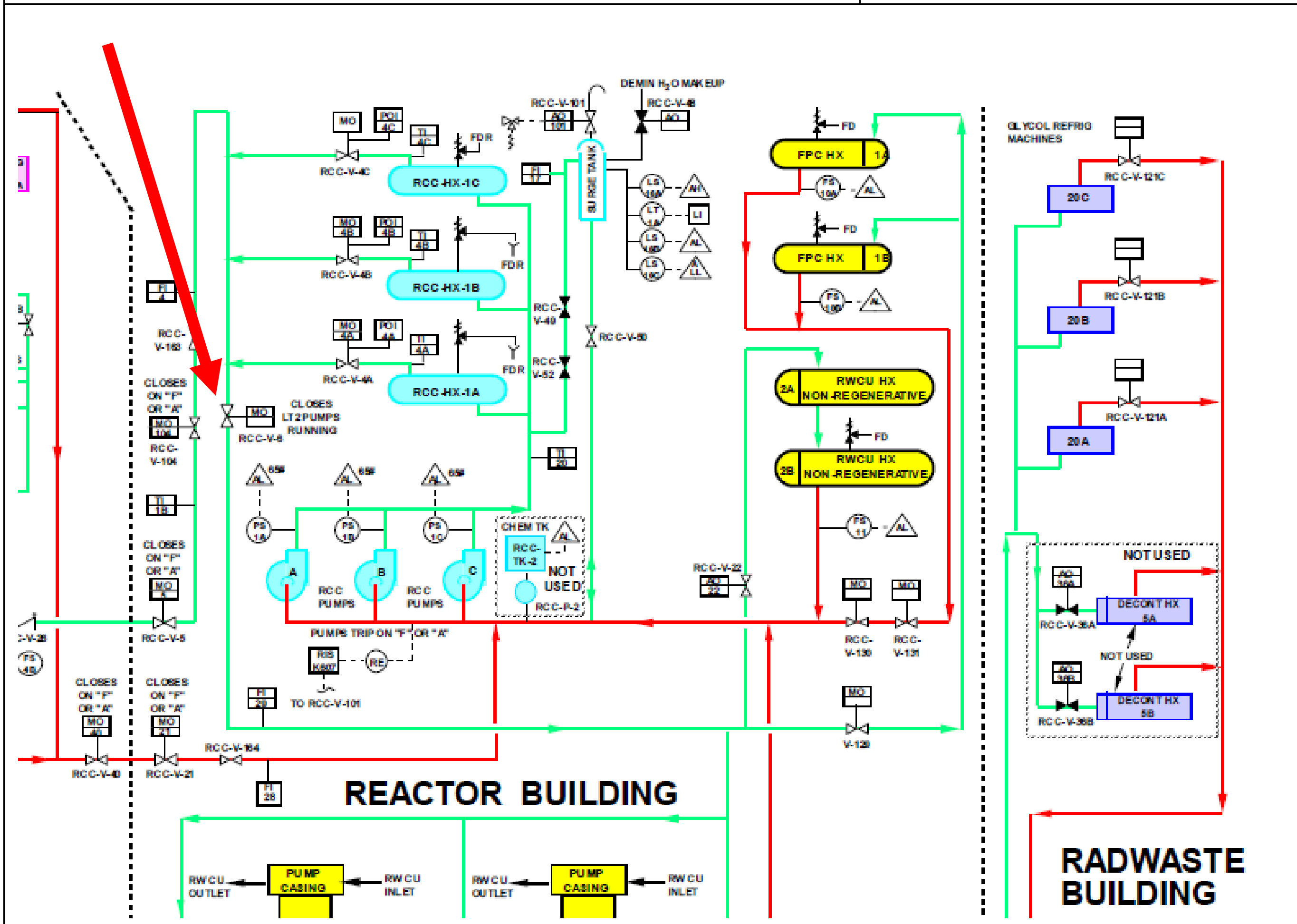
NOR - the valve closes automatically if less than two pumps are on line as sensed by breaker position (plus a 10 second time delay).

OPEN - the valve opens if at least two pumps are on line as sensed by breaker position

This valve closes if any of the following actions are done:

- Removing the control power fuses from RCC-P-1A or from valve RCC-V-6 itself
- Closing the RCC-V-6 breaker

Comments / Reference: SD000196	Revision: 14 mr 1



Examination Outline Cross-reference:	Level	RO		SRO
Rev. 0 Date: 12/16/2016	Tier	2		
	Group	2		
	K/A	201002.A1.03		
Level of Difficulty: 4	Importance Rating	3.0		

Reactor Manual Control System: Ability to predict and/or monitor changes in parameters associated with operating the REACTOR MANUAL CONTROL SYSTEM controls including: Rod movement sequence lights.

Question # 54

Given the following:

- A control rod is inserted using the Continuous Insert Pushbutton.
- The ROD DRIFT annunciator on H13-P603 alarms after releasing the Continuous Insert Pushbutton.
- The red DRIFT light associated with the selected control rod is lit.

Which of the following caused these indications?

A. RMCS has detected a reed switch that failed to close at some point during the requested control rod motion.

B. Control Rod Drive Hydraulic Cooling Water flow is set too low allowing the selected control rod to insert past the requested position.

C. Control Rod Drive Hydraulic Drive water d/p is set too high causing the control rod to settle after the RMCS rod motion timer has timed out.

D. The Continuous Insert Pushbutton bypasses the settle function and an odd reed switch was made up before the rod settled.

Answer: D

K/A Match:

Question presents a situation where rods were inserted and provides indications. The question determines if the candidate understands the associated indications (ability to monitor).

SRO Only:

N/A

Explanation:

The rod settle function is bypassed when using the Continuous Insert Pushbutton. If the pushbutton is released when the rod is near an odd reed switch, the drift light will be activated.

A. Incorrect. A reed switch position failure would not result in a rod drift alarm. Plausible because a rod drift alarm is caused by a rod passing an odd reed switch when the rod is not in the driving cycle.

B. Incorrect. CRD flow being too low does not cause a rod drift condition. Plausible because CRD flow being too high will cause a rod drift condition.

C. Incorrect. High water D/P does not cause a rod drift condition. Plausible since CRD water is kept at a higher pressure then RPV pressure.

D. Correct.

Technical Reference(s)		Attached w/ Revision # See Comments / Reference
PPM 4.603.A7, Annunciator Response		
SD000148, CGS System Description, Volume 5, Chapter 6, Reactor Manual Control		

Proposed references to be provided during examination: None

Learning Objective: 5792 – State the functions and interrelationships of these P603 controls: (a) Insert Pushbutton (b) Continuous Insert Pushbutton

Question Source: Bank # LO02183
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 41.5
55.43 ______

Comments / Reference: 4.603.A7	Revision: 51

Number: 4.603.A7	Use Category: CONTINUOUS	Major Rev: 051 Minor Rev: N/A Page: 55 of 71
Title: 603.A7 Annunciator Panel Alarms		

5-7 ROD DRIFT

5-7 WINDOW	SOURCE	AUTOMATIC ACTIONS
ROD DRIFT	Odd Reed Switch closed and no ROD DRIVING signal present	• Withdrawal block if withdrawal limit exceeded and Reactor power is below the RWM LPSP. • Insert block if insert limit is exceeded and Reactor power is below the RWM LPSP.

NOTE: The rod drift annunciator may alarm during control rod manipulations due to the rod settling to the next notch. If this happens, verify the correct control rod position, but DO NOT enter ABN-ROD, Control Rod Faults.

NOTE: Transponder card failures may result in an inward drift of the associated control rod.

1. **VERIFY** correct control rod position.
2. IF a control rod drift is verified,
   THEN **REFER** to ABN-ROD, Control Rod Faults.
3. **RESET** the alarm.

Comments / Reference: SD000148 pages 11 and 12	Revision: 14 mr 1

V. CONTROL THEORY AND INTERLOCKS

A. RMCS Controls Located on P603 (Figures 1, 1A, 2, 2A)

1. Rod motion controls - momentary contact pushbuttons

a) INSERT Button LO-5792a

Momentary depression of this button will cause the selected rod to move in one notch with the settle function. ("Momentary" depression should be made until the INSERT indicator has illuminated in order to comply with the recommendations of GE SIL 626.) Continuous depression of this button will cause the rod to move in until the button

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is released or some other motion terminating event such as a rod insertion block occurs at which time the rod will complete the settle function.

b) CONTINUOUS INSERT Button (Figure 9) LO-5792b

This button does the same thing as the Insert button but bypasses the activity control timer such that the "insert valves", valves 121 and 123 will open immediately when the button is depressed and close immediately when the button is released. There is no rod settle function when using this button. Depression of this button also causes the timer to reset. The primary purpose of this button is to provide the reactor operator with a means of driving in control rods even in the presence of a faulty timer.

c) WITHDRAW Button (Figure 10) LO-5792c

Comments / Reference: SD000148 page 17	Revision: 14 mr 1
5. **DRIFT** (Red) If an odd position indicating switch is picked up and the rod is not in a driving cycle, the drift light will be activated. This could be indicative of excessive CRDH cooling water flow, or a rod that has scrammed, or rods have been moved using the Continuous In button and are now settling past an odd reed switch. Drifting control rods are also annotated on the RWM touchscreen. A drifting control rod is annunciated on Panel A603-A7. Depression of the Rod Drift Test button while a control rod is being moved simulates a "no command for rod motion" condition, triggering a rod drift alarm when the control rod passes an odd position indicating switch. 6. **XX-YY** (Rod coordinates) (White)	LO-5798e LO-5793 LO-7750 LO-5798f

Examination Outline Cross-reference:	Level	RO		SRO
Rev. 0 Date: 12/16/2016	Tier	2		
	Group	2		
	K/A	201003.K5.08		
Level of Difficulty:	Importance Rating	3.1		

Control Rod and Drive Mechanism: Knowledge of the operational implications of the following concepts as they apply to CONTROL ROD AND DRIVE MECHANISM: How control rods affect shutdown margin.

Question # 55

CGS is operating in Mode 1.

- A manual reactor scram has been initiated due to a casualty.
- The reactor mode switch is in shutdown.
- CRO1 has completed the scram report.

Which of the following combinations would NOT require a transition to PPM 5.1.2, RPV Control ATWS, by ensuring that there is sufficient shutdown margin to assure the reactor is shutdown under all conditions?

1. one control rod at position 48
2. one control rod at position 08
3. two control rods at position 04
4. two control rods at position 02
5. all other Control Rods at position 04
6. all other Control Rods at position 02
7. all other Control Rods at position 00

A. 1 and 2 and 7

B. 2 and 3 and 6

C. 1 and 4 and 5

D. 2 and 4 and 7

Answer: D

K/A Match:
Question presents a situation where not all rods are inserted fully following a SCRAM and determines if candidate can identify which conditions would result in entry into the ATWS EOP (due to inadequate shutdown margin).

SRO Only:
N/A

Explanation:

A. Incorrect. Plausible since one control rod is at position 2. However, two control rods are greater than position 2 and Maximum Subcritical Bank Withdrawal Position (MSBWP) is not met.

B. Incorrect. Plausible since most control rods are at position 2. However, three two control rods are greater than position 2 and Maximum Subcritical Bank Withdrawal Position (MSBWP) is not met.

C. Incorrect. Plausible since one control rod is full out. However, multiple rods are greater than position 2 and Maximum Subcritical Bank Withdrawal Position (MSBWP) is not met.

D. Correct. This meets the criteria for assuring the reactor is shutdown with control rod position alone: *"Maximum Subcritical Banked Withdrawal Position (MSBWP): No more than one control rod greater than Notch 02 and all other rods at Notch 02 or less".*

Technical Reference(s)		Attached w/ Revision # See Comments / Reference
PPM 5.0.10, Flowchart Training Manual		

Proposed references to be provided during examination: None

Learning Objective: 8182: Given a list, identify the criteria that must be met to ensure that the reactor is shutdown with no boron injected.

Question Source: Bank # LO01786
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 41.5
55.43 ______

Comments / Reference: PPM 5.0.10, Flowchart Training Manual	Revision: Major 021 Minor 001

3) Actions to defeat the automatic ADS function through one-time performance of a single operation precludes requiring an operator to repeatedly reset the ADS timer while concurrently carrying out other actions specified in the EOP flowcharts.

4) Action to take manual control of HPCS is directed to ensure that the operation of HPCS is controlled so that the system will only be used for RPV level control with Table 5 systems if SLC pump(s) is operating, reactor power is above 2% and reactor power is not increasing due to the HPCS injection.

c. Override RC-2:

1) This override is applicable to all subsequent steps of PPM 5.1.1.

2) When the condition which required entry to PPM 5.1.2 no longer exists (i.e., the reactor will remain shutdown on control rod pattern alone for all conditions of boron concentration and RPV water temperature), it is appropriate to terminate boron injection and return to PPM 5.1.1, RPV Control, in order to continue control of RPV parameters. Entry to PPM 5.1.1 at Transition Point 1 is specified (rather than direct entry to the RPV water level control flowpath at Step L-1) to simplify transfer of parameter control between the RPV Control flowcharts.

3) Boron injection is initiated under the reactor power control flowpath. Its termination is specified in this override to ensure boron injection is stopped when the transition to PPM 5.1.1 is made.

4) Positive confirmation that the reactor will remain shutdown under all conditions is best obtained by determining that all control rods are full in. Green full-in lights on the full core display, RSCS, GDS, plant process computer, and RWM display on panel H13-P603 provide indication that all rods are full in. It is appropriate to continue normal RPV control guidance for this condition.

5) Criteria other than all rods full in may be used to determine that the existing control rod position alone will always assure reactor shutdown. These include:

- Core design basis shutdown margin with a single control rod full out and all other control rods full in.
- Maximum Subcritical Banked Withdrawal Position (MSBWP): No more than one control rod greater than Notch 02 and all other rods at Notch 02 or less.
- Reactor engineer evaluation of shutdown margin.

Number: 5.0.10	Use Category: INFORMATION	Major Rev: 021 Minor Rev: 001 Page: 99 of 320
Title: Flowchart Training Manual		

3) Positive confirmation that the reactor will remain shutdown under all conditions is best obtained by determining that all control rods are full in. Green full-in lights on the full core display, GDS, plant process computer, and RWM display on panel H13-P603 provide indication that all rods are full in. It is appropriate to continue non-failure-to-scram RPV control guidance for this condition.

4) Criteria other than all rods full in may be used to determine that the existing control rod position alone will always assure reactor shutdown. These include:

- Core design basis shutdown margin with a single control rod full out and all other control rods full in.

- Maximum Subcritical Banked Withdrawal Position (MSBWP): No more than one control rod greater than Notch 02 and all other rods at Notch 02 or less.

- Reactor engineer evaluation of shutdown margin.

Examination Outline Cross-reference:	Level	RO		SRO
Rev. 0 Date: 12/16/2016	Tier	2		
	Group	2		
	K/A	202002.K6.05		
Level of Difficulty: 3	Importance Rating	3.1		

Recirculation Flow Control System: Knowledge of the effect that a loss or malfunction of the following will have on the RECIRCULATION FLOW CONTROL SYSTEM: Reactor water level

Question # 56

CGS is operating in Mode 1 with both Reactor Recirculation (RRC) pumps at 60 Hz when the following alarm was received:

- P602.A6.5-1, LOOP A ASD CHANNEL FAILURE LIMIT

Approximately 1 minute later, Reactor Feed Pump Turbine (RFPT) 1A tripped. Plant conditions 1 minute after the turbine trip include:

- Annunciator P603.A8.3-7, REACTOR PRESSURE VESSEL LEVEL HIGH/LOW ALERT, in alarm.
- RPV level: 30.5 inches up slow.
- Lowest RPV level reached: 26.0 inches.

What is the current operating frequency for both RRC pumps?

A. RRC-P-1A: 15 Hz
RRC-P-1B: 15 Hz

B. RRC-P-1A: 30 Hz
RRC-P-1B: 30 Hz

C. RRC-P-1A: 51 Hz
RRC-P-1B: 15 Hz

D. RRC-P-1A: 51 Hz
RRC-P-1B: 30 Hz

Answer: B

K/A Match:

The question requires the candidate to understand the loss of actual reactor water level affects the Reactor Recirculation (RRC) system. A malfunction of a single RPV level channel would not affect the RRC system since the output of the failed channel would be substituted with a functioning channel's output automatically.

SRO Only:

N/A

Explanation:

A. Incorrect. Plausible since the Low RPV level runback (-13 inches, Ref. A, section V.C.1.e.(3)) will reduce both RRC pumps to 15 Hz. This could occur during a reactor scram. Information in the stem indicates that a reactor scram did not occur since RPV level on a reactor scram goes below 0 inches prior to recovery.

B. Correct. A Reactor Feed Pump Turbine (RFPT) trip along with RPV level below the Low Level Alert setpoint (31.5 inches, Ref. C) will cause both Reactor Recirculation (RRC) pumps to runback to 30 Hz (see Ref. A, section V.C.1.e.(2)). Although RRC-P-1A was operating in Manual at a reduced frequency of 51Hz due to an earlier loss of ASD Channel runback, the pump will runback to 30 Hz when the Loss of RFPT runback initiates.

C. Incorrect. Plausible since RRC-P-1A did initially runback to 51Hz due to the loss of an ASD channel (Ref. B). See distractor A plausibility statement above for further information. However, the RFPT runback will reduce BOTH RRC pump operating frequencies to 30 Hz (see Ref. A, section V.C.1.e.(2)).

D. Incorrect. Plausible since RRC-P-1A did initially runback to 51Hz due to the loss of an ASD channel (Ref. B). However, the RFPT runback will reduce BOTH RRC pump operating frequencies to 30 Hz (see Ref. A, section V.C.1.e.(2)).

Technical Reference(s)	
SD000184, CGS System Description, Volume 5, Chapter 5, Reactor Recirculation Flow Control (ASD)	Attached w/ Revision # See Comments / Reference
PPM 4.602.A6, Annunciator Panel Alarms, Window 5-1, LOOP A ASD CHANNEL FAILURE LIMIT	
PPM 4.603.A8, Annunciator Panel Alarms, Window 3-7, REACTOR PRESSURE VESSEL LEVEL HIGH/LOW ALERT	

Proposed references to be provided during examination: None

Learning Objective: 5022 – Describe the physical and/or cause-and-effect relationship between the RRC system and the following: (k) Reactor water level

Question Source: Bank # ____

Modified Bank # ____ (Note changes or attach parent)

New X

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 41.7
55.43 ______

Comments / Reference: SD000184, Section V.C.1, Runbacks	Revision: Major 19, Minor 001

C. Interlocks

1. ASD Automatic Runbacks

a) For each of the runback conditions listed below, the runback will seal in for the completion of the runback but will automatically clear when the initiating condition has cleared and the runback limit frequency has been reached. That is, once a runback has been initiated, it will always complete regardless of the duration of the initiating condition. LO-9682 LO-9683 LO-9684 LO-9686 NLO-12356

b) Rate of frequency decrease for all runbacks is **3 Hz/sec**.

c) ASD operation will be limited to the runback frequency limit as long as the runback condition exists.

d) ASD frequency will not change if operating at or below the runback frequency limit when the runback occurs.

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ASD SD000184, r19 mr1

e) RUNBACKS:

(1) Loss of a single ASD channel. Runback (of the affected pump only) to 51 Hz. LO-9682

Note: Only the RRC Loop operating on a single channel is affected. For Example, if both A and B are operating at 60 Hz in AUTO and the A2 Channel fails, the "A" Control Station will shift to MANUAL and "A" ASD will runback to 51 Hz. "B" will continue to operate at 60 Hz in AUTO.

(2) Reactor Feed Pump Turbine Trip with RPV Low Level Alarm. Runback of both ASDs to 30 Hz. LO-9683

At least one RFPT Tripped and RPV Level 4 (31.5")

(3) RPV Low Level. Runback of both ASDs to 15 Hz. LO-9684

RPV Level 3 (13")

Comments / Reference: PPM 4.602.A6, Annunciator Panel Alarms, Window 5-1, LOOP A ASD CHANNEL FAILURE LIMIT	Revision: 32

Number: 4.602.A6	Use Category: CONTINUOUS	Major Rev: 032 Minor Rev: N/A Page: 41 of 55
Title: 602.A6 Annunciator Panel Alarms		

5-1 LOOP A ASD CHANNEL FAILURE LIMIT

5-1 WINDOW	SOURCE	AUTOMATIC ACTIONS
LOOP A ASD CHANNEL FAILURE LIMIT	RRC-RLY-K9A	• If RRC A pump is GT 51 Hz, pump speed drops to 51 Hz. • RRC A speed limited to maximum of 51 Hz • RRC-M/A-676A transfers to manual

> NOTE: This is an expected alarm when a channel is tripped or shutdown.

1. **VERIFY** RRC-MA-R676A in **MANUAL**.
2. **VERIFY** associated ASD Loop Channel System Ready lamp and Run lamp extinguished.
3. **VERIFY** associated ASD Loop Channel System Stop lamp illuminated.
4. IF RRC-P-1A was GT 51 Hz,
   THEN **CHECK** RRC-P-1A speed has dropped to approximately 51 Hz.
5. **BALANCE** RRC loop flows as necessary.
6. **CHECK** ASD Video Display Unit for source of alarm.
7. IF RRC-P-1A was operating GT 51 Hz,
   THEN **VERIFY** RRC-MA-R676A reference demand has reset to approximately 51 Hz.

> NOTE: A corresponding indication of approximately 5600 Drive horsepower is achieved at approximately 51 Hz (50-51Hz), as indicated by frequency demand.

8. **COMPARE** local operating Drive horsepower to frequency (ASD Building)
9. **INITIATE** a Work Request.
10. WHEN the cause of the trip has been corrected,
    THEN **REFER** to SOP-RRC-ASD, Reactor Recirculation ASD Operation to restart the drive.

REFERENCES: EWD-49E-051
EWD-3E-088

Comments / Reference: PPM 4.603.A8, Annunciator Panel Alarms, Window 3-7, REACTOR PRESSURE VESSEL LEVEL HIGH/LOW ALERT	Revision: 36

Number: 4.603.A8	Use Category: CONTINUOUS	Major Rev: 036 Minor Rev: N/A Page: 34 of 65
Title: 603.A8 Annunciator Panel Alarms		

3-7 REACTOR PRESSURE VESSEL LEVEL HIGH/LOW ALERT

3-7 WINDOW	SOURCE	AUTOMATIC ACTIONS
RPV LEVEL HIGH/LOW ALERT	High/low Level Alarm Module (High 40.5") RFW-CRM-L108 (Low 31.5") OR RFW-CRM-L208 OR RFW-CRM-L308 OR RFW-CRM-L408	None

NOTE: Operating a single feedwater pump while below the low level alert point causes an RRC pump runback if RRC pumps are running at GT 30 Hz (~54% flow).

1. **CONFIRM** the alarm by checking RPV Level Indicators RFW-LI-606A, B, and C, Reactor Level.

2. IF a high level alarm is indicated,
   AND level continues to trend upward,
   THEN **REFER** to ABN-LEVEL, Reactor Vessel High Water Level.

3. IF a low water level alarm is indicated,
   AND level continues to trend downward,
   THEN take actions necessary to **RESTORE** RPV Level to normal.

4. **DETERMINE** and **CORRECT** the cause of the loss of proper RPV level control.

5. IF Reactor power starts to oscillate,
   THEN **REFER** to ABN-CORE, Unplanned Core Operating Conditions.

REFERENCES: EWD-72I-028
EWD-72I-029
EWD-72I-030

Examination Outline Cross-reference:	Level	RO		SRO
Rev. 0 Date: 12/16/2016	Tier	2		
	Group	2		
	K/A	204000.K4.02		
Level of Difficulty: 3	Importance Rating	2.7		

Reactor Water Cleanup System: Knowledge of REACTOR WATER CLEANUP SYSTEM design feature(s) and/or interlocks which provide for the following: Piping over-pressurization protection

Question # 57

During a plant shutdown and cooldown per PPM 3.2.1, under which of the following operating conditions may RWCU-V-31, Blowdown Orifice Bypass Valve FIRST be opened?

A. When MODE 4 is entered.

B. When RCS coolant temperature is 189°F.

C. At any reactor pressure when using RWCU-MOV-34 (Discharge to Main Condenser)

D. When RPV pressure is LT 125 psig.

Answer: D

K/A Match:
Question ensures candidate understands the design limitation of the RWCU system with regard to RWCU-V-31 to prevent piping over pressurization.

SRO Only:
N/A

Explanation:
Per PPM 3.1.2 and the system description, RWCU-V-31 should not be open above 125psig RPV pressure to avoid over pressurization.

A. Incorrect. RPV pressure must be below 125psig. Plausible because MODE 4 is a shutdown condition where temperature is below 200F.

B. Incorrect. Plausible because high temperatures in the RWCU system could result in damage to piping and valves associated with the filter-demineralizers.

C. Incorrect. Plausible because RWCU-V-34 is in the same flowpath as RWCU-V-31 and would impact flow rates and discharge location.

D. Correct.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
PPM 3.1.2, Normal Plant Startup	
SD000190, CGS System Description, Volume 4, Chapter 4, Reactor Water Cleanup	

Proposed references to be provided during examination: None

Learning Objective: 5034 – State the function of the following components: (c) Orifice Bypass Valve (RWCU-V-31)

Question Source: Bank # LO03309
Modified Bank # ____ (Note changes or attach parent)
New ____

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ____

10 CFR Part 55 Content: 55.41 41.7
55.43 ____

Comments / Reference: PPM 3.1.2	Revision: 81 mr 2

Number: 3.1.2	Use Category: CONTINUOUS	Major Rev: 081 Minor Rev: 002 Page: 20 of 27
Title: Reactor Plant Startup		

5.0 PROCEDURE BASIS

5.1 Level Control Basis

L2 Short Cycle cleanup lines up Condensate to the RFW-V10A and RFW-V-10B.

L3 Normal level control at this point is CRD makeup with RWCU blowdown.

L8 Condensate Booster Pump prestart checks are manpower intensive. If time allows and manpower is available, then prestart checks should be performed on all three Condensate Booster Pumps.

L10 Reactor Feed Pump prestart checks are manpower intensive. If time allows and manpower is available, then prestart checks should be performed on both Reactor Feed Pumps.

During reactor start up, the plant is highly vulnerable to scram from a single RFW pump trip. Therefore, the second RFW pump is being started earlier in the startup to minimize single point vulnerability.

L13 Until the RPV begins to boil off inventory Level control with the RFW-LIC-620 will not be stable. If excessive level fluctuations occur in **AUTO** (A) then consider returning RFW-LIC-620 to **MANUAL** (M)

L14a RWCU-V-31, Orifice Bypass valve, shall not be open with Reactor pressure GT 125 psig, to prevent over pressurization of the RWCU blow down piping.

L15 Prior to initiating warming of the RFT's condenser vacuum has to be established.

L25 This requires Single Element feed flow control and a feed flow signal manually inserted into the process computer.

Comments / Reference: SD000190 (page 5)	Revision: 14 mr 1
The restricting orifice bypass valve, RWCU-V-31, is opened to establish reasonable dump rates from the RPV during refueling when coolant temperature is low. Using the orifice bypass valve is not allowed when RPV pressure is above 125 psig. 4. Throttleable Valves	LO 5034c

Examination Outline Cross-reference:	Level	RO		SRO
Rev. 0 Date: 12/16/2016	Tier	2		
	Group	2		
	K/A	215001.A2.01		
Level of Difficulty: 3	Importance Rating	2.7		

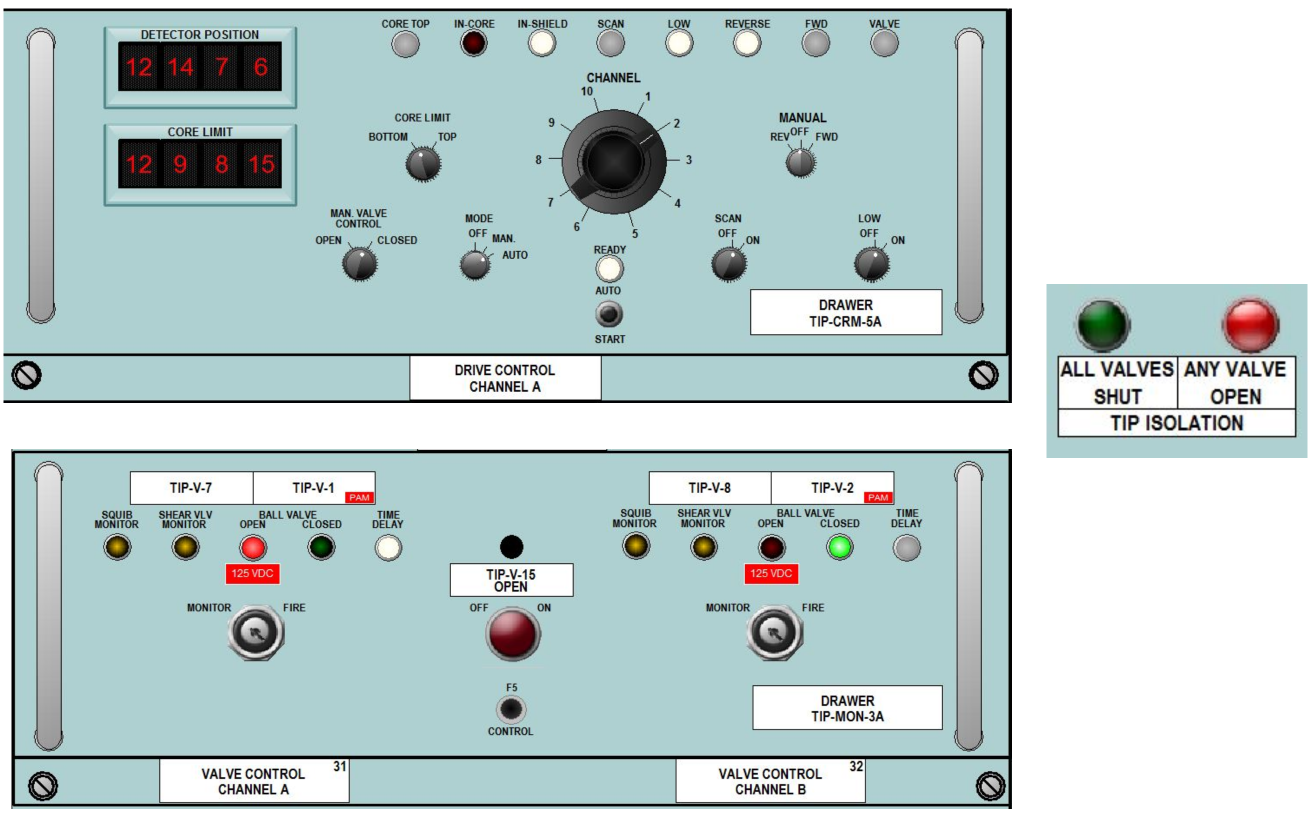
Traversing In-Core Probe: Ability to (a) predict the impacts of the following on the TRAVERSING IN-CORE PROBE ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Low reactor water level: Mark-I&II(Not-BWR1)

Question # 58

CGS is operating in Mode 1 with a Traversing In-Core Probe (TIP) trace is in progress.

- A transient occurs.
- RPV level: -65 inches, up slow.
- Drywell pressure: 1.45 psig, down slow.
- Highest drywell pressure: 1.55 psig.

Current condition of the TIP system:



How should the crew respond to these indications?

A. Place the 'MODE" switch in 'MAN' and retract the TIP probe in manual mode. TIP ball valve, TIP-V-1, will automatically close.

B. Press the 'AUTO START' pushbutton to automatically withdraw the TIP probe and take the Shear Valve Control keyswitch on Valve Control Channel A to 'Fire'.

C. Verify that the TIP probe automatically retracted and take the Shear Valve Control keyswitch on Valve Control Channel A to ‘Fire’.

D. Use the manual hand crank to retract the TIP probe. TIP ball valve, TIP-V-1, will automatically close.

Answer: C	

K/A Match:

This question requires the candidate to demonstrate understanding of the expected response of the Traversing In-Core Probe system (TIPS) to a low RPV level and the actions required if TIPS does not respond as expected.

SRO Only:

N/A

Explanation:

A. Incorrect. Plausible since this is a possible action if the probe did not automatically retract. However, the “IN-SHIELD” light lit on Control Drawer A indicates that all probes are in the in-shield position and the ball valve, TIP-V-1, failed to automatically shut.

B. Incorrect. Plausible since it is necessary to activate the shear valve, TIP-V-7, to isolate the TIP system. However, the “IN-SHIELD” light lit on Control Drawer A indicates that all probes are in the in-shield position and no action is necessary to withdraw the probe.

C. Correct. The “IN-SHIELD” light lit on Control Drawer A indicates that all probes are in the in-shield position, but the ball valve, TIP-V-1, failed to automatically shut. It is necessary to activate the shear valve, TIP-V-7, to isolate the TIP system.

D. Incorrect. Plausible since this is a possible action if the probe did not automatically retract or could not be retracted electrically. However, the “IN-SHIELD” light lit on Control Drawer A indicates that all probes are in the in-shield position and the ball valve, TIP-V-1, failed to automatically shut.

Technical Reference(s)		Attached w/ Revision # See Comments / Reference
ABN-TIPS, TIP System Failure to Isolate		
SD000155, CGS System Description, Vol 6, Chapter 7, Traversing In-Core Probe (TIP).		

Proposed references to be provided during examination: None

Learning Objective: 6989 – Explain the TIP system response to an “FA” (LOCA) signal.

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 41.5
55.43 ____

Comments / Reference: TIP System Description	Revision: 13

COLUMBIA SYSTEMS JUNE 2010
TIP SD000155, r13 mr0

V. CONTROL THEORY AND INTERLOCKS

The TIP and Control and Monitoring Instrument Panel P607 located in the Main Control Room, houses five Drive Control Units (one for each channel) a Flux Probing Monitor, three Valve Control Monitor panels, and one X-Y recorder.

A. Control Room Controls

1. Drive Control Units (Panel P607) (See Figure 6, 7)

a) DETECTOR POSITION digital display

Continuous digital display of detector position (units of inches).

0001 = **Reference point**, just behind the indexer (away from the core).

9811 to 9816 = In-shield position (value different for each machine). Counter increases as detector is driven forward (toward core) passing reference (0001) at indexer.

b) CORE LIMIT digital display

Static display of pre-programmed core top or bottom limits of selected channel. The BOTTOM or TOP limit indication is selected by the CORE LIMIT switch position. (The Core Bottom limit can range from 0708 to 0861, depending on the length of the tubing from the indexer to the bottom of the LPRM, and the Core Top limit can vary from 0849 to 1007. The values of these limits are posted on an Operator Aid on Panel P607).

c) CORE TOP (white light)

Indicates when the detector is approximately 5" below the top of the LPRM tubing (effectively at the top of active fuel) as sensed by core limit circuit.

d) IN-CORE (red light)

Indicates when the detector is at the bottom of the core as sensed by the core limit circuit.

e) IN-SHIELD (white light)

Indicates the detector has entered the shield chamber.

b) Meter Selector Switch 9 position maintain contact

A BUS-	
A BUS +	Connects % POWER/VOLTS meter to the
B BUS -	internal ± 10 volt power supplies
B BUS +	
CH 1 - 5	Connects % POWER/VOLTS meter to the selected channel's flux amplifier output

c) Channel Selector Switch 5 position maintain contact

OFF	Disconnects X-Y Recorder from TIP system.
CH 1 - 5	Connects the X-Y Recorder to the selected channel

3. Valve Control; Panels (Panel P607) LO-6985

a) SQUIB MONITOR (yellow light)

Normally off, lights to indicate squib (for shear valve) has been detonated or squib circuit is open.

b) SHEAR VALVE MONITOR (yellow light)

Normally off, lights to indicate shear valve is closed.

c) BALL VALVE (red & white lights)

OPEN (red) - Indicates ball valve not fully-closed

CLOSED (white) - Indicates ball valve closed

6. Containment Isolation Signal (LOCA) LO-6989

Upon receipt of a containment isolation signal, High drywell pressure of 1.68 psig ("F" signal) or Low RPV water level of -50" ("A" signal) (one out of two, taken twice), any TIP detector inserted past its shield chamber will automatically withdraw and the associated ball valve will close once the IN-SHIELD relay is picked up. This actuation seals in until it is reset. TIP purge isolation valve TIP-V-15 will automatically close to isolate its purge supply line from the Containment. If TIP-V-15 is closed, for any reason, the same action will take place, except a reset is not required when TIP-V-15 is reopened.

When the subsystem is operated in the Manual Mode of operation, the detector can be stopped at any time, or its travel direction changed by selecting from Manual Forward to Manual Off, or to Manual Reverse.

B. Abnormal Operation

1. Manual (Hand Crank) Drive (See Figure 3)

Manual Drive operation permits the electric motor in the drive mechanism to be disconnected from the detector drive, and a hand crank substituted to provide driving power. This method is used primarily for probing the guide tube runs to determine the distance to the end of the guide tube so that the core top limits can be programmed. These limits are unique to each drive mechanism and each detector channel therefore replacement of the detector drive cable or guide tubing requires reprogramming all of the core top and bottom limits. Manual Drive may also be used in an emergency if the motor failed to withdraw a detector following an isolation signal so that the ball valve may be manually closed.

Manual hand crank for isolation purposes is considered after other remote means of isolation fail. See ABN-TIPS.

2. TIP Containment Isolation LO-7634

The TIP System is required to be isolated if an F or A signal is received. Additionally if the TIP System becomes a discharge path between the RPV and the Reactor Building the EOPs also direct isolation. If any detector has not moved to the storage (IN-SHIELD) position then an attempt is made to withdraw the detector either with the manual reverse or if possible by the manual hand crank (high-dose job) to clear the ball valves. The ball valves should close when the detector enters the shield. If all the ball valves do not indicate closed, then the associated Squib Shear Valve is closed by taking the Shear Valve Control keyswitch on P607 to the FIRE position.

Comments / Reference: ABN-TIPS	Revision: Major 002 Minor 003

Number: ABN-TIPS	Use Category: CONTINUOUS	Major Rev: 002 Minor Rev: 003 Page: 2 of 4
Title: TIP SYSTEM FAILURE TO ISOLATE		

1.0 ENTRY CONDITIONS

A valid Containment Isolation Signal is present and the TIP Ball Valve(s) have failed to close as indicated at H13-P607 and H13-P601.

2.0 AUTOMATIC ACTIONS

None

3.0 IMMEDIATE OPERATOR ACTIONS

None

4.0 SUBSEQUENT OPERATOR ACTIONS

NOTE: If the failure to isolate occurred during TIP operation, this procedure may be entered at step 4.5.

NOTE: If the affected drive unit is not known, then it may require that all of the Tip units need to be checked.

Number: ABN-TIPS	Use Category: CONTINUOUS	Major Rev: 002 Minor Rev: 003 Page: 3 of 4
Title: TIP SYSTEM FAILURE TO ISOLATE		

4.4 **VERIFY** the following on the associated drive unit(s):

- The READY light is illuminated. _____
- The IN-SHIELD light is illuminated. (If any of the detectors are NOT IN-SHIELD, proceed to the following step). _____
- The detector position is at the posted IN-SHIELD location, ±1". _____

4.5 **REFER** to Technical Specification 3.6.1.3. _____

4.6 IF any detector is NOT IN-SHIELD,
THEN **RETRACT** each affected detector to the IN-SHIELD position as follows:

4.6.1 **PLACE** the Manual Drive Control switch on the appropriate Drive Control Unit to the **REV** position. _____

4.6.2 **VERIFY** the IN-SHIELD light is illuminated. _____

4.7 IF the detector is still not IN-SHIELD,
THEN **CONSIDER MANUALLY CRANKING** the affected detector to the IN-SHIELD position from the Drive Mechanism per PPM 10.24.187. _____

4.8 IF the detector is still not IN-SHIELD,
OR the isolation valve has failed to close,
THEN **ISOLATE** the affected TIP line(s) as follows:

4.8.1 **OBTAIN** permission from the CRS/Shift Manager to fire the applicable squib valve(s). _____

4.8.2 **PLACE** the keylock valve control switch (key number 31, 32, 33, 34, 35) on the appropriate valve control drawer to the **FIRE** position for the channel(s) that did not isolate. _____

Examination Outline Cross-reference:	Level	RO		SRO
Rev. 0 Date: 12/16/2016	Tier	2		
	Group	2		
	K/A	215002.A3.04		
Level of Difficulty: 3	Importance Rating	3.6		

Rod Block Monitor System: Ability to monitor automatic operations of the ROD BLOCK MONITOR SYSTEM including: Verification of proper functioning/ operability: BWR-3,4,5

Question # 59

The reactor is operating in Mode 1 with the following conditions:

- APRM power levels:
  - APRM 1: 24%
  - APRM 2: 31%
  - APRM 3: Bypassed
  - APRM 4: 30%
- A center control rod is selected on the Rod Select Matrix. Selected rod position is 32.

What is the status of the Rod Block Monitors (RBM)?

A. RBM A is NORMAL and RBM B is BYPASSED.

B. RBM A is NORMAL and RBM B is NORMAL.

C. RBM A is BYPASSED and RBM B is BYPASSED.

D. RBM A is BYPASSED and RBM B is NORMAL.

Answer: D

K/A Match:
The question requires demonstrating knowledge of Rod Block Monitor (RBM) automatic actions when Average Power Range Monitor (APRM) inputs are below the Low Power Setpoint (LPSP) or bypassed.

SRO Only:
N/A

Explanation:

A. Incorrect. Plausible if student does not understand where the LPSP is derived.

B. Incorrect. Plausible if student does not understand where the LPSP is derived or the LPSP value.

C. Incorrect. Plausible if student does not understand where the LPSP is derived or that RBM is not bypassed when a center rod is selected.

D. Correct. RBM A derives Simulate Thermal Power (STP) from APRM 1. Since APRM 1 is below the Low Power Setpoint (LPSP) of 26%, RBM A is bypassed. RBM B derives Simulate Thermal Power (STP) from APRM 2. Since APRM 2 is above the Low Power Setpoint (LPSP) of 26%, RBM B is in normal operation.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
SD001819, CGS System Description, Vol. 6, Chap. 10, Power Range Neutron Monitor	
CGS Licensee Controlled Specifications (LCS)	
ISP-RBM-B301, RBM-CHS-A Calibration	

Proposed references to be provided during examination: None

Learning Objective: 5699 Explain how the following systems interrelate with the RBM:
a. APRM System

Question Source:
- Bank #
- Modified Bank # (Note changes or attach parent)
- New X

Question History: Last NRC Exam N/A

Question Cognitive Level:
- Memory or Fundamental Knowledge
- Comprehension or Analysis X

10 CFR Part 55 Content:
- 55.41 41.7
- 55.43

Comments / Reference: SD001819	Revision: 2

e) Compares Reactor Recirculation flow signals from all four APRMs and provides alarm at the RBM Module in the event of a >5% mismatch between APRM core flow signals.

(1) "FLOW COMPARE" inverse video alarm received on associated RBM module.

(2) 4.603.A8 (Drop 3-6) "Flow Reference Off Normal".

f) Provides input into two of the four Operator Display Assemblies (ODAs) on Main Control Room Panel H13-P603.

2. General Description

a) RBM consists of two redundant channels for monitoring of reactor power in the immediate vicinity of a control rod selected for movement.

(1) Labeled as RBM 'A' and RBM 'B'.

b) RBM is active above LPSP as determined by the reference APRM for the associated RBM Channel. LO-5085

(1) 'A' RBM receives Simulated Thermal Power (STP) input from APRM #1 with alternate APRM being APRM #3 and second alternate channel being APRM #4.

(a) Alternate APRM is automatically selected when associated primary APRM is bypassed.

(2) 'B' RBM receives Simulated Thermal Power (STP) input from APRM #2 with alternate APRM being APRM #4 and second alternate channel being APRM #3

(a) Alternate APRM is automatically selected when associated primary APRM is bypassed.

(3) Both RBM channels are bypassed when APRM reference power is less than LPSP.

(4) Both RBM channels are also bypassed when a peripheral (edge) control rod is selected.

(a) RBM must have an internal control rod selected to be active.

Comments / Reference: LCS 1.3.2.1, Trip Setpoints	Revision: 92

Table 1.3.2.1-2 (page 1 of 1)
Rod Block Monitoring Instrumentation Trip Setpoints

-----NOTE-----
Table 1.3.2.1-2 lists required instrument setpoints to support OPERABILITY for LCO 3.3.2.1. See Technical Specification 3.3.2.1 and the applicable Bases for further application details.

FUNCTION	APPLICABLE SPECIFIED CONDITION SETPOINT	TRIP SETPOINT
1. Rod Block Monitor		
a. Low Power Range - Upscale	(a)	Values specified in COLR
b. Intermediate Power Range – Upscale	(b)	Values specified in COLR
c. High Power Range - Upscale	(c)	Values specified in COLR
d. Inop	≥ 26.0% RTP	NA

(a) Thermal Power ≥ 26.0% and ≤ 61.0% RTP
(b) Thermal Power > 61.0% and ≤ 81.0% RTP
(c) Thermal Power > 81.0% RTP

Comments / Reference: ISP-RBM-B301	Revision: 001

Number: ISP-RBM-B301	Use Category: CONTINUOUS	Major Rev: 001 Minor Rev: N/A Page: 22 of 29
Title: RBM-CHS-A Calibration		

7.7 Parameter Check

7.7.1 **NAVIGATE** to SHOW PARAMETERS soft key. _____

7.7.2 **PRESS** SHOW PARAMETERS soft key. _____

7.7.3 **RECORD** parameters in As-Found column of Table 7.7a. _____

Table 7.7a

Parameter	Desired	As-Found	As-Left
Downscale Alarm Setpoint	5.0%		
Flow Compare Alarm	5.0%		
Low Power Setpoint	26.0%		
Low Trip Setpoint	121.2%		
Intermediate Power Setpoint	61.0%		
Intermediate Trip Setpoint	116.2%		
High Power Setpoint	81.0%		
High Trip Setpoint	111.2%		
LPRM Auto Bypass	5.0%		

Examination Outline Cross-reference:	Level	RO		SRO
Rev. 0 Date: 12/16/2016	Tier	2		
	Group	2		
	K/A	226001 A4.19		
Level of Difficulty: 3	Importance Rating	3.4		

RHR/LPCI: Containment Spray System Mode: Ability to manually operate and/or monitor in the control room: Drywell Temperature

Question # 60

Columbia is operating at 100% power.

- A large LOCA occurs.
- RPV level lowers to -159" and is rising slowly.
- RPV pressure is 200 psig and lowering.
- The CRS directs lowering drywell temperature by spraying the drywell with RHR B.

What action needs to be taken before opening RHR-V-17B (Upper Drywell Spray Isolation Valve) and RHR-V-16B (Upper Drywell Spray Isolation Valve)?

A. STOP RRC Pumps, CLOSE RHR-V-42B (LPCI Injection).

B. STOP RRC Pumps, CLOSE RHR-V-24B (Suppression Pool Cooling/Test Return).

C. STOP ALL Drywell Cooling Fans, CLOSE RHR-V-42B (LPCI Injection).

D. STOP ALL Drywell Cooling Fans, CLOSE RHR-V-24B (Suppression Pool Cooling/Test Return).

Answer: C

K/A Match:

The question ensures the candidate knows the actions needed to align RHR B for drywell spray which will lower and control drywell temperature.

SRO Only:

N/A

Explanation:

Due to the low RPV level, RRC pumps have tripped and do not need to be secured. Drywell cooling flow (RCC) is lost as well, but drywell fans are still running and need to be secured. Based on conditions, RHR B would be injecting and RHR-V-42B will need to be closed.

A. Incorrect. RRC pumps are already secured. Plausible because PPM 5.2.1 directs securing RRC pumps prior to initiating spray. Plausible because RHR-V-42B needs to be closed.

B. Incorrect. RRC pumps are already secured. Plausible because PPM 5.2.1 directs securing RRC pumps prior to initiating spray. Plausible because RHR-V-24B would need to be closed if OPEN, but isn't OPEN in this condition.

C. Correct Answer.

D. Incorrect. RHR-V-24B does not need to be closed. Plausible because RHR-V-24B would need to be closed if it was open.

Technical Reference(s)		Attached w/ Revision # See Comments / Reference
PPM 5.2.1, Primary Containment Control		
SOP-RHR-SPRAY-DW-QC		

Proposed references to be provided during examination: None

Learning Objective: 5774 – Describe the flow path within the appropriate RHR system for each of the following: (e) Drywell spray

Question Source:
- Bank #
- Modified Bank # (Note changes or attach parent)
- New X

Question History: Last NRC Exam N/A

Question Cognitive Level:
- Memory or Fundamental Knowledge
- Comprehension or Analysis X

10 CFR Part 55 Content:
- 55.41 41.7
- 55.43

Comments / Reference: SOP-RHR-SPRAY-DW-QC	Revision: 3 mr 7

Number: SOP-RHR-SPRAY-DW-QC	Use Category: CONTINUOUS	Major Rev: 003 Minor Rev: 007 Page: 4 of 5
Title: Initiation of Drywell Spray - Quick Card		

2.0 PROCEDURE

2.1 Initiation of Drywell Sprays During EOPs

2.1.1 **VERIFY** RHR-P-2A(B) running. _____

2.1.2 **VERIFY** RHR-V-42A(B) **CLOSED** (LPCI Injection). _____

CAUTION

Operate Drywell sprays and Wetwell sprays on opposite loops if possible.
DO NOT initiate multiple loops of containment sprays simultaneously.

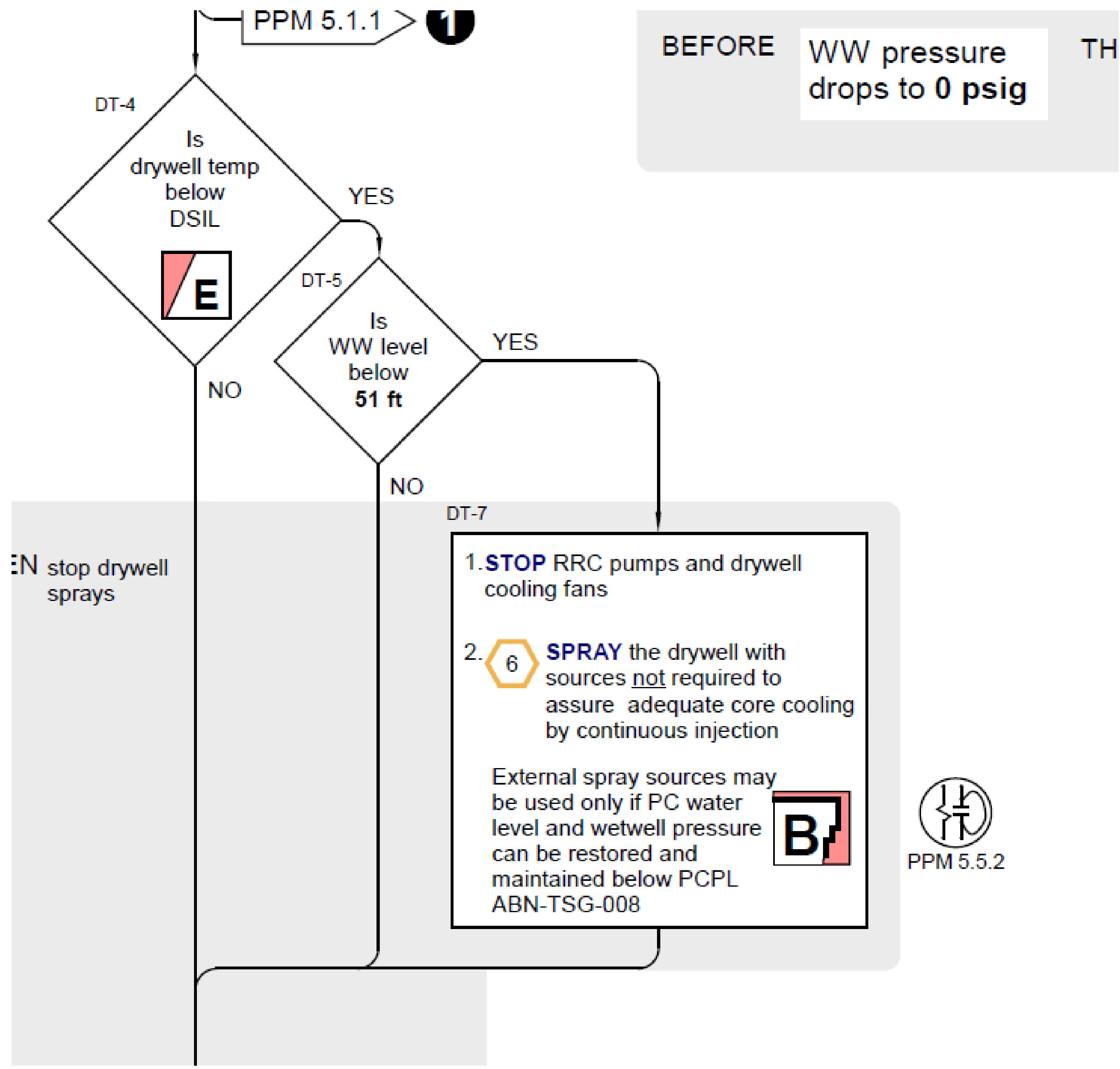
2.1.3 **OPEN** the following to spray the Drywell:

- RHR-V-17A(B) (Drywell Spray Inboard Isolation). _____
- RHR-V-16A(B) (Drywell Spray Outboard Isolation). _____

2.1.4 BEFORE Drywell pressure drops below 0.0 psig,
OR when directed by the CRS,
THEN **CLOSE** the following:

- RHR-V-16A(B). _____
- RHR-V-17A(B). _____

Comments / Reference: PPM 5.2.1	Revision: 23



Examination Outline Cross-reference:	Level	RO		SRO
Rev. 0 Date: 12/16/2016	Tier	2		
	Group	2		
	K/A	239001.2.2.42		
Level of Difficulty: 2	Importance Rating	3.9		

Main and Reheat Steam: Ability to recognize system parameters that are entry-level conditions for Technical Specifications.

Question # 61

CGS is operating in Mode 1. Reactor power is 100%.

How many Safety Relief Valves (SRVs) must be operable to continue unrestricted operation with the current plant conditions.

A. 4

B. 7

C. 12

D. 18

Answer: C

K/A Match:
This question requires the candidate to demonstrate knowledge of the entry conditions for .

SRO Only:
N/A

Explanation:

A. Incorrect. Plausible since only four SRVs are required to be operable when reactor power is < 25%. However, twelve SRVs are required to be operable for the given plant conditions.

B. Incorrect. Plausible if it believed that only the seven ADS related SRVs are required to be operable. However, twelve SRVs are required to be operable for the given plant conditions.

C. Correct. When reactor power is ≥ 25%, twelve SRVs are required to be operable.

D. Incorrect. Plausible if it believed that all SRVs are required to be operable. However, twelve SRVs are required to be operable for the given plant conditions.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
CGS Technical Specifications, LCO 3.4.3, Safety/Relief Valves (SRVs) - ≥ 25%	
CGS Technical Specifications, LCO 3.4.4, Safety/Relief Valves (SRVs) - < 25%	
SD000128, CGS System Description, Vol. 2 Chap. 1, Main Steam	

Proposed references to be provided during examination: None

Learning Objective: 5545 - Referencing Columbia Generating Station Technical Specifications (section 3 only for initial license candidates) associated with the Main Steam System and a set of plant conditions, determine as applicable the LSSS, the LCO, the action statement, and the appropriate bases.

Question Source:
- Bank # ____
- Modified Bank # ____ (Note changes or attach parent)
- New X

Question History: Last NRC Exam N/A

Question Cognitive Level:
- Memory or Fundamental Knowledge X
- Comprehension or Analysis ____

10 CFR Part 55 Content:
- 55.41 41.7
- 55.43 ____

Comments / Reference: LCO 3.4.3	Revision: 237

SRVs - ≥ 25% RTP
3.4.3

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.3 Safety/Relief Valves (SRVs) - ≥ 25% RTP

LCO 3.4.3 The safety function of 12 SRVs shall be OPERABLE, with two SRVs in the lowest two lift setpoint groups OPERABLE.

APPLICABILITY: THERMAL POWER ≥ 25% RTP.

Comments / Reference: LCO 3.4.4	Revision: 237

SRVs - < 25% RTP
3.4.4

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.4 Safety/Relief Valves (SRVs) - < 25% RTP

LCO 3.4.4 The safety function of four SRVs shall be OPERABLE.

APPLICABILITY: MODE 1 with THERMAL POWER < 25% RTP,
MODES 2 and 3.

Comments / Reference: SD000128	Revision: 12

COLUMBIA SYSTEMS September 2014
MAIN STEAM SD000128, r12 mr0

D. Safety/Relief Valves (Figure 2 and 3)

1. Eighteen safety/relief valves (SRVs), located on horizontal runs of main steam piping within the primary containment, discharge into the suppression pool. The SRVs are distributed on the Main Steam Lines as follows: LO-5528
   a) MSL A - 4 VALVES
   b) MSL B - 5 VALVES
   c) MSL C - 5 VALVES
   d) MSL D - 4 VALVES

2. All eighteen safety/relief valves (SRVs) operate in the following modes: LO-5527 NLO-12668b
   a) Automatic overpressure protection of the RPV in the Relief and Safety Modes of operation with setpoints and capacities as follows:

(RELIEF MODE)	(SAFETY MODE)	(LBM/HR/SRV)
2 valves @ 1091	2 valves @ 1165	876,118
4 valves @ 1101	4 valves @ 1175	883,950
4 valves @ 1111	4 valves @ 1185	891,380
4 valves @ 1121	4 valves @ 1195	898,800
4 valves @ 1131	4 valves @ 1205	906,250

   The Relief Mode prevents nuclear boiler overpressurization. Pressure Switches energize the SRV "C" solenoid which allows N_2 to enter a cylinder and mechanically open the valve. The relief setpoints are at a lower pressure than the safety setpoints.

   The Safety Mode protects the nuclear boiler from overpressurization that could lead to failure of the reactor coolant pressure boundary. The safety mode of the valve is actuated directly by the force exerted upon the valve spring by reactor pressure. By adjusting the spring compression the valve opening setpoint may be altered. The safety mode is a backup to the relief function.

   b) All SRVs can be opened from panel H13-P601 by controlling the "C" solenoid pilot valves.

3. Seven SRVs are designated to serve the vessel depressurization function.

Examination Outline Cross-reference:	Level	RO		SRO
Rev. 0 Date: 12/16/2016	Tier	2		
	Group	2		
	K/A	259001.K1.11		
Level of Difficulty: 3	Importance Rating	2.7		

Reactor Feedwater System: Knowledge of the physical connections and/or cause-effect relationships between REACTOR FEEDWATER SYSTEM and the following: RFP lube oil system

Question # 62

CGS is operating in Mode 1.

The following RFW-P-1A annunciators are alarming:

- P840.A1.7-1, TURBINE A VIBRATION HIGH
- P840.A1.8-1, TURBINE A THRUST BEARING WEAR HIGH

Which of the following is true if turbine thrust bearing wear rises to 15 mils?

The turbine trip oil header for RFW-P-1A will...

A. direct oil to the tops of the high pressure and low pressure stop valves causing them to close.

B. pressurize to align the high pressure and low pressure stop valve drains to the oil reservoir.

C. depressurize allowing the high pressure and low pressure stop valves to close.

D. remain pressurized. Thrust bearing wear does not cause a feed turbine trip.

Answer: C

K/A Match:

Type a brief explanation as to why the question is an exact match for the chosen K/A. If this cannot be easily written, the match is questionable and should be reviewed for K/A replacement or question substitution.

SRO Only:

N/A

Explanation:

A. Incorrect. Plausible since pressure on the top of the valves would cause them to close. However, the stop valves are hydraulically open, spring shut. They are tripped by venting control oil pressure from the bottom of the valves.

B. Incorrect. Plausible since the stop valves are tripped by venting control oil pressure from the bottom of the valves. However, this is accomplished by depressurizing the control oil header.

C. Correct. When Turbine Thrust Bearing wear reaches 15 mils, the turbine trip oil system depressurizes the oil lines leading to the high and low pressure stop valves. These valves close by spring force within 0.5 seconds.

D. Incorrect. Plausible since pump thrust bearing will actuate the Turbine A Vibration High alarm. However, high wear on the pump thrust bearing will not cause a turbine trip.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
SD000151, CGS System Description, Vol. 2 Chap. 3, Reactor Feedwater	
P840.A1.7-1, TURBINE A VIBRATION HIGH, Annunciator Response	
P840.A1.8-1, TURBINE A THRUST BEARING WEAR HIGH, Annunciator Response	

Proposed references to be provided during examination: None

Learning Objective: 5750 - Explain the following pertaining to the Turbine Oil System:
c. Basic relationship between trip oil pressure and steam stops/ hydraulic trip valve during normal and tripped conditions.
d. How RFP trip oil responds to a turbine trip and turbine reset.

Question Source: Bank # LO01945
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 41.2 to 41.9
55.43 ______

Comments / Reference: SD000151	Revision: Major 13 Minor 003

RFT Oil System (Figure 4)

The purpose of the RFT Oil System is to provide lubrication to the turbine and pump bearings, to provide high pressure control oil for the turbine control and trip systems operation, and to provide lubrication for the turbine turning gear. LO-5749f NLO-1273 NLO-1273

Turbine Trip Oil (Figure 5)

The turbine trip oil system, provides a means of quickly depressurizing the oil lines leading to the HP stop valve (MS-V-172A/B) and LP stop valve (BS-V-60A/B), such that these valves rapidly close in the event of a turbine trip. The turbine trip system receives high pressure oil from the control oil section through a 3/16" orifice. This small orifice allows the trip oil header to slowly pressurize and open the HP and LP Stop Valves when the turbine is reset. When the turbine is tripped, the trip oil header is dumped to the turbine oil reservoir through one or more flow paths, and the 3/16" orifice prevents the control oil system from recharging the trip oil header portion faster than it is being dumped. The loss of pressure allows a spring loaded slave piston in the HP and LP stop valve actuator assemblies to open which in turn removes pressure from the valve actuator spring loaded main pistons enabling the stop valves to rapidly close (LT 0.5 sec). LO-5750d

There are two solenoid trip valves (20TPF and 20TT), two manual hydraulic trip valves (RFT-V-22A/B and RFT-V-23A/B), and one mechanical overspeed trip device in the trip oil system for each RFT.

The primary solenoid trip valve (20TT) energizes to dump the trip oil header to the reservoir, closing the HP and LP stop valves, on any electrically generated NLO-12738a-f

Steam Stop Valve (Figures 6, 8, and 9)

Each turbine is equipped with one HP and one LP hydraulically (oil) operated steam stop valve. They enable rapid isolation of the HP and LP steam supplies to the turbine in case of a turbine trip. The HP stop valve (MS-V-172A/B) is bolted to the HP steam inlet section of the turbine governor. The LP stop valve (BS-V-60A/B) is bolted to the LP steam inlet on the opposite side of the turbine from the HP stop valve. The stop valves are hydraulic-to-open and spring-to-close. They are capable of closing in less then 0.5 seconds. LO-5749i NLO-12734i

Comments / Reference: P840.A1.7-1, Annunciator Response	Revision: Major 022 Minor 002

Number: 4.840.A1	Use Category: CONTINUOUS	Major Rev: 022 Minor Rev: 002 Page: 50 of 74
Title: 840.A1 Annunciator Panel Alarms		

7-1 TURBINE A VIBRATION HIGH

7-1 WINDOW	SOURCE	AUTOMATIC ACTIONS
TURB A VIB TROUBLE	Any of the sources listed in the table below.	None

SOURCES	ALERT SETPOINT	DANGER SETPOINT
RFW-VBI-XE/P1A - Pump Axial Thrust Displacement (second NOTE)	± 15 mls	± 18 mls
RFW-VBI-1A/XS/P0BXY - Pump Radial Outboard Bearing Vibration	3 mls	4.5 mls
RFW-VBI-1A/XS/P1BXY - Pump Radial Inboard Bearing Vibration (see NOTE)	3 mls	4.5 mls
RFW-VBI-XE/T1A - Turbine Axial Thrust Displacement (first NOTE)	± 12 mls	± 15 mls
RFW-VBI-1A/XS/T1BXY - Turbine Radial Inboard Bearing Vibration	3 mls	4.5 mls
RFW-VBI-1A/XS/T01BXY - Turbine Radial Outboard Bearing Vibration	3 mls	4.5 mls
RFW-VBI-1A/XS/TE - Turbine Eccentricity	4 mls	6 mls

NOTE: Turbine trips if Turbine Axial Thrust Displacement is 15 mils in the normal or 15 mils in the counter thrust direction (RFT-RLY-TBT/1A).

NOTE: Disregard the Pump Axial Thrust Displacement Alarm if it is in the counter thrust direction. Thrust displacement is only a problem in the normal direction.

NOTE: Turbine radial vibration may rise to the ALERT setpoint on some instruments during prolonged operation at 2400 - 3200 rpm, but should not rise to the DANGER setpoint(s) at any time.

Comments / Reference: P840.A1.8-1, Annunciator Response	Revision: Major 022 Minor 002

Number: 4.840.A1	Use Category: CONTINUOUS	Major Rev: 022 Minor Rev: 002 Page: 59 of 74
Title: 840.A1 Annunciator Panel Alarms		

8-1 TURBINE A THRUST BEARING WEAR HIGH

8-1 WINDOW	SOURCE	AUTOMATIC ACTIONS
TURB A THRUST BRG WEAR HIGH	RFT-VMP-1 (RFT-RLY-TBX/1A) (12 mils in the normal or 12 mils in the counter thrust direction)	None

NOTE: Turbine trips if thrust is 15 mils in the normal or 15 mils in the counter thrust direction (RFT-RLY-TBT/1A).

Examination Outline Cross-reference:	Level	RO		SRO
Rev. 0 Date: 12/16/2016	Tier	2		
	Group	2		
	K/A	268000.K3.04		
Level of Difficulty: 3	Importance Rating	2.7		

Radwaste: Knowledge of the effect that a loss or malfunction of the RADWASTE will have on following: Drain sumps

Question # 63

On a loss of control air to the Reactor Building, Reactor Building radioactive floor drain (FDR) sumps...

A. cannot be pumped down.

B. may be pumped to the Waste Collector Tank only.

C. may be pumped to the Floor Drain Collector Tank only.

D. may be pumped to either the Floor Drain Collector Tank or the Waste Collector Tank.

Answer: A

K/A Match:
The question determines if candidates understand the impact of a malfunction (Loss of air) on the Radwaste system with regard to floor drains.

SRO Only:
N/A

Explanation:

A. Correct. When control air is lost, sump pump discharge isolation valves FDR-V-219/220/221/222 fail closed. Reactor building sumps cannot be pumped down.

B. Incorrect. Plausible since the Floor Drain Collector Tank to Waste Collector Tank crosstie, FDR-V-33, fails as-is on a loss of control air. However, all reactor building sump pump discharge valves fail close and the sumps cannot be pumped down.

C. Incorrect. Plausible if it is believed that the sump pump discharge isolation valves FDR-V-219/220/221/222 fail as-is or open and the Floor Drain Collector Tank to Waste Collector Tank crosstie, FDR-V-33, fails closed. However, when control air is lost, sump pump discharge isolation valves FDR-V-219/220/221/222 fail closed. Reactor building sumps cannot be pumped down.

D. Incorrect. Plausible if it is believed that the sump pump discharge isolation valves FDR-V-219/220/221/222 fail as-is or open. However, when control air is lost, sump pump discharge isolation valves FDR-V-219/220/221/222 fail closed. Reactor building sumps cannot be pumped down.

Technical Reference(s)		Attached w/ Revision # See Comments / Reference
ABN-CAS, Control Air System Failure		
SD000130, CGS System Description, Vol. 9, Chap. 7, Plant Drains		

Proposed references to be provided during examination: None

Learning Objective: 12507 – Explain he interrelationships between the Plant Drains system and the following systems: (c) Control and Service Air

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ____

10 CFR Part 55 Content: 55.41 41.5
55.43 ____

Comments / Reference: ABN-CAS	Revision: 009

Number: ABN-CAS	Use Category: CONTINUOUS	Major Rev: 009 Minor Rev: N/A Page: 18 of 27
Title: CONTROL AIR SYSTEM FAILURE		

NOTE: Valves marked with an asterisk (*) are Safety-Related. {P-104643}

VALVE	BUILDING {P-90496}	FAILURE MODE	DESCRIPTION
FDR-V-105	RW	CLOSED	FDR-P-17 Discharge
FDR-V-111	RW	CLOSED	FDR-FU-19 Vent
FDR-V-118	RW	CLOSED	FDR-FU-19 Vent
FDR-V-171	RW	OPEN	FDR-DM-111 to FDR-TK-6
FDR-FCV-189	RW	CLOSED	FDR Discharge to CW
FDR-FCV-190	RW	CLOSED	FDR Discharge to CW
FDR-V-33	RW	AS IS	FDR-TK-6/EDR-TK-2 Crosstie
FDR-FCV-332	RW	CLOSED	FDR-P-23 Discharge
FDR-V-443	RW	CLOSED	FDR-P-23 Suction
FDR-V-445	RW	AS IS	COND Flush to FDR-P-23
FDR-V-448A (B)	RW	CLOSED	Waste Sludge Phase Separator Inlet
FDR-V-451	RW	CLOSED	FDR-TK-22 Recirc
FDR-V-457	RW	CLOSED	FDR-P-16 to FDR-FU-19
FDR-V-467	RW	CLOSED	FDR-DM-111 to Spent Resin Tank
FDR-V-476	RW	CLOSED	FDR-P-36 Suction
FDR-V-478	RW	CLOSED	FDR-P-36 Discharge
FDR-V-480	RW	CLOSED	FDR-P-23 Recirc
FDR-V-481	RW	CLOSED	FDR-P-23 Suction
FDR-V-219 *	RB	CLOSED	FDR-P-4A (B) Discharge Isolation
FDR-V-220 *	RB	CLOSED	FDR-P-4A (B) Discharge Isolation
FDR-V-221 *	RB	CLOSED	FDR-P-1A (B) Discharge Isolation
FDR-V-222 *	RB	CLOSED	FDR-P-1A (B) Discharge Isolation

Attachment 7.1, Loss of Control Air, Valve Failure Mode

Comments / Reference: Plant Drains System Description	Revision:

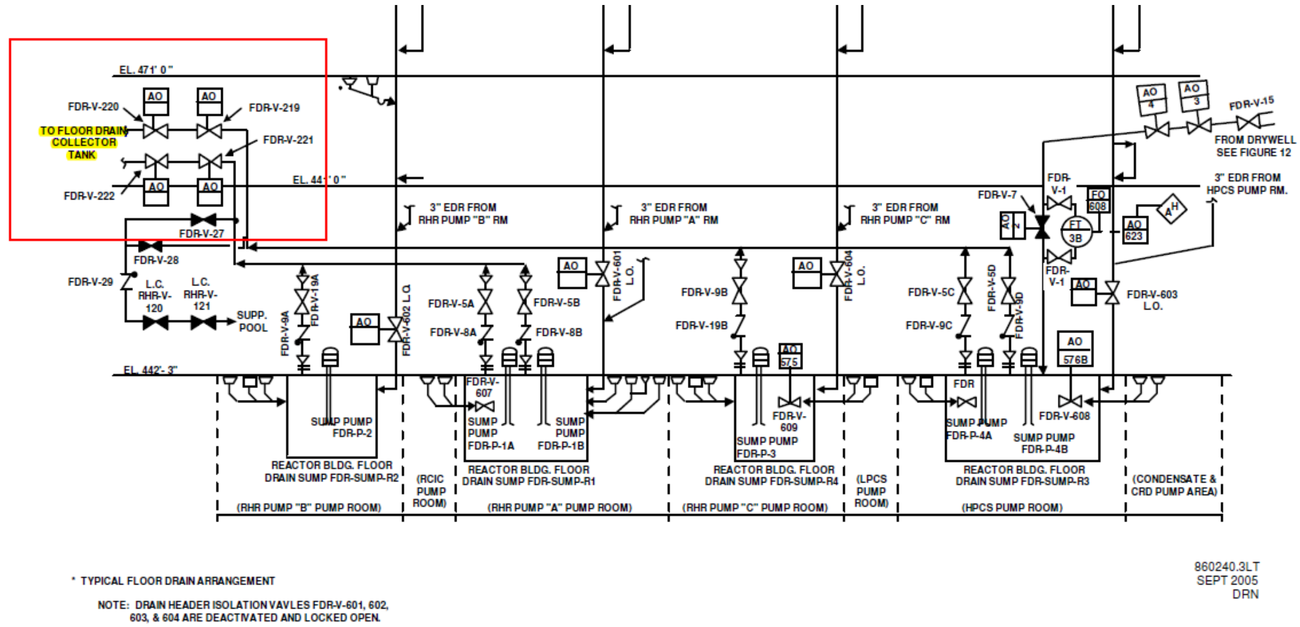


FIGURE 13. REACTOR BUILDING FDR

Examination Outline Cross-reference:	Level	RO		SRO
Rev. 0 Date: 12/16/2016	Tier	2		
	Group	2		
	K/A	272000.K4.02		
Level of Difficulty: 3	Importance Rating	3.7		

Radiation Monitoring System: Knowledge of RADIATION MONITORING System design feature(s) and/or interlocks which provide for the following: Automatic actions to contain the radioactive release in the event that the predetermined release rates are exceeded.

Question # 64

CGS is operating in Mode 1.

The following annunciators are alarming:

- P602.A5.1-5: REACTOR BLDG EXH PLENUM RAD HIGH
- P602.A5.1-4: REACTOR BLDG EXH PLENUM RAD HI-HI

Reactor building exhaust plenum radiation levels read:

- REA-RIS-609A: 15.5Mr/hr
- REA-RIS-609B: 16.1Mr/hr
- REA-RIS-609C: 14.8Mr/hr
- REA-RIS-609D: 15.6Mr/hr

Which of the following will occur in response to these conditions?

A. Control room ventilation fans trip.

B. Offgas system discharge isolates.

C. Reactor water sample valves close.

D. Containment Nitrogen Makeup isolates.

Answer: D

K/A Match:
Question evaluates whether the candidate understands automatic actions that will occur as a result on high radiation detected in the reactor building exhaust plenum

SRO Only:
N/A

Explanation:
High reactor building exhaust plenum levels cause a Z signal when above 13mr/hr. Per ABN-FAZ-QC, Containment Nitrogen makeup isolates on a Z signal.

A. Incorrect. Control room ventilation fans do not trip. Instead the CR emergency filtration systems start and align air to the remote intakes. Plausible because automatic actions are associated with the control room ventilation system.

B. Incorrect. Offgas does not isolate on a Z signal. Plausible because the offgas system processes radiative non-condensable gases from the condenser during normal operations.

C. Incorrect. Reactor water sample valves do not isolate on a Z signal. Plausible because they do isolate on an F or A signal.

D. Correct.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
ABN-FAZ-QC FAZ Automatic Actions – Quick Card	
PPM 4.602.A5, Annunciator Response Procedure	

Proposed references to be provided during examination: None

Learning Objective: 11937 – Describe the effect that a loss or malfunction of each of the following would have on the NS4 system. (b) RB Exhaust Plenum Radiation Monitor

Question Source: Bank # LX00825
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 41.7
55.43 ______

Comments / Reference: 4.602.A5	Revision: 46

Number: 4.602.A5	Use Category: CONTINUOUS	Major Rev: 046 Minor Rev: N/A Page: 8 of 63
Title: 602.A5 Annunciator Panel Alarms		

1-4 REACTOR BUILDING EXHAUST PLENUM RADIATION HIGH-HIGH

1-4 WINDOW	SOURCE	AUTOMATIC ACTIONS
REACTOR BLDG EXH PLENUM RAD HI-HI	Any of the following: • REA-RIS-609A (H13-P606) • REA-RIS-609B (H13-P606) • REA-RIS-609C (H13-P633) • REA-RIS-609D (H13-P633) (GE 13 Mr/hr)	The following may occur: • Primary Containment Vent/Purge Isolates • Secondary Containment EDR/FDR Lines Isolate • Secondary Containment Supply/Exhaust Dampers Isolate • Reactor Bldg Supply Fans Trip • Reactor Bldg Exhaust Fans Trip • Standby Gas Treatment Initiates • Control Room HVAC Outside Air Dampers Close • Control Room Emergency Filtration Initiates • Rx Building Quench Initiates • The Reactor Bldg Sump Exhaust Fans stop • The Reactor Bldg Room Cooling Fans start

Comments / Reference: ABN-FAZ-QC	Revision: 3 mr 1

Number: ABN-FAZ-QC	Use Category: REFERENCE	Major Rev: 003 Minor Rev: 001 Page: 4 of 5
Title: FAZ Automatic Actions - Quick Card		

2.0 FAZ AUTOMATIC ACTIONS

- F= High DW Pressure (1.68 psig)
- A= RPV Low Level 2 (-50")
- Z= High RBHV Exhaust Plenum Radiation Level (13 mr/hr)

ACTION	F	A	Z	✓
HPCS Diesel starts, HPCS-P-2 starts (H13-P601)	•	•		
HPCS-P-1 starts (H13-P601)	•	•		
LPCS-P-1 starts (H13-P601)	•			
RHR-P-2A starts (H13-P601)	•			
RHR-P-2B starts (H13-P601)	•			
RHR-P-2C starts (H13-P601)	•			
Drywell EDR and FDR sumps isolate (EDR-V-19,20)(FDR-V-3,4) (H13-P601)	•	•		
Miscellaneous RHR Valves isolate: Sample valves (1.68#, +13"), Disch to RW (1.68#, +13"), SDC (+13")(H13-P601)	•			
RRC sample lines isolate (RRC-V-19,20) (H13-P601)		•		
RWCU isolates (H13-P601)		•		
RCIC starts (H13-P601)		•		
RRC pumps trip (H13-P602)		•		
RWCU pumps trip(H13-P602)		•		
Reactor Scrams (at 1.68 psig or +13") (H13-P603)	•	•		
CW-P-1B and CW-P-1C trips (H13-P840)	•	•		
If BOTH TSW pumps were running, TSW-P-1B trips (H13-P840)	•	•		

Comments / Reference: ABN-FAZ-QC	Revision: 3 mr 1

Number: ABN-FAZ-QC	Use Category: REFERENCE	Major Rev: 003 Minor Rev: 001 Page: 5 of 5
Title: FAZ Automatic Actions - Quick Card		

ACTION	F	A	Z	✓
Div 1 and 2 SW start (H13-P840 & H13-P820)	•			
RCC pumps trip (H13-P840 & H13-P820)	•	•		
Div 1 and 2 Diesel Generators start (H13-P800)	•			
CB-75/72, CB-85/82 trip(H13-P800)	•	•		
MC-7C, 7E, 8C, and 8E trip: • RWCU pumps trip • MSL Tunnel Cooling fans trip • FPC Suppression Pool Cleanup mode isolates and pump trips • RWHV Exhaust Fans trip • CJW-P-1A trips	•	•		
Reactor Building EDR and FDR sump pump discharge headers isolate (H13-P632)	•	•	•	
TIP withdraws and isolates(H13-P607)	•	•		
SGT Systems start and maintain secondary containment pressure at -1.7" wg.(H13-P811 & H13-P827)	•	•	•	
RBHV isolates and fans trip (H13-P812)	•	•	•	
Reactor Building Emergency Room Coolers start (H13-P812)	•	•	•	
Reactor Building Lighting quenches (H13-P812)	•	•	•	
CSP/CEP isolate (H13-P813)	•	•	•	
Containment Nitrogen makeup isolates (H13-P813)	•	•	•	
CR and TSC Emergency Filtration systems start and align to remote intakes(H13-P826)	•	•	•	
RCC Containment isolation (H13-P825)	•	•		
CMS-SR-20 and 21 isolate (BD-RAD-22 & 23)	•	•		

Examination Outline Cross-reference:	Level	RO		SRO
Rev. 0 Date: 12/16/2016	Tier	2		
	Group	2		
	K/A	286000.K2.02		
Level of Difficulty: 2	Importance Rating	2.9		

Fire Protection System: Knowledge of electrical power supplies to the following: Pumps

Question # 65

What is the power supply to FP-P-2A, Fire Protection Pump 2A?

A. MC-5N

B. MC-3C

C. MC-7A

D. Diesel Engine

Answer: A

K/A Match:
Question determines if candidates know the power supply to fire protection pumps.

SRO Only:
N/A

Explanation:

FP-P-2A is powered by MC-5N

A. Correct.

B. Incorrect. Plausible because FP-TK-1 (cardox unit) is powered by MC-3C.

C. Incorrect. Plausible because FP-P-2B is powered by MC-6N.

D. Incorrect. Plausible because FP-P-110 is a diesel powered pump.

Technical Reference(s)		Attached w/ Revision # See Comments / Reference
A	SD000177, CGS System Description, Volume 3, Chapter 2, Fire Protection	

Proposed references to be provided during examination: None

Learning Objective: 12271 Explain the function and operation of the following Fire Protection System components, including any automatic features or interlocks: Fire Pumps

Question Source: Bank # LO03268
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 41.7
55.43 ______

Comments / Reference: SD000177	Revision: 16 mr 1

IX. <u>POWER SUPPLIES</u>

FP-P-1	Diesel Engine
FP-P-2A	MC-5N
FP-P-2B	MC-6N
FP-P-3	MC-6N
FP-P-110	Diesel Engine
FP-P-111	PDP-FP-7A
FP-TK-1 Cardox Unit	MC-3C
FP-VZ-1 (CO2 Vaporizer)	MC-3C
FCP-1/-2/-3	E-PP-8AA

Examination Outline Cross-reference:	Level	RO		SRO
Rev. 0 Date: 12/16/2016	Tier	3		
	Group			
	K/A	2.1.2		
Level of Difficulty: 2	Importance Rating	4.1		

Knowledge of operator responsibilities during all modes of plant operation.

Question # 66

Which of the following is correct regarding licensed operator responsibilities?

1. The CRS shall be in the Main Control Room while in MODEs 1, 2, or 3, unless relieved by the SM.
2. During periods when reactivity manipulations are in progress, the CRS is to assume the responsibility of the Reactivity Manager.
3. At least two licensed Control Room Operators (CRO) shall be at the Controls in the Control Room at all times.
4. The CRO1 should be within the Operator at the Controls Zone at all times.
5. The Control Room Operator(s) shall have an active RO license.

A. 1, 2, and 4

B. 1, 2 and 3

C. 1, 5 only

D. 2, 3 and 4

Answer: A

K/A Match:
Question determines if operators understand general licensed operator responsibilities related to control room staffing.

SRO Only:
N/A

Explanation:

A. Correct.

B. Incorrect. Plausible since 1 and 2 are correct. However, 3 is not correct.

C. Incorrect. Plausible since 1 is correct. However, 5 is not correct since control room operators may have and SRO OR RO license.

D. Incorrect. Plausible since 2 and 4 are correct. However, 3 is not correct.

Technical Reference(s)		Attached w/ Revision # See Comments / Reference
PPM 1.3.1 Operating Policies, Programs, and Practices.		

Proposed references to be provided during examination: None

Learning Objective: 6095 – State who shall be "at the controls" in the Control Room at all times.
6094 – State the responsibilities of the Control Room Operator (CRO)

Question Source: Bank # LO01463
Modified Bank # ____ (Note changes or attach parent)
New ____

Question History: Last NRC Exam ____

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ____

10 CFR Part 55 Content: 55.41 41.10
55.43 ____

Comments / Reference: PPM 1.3.1	Revision: Major 120 Minor 005

Number: 1.3.1	Use Category: INFORMATION	Major Rev: 120 Minor Rev: 005 Page: 43 of 98
Title: Operating Policies, Programs and Practices		

4.12.6 Control Room Operator

a. At least one licensed Control Room Operator (CRO) shall be at the Controls in the Control Room at all times. The Control Room Operator(s) shall have an SRO or RO license. The CRO1 should be within the Operator-at-the-Controls Zone at all times (Attachment 6.1). {R-253}, {R-1669}, {R-4888}, {R-7159}

b. Maintains responsibility for overall plant operations. The principal concern is monitoring of key primary plant parameters relating to reactivity control, vessel level control, and decay heat removal. Computer CRTs should be used, when

Number: 1.3.1	Use Category: INFORMATION	Major Rev: 120 Minor Rev: 005 Page: 42 of 98
Title: Operating Policies, Programs and Practices		

4.12.4 Control Room Supervisor {P-149175}

a. The CRS assists the Shift Manager in the performance of his duties and performs those duties when the Shift Manager is not available. {R-6970}

b. The CRS is responsible for supervising the activities of the Control Room Operators and other assigned personnel required to operate the plant safely and efficiently.

c. The CRS shall be in the Main Control Room while in Operational Conditions 1, 2, or 3 and shall have the Control Room Command Function unless relieved of the Control Room Command Function by the Shift Manager. If relieved by another valid SRO licensed individual (other than the STA) he is relieved of both the Control Room Command Function and the Control Room Supervisor Watch. The CRS shall have an SRO license. {R-253},{R-4824},{R-4888},{R-6971}

d. Direct the operation of the plant in accordance with Technical Specifications and approved plant procedures.

Examination Outline Cross-reference:	Level	RO		SRO
Rev. 0 Date: 12/16/2016	Tier	3		
	Group			
	K/A	2.1.20		
Level of Difficulty: 3	Importance Rating	4.6		

Ability to interpret and execute procedure steps.	
Question # 67	

Control room operators are performing a continuous use procedure when they come to a conditional step that does not apply to the current plant conditions. The operators mark the step “N/A”.

What is required to mark this step “N/A”?

A. Obtain CRS initials next to the step.

B. Obtain SM initials next to the step.

C. Obtain CRS and SM initials next to the step.

D. No additional justification or approval is required.

Answer: D	

K/A Match:
Question provides a reference and plant conditions and determines if the candidate can correctly interpret and execute procedural steps.

SRO Only:
N/A

Explanation:
Student must interpret PPM 3.1.2 Attachment 7.1.

A. Incorrect. Plausible since CRS initials are required to N/A non-conditional steps. However, for conditional steps that do not apply to the current plant conditions, no other approval is required.

B. Incorrect. Plausible since SM initials are required to N/A non-conditional steps. However, for conditional steps that do not apply to the current plant conditions, no other approval is required.

C. Incorrect. Plausible since CRS and SM initials are required to N/A non-conditional steps. However, for conditional steps that do not apply to the current plant conditions, no other approval is required.

D. Correct. When a conditional step does not apply to the current plant conditions, it may be marked N/A and no additional justification or approval is required.

Technical Reference(s)		Attached w/ Revision # See Comments / Reference
SWP-PRO-01, Procedure and Work Instruction Use and Adherence		

Proposed references to be provided during examination: None

Learning Objective: 10774 – Using the appropriate procedures, discuss procedure use and adherence at Columbia.

Question Source: Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 41.10
55.43 ______

Comments / Reference: SWP-PRO-01, section 4.7	Revision: 30

Number: SWP-PRO-01	Use Category: INFORMATION	Major Rev: 030 Minor Rev: N/A Page: 17 of 34
Title: Procedure and Work Instruction Use and Adherence		

4.7 Use of Not Applicable (NA) and Out of Sequence

NOTE: The use of NA on non-conditional step(s) and the use of out of sequence are the exception, not the normal practice.

NOTE: If an entire section of a work instruction or procedure is not performed, documentation of NA is not required.

NOTE: Not all prerequisites may be applicable under all plant conditions, or if only a portion of the procedure is to be performed. The Shift Manager may NA prerequisites that are not applicable.

NOTE: If a partial procedure is to be performed (one section of a Surveillance, System Operating Procedures (SOP), etc.), and the remaining step(s) or sections are NA'd, then an AR Type CR is not required.

NOTE: If some step(s) or section(s) of a surveillance procedure are to be performed for an operability retest following maintenance, (e.g., to re-perform a valve stroke time that was in the Alert range, etc.), and the remaining step(s) or sections are NA'd, then an AR Type CR is not required.

4.7.1 Use of Not Applicable (NA)

NOTE: NA is not to be used to bypass step(s) that are inadequately or improperly written and is not to be used in lieu of an approved change.

a. Continuous or Reference Use – Conditional Step(s) or Specific Condition Statements

MARK step(s) NA (no additional justification or approval is required)

b. Continuous or Reference Use – Non-Conditional Step(s) {C-7.6}

1) **VERIFY** ALL the following criteria are met prior to marking the step NA:

- The step or data entry to be NA'd is not a critical step or data entry (required for the successful completion of the procedure).
- The step is not needed for the current mode, condition or configuration of the plant.
- The intent (method of operation or the results) of the step(s) or sections does not change.
- An unsafe condition is not created.
- The initial conditions, precautions, or prerequisites sections are not violated; however, the Shift Manager may NA prerequisites that are not applicable.
- The method by which processes are performed that may affect quality (safety) structures, systems, and components is not changed without Shift Manager approval.

2) **DOCUMENT** the reason on the front cover or next to the step.

NOTE:	The purpose of the asterisk with the number is to tie the particular step to the associated comment on the front page. {C-7.6}

3) IF documenting the reason on the front cover, {C-7.6}
THEN **PLACE** an asterisk with a number on the cover page and by the step being NA'd
AND **INCLUDE** the reason.

NOTE:	Managers and Supervisors are responsible for ensuring they are technically knowledgeable of the task, or obtain concurrence from an individual who is technically knowledgeable, ensuring the explanation for marking the step(s) as NA is documented and appropriate.

4) **OBTAIN** supervisor initials to show approval, next to the explanation for the step being marked NA.

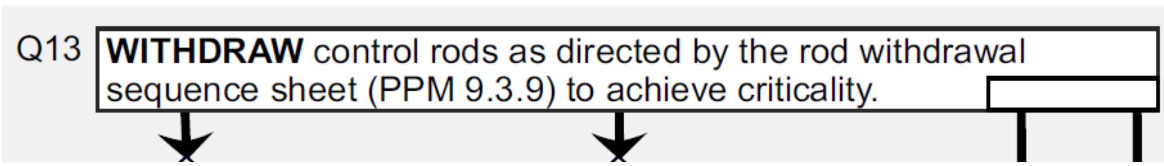
5) IF the procedure or work instruction is Continuous Use, {C-7.6}
pertain to Safety-Related equipment, Technical Specifications, or Licensee Controlled Specification procedure,
THEN **OBTAIN** Shift Manager initials to show approval next to the explanation.

Examination Outline Cross-reference:	Level	RO		SRO
Rev. 0 Date: 12/16/2016	Tier	3		
	Group			
	K/A	2.1.23		
Level of Difficulty: 3	Importance Rating	4.3		

Ability to perform specific system and integrated plant procedures during all modes of plant operation.	
Question # 68	

CGS is operating in Mode 3.

- A reactor startup is in progress in accordance with PPM 3.1.2, Start Up Flow Chart.
- The following step is being performed:



What indications are used to determine criticality?

Criticality shall be identified by increasing neutron level, …

A. a constant positive period and no change in feed flow.

B. no simultaneous control rod motion and no change in feed flow.

C. a constant positive period and no simultaneous control rod motion.

D. positive reactor period getting longer and no simultaneous control rod motion.

Answer: C	

K/A Match:
Question determines if candidates can correctly interpret and execute general (not overly specific) procedural requirements. Procedural step is from PPM 3.1.2, Plant Startup, which is an integrated plant procedure.

SRO Only:
N/A

Explanation:
Candidates must identify the correct indications for criticality which are a constant steady period and no simultaneous rod motion.

A. Incorrect. Feed flow is not used when determining criticality. Plausible because observing a constant steady period is required.

B. Incorrect. A constant steady period must be observed to verify criticality. Plausible because there should be no change in reactivity when validating a reactor is critical.

C. Correct.

D. Incorrect. Reactor period getting longer would indicate a subcritical reactor. Plausible because reactor period is observed and no rod motion is verified as part of the check for criticality.

Technical Reference(s)		Attached w/ Revision # See Comments / Reference
PPM 3.1.2, Startup Flowchart, Attachment 7.3		

Proposed references to be provided during examination: None

Learning Objective: 6651 – With procedures available, determine how criticality is identified.

Question Source: Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 41.10
55.43 ______

Comments / Reference: PPM 3.1.2, Attachment 7.3	Revision: Major 81 Minor 001

Notes

N1	The CRS/Shift Manager may authorize steps to be performed out of sequence or marked N/A to take into account current Plant configuration or conditions.
N2	MSIV wetting may be required prior to opening MSIVs. {2.6}
N3	Start daily performance of OSP-RRC-D701. Appropriate sections to be performed will be identified as plant conditions change.
N4	RHR-V-9 is required to be closed, with the RHR-V-9 power disconnect (RHR-DISC-V/9) in the OFF position in Modes 1,2,3 when RPV pressure is GT 135 psig.
N5	Placing the Mode switch in START/HOT STBY defines entry into MODE 2.
N6	Criticality usually occurs in the source range between $1x10^3$ and $1x10^4$ cps. For purposes of this procedure, criticality shall be identified by increasing neutron level, a constant steady period and no simultaneous control rod motion. {P-104550}

Examination Outline Cross-reference:	Level	RO		SRO
Rev. 0 Date: 12/16/2016	Tier	3		
	Group			
	K/A	2.2.6		
Level of Difficulty: 3	Importance Rating	3.0		

Knowledge of the process for making changes to procedures.

Question # 69

CGS is operating in Mode 3.

- A reactor startup is in progress.
- CRO2 is preparing to start RCC-P-1B per SOP-RRC-START.
- Prior to giving direction for OPS2 to close RCC-V-2B (pump discharge valve), CRO2 notes that the procedure contains an error. RCC-V-2B is incorrectly listed as RCC-V-2A in the procedure.

How should this error be resolved?

RCC startup may continue…

A. after receiving a peer check on the equipment EPN, revising the erroneous EPN using pen and ink, and initiating a condition report.

B. after an electronic Procedure Change Notification (PCN) has been initiated and approved by the Shift Manager and the Operations Manager.

C. after a Minor Revision Procedure Change Notification (PCN) has been initiated electronically and approved by the Plant Operations Committee (POC).

D. after receiving verbal approval for the procedure change (PCN) from the CRS and Shift Manager. Document the verbal approval in the procedure.

Answer: D

K/A Match:
Question presents an operationally valid situation where a minor error in the procedure needs to be resolved and determines if the candidates have knowledge of the process to do so

SRO Only:
N/A

Explanation:

Per SWP-PRO-02, once a verbal PCN has been completed per section 5.5.1, work may continue.

A. Incorrect. The PCN process should be used. Plausible because the procedural error is minor. Also, when a procedure is not currently in use and an error is identified, it is appropriate to write a CR.

B. Incorrect. Operations manager approval is not required for PCNs. Plausible because a PCN should be initiated to correct the procedural error.

C. Incorrect. POC review and approval is NOT required prior to continuing work provided the verbal PCN process is used. Plausible because POC approval may be required when submitting the procedure for revision depending on the significance of the change.

D. Correct.

Technical Reference(s)		Attached w/ Revision # See Comments / Reference
SWP-PRO-02 Preperation, Review, Approval and Distribution of Procedures.		

Proposed references to be provided during examination: None

Learning Objective: 6064 – State when a verbal temporary change to a procedure can be used.

Question Source: Bank # LO02784
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 41.10
55.43 ______

Comments / Reference: SWP-PRO-02	Revision: 45

Number: SWP-PRO-02	Use Category: INFORMATION	Major Rev: 045 Minor Rev: N/A Page: 31 of 63
Title: Preparation, Review, Approval and Distribution of Procedures		

h. IF there was a previous hold on the revision,
THEN **SELECT** Sign for EC Release
AND **SELECT** Sign
AND **ENTER** your Organization in the title Field
AND **ENTER** the appropriate credentials
AND **SELECT** Sign.

5.5 Procedure Change Notice (PCN)

> NOTE: Revisions may be processed afterhours in accordance with Section 5.9.
>
> NOTE: Revisions may be handwritten if an electronic format is not possible.

5.5.1 Verbal Procedure Change Notices (PCNs)

> NOTE: Applicable to physical work, testing or processes involving or directly affecting plant systems, structures or components: When, in the judgment of the Control Room Supervisor (CRS)/Shift Manager, the procedure changes are minor and easily understood and completion of the work is essential to Plant operation, the completion of the PCN paperwork may be delayed provided it has been approved verbally by the CRS/Shift Manager and one other member of the Management/Supervisory staff and it has been logged in the Control Room Log. {R-7.6}, {C-7.13}
>
> NOTE: Verbal deviations are allowed for administrative procedures (e.g., Site-Wides) when a Condition Report has been generated. No PCN is required.

a. **ENSURE** the minor changes do not alter the intent or scope of the procedure. {R-7.6}, {R-7.10}

b. The CRS/Shift Manager **AUTHORIZES** verbal PCNs related to station work.

c. **DOCUMENT** the verbal PCN by pen and ink or electronically, to the original document.

d. **DOCUMENT** verbal PCN within 12 hours in the Automated Procedure Process Workflows as described in Section 5.5.2. {C-7.13}

Examination Outline Cross-reference:	Level	RO		SRO
Rev. 0 Date: 12/16/2016	Tier	3		
	Group	2		
	K/A	2.2.35		
Level of Difficulty: 2	Importance Rating	3.6		

Ability to determine Technical Specification Mode of Operation

Question # 70

Given the following:

- All reactor vessel head closure bolts are fully tensioned.
- The mode switch in shutdown.
- Average reactor coolant temperature at 135 degrees.

The reactor is in Mode…

A. 2

B. 3

C. 4

D. 5

Answer: C

K/A Match:
The question provides plant conditions and determines if the candidate can select the correct MODE of operation based on those conditions.

SRO Only:
N/A

Explanation:
With temperature less than or equal to 200 degrees F and all vessel head closure bolts tensioned, the reactor vessel is in Mode 4.

A. Incorrect. Plausible if the mode switch was in “Refuel”.

B. Incorrect. Plausible if RCS temperature were above 200 degrees F.

C. Correct.

D. Incorrect. Plausible if one or more reactor vessel head closure bolts was less than fully tensioned.

Technical Reference(s)		Attached w/ Revision # See Comments / Reference
CGS Technical Specifications, Table 1.1-1, Modes		

Proposed references to be provided during examination: None

Learning Objective: 10297: Define: Mode 1, Mode 2, Mode 3, Mode 4, and Mode 5.

Question Source: Bank #
Modified Bank # LO01576 (Note changes or attach parent)
New

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 41.7, 41.10
55.43

Comments / Reference: CGS TS table 1.1-1, Modes	Revision: 237

1.1

Table 1.1-1 (page 1 of 1)
MODES

MODE	TITLE	REACTOR MODE SWITCH POSITION	AVERAGE REACTOR COOLANT TEMPERATURE (°F)
1	Power Operation	Run	NA
2	Startup	Refuel[a] or Startup/Hot Standby	NA
3	Hot Shutdown[a]	Shutdown	> 200
4	Cold Shutdown[a]	Shutdown	≤ 200
5	Refueling[b]	Shutdown or Refuel	NA

(a) All reactor vessel head closure bolts fully tensioned.

(b) One or more reactor vessel head closure bolts less than fully tensioned.

Comments / Reference: Question LO01576	Revision:

With the mode switch in shutdown and reactor pressure at 135 psig, the reactor would be in Mode:

A. 2

B. 3

C. 4

D. 5

Answer: B□

Examination Outline Cross-reference:	Level	RO		SRO
Rev. 0 Date: 12/16/2016	Tier	3		
	Group	2		
	K/A	2.2.12		
Level of Difficulty: 2	Importance Rating	3.7		

Knowledge of surveillance procedures.

Question # 71

When reading surveillance procedures, which symbol indicates steps that contain technical specification surveillance acceptance criteria?

A. #

B. *

C. $

D. →

Answer: A

K/A Match:
The question determines whether the candidate knows about important symbols contained throughout surveillance procedures. Since the symbol applies to all surveillance procedures, it is generic in nature and meets the K/A.

SRO Only:
N/A

Explanation:

SWP-PRO-03 states the following:

*“**IDENTIFY** steps that contain or satisfy Technical Specification Surveillance acceptance criteria with a # sign to the left of the step number.”*

A. Correct.

B. Incorrect. Plausible because it is a commonly used symbol in technical writing.

C. Incorrect. Plausible because it is a commonly used symbol (indicates ODCM and LCS criteria).

D. Incorrect. Plausible because it is a commonly used symbol in technical writing.

Technical Reference(s)		Attached w/ Revision # See Comments / Reference
SWP-PRO-03, Writer’s Manual		

Proposed references to be provided during examination: None

Learning Objective: 13559 - Given copies of plant procedures, locate and demonstrate an understanding of the procedural steps that apply to performing system tasks, including precautions and limitations, and Operations responsibilities.

Question Source: Bank # ___
Modified Bank # ___ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ___

10 CFR Part 55 Content: 55.41 41.10
55.43 ___

Comments / Reference: SWP-PRO-3	Revision: 22

Number: SWP-PRO-03	Use Category: INFORMATION	Major Rev: 022 Minor Rev: N/A Page: 22 of 71
Title: Writer's Manual		

2) **USE** Individual step sign-offs (initial blanks) in procedures to provide documentation of step completion or place-keeping during performance, if desired.

3) **ENSURE** sign-off spaces consist of 10 spaces underscored, with a right tab at 7.0″ at the end of the step or, if space is a problem, one hard return below the text.

4) **ENSURE** that when safety-related components are returned to service after maintenance or testing, verification is required. {R-6403}

5) IF it is important to document the date a step was performed, or to document the date of M&TE usage,
THEN **ENSURE** a date blank accompanies the initial blank.

6) **ENSURE** date blank consists of 10 spaces underscored, with a right tab set at 7.0" at the end of the step or, if space is a problem, one hard return below the text.

NOTE: A reviewer signature is usually provided on the coversheet.

7) WHEN performing work where step sign-offs document step completion,
THEN **ENSURE** a space for the signature of the procedure reviewer is provided.

8) **IDENTIFY** steps that contain or satisfy Technical Specification Surveillance acceptance criteria with a # sign to the left of the step number.

9) **IDENTIFY** individual steps that contain or satisfy Licensee Controlled

Examination Outline Cross-reference:	Level	RO		SRO
Rev. 0 Date: 12/16/2016	Tier	3		
	Group	3		
	K/A	2.3.12		
Level of Difficulty: 3	Importance Rating	3.2		

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.

Question # 72

CGS is in Mode 2, performing a reactor startup following a refueling outage.

- Reactor power is 5% and steady.
- An equipment configuration issue is identified which requires a containment entry to realign equipment.
- Two operators will enter to complete the equipment lineup.

Who must give permission for this containment entry?

Permission for containment entry is <u>required</u> from the Shift Manager (SM)…

A. only.

B. and Plant General Manager (PGM) only.

C. and Radiation Protection Manager (RPM) only.

D. and Radiation Protection Manager (RPM) and Plant General Manager (PGM) only.

Answer: D

K/A Match:

The K/A asks for knowledge of radiation principles, such as containment entry requirements. This question requires the candidate to differentiate between the required approvals for various containment entries.

SRO Only:

N/A

Explanation:

A. Incorrect. Plausible if the containment entry was not the initial containment entry. However, since this the first entry since de-inerting containment following a reactor startup (see Ref. A, section 2.4 below), this entry is classified as an initial entry and additional permissions are required.

B. Incorrect. Plausible if this containment entry is being performed at power. However, plant conditions given show the reactor is shutdown. Therefore, plant general manager permission to enter containment is not required.

C. Incorrect. Per Ref. A, step 6.1.5, initial containment entry with the reactor shutdown requires Shift Manager and Radiation Protection Manager permission only.

D. Correct. Step 4.1.2 or Ref. A requires Plant General Manager's permission to enter containment at power.

Technical Reference(s)	
SOP-ENTRY-DW, Personnel Entry Into Drywell	Attached w/ Revision # See Comments / Reference
PPM 11.2.7.3, High Radiation Area, Locked High Radiation Area, and Very High Radiation Area Controls	

Proposed references to be provided during examination: None

Learning Objective: 13262 – Identify the requirements necessary to coordinate an initial DW entry.

Question Source: Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 41.12
55.43 ______

Comments / Reference: SOP-ENTRY-DW, section 6.1, Drywell Entry Preparation, step 6.1.5	Revision: Major: 023, Minor 001

Number: SOP-ENTRY-DW	Use Category: CONTINUOUS	Major Rev: 023 Minor Rev: 001 Page: 17 of 43
Title: Personnel Entry Into Drywell		

- The following conditions which will warrant immediate evacuation of the Drywell.
  - Radiological stay time is in danger of being exceeded _____
  - Oxygen or combustible gas concentrations exceed acceptable limits _____
  - HP Technician judges radiation levels excessive _____
  - Plant status change which could threaten the safety of the Initial Entry Team _____
  - Any member of the Initial Entry Team is injured _____
  - Any other signs of problems with personnel or equipment are encountered that would warrant evacuation _____

6.1.5 **OBTAIN** approval from the following individuals:

- CRS/Shift Manager _____ CRS/SM

> NOTE: The Shift Manager may obtain the following two approvals by telephone and sign for them.

- Radiation Protection Manager _____ RPM
- IF Reactor is critical,
  THEN approval from Plant General Manager. _____ PGM

Comments / Reference: A, section 6.3, Subsequent Drywell Entry, step 6.3.3	Revision: Major: 023, Minor 001

Number: SOP-ENTRY-DW	Use Category: CONTINUOUS	Major Rev: 023 Minor Rev: 001 Page: 22 of 43
Title: Personnel Entry Into Drywell		

6.3.3 **PERFORM** the following:

- **OBTAIN** approval from CRS/Shift Manager. ______ CRS/SM
- **VERIFY** an RWP is available for the Drywell entry. ______
- IF breaking Primary Containment Integrity,
  THEN **INITIATE** Barrier Impairment permit per PPM 1.3.57. {P-158665} ______
- IF required,
  THEN **ESTABLISH** the Drywell as a Foreign Material Exclusion Boundary. (This requires logging all tools and equipment in and out of the Drywell per PPM 10.1.13). {R-2507}, {P-113100} ______

CAUTION

Failure to follow the guidance provided in SOP-DOOR/HATCH-OPS when operating the Airlock Doors can cause severe damage to the operating mechanism and interlock.

- **OPEN** the Outer Airlock Door per SOP-DOOR/HATCH-OPS. ______
- IF required,
  THEN **VERIFY** all Entry Team members (including Attendant) **DON** respiratory equipment. ______
- **VERIFY** all Entry Team members **DON** the required protective clothing. ______

6.3.4 All personnel making the Subsequent Entry and the Airlock Operator **ENTER** the Airlock. ______

6.3.5 **CLOSE** Outer Airlock Door. ______

6.3.6 **OPEN** the Inner Airlock Door per SOP-DOOR/HATCH-OPS. ______

Comments / Reference: SOP-ENTRY-DW, section 2.4	Revision: Major: 023, Minor: 001

Number: SOP-ENTRY-DW	Use Category: CONTINUOUS	Major Rev: 023 Minor Rev: 001 Page: 4 of 43
Title: Personnel Entry Into Drywell		

1.0 PURPOSE

Provide a safe method for Operations and other personnel to enter the Drywell, both for Initial Entry and Subsequent Entry. As this is a stand-alone procedure, ISPM-3, Confined Space Entry, does not need to be performed concurrently with this procedure.

2.0 DEFINITIONS

2.1 Airlock Operator - Qualified Operator that remains in the Airlock during the entire time other members of the Initial Entry Team are inside the Drywell. This Operator should be the only person to operate the Airlock doors and should also serve as the communications link between the personnel inside the Drywell and the Attendant.

2.2 Attendant - This person is present outside the Drywell Access Point, monitoring entry activities This person will be in communication with the Airlock Operator during Initial Entry, and will call for assistance if an emergency condition exists.

2.3 Drywell Entry - Drywell Entry is defined as any time the Inner Drywell Door is open.

2.4 Initial Entry - First entry into the Drywell following de-inerting or the Drywell has been closed for GT 24 hours and Containment purging/ventilation is secured.

Comments / Reference: PPM 11.2.7.3, section 6.6, steps 6.6.1 & 6.6.2	Revision: Major: 041, Minor: 001

Number: 11.2.7.3	Use Category: INFORMATION	Major Rev: 041 Minor Rev: 001 Page: 20 of 35
Title: High Radiation Area, Locked High Radiation Area, and Very High Radiation Area Controls		

6.6 Controlling Access to Very High Radiation Areas

6.6.1 Barricades, locks, and postings to prevent unauthorized and inadvertent entry

a. Each entryway to a VHRA shall be conspicuously posted as outlined in PPM 11.2.7.1, and shall be provided with a locked or continuously guarded door or gate that prevents unauthorized or inadvertent entry. The door or gate locks shall consist of at least two locks, one of which shall be a VHRA lock controlled only by RP. {R-81}

b. Physical barriers used for preventing access to a VHRA should provide a greater level of control than those for a HRA or LHRA. They should be installed such that the area is completely surrounded and inadvertent entry cannot be made. For example, a six-foot high fence may be adequate to barrier a HRA or LHRA, but a fence used to barricade a VHRA should extend all the way to the ceiling of the area, if this is necessary to provide the level of control required.

c. VHRA controls and posting should be implemented prior to starting operations with the potential for creating VHRAs. Reasonable provisions should be made to anticipate, control, and provide for ready evacuation of these areas. These areas may include:

1) Drywell when the reactor is at power. No entries should be made when the reactor is at GTE 10% power.

2) Locations where highly radioactive materials may be accessible to personnel, such as spent fuel storage or radwaste processing containers. These should be evaluated on a case-by case basis.

d. VHRA controls and posting need not be implemented for highly radioactive materials in the Spent Fuel Pool, Reactor Cavity, Equipment Pool where the materials are submerged and inaccessible to personnel in the area, if measures are in place to ensure that the materials are not inadvertently raised near the surface or out of the water.

6.6.2 Unlocking a VHRA

a. The requirements for unlocking a VHRA are the same as those outlined for unlocking a LHRA, with the following differences:

1) Issuance of the VHRA key is documented in a VHRA Key Log (Form 26436) rather than the LHRA Key Log.

2) The Plant General Manager (PGM) and RPM should authorize issuance of the VHRA key.

Comments / Reference: PPM 11.2.7.3, section 6.6, step 6.6.3	Revision: Major: 041, Minor: 001

Number: 11.2.7.3	Use Category: INFORMATION	Major Rev: 041 Minor Rev: 001 Page: 21 of 35
Title: High Radiation Area, Locked High Radiation Area, and Very High Radiation Area Controls		

3) Identify the individuals (i.e., PGM and RPM) authorizing issue of the VHRA key, by name, in the VHRA Key Log. {C-9694}

6.6.3 Requirements for entry into a VHRA

a. Access to VHRAs is prohibited except under the following circumstances:

1) Initial entries made by individuals qualified in radiation protection procedures, and personnel continuously escorted by such individuals, for the purpose of surveying and down-posting the VHRA.
2) For an operational or safety reason as specifically authorized and approved by the PGM and RPM.

Examination Outline Cross-reference:	Level	RO		SRO
Rev. 0 Date: 12/16/2016	Tier	3		
	Group	3		
	K/A	2.3.14		
Level of Difficulty: 2	Importance Rating	3.4		

Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities.

Question # 73

CGS is in Mode 4.

- An event requires operators to enter the Reactor Building to perform a system isolation that will protect some valuable equipment.
- An individual has volunteered to perform the task.

At what classification are Energy Northwest's administrative exposure hold points waived and, what is the maximum dose that the Emergency Director may authorize the individual to receive?

A. Alert; 10 rem TEDE

B. Unusual Event; 5 rem TEDE

C. Alert; 25 rem TEDE

D. Site Area Emergency; 50 rem TEDE

Answer: A

K/A Match:
This question requires the candidate to demonstrate knowledge of the emergency allowable dose limits and when the Energy Northwest dose hold points are waived.

SRO Only:
N/A

Explanation:

A. Correct. A total dose of 10 rem TEDE may be authorized to save valuable equipment. Energy Northwest administrative exposure hold points are automatically waived at an Alert classification.

B. Incorrect. Plausible since 5 rem is the normal federal exposure limit. However, a total dose of 10 rem TEDE may be authorized to save valuable equipment.

C. Incorrect. Plausible since 25 rem is the emergency exposure limit for life saving. However, a total dose of 10 rem TEDE may be authorized to save valuable equipment.

D. Incorrect. Plausible since > 25 rem is the emergency exposure limit for life saving on a volunteer basis. However, a total dose of 10 rem TEDE may be authorized to save valuable equipment.

Technical Reference(s)		Attached w/ Revision # See Comments / Reference
PPM 13.2.1, Emergency Exposure Levels / Protective Action Guides		

Proposed references to be provided during examination: None

Learning Objective: 11258 – Knowledge of the facility ALARA program

Question Source: Bank # LO01923
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam 2009, #23

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 41.2
55.43 ______

Comments / Reference: PPM 13.2.1, section 2.2, Emergency Exposure Controls	Revision: 22

Number: 13.2.1	Use Category: REFERENCE	Major Rev: 022 Minor Rev: N/A Page: 5 of 18
Title: Emergency Exposure Levels / Protective Action Guides		

1.0 PURPOSE

This procedure outlines the authority and process for exceeding annual administrative exposure holdpoints and implementing EPA-400 limits for emergency worker Protective Action Guides (PAGs). Additionally, it provides guidance for administration of potassium iodide (KI) and authorization of Emergency Exposures above EPA-400 limits during emergency situations. {R-1599}

2.0 DISCUSSION

2.1 Precautions and Limitations

2.1.1 If respiratory protection equipment is not prescribed, administer potassium iodide (KI) as outlined in Attachment 7.3. {2.1}

2.2 Emergency Exposure Controls

2.2.1 Refer to Attachment 7.1, Federal Personnel Dose Limits (10 CFR 20) for normal worker dose limits.

2.2.2 Declaration of an Alert or higher emergency classification automatically waives Energy Northwest administrative exposure hold points.

Comments / Reference: PPM 13.2.1, EPA 400 Protective Action Guides for Emergency Workers	Revision: 22

Number: 13.2.1	Use Category: REFERENCE	Major Rev: 022 Minor Rev: N/A Page: 11 of 18
Title: Emergency Exposure Levels / Protective Action Guides		

EPA 400 PROTECTIVE ACTION GUIDES FOR EMERGENCY WORKERS

DOSE LIMIT (TEDE)+	ACTIVITY	PROTECTIVE ACTIONS
5 rem	ALL	
10 rem	PROTECTING VALUABLE PROPERTY	Lower dose not practicable
25 rem	LIFE-SAVING OR PROTECTION OF LARGE POPULATIONS	Lower dose not practicable
>25 rem	LIFE-SAVING OR PROTECTION OF LARGE POPULATIONS	Only on a voluntary basis to persons fully aware of the risks involved.

Refer to Attachment 7.3 for information concerning the administration of Potassium Iodide (KI).

\+ Sum of external effective dose equivalent and committed effective dose equivalent to nonpregnant adults from exposure and intake during an emergency situation. Workers performing services during emergencies should limit dose to the lens of the eye to three times the listed value and doses to any other organ (including skin and body extremities) to ten times the listed value. These limits apply to all doses from an incident, except those received in an unrestricted area as members of the public during the intermediate phase of the incident.

Examination Outline Cross-reference:	Level	RO		SRO
Rev. 0 Date: 12/16/2016	Tier	3		
	Group	4		
	K/A	2.4.4		
Level of Difficulty: 3	Importance Rating	4.5		

Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures.

Question # 74

The plant is in MODE 1.

- The crew has identified a slow increase in drywell unidentified leakage.
- Drywell pressure: 0.7 psig, up slow.

A loss of IN-1 occurs, resulting in a loss of power to US-PP.

- At a Drywell pressure of 1.0 psig, the CRS orders a reactor scram.
- RPV level: -20 inches, up slow.
- Drywell pressure: 1.5 psig, up slow
- Drywell temperature: 125°F, up slow
- Reactor Building exhaust plenum radiation level: 2.5 mr/hr, up slow.

Which EOP(s) should be entered?

A. Enter PPM 5.1.1 RPV Control only.

B. Enter PPM 5.1.1 RPV Control and PPM 5.2.1 Primary Containment Control.

C. Enter PPM 5.1.1 RPV Control and transition to PPM 5.1.2 RPV Control-ATWS.

D. Enter PPM 5.1.1 RPV Control and PPM 5.3.1 Secondary Containment Control.

Answer: A

K/A Match:
This question requires the candidate to demonstrate knowledge of EOP entry conditions.

SRO Only:
N/A.

Explanation:

A. Correct. Entry into PPM 5.1.1 is required due to RPV level below +13 inches.

B. Incorrect. Plausible since Drywell pressure and temperature are rising. However, both parameters do not meet the requirements to enter PPM 5.2.1.

C. Incorrect. Plausible since the loss of IN-1 will cause a loss of the full-core display. However, all rod positions may be verified with the Rod Worth Minimizer (RWM) and rods may be verified to be fully inserted. Therefore, entry into PPM 5.1.2 is not required.

D. Incorrect. Plausible since RB exhaust plenum radiation level is rising. However, the level does not meet the requirement to enter PPM 5.3.1.

Technical Reference(s)		Attached w/ Revision # See Comments / Reference
ABN-ELEC-INV, 120 VAC Critical Distribution System Failures		
PPM 5.1.1, RPV Control		

Proposed references to be provided during examination: None

Learning Objective: 8017: Given plant conditions, recognize an EOP entry condition(s) and enter the appropriate flow chart.

Question Source: Bank # LR00114
Modified Bank # ____ (Note changes or attach parent)
New ____

Question History: Last NRC Exam ____

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ____

10 CFR Part 55 Content: 55.41 Best match only
55.43 ____

Comments / Reference: ABN-ELEC-INV	Revision: Major 13 Minor 001

Number: ABN-ELEC-INV	Use Category: CONTINUOUS	Major Rev: 013 Minor Rev: 001 Page: 3 of 17
Title: 120 VAC Critical Distribution System Failures		

1.0 ENTRY CONDITIONS

1.1 E-IN-1 Loss of IN-1 output voltage any time IN-1 is required to be in operation.

1.2 E-IN-2A or E-IN-2B Loss of IN-2A or IN-2B output voltage any time E-IN-2A or IN-2B is required to be in operation.

1.3 E-IN-3A or E-IN-3B Loss of IN-3A or IN-3B output voltage any time IN-3A or IN-3B is required to be in operation.

1.4 E-IN-5 Loss of E-IN-5 output voltage any time IN-5 is required to be in operation.

2.0 AUTOMATIC ACTIONS

2.1 IN-1

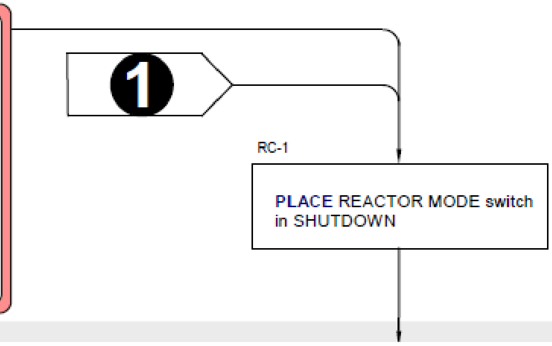
2.1.1 Loss of IN-1 output causes the static switch to transfer to ALT AC Input (E-MC-7F).

2.1.2 If the inverter failed to auto transfer, the following actions will occur, due to a loss of E-PP-US.

- Feedwater heater controllers lose power and the valves fail open
- Power is lost to the Full Core Display.
- DEH will lose Primary Power Supply
- Power is lost to OG-RIS-601A
- Loss of power to multiple rad monitors on Rad Board 24
- Power is lost to RDCS preventing normal rod motion except by scram.

Comments / Reference: PPM 5.1.1 Entry Requirements	Revision: 021

- RPV level below **+13 in.**
- RPV pressure above **1060 psig**
- Drywell pressure above **1.68 psig**
- Both:
  - a reactor scram is required
  - AND
  - reactor power is above **5%** or cannot be determined



Examination Outline Cross-reference:	Level	RO		SRO
Rev. 0 Date: 12/16/2016	Tier	3		
	Group	4		
	K/A		2.4.5	
Level of Difficulty: 3	Importance Rating	3.7		

Type the K/A System or Condition Here: Knowledge of the organization of the operating procedures network for normal, abnormal, and emergency evolutions.

Question # 75

CGS was operating in Mode 1 when a condition caused the crew to enter an abnormal procedure (ABN). Conditions degraded such that the crew initiated a manual reactor scram and entered the reactor scram procedure and an emergency operating procedure (EOP) prior to completing all steps in the ABN.

Which of the following is correct concerning continuing execution of ABN steps?

Steps in the ABN may be executed…

A. concurrently with the EOP ONLY IF specifically called out by the EOP.

B. concurrently with the EOP ONLY IF ABN actions do not conflict with EOP actions.

C. ONLY AFTER the shift manager has determined that an emergency no longer exists.

D. ONLY AFTER applicable steps of the reactor scram procedure have been completed.

Answer: B

K/A Match:

This questions tests the candidate's knowledge of the hierarchy of use for operating procedures, including emergency, abnormal and normal procedures.

SRO Only:

N/A

Explanation:

A. Incorrect. Plausible since Ref. A, section 4.8.3.a states that *"All required concurrent execution of any Volume 2, 3, or 4 Procedure are specifically called out by the EOPs"*. However, the section continues by stating *"This does not mean that a Volume 4, Abnormal Procedure, cannot otherwise be concurrently executed with the EOPs so long as its specified actions do not conflict with the direction given by the EOPs".* Therefore, ABN actions may be performed concurrently with EOP actions without being specifically referenced in the EOP.

B. Correct. As delineated in Ref. A, section 4.8.3.a, Abnormal procedure (ABN) actions may be executed concurrently with Emergency procedure (EOP) actions as long as the ABN actions *"do not conflict with the direction given by the EOPs".*

C. Incorrect. Plausible since Ref. A, section 4.8.1.a states *"The Volume 5 Emergency Operating Procedures (EOPs) and the actions specified therein have priority/precedence over all Volume 2, 3, and 4 Procedures when an emergency exists (EOP entry condition(s) is/are met)"*. This statement implies that once an EOP is entered, actions from lower tier procedures are stopped. Additionally, Ref. A, section 4.8.2.e states *"EOPs are exited only if the Shift Manager determines that an emergency no longer exists and directs EOP exit or the EOPs direct exit to appropriate plant procedures"*. This distractor infers that the EOPs must be exited prior to completing ABN actions. This is not the case, however. See explanation for answer 'B' above.

D. Incorrect. Plausible since Ref. A, section 4.8.1.b states *"During a transient, and as plant conditions continue to degrade, the flow path of procedure usage is from the Volume 2 and 3 Procedures to the Volume 4, Abnormal Procedures, and then to the EOPs"*. This infers that actions from a Volume 3, General Operating Procedure, such as PPM 3.3.1, Reactor Scram, should be completed prior to completing actions from a Volume 4 procedure, such as ABNs. This is not the case. See the explanation for answers 'B' and 'C' above.

Technical Reference(s)		Attached w/ Revision # See Comments / Reference
PPM 1.3.1, Operating Policies, Programs and Practices		

Proposed references to be provided during examination: None

Learning Objective: 6105 – State which procedures have priority/precedence over all other operating procedures when an emergency exists.

Question Source: Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 41.10
55.43 ______

Comments / Reference: PPM 1.3.1, section 4.8.3, Procedure Use During an Emergency	Revision: 120, Minor 003

Number: 1.3.1	Use Category: INFORMATION	Major Rev: 120 Minor Rev: 003 Page: 26 of 98
Title: Operating Policies, Programs and Practices		

4.8.3 Procedure Usage During an Emergency

a. The EOPs direct the operator to execute a particular section or an entire Volume 2, 3 or 4 Procedure. All required concurrent execution of any Volume 2, 3, or 4 Procedure are specifically called out by the EOPs.

This does not mean that a Volume 4, Abnormal Procedure, cannot otherwise be concurrently executed with the EOPs so long as its specified actions do not conflict with the direction given by the EOPs.

b. Once EPIPs have been entered (Emergency Classification occurs), recovery actions not specifically authorized by plant procedures which have a potential for radioactive release to the environment require Emergency Director concurrence (this includes items listed in FAZ Procedures).

c. If it becomes necessary to take actions other than those prescribed by the License Basis Documents or by plant procedures; (to protect the immediate health and safety of the public) those actions shall be implemented using 10 CFR 50.54(x) per Section 4.3.

d. If a transient is in progress or an emergency condition dictates, take immediate corrective action up to and including power reduction or SCRAM to stabilize the situation. Then, as soon as possible, refer to appropriate procedures. Appropriate times to exercise this option include:

- To maintain the safety of the public
- To protect the safety of individuals (e.g., deenergize equipment that is causing an electrocution or stop a leak that is endangering an individual)
- To preclude unnecessary equipment damage (e.g., stop a piece of equipment that is quickly degrading or is in imminent danger)
- SCRAM prior to exceeding an RPS limit

Comments / Reference: PPM 1.3.1, section 4.8.4, Abnormal Procedures	Revision: 120, Minor 003

Number: 1.3.1	Use Category: INFORMATION	Major Rev: 120 Minor Rev: 003 Page: 27 of 98
Title: Operating Policies, Programs and Practices		

4.8.4 Abnormal Procedures

NOTE: These procedures are not Emergency procedures, but may be executed concurrently with EOPs if, following diagnosis of plant conditions, the Abnormal Procedure is applicable. If any conflict in the required operator actions exist between the Abnormal Procedure and the EOPs, the actions specified in the EOPs should have priority/precedence. If a Control Room evacuation is required, ABN-CR-EVAC will supersede the EOPs, once the Control Room is evacuated.

NOTE: Refer to PPM 1.3.82 for reportability requirements. if a plant transient or equipment issue has resulted in a change in CGS real output generating capability.

a. Abnormal Procedures specify operator actions for restoring equipment or systems to their normal controlled status upon a failure or to restore normal operating conditions following a perturbation.

b. If an abnormal condition exists, immediate operator actions shall be implemented from memory. These procedures should be referred to as soon as practical following stabilization of the plant to ensure immediate actions have been performed and to execute subsequent operator actions.

c. Abnormal procedure subsequent actions should be performed in order. Subsequent operator actions initially deemed not applicable based on plant conditions should be reevaluated for applicability, and performed as appropriate, as plant conditions change.

d. If an abnormal condition is corrected prior to procedure implementation, the abnormal procedure should be referred to and implemented, to ensure that the system or component is returned to normal operation and that all required actions have been completed.

e. When the abnormal condition no longer exists, the appropriate system operating procedures should be entered.

f. Abnormal procedures may also have an associated bases document. The bases document provides the reason for the steps performed in the abnormal procedure. The bases document is for information only and is not intended to be used for directing plant actions.

Comments / Reference: PPM 1.3.1, section 4.8.1, Procedure Hierarchy	Revision: 120, Minor 005

Number: 1.3.1	Use Category: INFORMATION	Major Rev: 120 Minor Rev: 005 Page: 25 of 98
Title: Operating Policies, Programs and Practices		

4.8 Procedure Usage

> NOTE: Use and adherence to approved Plant Procedures is controlled by the SWP-PRO-01.

4.8.1 Procedure Hierarchy

a. The Volume 5 Emergency Operating Procedures (EOPs) and the actions specified therein have priority/precedence over all Volume 2, 3, and 4 Procedures when an emergency exists (EOP entry condition(s) is/are met).

b. During a transient, and as plant conditions continue to degrade, the flow path of procedure usage is from the Volume 2 and 3 Procedures to the Volume 4, Abnormal Procedures, and then to the EOPs.

Comments / Reference: PPM 1.3.1, section 4.8.2, Volume 5 Emergency Operating Procedures (EOPs)	Revision: 120, Minor 005

4.8.2 Volume 5 Emergency Operating Procedures (EOPs)

NOTE: EOPs are symptom oriented rather then event based procedures. Operators need not understand the event in order to successfully respond using symptom based EOPs.

a. Addresses other than normal conditions which, if not corrected, could result in core damage, loss of containment integrity or an uncontrolled radioactivity release to the environment.

b. These procedures provide the necessary direction to shutdown and cooldown the reactor during emergencies and should be entered whenever any EOP entry condition exists.

c. If following entry into an EOP, an additional entry condition is met, the EOP should be reentered at the beginning.

d. Operator Actions are given in a logical sequence and should be performed in order, if possible.

- Conditional WHEN/THEN statements act as stop signs and subsequent actions do not have to be performed until the condition has been met
- Conditional IF/THEN statements act as stop signs and the condition should be determined before proceeding
- Command steps may be performed in parallel, when appropriate
- Overrides should be frequently checked since they become effective when read and remain effective in the flow chart area identified.

e. EOPs are exited only if the Shift Manager determines that an emergency no longer exists and directs EOP exit or the EOPs direct exit to appropriate plant procedures

Examination Outline Cross-reference:	Level	RO		SRO
Rev. 0 Date: 12/16/2016	Tier			1
	Group			1
	K/A	295001.AA2.03		
Level of Difficulty: 3	Importance Rating			3.3

<u>PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION</u>: Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION: Actual core flow

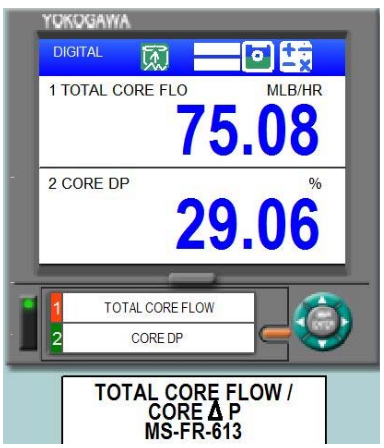
Question # 76

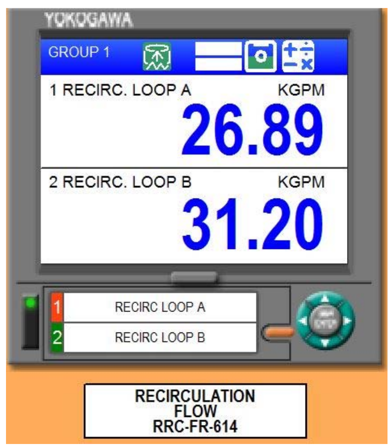
CGS is in Mode 1. Reactor power is 100%.

An event causes both RRC pumps to run back.

Current conditions:

- Reactor power: 80%
- Core flow:





What is the <u>earliest</u> action required to satisfy Technical Specifications?

A. Declare RRC Loop ‘A’ inoperable in 2 hours.

B. Shutdown the reactor to Mode 3 in 12 hours.

C. Reduce reactor power to less than 70% in 14 hours.

D. Secure RRC-P-1A and enter single loop operation within 16 hours.

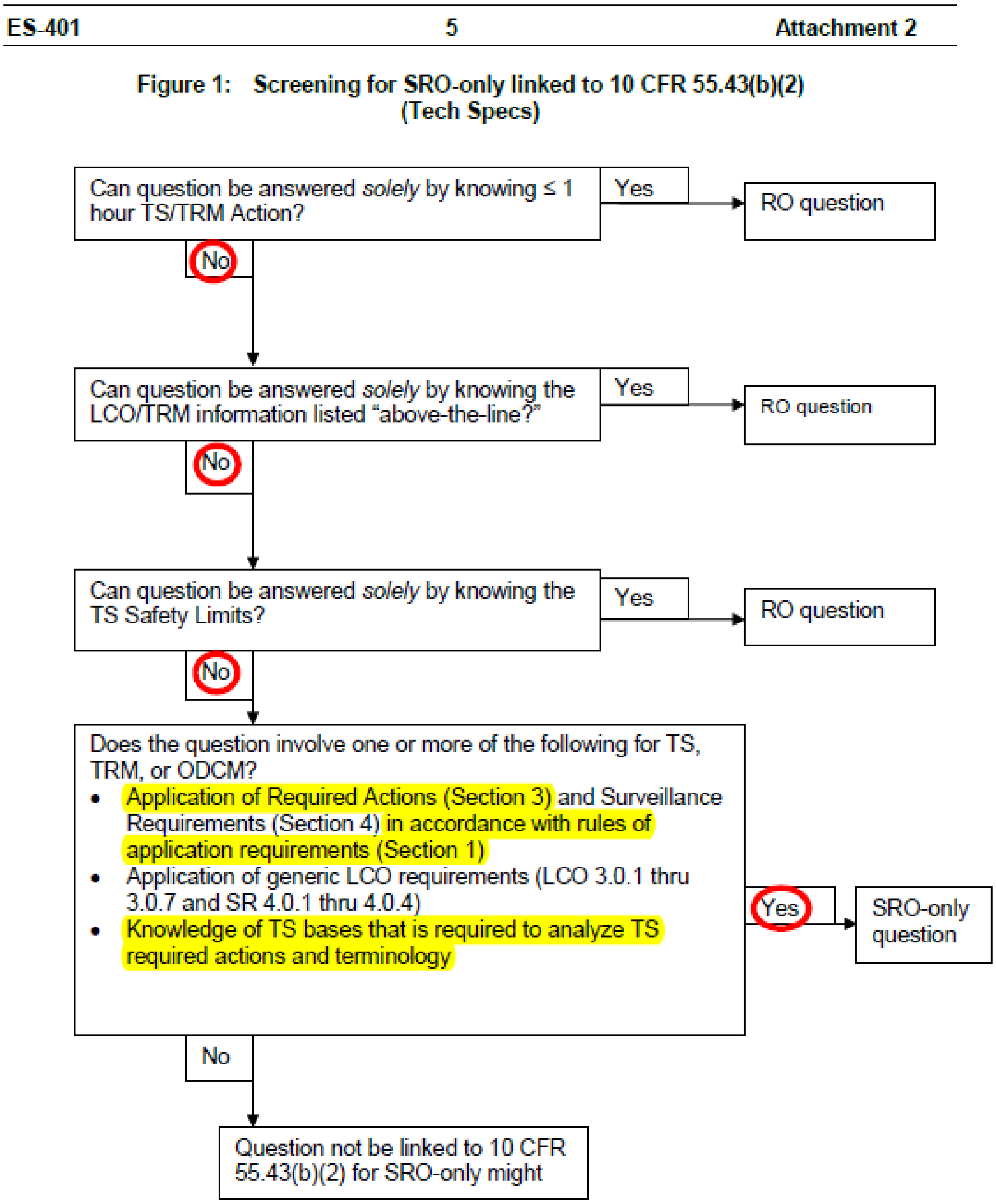
Answer: A

K/A Match:

Requires the candidate to interpret core flow indications during a partial loss of core flow and determine actions necessary to mitigate the casualty.

SRO Only:

K/A is an "A2" Statement.



Explanation:

A. Correct. Total core flow is below 70% of rated core flow (75.95 Mlb/hr). In accordance with TS bases (Ref. B), and the daily surveillance (Ref. C), Jet Loop Flow mismatch must be LE 4172 gpm. From indications in the stem, the mismatch is 4310 gpm, which does not meet the criterion established in surveillance 3.4.1.1. Condition A of LCO 3.4.1 (Ref. A) applies, and the Loop with the lowest flow (Loop A) should be declared inoperable within 2 hours.

B. Incorrect. In accordance with LCO 3.4.1, condition C (Ref. A), placing the reactor in Mode 3 is a correct action if conditions A & B are not complete. However, the stem asked for the earliest action to meet TS, which is answer ‘A’.

C. Incorrect. Plausible if candidate believes that loop flow deviation limit of LE 10% is applicable > 70% power vice 70% rated flow.

D. Incorrect. Plausible since securing RRC-P-1A and entering single loop operations would meet the requirements of LCO 3.4.1. However, However, the stem asked for the earliest action to meet TS, which is answer ‘A’.

Technical Reference(s)		
A	Technical Specification 3.4.1, Recirculation Loops Operating	Attached w/ Revision # See Comments / Reference
B	Technical Specifications Bases for TS 3.4.1	
C	OSP-RRC-D701, Jet Pump Operability and Recirculation Loop Flow Mismatch, Two Loop Operation	

Proposed references to be provided during examination: None

Learning Objective: 5031 - Referencing Columbia Generating Station Technical Specifications associated with the Reactor Recirculation System and a set of plant conditions, determine as applicable the LSSS, the LCO, the action statement, and the appropriate bases.

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 ____
55.43 43.5

Comments / Reference: TS 3.4.1, Page 1	Revision: Amendment 237

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 Recirculation Loops Operating

LCO 3.4.1 Two recirculation loops with matched flows shall be in operation.

OR

One recirculation loop shall be in operation provided that the following limits are applied when the associated LCO is applicable:

a. LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," single loop operation limits specified in the COLR;

b. LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," single loop operation limits specified in the COLR; and

c. LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," Function 2.b (Average Power Range Monitors, Simulated Thermal Power - High), Allowable Value of Table 3.3.1.1-1 is reset for single loop operation.

APPLICABILITY: MODES 1 and 2 after implementation of Power Range Neutron Monitor (PRNM) upgrade.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Recirculation loop flow mismatch not within limits.	A.1	Declare the recirculation loop with lower flow to be "not in operation."	2 hours
B. Requirements of the LCO not met for reasons other than Condition A.	B.1	Satisfy the requirements of the LCO.	4 hours

Comments / Reference: TS 3.4.1, Page 2	Revision: Amendment 237

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A or B not met. <u>OR</u> No recirculation loops in operation.	C.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 drive flow mismatch with both recirculation loops in operation is: a. $\leq$ 10% of rated recirculation loop drive flow when operating at < 70% of rated core flow; and b. $\leq$ 5% of rated recirculation loop drive flow when operating at $\geq$ 70% of rated core flow.	24 hours

Comments / Reference: TS Bases for TS 3.4.1	Revision: 92

BASES

ACTIONS (continued)

reverse flow is detected, the condition should be alleviated by changing pump speeds to re-establish forward flow or by tripping the pump.

With the requirements of the LCO not met for reasons other than Condition A (e.g., one loop is "not in operation"), the recirculation loops must be restored to operation with matched flows within 4 hours. A recirculation loop is considered not in operation when the pump in that loop is idle or when the mismatch between total jet pump flows of the two loops is greater than required limits for greater than 2 hours (i.e., Required Action A.1 has been taken). Should a LOCA occur with one recirculation loop not in operation, the core flow coastdown and resultant core response may not be bounded by the LOCA analyses. Therefore, only a limited time is allowed to restore the inoperable loop to operating status.

Alternatively, if the single loop requirements of the LCO are applied to operating limits and RPS setpoints, operation with only one recirculation loop would satisfy the requirements of the LCO and the initial conditions of the accident sequence.

The 2 and 4 hour Completion Times are based on the low probability of an accident occurring during this time period, on a reasonable time to complete the Required Action, and on frequent core monitoring by operators allowing abrupt changes in core flow conditions to be quickly detected.

C.1

With the Required Action and associated Completion Time of Condition A or B not met, the unit is required to be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 12 hours. In this condition, the recirculation loops are not required to be operating because of the reduced severity of DBAs and minimal dependence on the recirculation loop coastdown characteristics. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.1.1

This SR ensures the recirculation loop flows are within the allowable limits for mismatch. At low core flow (i.e., < 70% of rated core flow, 75.95 x 10^6 lbm/hr), the MCPR requirements provide larger margins to the fuel cladding integrity Safety Limit such that the potential adverse effect of

Comments / Reference: OSP-RRC-D701, Daily Surveillance	Revision: Major 018, Minor 002

Number: OSP-RRC-D701	Use Category: CONTINUOUS	Major Rev: 018 Minor Rev: 002 Page: 8 of 18
Title: Jet Pump Operability and Recirculation Loop Flow Mismatch, Two Loop Operation		

7.2 **CALCULATE** Recirculation Loop Flow Mismatch as follows:

7.2.1 IF Plant Process Computer is operable,
THEN **PERFORM** the following: N/A if already performed.

a. **DEMAND** Option (L2) JET PUMP OPERABILITY from the logging and report menu. ______

b. **OBTAIN** a hard copy of the display (by entering a CTRL P, or print from Plant Process Computer Terminal), ______
AND **ATTACH** the hard copy to the procedure. ______

7.2.2 IF Plant Process Computer is inoperable,
THEN **RECORD** the following information (from MS-FR-613, RRC-FI-676A(B), TDAS pts. X036, X498 or X505, sum of TDAS pts (X034 + X049), or other instruments monitoring same parameter).

- Total Core Flow: __________ Instrument Used: __________ ______
- Indicated Loop A Drive Flow: __________ Instrument Used: __________ ______
- Indicated Loop B Drive Flow: __________ Instrument Used: __________ ______
- Loop Flow Mismatch:

  $|\text{Loop A flow} - \text{Loop B flow}|$ = __________ gpm ______

\# 7.2.3 IF Total Core Flow is GE 75.95 Mlbm/hr (70% of rated core flow),
THEN **VERIFY** loop flow mismatch is LE 2086 gpm. ______

\# 7.2.4 IF Total Core Flow is LT 75.95 Mlbm/hr (70% of rated core flow),
THEN **VERIFY** loop flow mismatch is LE 4172 gpm. ______

Examination Outline Cross-reference:	Level	RO		SRO
Rev. 0 Date: 12/16/2016	Tier			1
	Group			1
	K/A	295003.2.2.36		
Level of Difficulty: 3	Importance Rating			4.2

Partial or Complete Loss of AC: Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operations.

Question # 77

CGS is in Mode 1. DG-2 is in a maintenance window.

- DG-2 was declared inoperable on Jan. 1, 2017, at 1600.
- A risk management action plan for the use of the alternate AC source, DG-4, has been established prior to the DG-2 maintenance window.
- On Jan. 8, 2017, at 2100, a fault caused a lock-out condition on the Backup Transformer, TR-B, and TR-B was declared inoperable.

Using the references provided, determine the next action required by technical specifications.

Declare…

A. TR-B operable prior to 2100 on Jan. 9, 2017.

B. DG-2 operable prior to 0900 on Jan. 9, 2017.

C. TR-B operable prior to 2100 on Jan. 11, 2017.

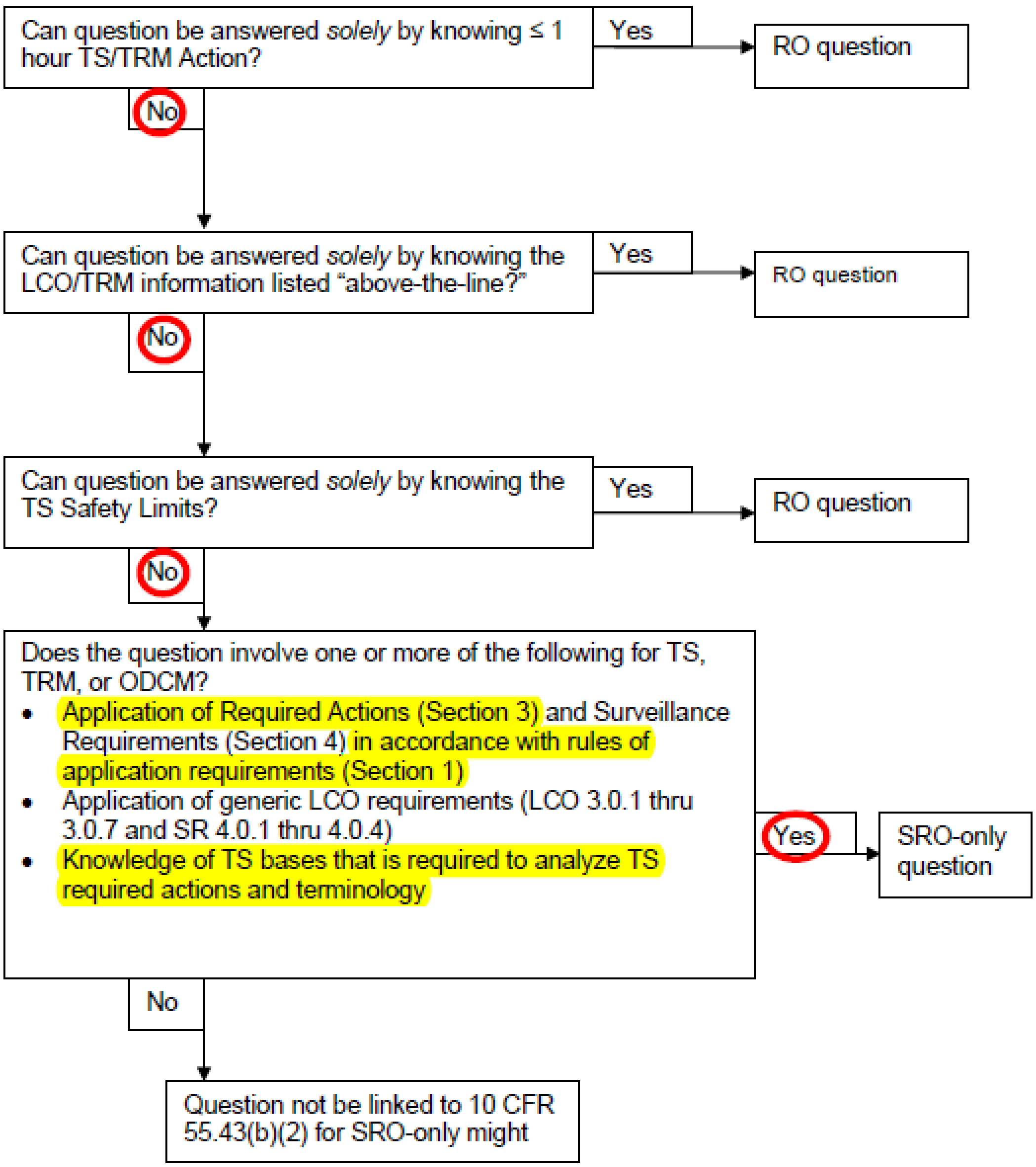
D. DG-2 operable prior to 1600 on Jan. 15, 2017.

Answer: B

K/A Match:
The question requires the candidate to understand how a diesel generator maintenance window affects LCO 3.8.1 with a concurrent loss of one qualified offsite circuit.

SRO Only:
K/A is a “G” statement linked to 10CFR 55.43.5

Figure 1: Screening for SRO-only linked to 10 CFR 55.43(b)(2) (Tech Specs)



Explanation:
A. Incorrect. Plausible since this action and time would be applicable if two offsite sources were lost (LCO 3.8.1, condition C). However, with one offsite source and one diesel generator inoperable, either one must be declared operable within 12 hours to meet the requirements of LCO 3.8.1 condition D. B. Correct. Since one DG and one qualified offsite circuit is inoperable, LCO 3.8.1 condition D is applicable. C. Incorrect. One inoperable qualified offsite circuit is normally required to be restored to operable within 72 hours per LCO 3.8.1, condition A. However, since one DG and one qualified offsite circuit is inoperable, LCO 3.8.1 condition D is applicable. D. Incorrect. With a risk management plan for the use of the alternate AC source, DG may be inoperable for 14 days per LCO 3.8.1, condition B. However, since one DG and one qualified offsite circuit is inoperable, LCO 3.8.1 condition D is applicable.

Technical Reference(s)		
A	Technical Specification 3.8.1, AC Sources-Operating	Attached w/ Revision # See Comments / Reference
B	Technical Specification 3.8.7, Distribution Systems-Operating	

Proposed references to be provided during examination: LCO 3.8.1 and LCO 3.8.7

Learning Objective: 5059 - Referencing Technical Specifications associated with the AC Distribution System and a set of plant conditions, determine as applicable the LCO, the action statement, and the appropriate bases.

Question Source: Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 ______
55.43 43.2

Comments / Reference: TS 3.8.1, AC Sources-Operating	Revision: Amendment 237

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 Electric Power Distribution System; and

b. Three diesel generators (DGs).

APPLICABILITY: MODES 1, 2, and 3.

-----NOTE-----
Division 3 AC electrical power sources are not required to be OPERABLE when High Pressure Core Spray System is inoperable.

ACTIONS

-----NOTE-----
LCO 3.0.4.b is not applicable to DGs.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One offsite circuit inoperable.	A.1 Perform SR 3.8.1.1 for OPERABLE offsite circuit. AND	1 hour AND Once per 8 hours thereafter

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.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>		
	A.3	Restore offsite circuit to OPERABLE status.	72 hours <u>AND</u> 6 days from discovery of failure to meet LCO when not associated with Required Action B.4.2.2 <u>AND</u> 17 days from discovery of failure to meet LCO
B. One required DG 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>		
	B.2	Declare required feature(s), supported by the inoperable DG, inoperable when the redundant required feature(s) are inoperable.	4 hours from discovery of Condition B concurrent with inoperability of redundant required feature(s)
	<u>AND</u>		

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.3.1 Determine OPERABLE DG(s) are not inoperable due to common cause failure.	24 hours
	OR	
	B.3.2 Perform SR 3.8.1.2 for OPERABLE DG(s).	24 hours if not performed within the past 24 hours
	AND	
	B.4.1 Restore required DG to OPERABLE status.	72 hours from discovery of an inoperable DG AND 6 days from discovery of failure to meet LCO
	OR	
	B.4.2.1 Establish risk management actions for the alternate AC sources.	72 hours
	AND	
	B.4.2.2 Restore required DG to OPERABLE status.	14 days AND 17 days from discovery of failure to meet LCO

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Two offsite circuits inoperable.	C.1 Declare required feature(s) inoperable when the redundant required feature(s) are inoperable.	12 hours from discovery of Condition C concurrent with inoperability of redundant required feature(s)
	AND	
	C.2 Restore one offsite circuit to OPERABLE status.	24 hours
D. One offsite circuit inoperable. AND One required DG inoperable.	----------------NOTE---------------- Enter applicable Conditions and Required Actions of LCO 3.8.7, "Distribution Systems - Operating," when Condition D is entered with no AC power source to any division. --	
	D.1 Restore offsite circuit to OPERABLE status.	12 hours
	OR	
	D.2 Restore required DG to OPERABLE status.	12 hours
E. Two required DGs inoperable.	E.1 Restore one required DG to OPERABLE status.	2 hours OR 24 hours if Division 3 DG is inoperable

AC Sources - Operating
3.8.1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. Required Action and associated Completion Time of Condition A, B, C, D, or E not met.	F.1 --------------NOTE------------- LCO 3.0.4.a is not applicable when entering MODE 3. ---------------------------------- Be in MODE 3.	12 hours
G. Three or more required AC sources inoperable.	G.1 Enter LCO 3.0.3.	Immediately

Comments / Reference: TS 3.8.7, Distribution Systems-Operating	Revision: Amendment 237

3.8 ELECTRICAL POWER SYSTEMS

3.8.7 Distribution Systems - Operating

LCO 3.8.7 The following AC and DC electrical power distribution subsystems shall be OPERABLE:

a. Division 1 and Division 2 AC electrical power distribution subsystems;

b. Division 1 and Division 2 125 V DC electrical power distribution subsystems;

c. Division 1 250 V DC electrical power distribution subsystem; and

d. Division 3 AC and DC electrical power distribution subsystems.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Division 1 or 2 AC electrical power distribution subsystem inoperable.	A.1 Restore Division 1 and 2 AC electrical power distribution subsystems to OPERABLE status.	8 hours <u>AND</u> 16 hours from discovery of failure to meet LCO 3.8.7.a or b
B. Division 1 or 2 125 V DC electrical power distribution subsystem inoperable.	B.1 Restore Division 1 and 2 125 V DC electrical power distribution subsystems to OPERABLE status.	2 hours <u>AND</u> 16 hours from discovery of failure to meet LCO 3.8.7.a or b

Distribution Systems - Operating
3.8.7

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. Required Action and associated Completion Time of Condition A or B not met.	C.1	------------NOTE------------ LCO 3.0.4.a is not applicable when entering MODE 3. Be in MODE 3.	12 hours
D. Division 1 250 V DC electrical power distribution subsystem inoperable.	D.1	Declare associated supported feature(s) inoperable.	Immediately
E. One or more Division 3 AC or DC electrical power distribution subsystems inoperable.	E.1	Declare High Pressure Core Spray System inoperable.	Immediately
F. Two or more divisions with inoperable electrical power distribution subsystems that result in a loss of function.	F.1	Enter LCO 3.0.3.	Immediately

Examination Outline Cross-reference:	Level	RO		SRO
Rev. 1 Date: 12/16/2016	Tier			1
	Group			1
	K/A	295004 AA2.01		
Level of Difficulty: 3	Importance Rating			3.6

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

Question # 78

CGS is operating in Mode 1.

- A LOCA signal is received.
- The crew enters PPM 3.3.1, Reactor Scram and PPM 5.1.1, RPV Control

While verifying auto actions, operators note the following:

- LPCS-P-1 and RHR-P-2A have 0 running amps.
- LPCS-P-1 and RHR-P-2A pump breaker position indication is lost.
- Wetwell level: 32 feet, up slow.
- RPV level is -160 inches, down slow
- RPV pressure is 1050 psig, up slow
- MSIVs have closed on low RPV level.

What has caused these indications and what actions should the CRS direct to mitigate this event?

A loss of 125 vdc Division 1 (E-DP-S1/1) occurred __(1)__ the LOCA. Reduce RPV pressure with __(2)__ per PPM 5.1.1, Table 4, Alternate Pressure Control Systems.

A. (1) after
(2) ADS SRVs

B. (1) before
(2) ADS SRVs

C. (1) after
(2) RCIC

D. (1) before
(2) RCIC

Answer: B

K/A Match:
This question requires candidates to demonstrate an understanding that a partial loss of 125 vdc has occurred and the EOP actions required to mitigate the event.

SRO Only:
K/A is an “A2” statement and

ES-401 **8** **Attachment 2**

Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5) (Assessment and selection of procedures)

Can the question be answered *solely* by knowing “systems knowledge”, i.e., how the system works, flowpath, logic, component location? — Yes → RO question

No

Can the question be answered *solely* by knowing immediate operator actions? — Yes → RO question

No

Can the question be answered *solely* by knowing entry conditions for AOPs or plant parameters that require direct entry to major EOPs? — Yes → RO question

No

Can the question be answered *solely* by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure? — Yes → RO question

No

Does the question require one or more of the following?

- Assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed
- Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps
- Knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific sub-procedures or emergency contingency procedures
- Knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures

Yes → SRO-only question

No → Question might not be linked to 10 CFR 55.43(b)(5) for SRO-only

Explanation:
A. Incorrect. Plausible since ADS SRVs will be used to control RPV pressure. However, the loss of division 1 DC occurred before the LOCA signal was processed, as evidenced by LPCS-P-1 and RHR-P-2A not starting the LOCA. B. Correct. A loss of division 1 125 vdc before the LOCA signal would not allow LPCS-P-1 and RHR-P-2A to start when the LOCA signal was received. Additionally, pump breaker position indication is lost. PPM 5.1.1 requires RPV pressure be controlled less than 1060 psig. Alternate methods are authorized to accomplish this. ADS SRVs will be used since the RCIC turbine trips on a loss of division 1 DC power. C. Incorrect. Plausible since a controlling RPV pressure using RCIC is authorized for the given plant conditions. However, A loss of division 1 125 vdc before the LOCA signal would not allow LPCS-P-1 and RHR-P-2A to start when the LOCA signal was received. Additionally, pump breaker position indication is lost. PPM 5.1.1 requires RPV pressure be controlled less than 1060 psig. Alternate methods are authorized to accomplish this. ADS SRVs will be used since the RCIC turbine trips on a loss of division 1 DC power. D. Incorrect. Plausible since, for the given plant conditions, ADS SRVs would be used to control RPV pressure. However, A loss of division 1 125 vdc before the LOCA signal would not allow LPCS-P-1 and RHR-P-2A to start when the LOCA signal was received. Additionally, pump breaker position indication is lost.

Technical Reference(s)		Attached w/ Revision # See Comments / Reference
ABN-ELEC-125VDC, Plant BOP, DIV 1,2 & 3 125 VDC Distribution System Failures		
PPM 5.1.1, RPV Control		

Proposed references to be provided during examination: None

Learning Objective: 7652: Predict the effects that a loss of DC bus S1-1 will have on b. RCIC, c. RHR, d. LPCS.

Question Source: Bank # ____
Modified Bank # LO01782 (Note changes or attach parent)
New ____

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 ____
55.43 43.5

Comments / Reference: ABN-ELEC-125VDC, Plant BOP, DIV 1,2 & 3 125 VDC Distribution System Failures	Revision: 014

Number: ABN-ELEC-125VDC	Use Category: CONTINUOUS	Major Rev: 014 Minor Rev: N/A Page: 12 of 35
Title: Plant BOP, DIV 1,2 & 3 125 VDC Distribution System Failures		

4.2 125 VDC Division 1

NOTE: The overall effect on Plant due to a loss of E-DP-S1/1 from 100% power is that the Plant continues at 100% with the operation of Div 1 equipment severely limited.

NOTE: A alphabetical list of major loads supplied by 125 VDC Div 1 distribution system and their status is provided in Attachment 7.2 and a complete list is found in SOP-ELEC-DC-LU.

NOTE: Either 125 VDC charger E-C1-1A or E-C1-1B is capable of supplying the Div 1 125 VDC bus E-DP-S1/1 with the 125 VDC battery B1-1 disconnected from the bus.

NOTE: IF DC Power is completely lost, THEN loss of indication and the ability to operate the following equipment is expected:

- H13-P601 SRV Relief Operation Solenoid)
- H13-P601 Div 1 ADS (A Solenoid)
- P628 SRV Div 1 ADS Operation (A Solenoid)
- On line inverter IN-3A or IN-3B (Will auto transfer to AC source)
- RCIC Flow Controller causing RCIC trip if operating
- DG-1 DC powered pumps
- RCIC (Div 1) motor operated valves
- DG 1 Control Panels and Swgr Control Power
- Remote Shutdown & Alt. Remote Shutdown Room DC loads
- Div 1 remotely operated circuit bkr position indication and closing/tripping power
- RHR-P-2A and LPCS-P-1 (Will continue to run if previously operating)
- SW-P-1A
- CRD-P-1A

NOTE: A loss of power to E-PP-7AA results in a loss of RCC to the Drywell due to a BOP outboard isolation from a loss of RC-1.

NOTE: A loss of power to E-PP-7AA causes an RC-1 half trip. Consider protecting the RC-2 logic train and operating equipment. Refer to PPM 1.3.83 and OI-41. {AR-238691}

NOTE: H13-P800.C1-8.3, 125 VDC IN-3A/3B TROUBLE alarming may be an indication of E-IN-3A or E-IN-3B failing to transfer to its alternate AC source, E-PP-7A, after a loss of E-DP-S1/1.

Number: ABN-ELEC-125VDC	Use Category: CONTINUOUS	Major Rev: 014 Minor Rev: N/A Page: 13 of 35
Title: Plant BOP, DIV 1,2 & 3 125 VDC Distribution System Failures		

4.2.2 IF E-IN-3A or E-IN-3B failed to automatically transfer to the Alternate AC source,
THEN **PERFORM** the following:

a. **REFER** to ABN-ELEC-INV. ____

b. **REFER** to ABN-FAZ, FAZ. ____

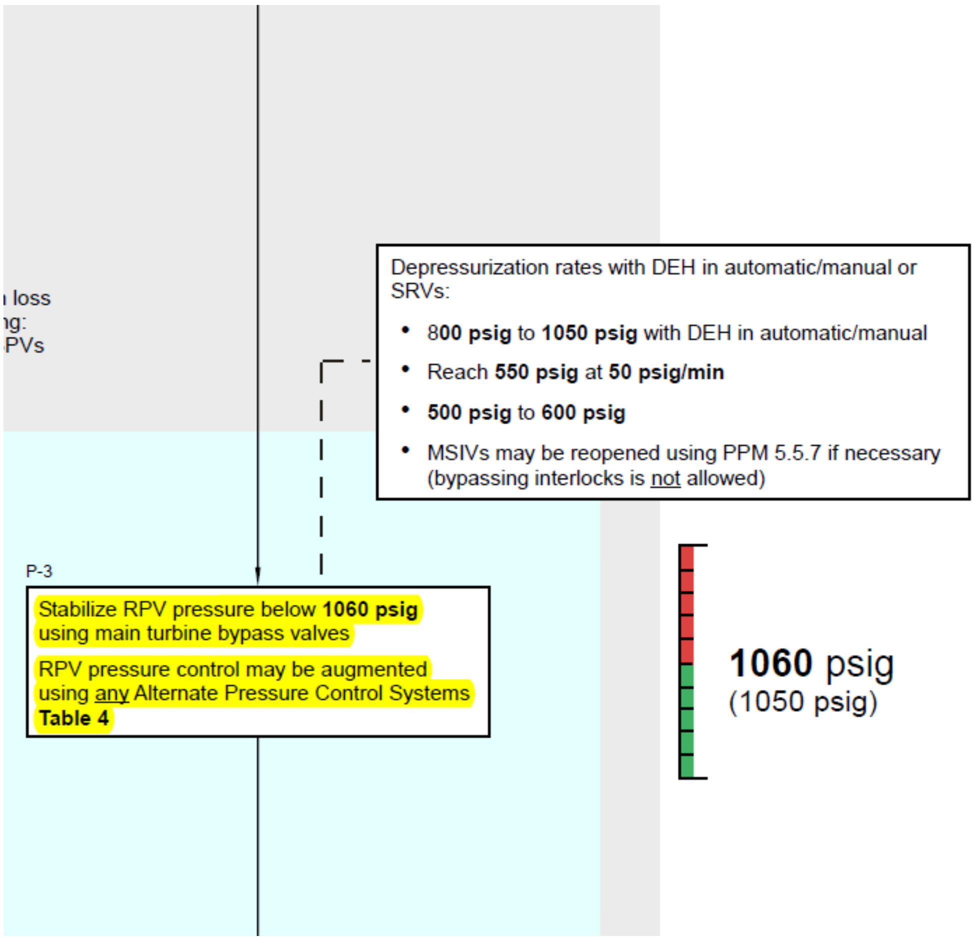
4.2.3 IF a total failure of a 125 VDC Div I (E-DP-S1/1) occurs,
THEN **PERFORM** the following:

a. **MONITOR** plant conditions and status of operating machinery, valves, breakers, etc., for which remote indications and annunciation are lost (Reference Attachments 7.2 and 7.3). ____

b. **PLACE** the following keys in the control switches for operating ADS SRVs (H13-P631).

- Key # 106 (MS-RV-4C) ____
- Key # 107 (MS-RV-5C) ____
- Key # 108 (MS-RV-4D) ____
- Key # 109 (MS-RV-4B) ____
- Key # 110 (MS-RV-4A) ____
- Key # 111 (MS-RV-5B) ____
- Key # 112 (MS-RV-3D) ____

Comments / Reference: PPM 5.1.1, RPV Control	Revision: 21



4 Alternate Pressure Control Systems

- SRVs only if wetwell level is above 17 ft
  (Follow SRV opening sequence, if possible)

  IF continuous SRV nitrogen supply is or becomes unavailable

  THEN - for pressure stabilization (P-3):
  place control switch for all SRVs to AUTO
  - for cooldown (P-5):
  conserve nitrogen by obtaining maximum pressure drop with each SRV operation

- RCIC 2 3 4
  with suction from CSTs if available

PPM 5.5.5

PPM 5.5.13 (high level)

- Main condenser
  - MSL drains
  - SJAE
  - Sealing Steam System
  - RFW turbines
  - Offgas preheaters

- SRVs from outside Control Room
  only if wetwell level is above 17 ft and a continuous nitrogen supply is available

Comments / Reference: Original Bank Question LO01782	Revision: N/A

The plant was operating at 99% power when a LOCA signal was received. While verifying auto actions, the CRO notes both LPCS-P-1 and RHR-P-2A have 0 running amps. Additionally, the CRO notes that neither system has pump breaker position indication on P601.

The CRO attempts to start both pumps but neither pump starts with the control switch on P601.

Which of the following is the correct explanation for these conditions?

A loss of….

A. both B1-1 and C1-1 before the LOCA signal.

B. both B1-1 and C1-1 after the LOCA signal.

C. both B1-2 and C1-2 before the LOCA signal.

D. both B1-2 and C1-2 after the LOCA signal.

Examination Outline Cross-reference:	Level	RO		SRO
Rev. 0 Date: 12/16/2016	Tier			1
	Group			1
	K/A	295016.2.4.35		
Level of Difficulty: 3	Importance Rating			4.0

Control Room Abandonment: Knowledge of local auxiliary operator tasks during an emergency and the resultant operational effects.

Question # 79

CGS is operating in Mode 1.

- The shift manager directs a control room evacuation due to a chemical spill that makes the control room uninhabitable.
- All immediate actions were completed before the control room crew evacuated the control room.
- Supplemental actions are in progress outside the control room.

Using the reference provided, what actions should the CRS direct the equipment operators to perform?

The CRS should direct the equipment operators to…

A. trip E-CB-B/8 and RPS breakers to ensure that the reactor is scrammed, all MSIVs are closed, and DG-2 remains available.

B. open circuit breakers and remove control power fuses for condensate and condensate booster pumps to prevent uncontrolled RPV injection.

C. trip and then restart DG-2 to ensure that it is started in the correct sequence for control room evacuation due to a control room fire.

D. isolate SM-8 and close breaker CB-8/DG2 to ensure DG-2 remains available while the control room is evacuated.

Answer: B

K/A Match:
This question requires the candidate to demonstrate knowledge of equipment operator actions during control room evacuation and the effects that these field actions have on mitigation strategies.

SRO Only:
K/A is “G” statement and

**Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5)
(Assessment and selection of procedures)**

Can the question be answered *solely* by knowing “systems knowledge”, i.e., how the system works, flowpath, logic, component location?

Yes → RO question

No

Can the question be answered *solely* by knowing immediate operator actions?

Yes → RO question

No

Can the question be answered *solely* by knowing entry conditions for AOPs or plant parameters that require direct entry to major EOPs?

Yes → RO question

No

Can the question be answered *solely* by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure?

Yes → RO question

No

Does the question require one or more of the following?

- Assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed
- Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps
- Knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific sub-procedures or emergency contingency procedures
- Knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures

Yes → SRO-only question

No

Question might not be linked to 10 CFR 55.43(b)(5) for SRO-only

Explanation:
A. Incorrect. Plausible since these actions are performed during control room evacuation due to a fire. However, a control room fire is not occurring and these actions are performed by a control room operator. B. Correct. In accordance with ABN-CR-EVAC, attachment 7.7, when the control room is evacuated, and equipment operator will open condensate and condensate booster pump breakers to prevent RPV overfill following emergency depressurization. C. Incorrect. Plausible since these actions are performed by an equipment operator during control room evacuation in accordance with ABN-CR-EVAC, attachment 7.5. However, they are only performed if the evacuation stems from a control room fire. D. Incorrect. Plausible since these actions are performed following a control room evacuation in accordance with ABN-CR-EVAC, attachment 7.3. However, they are performed by a control room operator only if the evacuation stems from a control room fire.

Technical Reference(s)		Attached w/ Revision # See Comments / Reference
ABN-CR-EVAC, control Room Evacuation and Remote Cooldown		

Proposed references to be provided during examination: ABN-CR-EVAC Flow Chart

Learning Objective: 11272 - Knowledge of RO tasks performed outside the main control room during emergency operations including system geography and system implications.

Question Source: Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

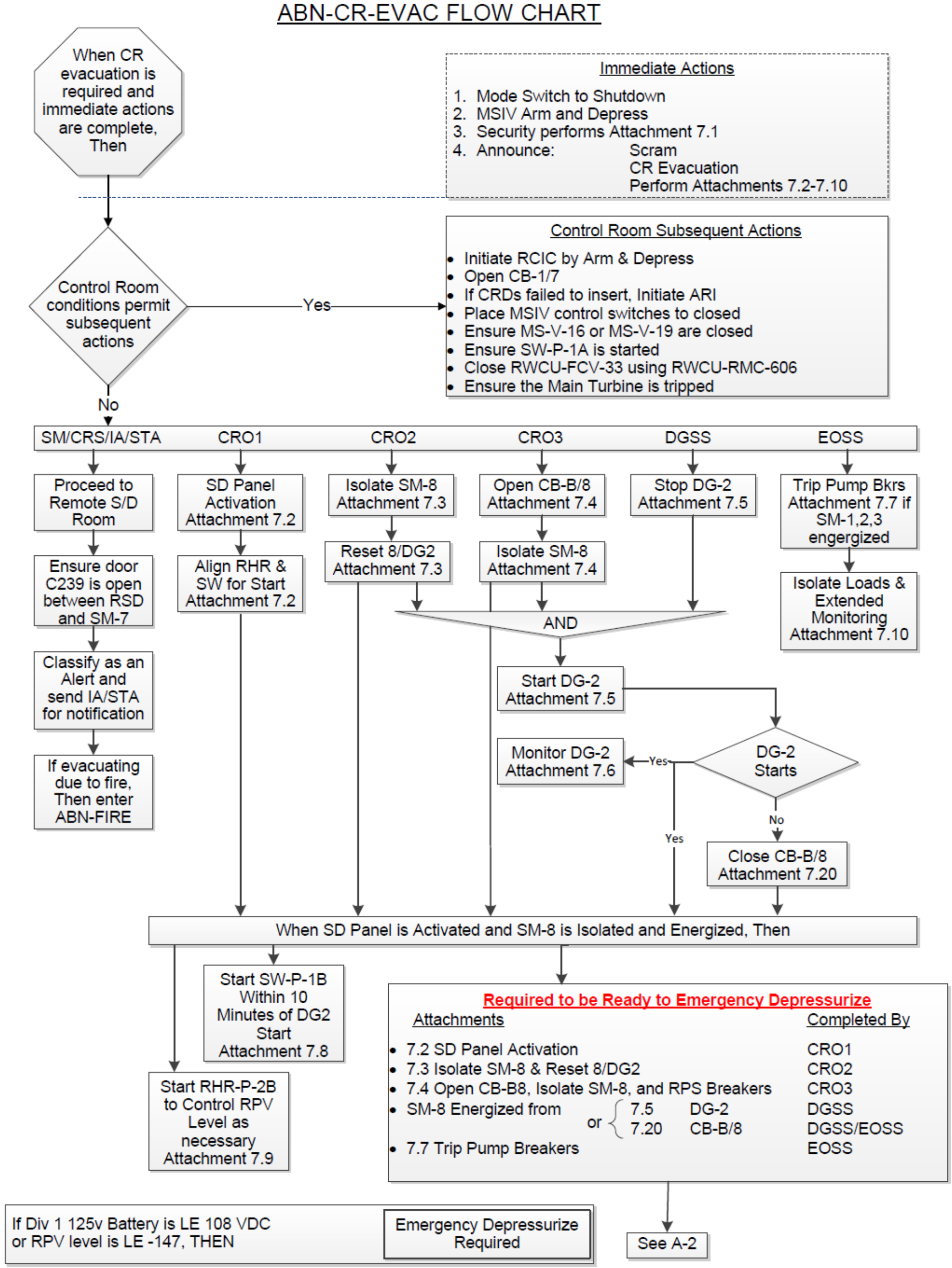
Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 ______
55.43 43.5

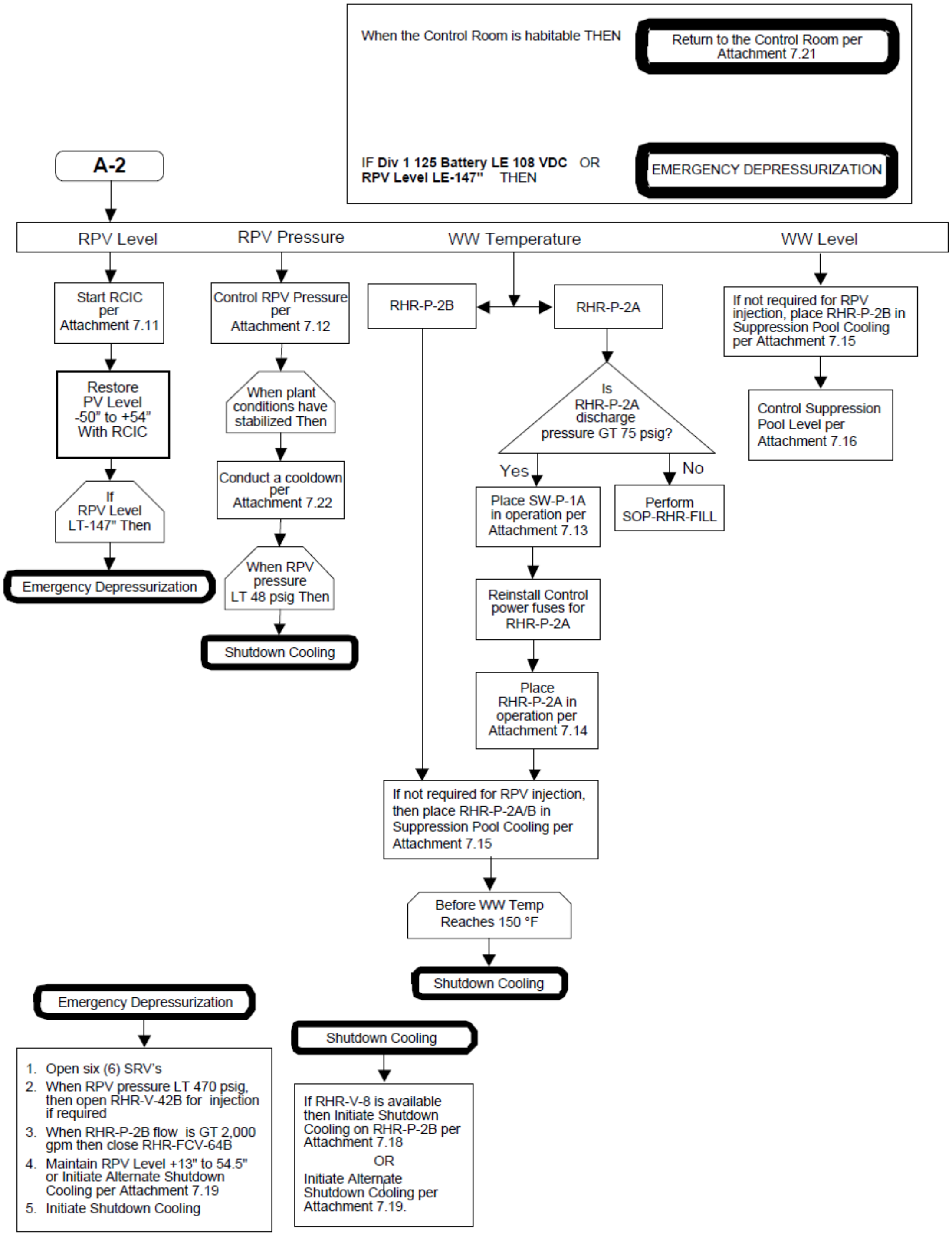
Comments / Reference: ABN-CR-EVAC, Flow Chart (Reference)	Revision: 035

Number: ABN-CR-EVAC	Use Category: CONTINUOUS	Major Rev: 035 Minor Rev: N/A Page: 4 of 63
Title: Control Room Evacuation and Remote Cooldown		

ABN-CR-EVAC FLOW CHART



Number: ABN-CR-EVAC	Use Category: CONTINUOUS	Major Rev: 035 Minor Rev: N/A Page: 5 of 63
Title: Control Room Evacuation and Remote Cooldown		



Comments / Reference: ABN-CR-EVAC, Bases	Revision: 035

Number: ABN-CR-EVAC	Use Category: CONTINUOUS	Major Rev: 035 Minor Rev: N/A Page: 15 of 63
Title: Control Room Evacuation and Remote Cooldown		

7.4 E-CB-B/8 is required to be isolated from SM-8 to prevent a fire in the Control Room from causing damage to SM-8. Removing fuses identified in the SM-8 aux compartment trips E-CB-B/8 if closed. E-CB-B/8 must be tripped within 10 minutes of Control Room fire to ensure DG-2 remains available.

Opening all six RPS EPA breakers will ensure the reactor is scrammed and the MSIVs are closed. A fire in the Control Room could prevent the Control Room switches from performing their functions. All RPS breakers must be tripped within 10 minutes of Control Room fire to ensure scram occurs.

7.5 If the evacuation is due to a Control Room fire, then DG-2 is the only credited source of power. It is critical to start DG-2 and load it in a timely manner in order to provide power to RHR-P-2B and SW-P-1B.

If DG-2 is already running due to under voltage on SM-8, it will need to be shutdown, then restarted in the correct sequence.

Placing DIV 2 DG control power transfer switch FRTS-7 to the **EMERG** position bypasses LOCA start signals; DG response to bus undervoltage is unchanged.

Division 3 equipment is not fire protected. HPCS-P-2 may be lost due to the Control Room fire. Therefore DG-3 and HPCS-P-1 can cause vessel overfill within 10 minutes following the scram based on GEH-0000-0075-4920.

7.7 Pulling the control power fuses and tripping the breakers for the pumps listed in this Attachment (7.7) are required to prevent uncontrolled injection into the RPV when the vessel is depressurized. If the pumps are not tripped, the subsequent injection through the relief valve could also fill the Suppression Pool. Also loss of keep fill pump(s) due to the fire could cause the unprotected ECCS systems to depressurize. Subsequent start of a voided ECCS system could cause a water hammer event. The Condensate and Condensate Booster pumps must be tripped within 10 minutes from the time the Shift Manager (or designee) orders a reactor scram due to a design basis fire to prevent RPV over-fill.

{AR-335854}

Examination Outline Cross-reference:	Level	RO		SRO
Rev. 0 Date: 12/16/2016	Tier			1
	Group			1
	K/A	295019 AA2.02		
Level of Difficulty: 3	Importance Rating			3.7

Partial or Total Loss of Inst. Air: Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF INTRUMENT AIR: Status of safety-related instrument air system loads.

Question # 80

CGS is operating in Mode 1.

- RHR-P-2A was declared inoperable due to an overcurrent trip at 1200, on 1/1/2017.
- The crew entered LCO 3.5.1, condition A, when RHR-P-2A tripped.
- At 1800, on 1/4/2017, an operator reported that ADS Division 1 Nitrogen Bank average pressure was 1600 psig, down slow.

With no other operator actions, using the reference provided, what is the latest time that the reactor must be in Mode 3 according to technical specifications?

A. 0100 on 1/5/2017.

B. 0600 on 1/5/2017.

C. 0600 on 1/8/2017.

D. 0000 on 1/9/2017.

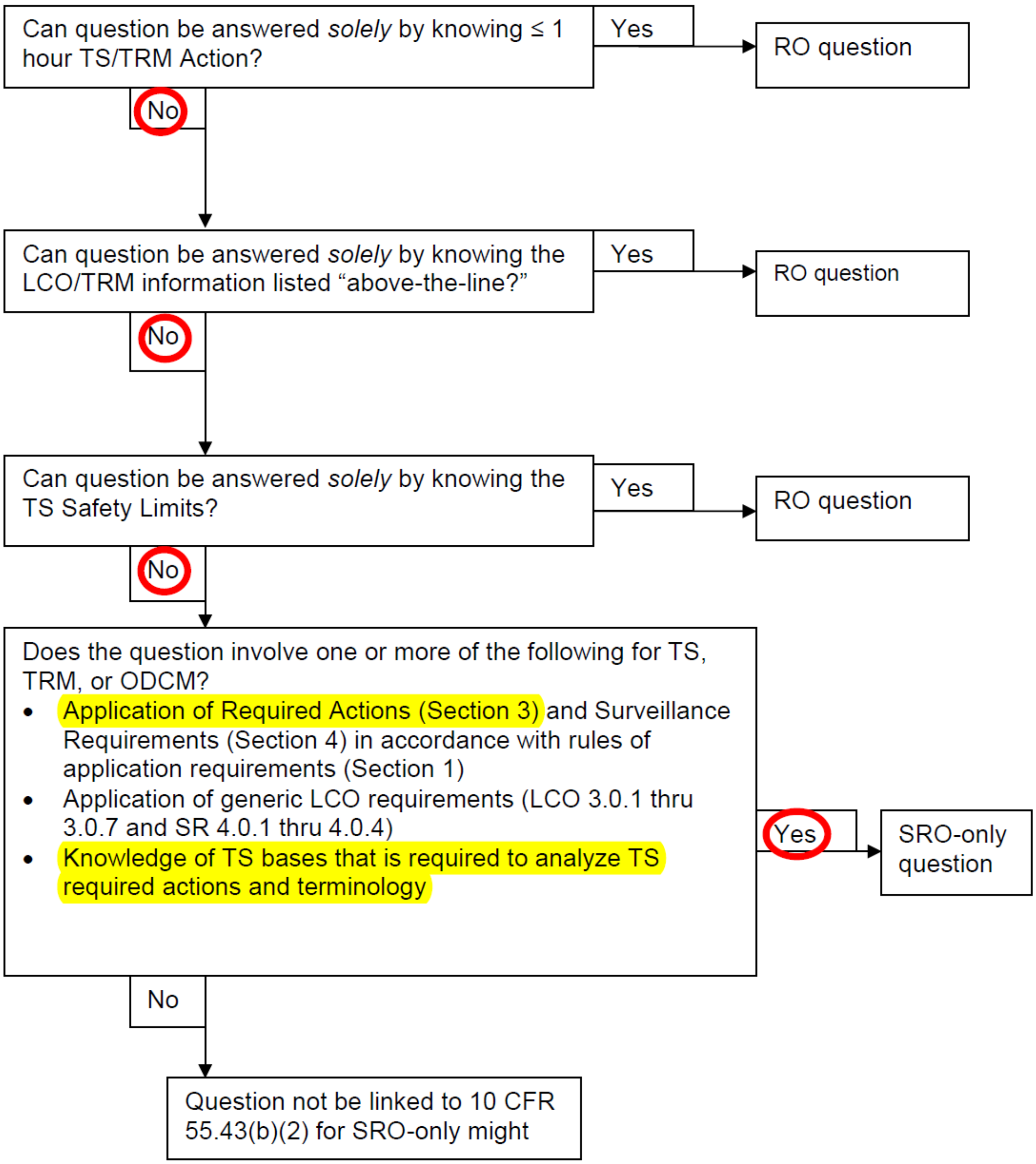
Answer: B

K/A Match:
The question requires the candidate to demonstrate knowledge of the relationship between the ADS nitrogen banks (part of the Containment Instrument Air system), and operability of Safety Relief Valves (SRVs).

SRO Only:
K/A is a “G” statement tied to 10CFR.55.43 and

ES-401	5	Attachment 2

Figure 1: Screening for SRO-only linked to 10 CFR 55.43(b)(2) (Tech Specs)



Explanation:
A. Incorrect. Plausible for LCO 3.5.1, condition H (ie - two additional ECCS subsystems inoperable, OR HPCS and LPCS inoperable, OR HPCS and ADS valve inoperable, etc.). aHowever, only one non-HPCS ECCS subsystem is inoperable. B. Correct. In accordance with TS 3.5.1 bases, an average pressure of ≥ 2200 psig is required in each ADS nitrogen bank to maintain operability. The division 1 nitrogen bank supplies 3 SRVs. Therefore, 2 of the required 6 ADS SRVs is inoperable. LCO 3.5.1 action G.1 applies and the reactor must be in Mode 3 12 hours from the discovery that the LCO is not met. C. Incorrect. Plausible for LCO 3.5.1 action statements F and G. D. Incorrect. Plausible for LCO 3.5.1, condition A and B, if it is believed that ADS SRVs are still operable with Containment Instrument Air (CIA) only. See distractor A and B explanations.

Technical Reference(s)		Attached w/ Revision # See Comments / Reference
TS 3.5.1, ECCS - Operating		
TS 3.5.1, ECCS – Operating, Bases		

Proposed references to be provided during examination: TS 3.5.1, Actions table.

Learning Objective: LO7748 - Predict the plant impact that a loss or malfunction of the Containment Instrument Air System will have on the following: SRVs

Question Source: Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 ______
55.43 43.5

Comments / Reference: TS 3.5.1, ECCS – Operating (Reference)	Revision: 237

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS) AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM

3.5.1 ECCS - Operating

LCO 3.5.1 Each ECCS injection/spray subsystem and the Automatic Depressurization System (ADS) function of six safety/relief valves shall be OPERABLE.

APPLICABILITY: MODE 1,
MODES 2 and 3, except ADS valves are not required to be OPERABLE with reactor steam dome pressure ≤ 150 psig.

ACTIONS

------------------------------------NOTE------------------------------------
LCO 3.0.4.b is not applicable to HPCS.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One low pressure ECCS injection/spray subsystem inoperable.	A.1 Restore low pressure ECCS injection/spray subsystem to OPERABLE status.	7 days[1]
B High Pressure Core Spray (HPCS) System inoperable.	B.1 Verify by administrative means RCIC System is OPERABLE when RCIC System is required to be OPERABLE. AND B.2 Restore HPCS System to OPERABLE status.	Immediately 14 days

[1] The Completion Time that one train of RHR (RHR-B) can be inoperable as specified by Required Action A.1 may be extended beyond the 7 day completion time up to 7 days to support restoration of RHR-B from the modification activity. Upon successful restoration of RHR-B, this footnote is no longer applicable and will expire at 05:00 PST on February 9, 2015.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. Two ECCS injection subsystems inoperable. <u>OR</u> One ECCS injection and one ECCS spray subsystem inoperable.	C.1	Restore ECCS injection/spray subsystem to OPERABLE status.	72 hours
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1	--------------NOTE-------------- LCO 3.0.4.a is not applicable when entering MODE 3. ------------------------------------- Be in MODE 3.	 12 hours
E. One required ADS valve inoperable.	E.1	Restore ADS valve to OPERABLE status.	14 days
F. One required ADS valve inoperable. <u>AND</u> One low pressure ECCS injection/spray subsystem inoperable.	F.1 <u>OR</u> F.2	Restore ADS valve to OPERABLE status. Restore low pressure ECCS injection/spray subsystem to OPERABLE status.	72 hours 72 hours

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
G. Required Action and associated Completion Time of Condition E or F not met. OR Two or more required ADS valves inoperable.	G.1	--------------NOTE-------------- LCO 3.0.4.a is not applicable when entering MODE 3. ------------------------------------- Be in MODE 3.	 12 hours
H. HPCS and Low Pressure Core Spray (LPCS) Systems inoperable. OR Three or more ECCS injection/spray subsystems inoperable. OR HPCS System and one or more required ADS valves inoperable. OR Two or more ECCS injection/spray subsystems and one or more required ADS valves inoperable.	H.1	Enter LCO 3.0.3.	Immediately

Comments / Reference: TS 3.5.1, ECCS – Operating Bases	Revision: 92

BASES

BACKGROUND (continued)

pressures, HPCS flow begins as soon as the necessary valves are open. A full flow test line is provided to route water to the CST to allow testing of the HPCS System during normal operation without spraying water into the RPV.

The ECCS pumps are provided with minimum flow bypass lines, which discharge to the suppression pool. The valves in these lines automatically open to prevent pump damage due to overheating when other discharge line valves are closed or RPV pressure is greater than the LPCS or LPCI pump discharge pressures following system initiation. To ensure rapid delivery of water to the RPV and to minimize water hammer effects, the ECCS discharge line "keep fill" systems are designed to maintain all pump discharge lines filled with water.

The ADS (Ref. 4) consists of 7 of the 18 SRVs. It is designed to provide depressurization of the primary system during a small break LOCA if HPCS fails or is unable to maintain required water level in the RPV. ADS operation reduces the RPV pressure to within the operating pressure range of the low pressure ECCS subsystems, so that these subsystems can provide core cooling. Each ADS valve is supplied with pneumatic power from a cryogenic nitrogen supply system, which includes accumulators located in the drywell. In addition, during post LOCA conditions, if the normal, non-safety related, nitrogen supply becomes unavailable, the gas supply piping to the ADS function accumulators will automatically isolate from the cryogenic nitrogen supply. The ADS accumulator backup compressed gas manifold subsystems will then provide a nominal pressure of 180 psig nitrogen from banks of high pressure compressed nitrogen cylinders. These cylinders provide a 30 day supply of nitrogen for the ADS function during a post LOCA condition.

BASES

SURVEILLANCE REQUIREMENTS (continued)

In MODE 3 with the reactor steam dome pressure less than 48 psig, the RHR System may be required to operate in the shutdown cooling mode to remove decay heat and sensible heat from the reactor. Therefore, this SR is modified by a Note that allows LPCI subsystems to be considered OPERABLE during alignment and operation for decay heat removal, if capable of being manually realigned (remote or local) to the LPCI mode and not otherwise inoperable. Alignment and operation for decay heat removal includes when the required RHR pump is not operating or when the system is being realigned from or to the RHR shutdown cooling mode. At the low pressures and decay heat loads associated with operation in MODE 3 with reactor steam dome pressure less than 48 psig, a reduced complement of low pressure ECCS subsystems should provide the required core cooling, thereby allowing operation of RHR shutdown cooling, when necessary.

SR 3.5.1.3

Verification every 31 days that ADS accumulator backup compressed gas system average pressure in the required bottles is ≥ 2200 psig assures an adequate and OPERABLE air supply to the ADS valves. The minimum number of required bottles is 14 bottles in Division 1 and 17 in Division 2. The remote nitrogen cylinder connection in the DG corridor may be used to make up the minimum number of required bottles, provided the bottle(s) is properly installed to satisfy the seismic Category 1 restraint requirements and the bottle(s) capacity is greater than or equal to the capacity of the bottle being replaced. The nitrogen banks are sized to provide a 30 day supply of nitrogen for the ADS function. The ADS function is required to provide a flow path for alternate shutdown cooling. Alternate shutdown cooling is accomplished utilizing one RHR subsystem and the ADS to provide a path to the suppression pool for decay heat removal.

Examination Outline Cross-reference:	Level	RO		SRO
Rev. 0 Date: 12/16/2016	Tier			1
	Group			1
	K/A	295021.2.1.19		
Level of Difficulty: 3	Importance Rating			3.8

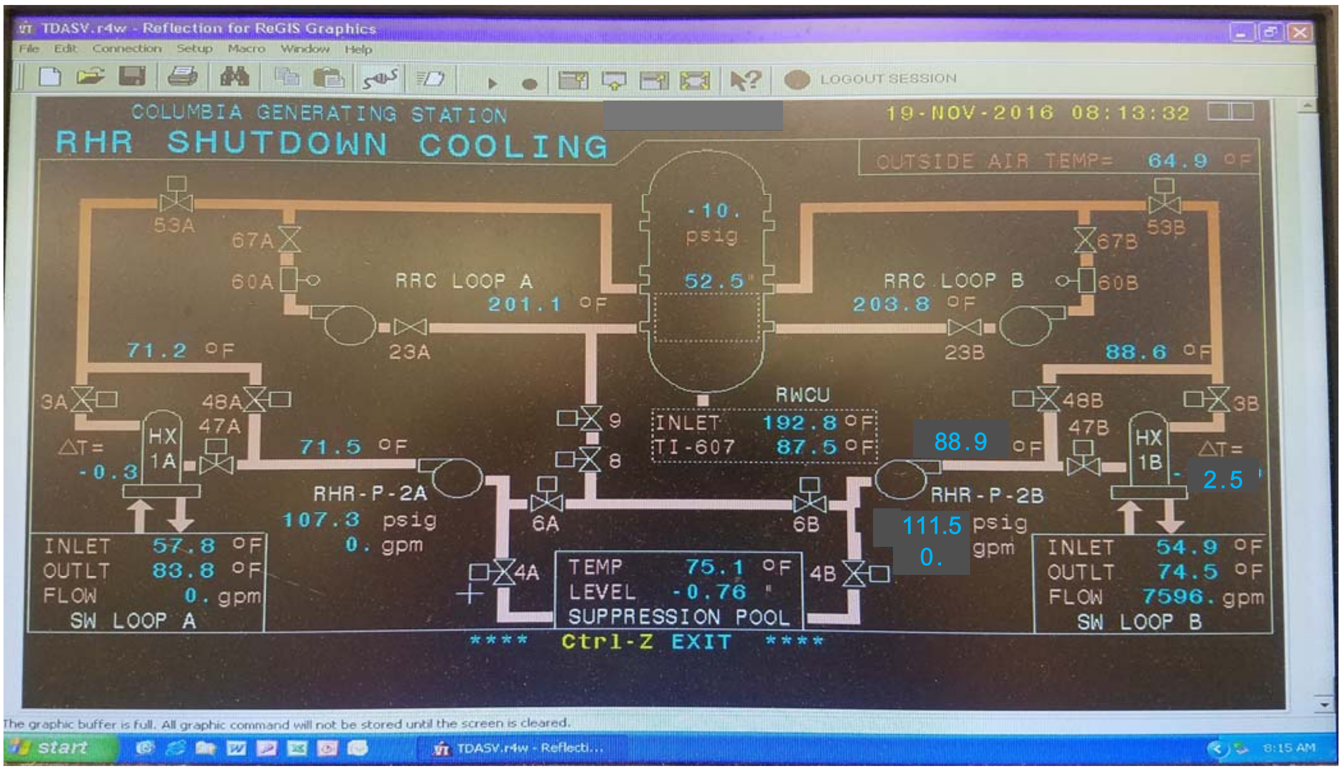
Loss of Shutdown Cooling: Ability to use plant computers to evaluate system or component status.	
Question # 81	

CGS is in Mode 4.

- RHR Loop B is in Shutdown Cooling (SDC).
- RHR-P-2B is inadvertently secured.

3 hours later, the crew is ready to restore SDC

Using the Plant Computer display below, how should SDC be restored?



The CRS should direct the crew to…

A. Place RHR Loop B in SDC in accordance with SOP-RHR-SDC, section 5.7, RHR Loop B Shutdown Cooling Quick Restart.

B. Place RHR Loop A in SDC in accordance with SOP-RHR-SDC, section 5.6, RHR Loop A Shutdown Cooling Quick Restart.

C. Place RHR Loop B in SDC in accordance with SOP-RHR-SDC, section 5.2, RHR Loop B Shutdown Cooling Initiation.

D. Establish SDC in accordance with ABN-RHR-SDC-ALT, section 7.1, Discharge Steam to the Main Condenser.

Answer: C	

K/A Match:
This question requires the candidate to use information from the plant computer to evaluate the appropriate method to restore shutdown cooling.

SRO Only:
K/A is a "G" statement and

Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5) (Assessment and selection of procedures)

Can the question be answered *solely* by knowing "systems knowledge", i.e., how the system works, flowpath, logic, component location?
Yes → RO question
No ↓

Can the question be answered *solely* by knowing immediate operator actions?
Yes → RO question
No ↓

Can the question be answered *solely* by knowing entry conditions for AOPs or plant parameters that require direct entry to major EOPs?
Yes → RO question
No ↓

Can the question be answered *solely* by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure?
Yes → RO question
No ↓

Does the question require one or more of the following?

- Assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed
- Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps
- Knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific sub-procedures or emergency contingency procedures
- Knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures

Yes → SRO-only question
No ↓

Question might not be linked to 10 CFR 55.43(b)(5) for SRO-only

Explanation:
A. Incorrect. Plausible since the quick start procedure could be used to restore SDC if the ΔT between RRC inlet temperature and RHR-P-2B outlet temperature were < 80°F. B. Incorrect. Plausible since the quick start procedure could be used to restore SDC if the ΔT between RRC inlet temperature and RHR-P-2A outlet temperature were < 80°F. Additionally, ABN-RHR-SDC-LOSS states that the SDC loop that was lost is the preferred loop to be restored. C. Correct. Since the ΔT between RRC inlet temperature and RHR-P-2B outlet temperature is > 80°F, SDC must be restored using the normal section of the procedure to warm up the RHR system. D. Incorrect. Plausible if RPV pressure was > 30 psig or if the normal SDC suction valves are inoperable.

Technical Reference(s)		Attached w/ Revision # See Comments / Reference
SOP-RHR-SDC, RHR Shutdown Cooling		
ABN-RHR-SDC-ALT, Residual Heat Removal Alternate Shutdown Cooling		

Proposed references to be provided during examination: None

Learning Objective: 11814 - Predict the impact of the following on the Residual Heat Removal System: b. Pump trips

Question Source: Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 ______
55.43 N/A (Group 2 Generic)

Comments / Reference: SOP-RHR-SDC, RHR Shutdown Cooling	Revision: Major 025 Minor 002

Number: SOP-RHR-SDC	Use Category: CONTINUOUS	Major Rev: 025 Minor Rev: 002 Page: 23 of 81
Title: RHR Shutdown Cooling		

5.2 RHR Loop B Shutdown Cooling Initiation

NOTE: This section is used for both EOP and non-EOP related SDC initiation. {R-6498}

CAUTION

To prevent failure of the RHR pumps due to excessive radiation exposure, alternate shutdown cooling, by Suppression Pool Cooling, is the only allowable mode for shutdown cooling once a degraded core condition has been identified. {R-9677}

CAUTION

Two Loop RHR Shutdown Cooling operations may cause actuation of Excess Flow Trip Isolation if ICP-RHR-Q901, RHR SDC Mode High Flow Isolation - CFT/CC, has not been completed within its required surveillance interval. {P-55450}

CAUTION

RHR Shutdown Cooling shall not be entered with reactor pressure above 48 psig except as provided for in PPM 3.2.3. This is due to operating limitations of RHR-V-8 & 9 and RHR system pipe supports. {2.3} {2.4}

Number: SOP-RHR-SDC	Use Category: CONTINUOUS	Major Rev: 025 Minor Rev: 002 Page: 40 of 81
Title: RHR Shutdown Cooling		

5.6 RHR Loop A Shutdown Cooling Quick Restart

CAUTION

To prevent failure of the RHR pumps due to excessive radiation exposure, alternate shutdown cooling, by Suppression Pool Cooling, is the only allowable mode for shutdown cooling once a degraded core condition has been identified. {R-9677}

CAUTION

Two Loop RHR Shutdown Cooling operations may cause actuation of Excess Flow Trip Isolation if ICP-RHR-Q901, RHR SDC Mode High Flow Isolation - CFT/CC, has not been completed within its required surveillance interval. {P-55450}

CAUTION

Failure to warm the RHR pump suction line may cause excessive thermal stress on the RHR injection line/Recirculation piping tee. {P-100924}

NOTE: This section is used if the ΔT between RHR A Heat Exchanger Outlet and RRC-P-1A Suction (RRC-TR-650, pt. 1, or TDAS pt. X292) is LT 80° F.

NOTE: If normal Shutdown Cooling can not be used, then refer to ABN-RHR-SDC-ALT.

NOTE: Technical Specifications require Reactor Vessel and head flange temperatures be maintained GT 80° F when the Vessel head bolting studs are being tensioned. (SR 3.4.11.7)

NOTE: If core decay heat is present, or if system metal temperature is high, a recirculation pump or RHR pump (do not use a recirculation pump if an RHR pump is available) along with a means of determining Reactor water temperature should be kept in service.

Number: SOP-RHR-SDC	Use Category: CONTINUOUS	Major Rev: 02
Title: RHR Shutdown Cooling		Minor Rev: 00 Page: 45 of 81

5.7 RHR Loop B Shutdown Cooling Quick Restart

CAUTION

To prevent failure of the RHR pumps due to excessive radiation exposure, alternate shutdown cooling, by Suppression Pool Cooling, is the only allowable mode for shutdown cooling once a degraded core condition has been identified. {R-9677}

CAUTION

Two Loop RHR Shutdown Cooling operations may cause actuation of Excess Flow Trip Isolation if ICP-RHR-Q901, RHR SDC Mode High Flow Isolation - CFT/CC, has not been completed within its required surveillance interval. {P-55450}

CAUTION

Failure to warm the RHR pump suction line may cause excessive thermal stress on the RHR injection line/Recirculation piping tee. {P-100924}

NOTE: This section is used if the ΔT between RHR B Heat Exchanger Outlet and RRC-P-1A Suction (RRC-TR-650, pt. 1, or TDAS pt. X292) is LT 80° F.

NOTE: If normal Shutdown Cooling cannot be used, then refer to ABN-RHR-SDC-ALT.

NOTE: Technical Specifications require Reactor Vessel and head flange temperatures be maintained GT 80° F when the Vessel head bolting studs are being tensioned. (SR 3.4.11.7)

NOTE: If core decay heat is present, or if system metal temperature is high, a recirculation pump or RHR pump (do not use a recirculation pump if an RHR pump is available) along with a means of determining Reactor water temperature should be kept in service.

Comments / Reference: ABN-RHR-SDC, Residual Heat Removal Alternate Shutdown Cooling	Revision: 013

Number: ABN-RHR-SDC-ALT	Use Category: CONTINUOUS	Major Rev: 013 Minor Rev: N/A Page: 3 of 54
Title: Residual Heat Removal Alternate Shutdown Cooling		

1.0 ENTRY CONDITIONS

Any of the following:

- RPV pressure is LT 135 psig, and normal Shutdown Cooling alignment is not available
- Normal Shutdown Cooling suction valves are inoperable
- RHR SDC is required to be restarted, but RPV pressure is GT 30 psig
- CRS/Shift Manager determine Alternate Shutdown Cooling operation is required during high heat load conditions

Examination Outline Cross-reference:	Level	RO		SRO
Rev. 0 Date:	Tier			1
	Group			1
	K/A	700000.AA2.08		
Level of Difficulty: 3	Importance Rating			4.4

Generator Voltage and Electric Grid Disturbances: Ability to determine and/or interpret the following as they apply to GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES: Criteria to trip the turbine or reactor.

Question # 82

CGS is operating in Mode 1.

Operators notice main generator parameters are fluctuating.

- Dittmer reports that there are no transmission system disturbances in progress.
- The Power System Stabilizer (PSS) has been turned off.
- The main generator voltage regulator is in TEST.

What actions should be taken?

The CRS will direct the operators to…

A. restore the PSS within 30 minutes if main generator fluctuations have not subsided.

B. scram the reactor if main generator power fluctuations reach 200 MWe peak to peak.

C. scram the reactor 3 minutes after main generator output voltage fluctuations reach 5 kv peak to peak.

D. restore the main generator voltage regulator to AUTO once parameters are within the Generator Capability Curve.

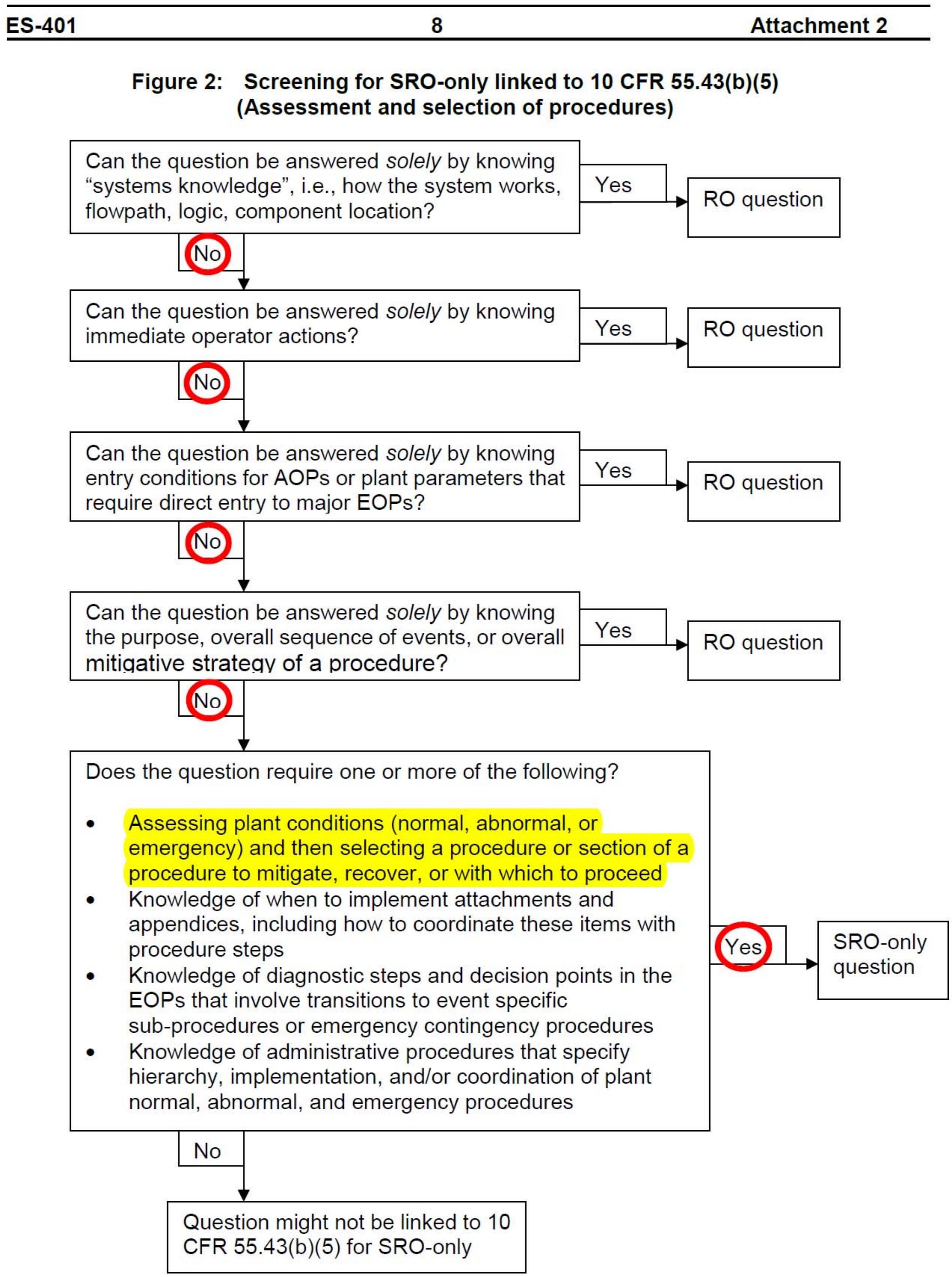
Answer: B

K/A Match:

The question requires the candidate to determine the reactor scram requirements during main generator output fluctuations.

SRO Only:

K/A is a "G" statement tied to 10CFR.55.43 and



ES-401 **8** **Attachment 2**

Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5) (Assessment and selection of procedures)

Can the question be answered *solely* by knowing "systems knowledge", i.e., how the system works, flowpath, logic, component location?

Yes → RO question

No

Can the question be answered *solely* by knowing immediate operator actions?

Yes → RO question

No

Can the question be answered *solely* by knowing entry conditions for AOPs or plant parameters that require direct entry to major EOPs?

Yes → RO question

No

Can the question be answered *solely* by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure?

Yes → RO question

No

Does the question require one or more of the following?

- Assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed
- Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps
- Knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific sub-procedures or emergency contingency procedures
- Knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures

Yes → SRO-only question

No

Question might not be linked to 10 CFR 55.43(b)(5) for SRO-only

Explanation:
A. Incorrect. Plausible since removing the PSS from service requires a report to Dittmer within 30 minutes. However, there is no requirement to restore the PSS to service with main generator fluctuations in progress. B. Correct. In accordance with ABN-GENERATOR, step 4.6.1.c.6), the reactor should be scrammed and the main generator tripped if main generator output power fluctuations reach 200 MWe peak to peak. C. Incorrect. Plausible since there is a requirement to scram the reactor if main generator output power fluctuations are 100 MWe peak to peak sustained for GT 3 minutes and main generator output voltage may be fluctuating. However, there is no requirement to scram the reactor on high output voltage fluctuations. D. Incorrect. Plausible since the voltage regulator is taken out of AUTO in an attempt to stabilize main generator output parameters. However, there is no direction to restore the voltage regulator to AUTO based on status of the Generator Capability curve.

Technical Reference(s)		Attached w/ Revision # See Comments / Reference
ABN-GENERATOR, Main Generator Trouble		

Proposed references to be provided during examination: None

Learning Objective: LO5531: Describe the effect that a loss or malfunction of the following will have on the Main Generator – Voltage regulation.

Question Source: Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 ______
55.43 43.5

Comments / Reference: ABN-GENERATOR, section 4.6, Main Generator Power Oscillations	Revision: Major 015 Minor 003

Number: ABN-GENERATOR	Use Category: CONTINUOUS	Major Rev: 015 Minor Rev: 003 Page: 12 of 32
Title: Main Generator Trouble		

4.6 Main Generator Power Oscillations

NOTE: The function of the Power System Stabilizer (PSS) is to assist other generators on the grid to dampen out frequency oscillations. The PSS should not be taken out of service unless absolutely necessary.

4.6.1 IF the power output of the Main Generator is fluctuating as observed on the field voltage meter, field current meter, MW meter, MVAR meter or the generator current and voltage meters, THEN **PERFORM** the following:

a. **NOTIFY** BPA to minimize near grid switching and voltage adjustments. {P-226157}_____

b. **CONTACT** Dittmer to ensure no transmission system disturbances are in progress. _____

NOTE: BPA must be notified as soon as practical, but within 30 minutes of the status change, of the Automatic Voltage Regulator or the Power System Stabilizer. This notification will include the expected duration of the change in status/capability of the Automatic Voltage Regulator and/or the Power System Stabilizer. The control room operator will log the time of notification so evidence of notification is maintained. {AR 176832}

c. IF no transmission system disturbances are in progress THEN **PERFORM** the following:

1) **PLACE** the Power System Stabilizer switch (E-RMS-FV/8) in the **OFF** position. _____

2) **RECORD** the time the Power System Stabilizer was placed in **OFF** position in the Control Room Log. {AR-176832} _____

3) **MONITOR** Main Generator parameters. _____

4) IF Main Generator parameters continue to fluctuate, THEN **SHIFT** Main Generator voltage regulator to **TEST** per Attachment 7.1. _____

5) IF Main Generator power output is fluctuating by GT +/- 50 MWe (100 MWe peak to peak) sustained for GT 3 minutes, THEN **PERFORM** the following:

a) **SCRAM the Reactor** per PPM 3.3.1. _____

b) **TRIP the Main Turbine**. _____

Number: ABN-GENERATOR	Use Category: CONTINUOUS	Major Rev: 015 Minor Rev: 002 Page: 13 of 32
Title: Main Generator Trouble		

6) IF Main Generator power output is fluctuating by GT +/- 100 MWe (200 MWe peak to peak), THEN **PERFORM** the following:

a) **SCRAM the Reactor** per PPM 3.3.1. _____

b) **TRIP the Main Turbine.** _____

4.6.2 IF the Power System Stabilizer has been removed from service, OR the Main Generator Voltage Regulator was placed in **TEST**, THEN PERFORM the following:

a. INFORM Dittmer within 30 minutes of the expected out of service time. {AR-176832} _____

b. ENTER in the Control Room Log the time Dittmer was notified. {AR-176832} _____

c. INITIATE a Condition Report to track out of service time. _____

Comments / Reference: ABN-GENERATOR, section 5.0, Bases for section 4.6	Revision: Major 015 Minor 003

Number: ABN-GENERATOR	Use Category: CONTINUOUS	Major Rev: 015 Minor Rev: 003 Page: 21 of 32
Title: Main Generator Trouble		

4.6 The function of the Power System Stabilizer (PSS) is to assist other generators on the grid to dampen out frequency oscillations. The performance of the stabilizer can be evaluated by observing no discernable fluctuations on the regulator balance meter. Oscillations on the exciter field voltage and current while the Generator output is relatively stable indicates the PSS is actively working through the voltage regulator to dampen a grid disturbance and the PSS should NOT be removed from service as it is actively working to maintain a steady Generator output. Should fluctuations of GT 2 volts for more than 3 minutes occur, the PSS may be taken out of service with concurrence from the System Engineer. Removing the PSS from service to facilitate repairs does not make the grid unstable but limits our generators contribution to dampen out any grid frequency disturbances. If system operation does not stabilize then direction is provided to initiate a work request to perform a calibration of the PSS per PPM 10.25.147. Subsequent actions are directed to shift the main generator voltage regulator control from Auto to Manual to stabilize operation in accordance with SOP-MT-START. If these actions do not stabilize the Main Generator power output, then an evaluation of the magnitude of the fluctuation is performed and if GT prescribed limits the reactor is scrammed and the Main Turbine is tripped.

Federal Energy Regulatory Commission (FERC) Standards require notification of Dittmer within 30 minutes of a status change of the Voltage Regulator or Power System Stabilizer. {AR-176832}

The Western Electrical Coordinating Council (WECC) limits out of service time for the Power System Stabilizer. {AR-176832}

CAUTION This caution reminds the operator that intentional entry into the region of potential core power instabilities during RRC flow reductions is not appropriate.

Examination Outline Cross-reference:	Level	RO		SRO
Rev. 1 Date: 12/16/2016	Tier			1
	Group			2
	K/A	295007.2.4.21		
Level of Difficulty: 2	Importance Rating			4.6

High Reactor Pressure: Knowledge of the parameters and logic used to assess the status of safety functions, such as reactivity control, core cooling and heat removal, reactor coolant system integrity, containment conditions, radioactivity release control, etc.

Question # 83

Why is reactor steam dome pressure limited to ≤ 1325 psig?

This limit is established to ensure that RCS pressure will not exceed __(1)__ of design pressure at the __(2)__ elevation in the RCS.

A. (1) 110%
(2) lowest

B. (1) 110%
(2) highest

C. (1) 125%
(2) lowest

D. (1) 125%
(2) highest

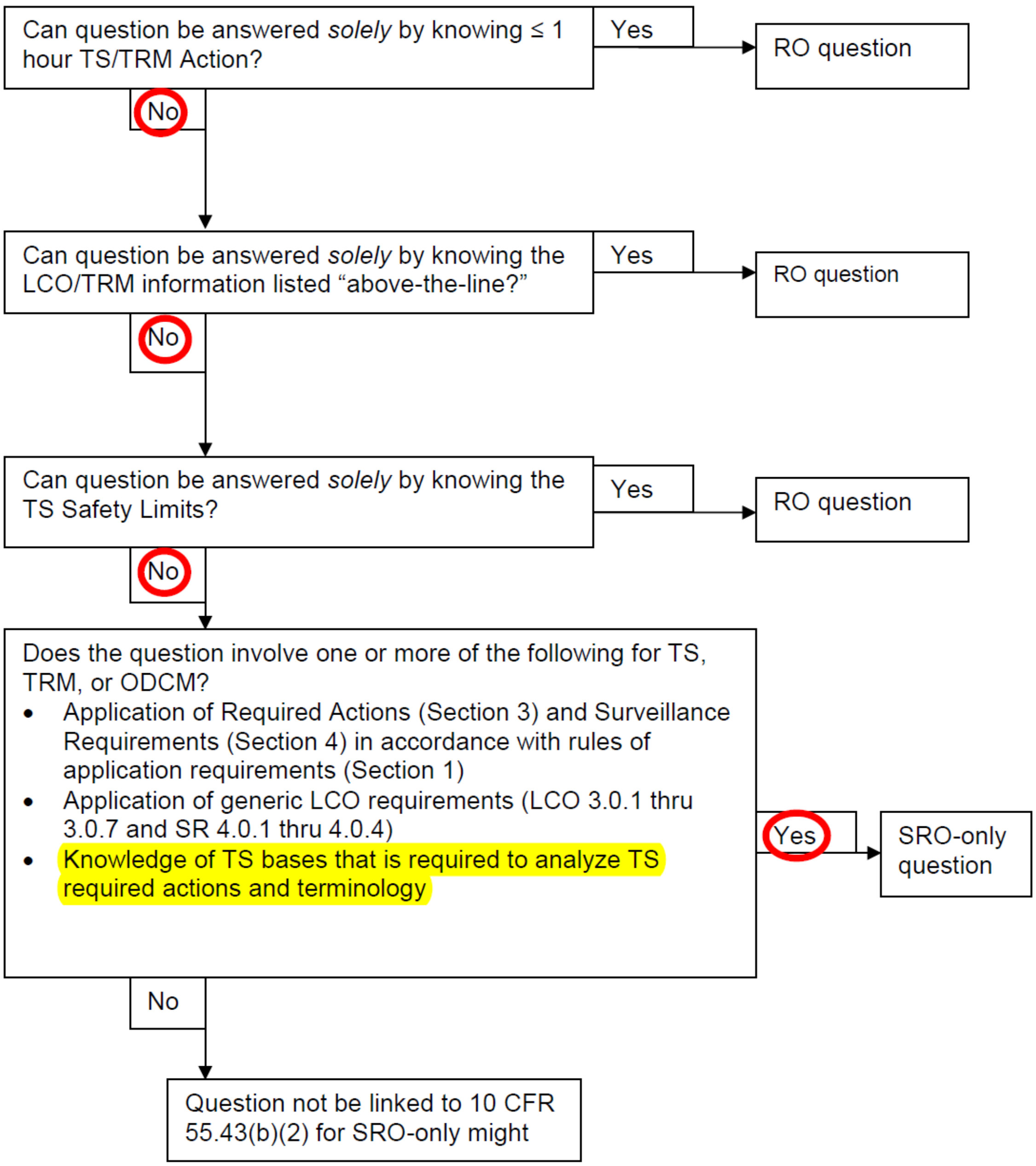
Answer: A

K/A Match:
The question requires the candidate to understand the logic used to establish the steam dome pressure safety limit, which maintains RCS integrity.

SRO Only:
K/A is a "G" statement tied to 10CFR.55.43 and

ES-401 **5** **Attachment 2**

Figure 1: Screening for SRO-only linked to 10 CFR 55.43(b)(2) (Tech Specs)



Explanation:

A. Correct. In accordance with TS bases for SL 2.1.2, the limit for reactor steam dome pressure ensures that pressure at the lowest point of the RCS is maintained below 110% of the design pressure of 1250 psig.

B. Incorrect. Plausible since the limit for reactor steam dome pressure is to ensures that pressure the highest RCS pressure point is maintained below 110% of the design pressure of 1250 psig. However, the highest RCS pressure is at the lowest point in the RCS.

C. Incorrect. Plausible since the steam dome safety limit will limit pressure below design pressure at the lowest point in the RCS. However, pressure is limited to 110% of design pressure at this point. The transient pressure limit for RCS piping and valves is 125% of design pressure. The safety limit is based on the most conservative limit, or 110% of design pressure.

D. Incorrect. Plausible The transient pressure limit for RCS piping and valves is 125% of design pressure. However the safety limit is based on the most limiting allowance, which is 110% of design pressure for the pressure vessel. Additionally, the limiting area is the lowest point in the RCS.

Technical Reference(s)		Attached w/ Revision # See Comments / Reference
Technical Specification Bases, 2.1.2, Reactor Coolant System (RCS) Pressure SL		

Proposed references to be provided during examination: None

Learning Objective: 13427 Describe the bases for the Reactor Steam Dome Pressure Safety Limit. [TS Bases] (SRO-only)

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New X

Question History: Last NRC Exam None

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis

10 CFR Part 55 Content: 55.41
55.43 43.5

Comments / Reference: Technical Specification Bases, 2.1.2, Reactor Coolant System (RCS) Pressure SL	Revision: 92

B 2.0 SAFETY LIMITS (SLs)

B 2.1.2 Reactor Coolant System (RCS) Pressure SL

BASES

BACKGROUND

The SL on reactor steam dome pressure protects the RCS against overpressurization. In the event of fuel cladding failure, fission products are released into the reactor coolant. The RCS then serves as the primary barrier in preventing the release of fission products into the atmosphere. Establishing an upper limit on reactor steam dome pressure ensures continued RCS integrity. According to 10 CFR 50, Appendix A, GDC 14, "Reactor Coolant Pressure Boundary," and GDC 15, "Reactor Coolant System Design" (Ref. 1), the reactor coolant pressure boundary (RCPB) shall be designed with sufficient margin to ensure that the design conditions are not exceeded during normal operation and anticipated operational occurrences (AOOs).

During normal operation and AOOs, RCS pressure is limited from exceeding the design pressure by more than 10%, in accordance with Section III of the ASME Code (Ref. 2). To ensure system integrity, all RCS components are hydrostatically tested at 125% of design pressure, in accordance with ASME Code requirements, prior to initial operation when there is no fuel in the core. Any further hydrostatic testing with fuel in the core may be done under LCO 3.10.1, "Inservice Leak and Hydrostatic Testing Operation." Following inception of unit operation, RCS components shall be pressure tested in accordance with the requirements of ASME Code, Section XI (Ref. 3).

Overpressurization of the RCS could result in a breach of the RCPB reducing the number of protective barriers designed to prevent radioactive releases from exceeding the limits specified in 10 CFR 50.67 (Ref. 4). If this occurred in conjunction with a fuel cladding failure, the number of protective barriers designed to prevent radioactive releases from exceeding the limits would be reduced.

APPLICABLE SAFETY ANALYSES

The RCS safety/relief valves and the Reactor Protection System Reactor Vessel Steam Dome Pressure - High Function have settings established to ensure that the RCS pressure SL will not be exceeded.

The RCS pressure SL has been selected such that it is at a pressure below which it can be shown that the integrity of the system is not endangered. The reactor pressure vessel is designed to ASME, Boiler and Pressure Vessel Code, Section III, 1971 Edition, including Addenda through the summer of 1971 (Ref. 5), which permits a maximum pressure transient of 110%, 1375 psig, of design pressure 1250 psig. The SL of 1325 psig, as measured in the reactor steam dome, is equivalent to

BASES

APPLICABLE SAFETY ANALYSES (continued)

1375 psig at the lowest elevation of the RCS. The RCS is designed to ASME Code, Section III, 1971 Edition, including Addenda through the summer of 1971 (Ref. 5), for the reactor recirculation piping, which permits a maximum pressure transient of 125% of design pressures of 1250 psig for suction piping and 1550 psig for discharge piping. The RCS pressure SL is selected to be the lowest transient overpressure allowed by the applicable codes.

SAFETY LIMITS

The maximum transient pressure allowable in the RCS pressure vessel under the ASME Code, Section III, is 110% of design pressure. The maximum transient pressure allowable in the RCS piping, valves, and fittings is 125% of design pressures of 1250 psig for suction piping and 1550 psig for discharge piping. The most limiting of these allowances is the 110% of design pressure; therefore, the SL on maximum allowable RCS pressure is established at 1325 psig as measured at the reactor steam dome.

Examination Outline Cross-reference:	Level	RO		SRO
Rev. 0 Date:	Tier			1
	Group			2
	K/A	295010.AA2.03		
Level of Difficulty: 3	Importance Rating			3.6

High Drywell Pressure: Ability to determine and/or interpret the following as they apply to HIGH DRYWELL PRESSURE: Drywell radiation levels.

Question # 84

CGS is operating in Mode 1.

- An event occurs that requires entry into the Emergency Operating Procedures (EOPs).
- Primary containment radiological conditions are greater than the ODCM RFO Offsite Radioactivity Release Limits.
- Drywell pressure is greater than 1.68 psig.
- PC pressure is below the Primary Containment Pressure Limit (PCPL).

Under what conditions should the CRS consider venting Primary Containment (PC) through the Standby Gas Treatment (SGT) system?

The CRS should consider venting the PC…

A. if PC hydrogen is detected, to prevent an explosive gas mixture from developing in containment.

B. if emergency depressurization (ED) is required and less than 7 SRVs are available, to ensure RPV pressure is below Decay Heat Removal Pressure (DHRP).

C. if Wetwell level exceeds the SRV Tail Pipe Level Limit (SRVTPL), to minimize potential for Drywell floor failure.

D. if significant fuel damage is anticipated, to reduce the overall radiation release by venting before containment atmosphere becomes more contaminated.

Answer: D

K/A Match:
This question requires the candidate to demonstrate understanding of the relationship between high drywell pressure and drywell radiological conditions and the strategies available to minimize off site radiation release.

SRO Only:
K/A is an "A2" statement and

ES-401 **8** **Attachment 2**

Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5) (Assessment and selection of procedures)

Can the question be answered *solely* by knowing "systems knowledge", i.e., how the system works, flowpath, logic, component location?

Yes → RO question

No

Can the question be answered *solely* by knowing immediate operator actions?

Yes → RO question

No

Can the question be answered *solely* by knowing entry conditions for AOPs or plant parameters that require direct entry to major EOPs?

Yes → RO question

No

Can the question be answered *solely* by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure?

Yes → RO question

No

Does the question require one or more of the following?

- Assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed
- Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps
- Knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific sub-procedures or emergency contingency procedures
- Knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures

Yes → SRO-only question

No

Question might not be linked to 10 CFR 55.43(b)(5) for SRO-only

Explanation:

A. Incorrect. Plausible since primary containment venting is allowed if hydrogen gas concentration is detected (> 0.6%). However, in this event, venting is not allowed if primary containment radiological conditions exceed the ODCM RFO Offsite Radioactivity Release Limits.

B. Incorrect. Plausible since this condition requires rapid RPV depressurization to reduce RPV to drywell differential pressure to allow RHR shutdown cooling. However, venting containment will exacerbate the d/p.

C. Incorrect. Plausible since this event requires the RPV to be depressurized via emergency depressurization. However, venting the primary containment will not mitigate this event.

D. Correct. In accordance with PPM 5.0.10, *“venting below the Primary Containment Pressure Limit may be appropriate to limit radioactivity release if significant fuel damage is anticipated. Reducing primary containment pressure while the primary containment atmosphere is still relatively clean increases the capacity of the containment to retain fission products. Later releases, after core damage has progressed, may thereby be avoided.”*

Technical Reference(s)		Attached w/ Revision # See Comments / Reference
PPM 5.0.10, Flowchart Training Manual		
PPM 5.2.1, Primary Containment Control		

Proposed references to be provided during examination: None

Learning Objective: ____

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 Best match only
55.43 ____

Comments / Reference: PPM 5.0.10 – Flowchart Training Manual	Revision: Major 21 Minor 001

Number: 5.0.10	Use Category: INFORMATION	Major Rev: 021 Minor Rev: 001 Page: 274 of 320
Title: Flowchart Training Manual		

3) For some events, an increasing drywell pressure trend may not produce a corresponding trend in Wetwell pressure until the drywell has pressurized sufficiently to clear the downcomer vents (~5 psig with Wetwell water level in the normal range). Since the need to initiate Wetwell sprays in Step P-5 is based on Wetwell pressure and not drywell pressure, the operator must ensure that Wetwell pressure is at least above 1.68 psig before initiating Wetwell sprays. Otherwise, the direction given in second IF/THEN of Override P-4 would force the operator to immediately terminate sprays.

c. Step P-3:

1) Wetwell sprays are started only if Wetwell water level is below the elevation of the Wetwell spray nozzles (52 ft 6 in.). If the Wetwell spray nozzles are submerged, no spray action will occur and therefore there is no benefit in starting Wetwell sprays. However, 51 ft is specified as the conditional water level limit for Wetwell spray initiation, because this is the highest indication of Wetwell water level.

d. Override P-4:

1) This override is applicable to the remaining steps in the primary containment pressure control flowpath.

2) First IF/THEN:

a) If, under extreme conditions beyond the plant design basis, primary containment pressure cannot be controlled using sprays, it may become necessary to evaluate the need for emergency RPV depressurization (Step P-12) and primary containment venting (Step P-13).

b) In general, primary containment venting should be considered as a last resort and only if primary containment pressure cannot be controlled using containment sprays. The volume of any release should be limited to the minimum required to maintain pressure below the Primary Containment Pressure Limit. However, in extreme cases, such as an extended station blackout beyond the plant design basis in which power is not restored, earlier or more extensive primary containment pressure reductions may be appropriate to restore and maintain adequate core cooling or limit the total radioactivity release.

Number: 5.0.10	Use Category: INFORMATION	Major Rev: 021 Minor Rev: 001 Page: 275 of 320
Title: Flowchart Training Manual		

c) Venting below the Primary Containment Pressure Limit may be appropriate to restore and maintain adequate core cooling if:

- Pressure must be reduced to permit RPV injection. If a primary system break exists, the primary system and primary containment are essentially a single volume. It may thus be necessary to vent the primary containment to permit injection from a portable pump or other alternative low pressure injection source.

- The Wetwell approaches saturation conditions and can no longer effectively condense steam discharged from RCIC. During a station blackout with RCIC available, local Wetwell temperatures are expected to increase as a result of RCIC operation. During an extended station blackout beyond the plant design basis, local temperatures could increase to the extent that the RCIC exhaust can no longer be completely condensed. It may then be appropriate to direct the relatively clean RCIC discharge through a containment vent path to maintain pressure suppression capability and avoid emergency RPV depressurization with subsequent loss of RCIC. Bulk suppression pool temperatures may remain low enough to provide pressure suppression capability even after local temperatures exceed the value at which RCIC exhaust can be effectively condensed.

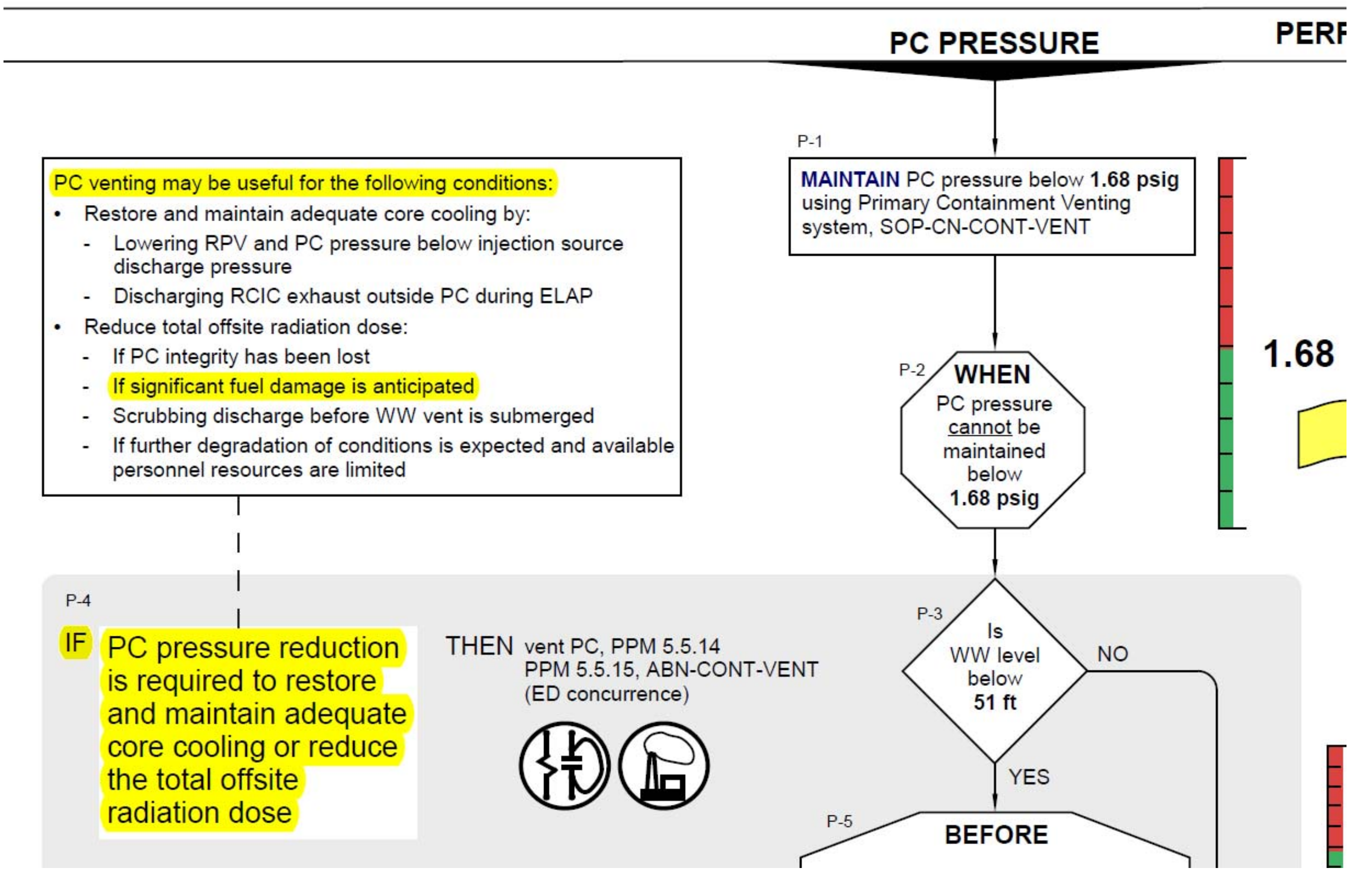
d) Venting below the Primary Containment Pressure Limit may be appropriate to limit radioactivity release if:

- A containment breach exists. If primary containment integrity has been lost, directing fission products through a filtered or elevated release point may reduce the total offsite dose and permit continued access to critical secondary containment areas.

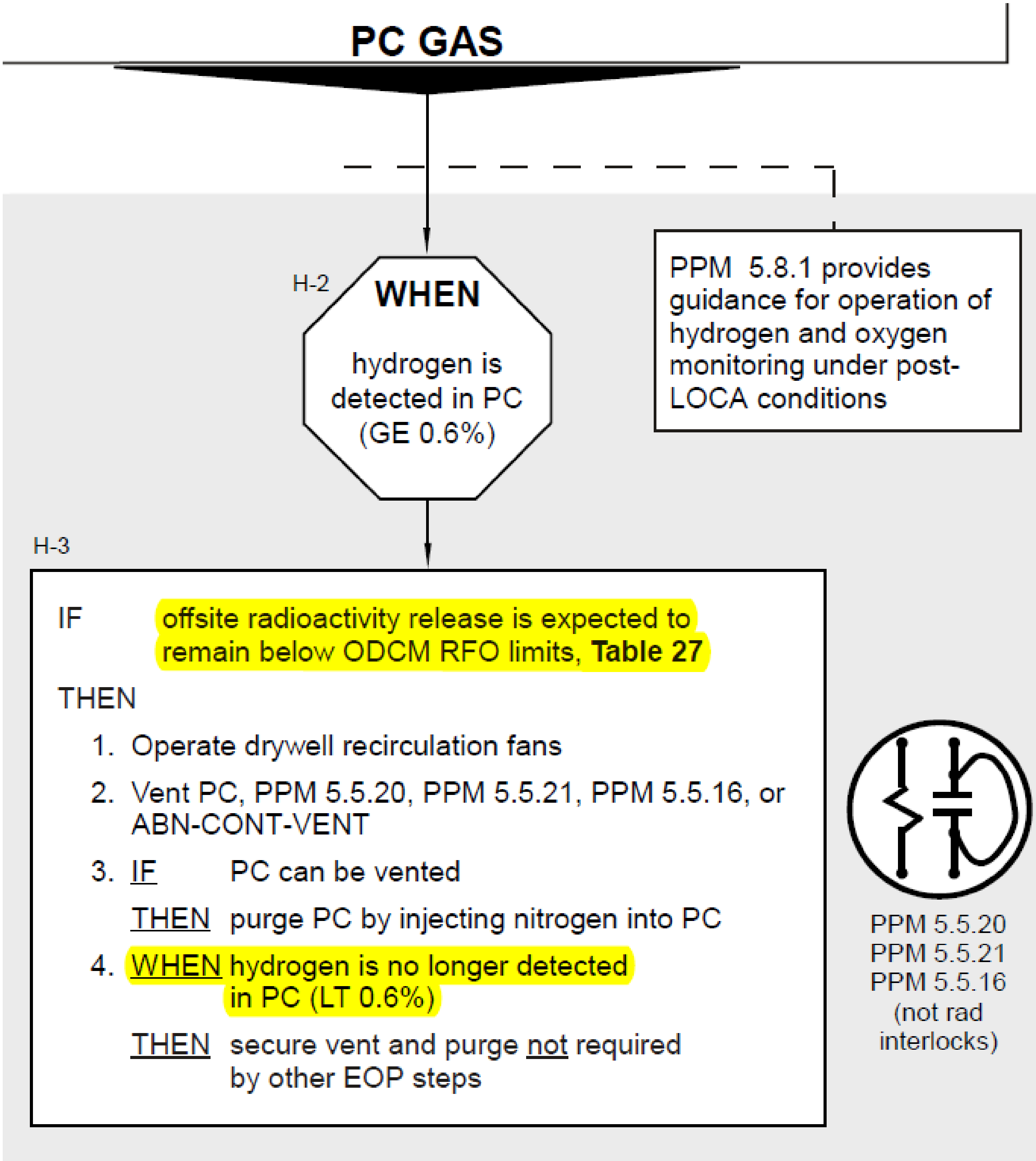
- Significant fuel damage is anticipated. Reducing primary containment pressure while the primary containment atmosphere is still relatively clean increases the capacity of the containment to retain fission products. Later releases, after core damage has progressed, may thereby be avoided.

- Wetwell level is elevated and rising. Reducing primary containment pressure while the Wetwell vent path is still available scrubs the discharge through the Wetwell, reducing any radioactivity release. Delaying venting until after the Wetwell vent penetration is submerged would necessitate use of the drywell vent path, increasing the resulting radioactivity release.

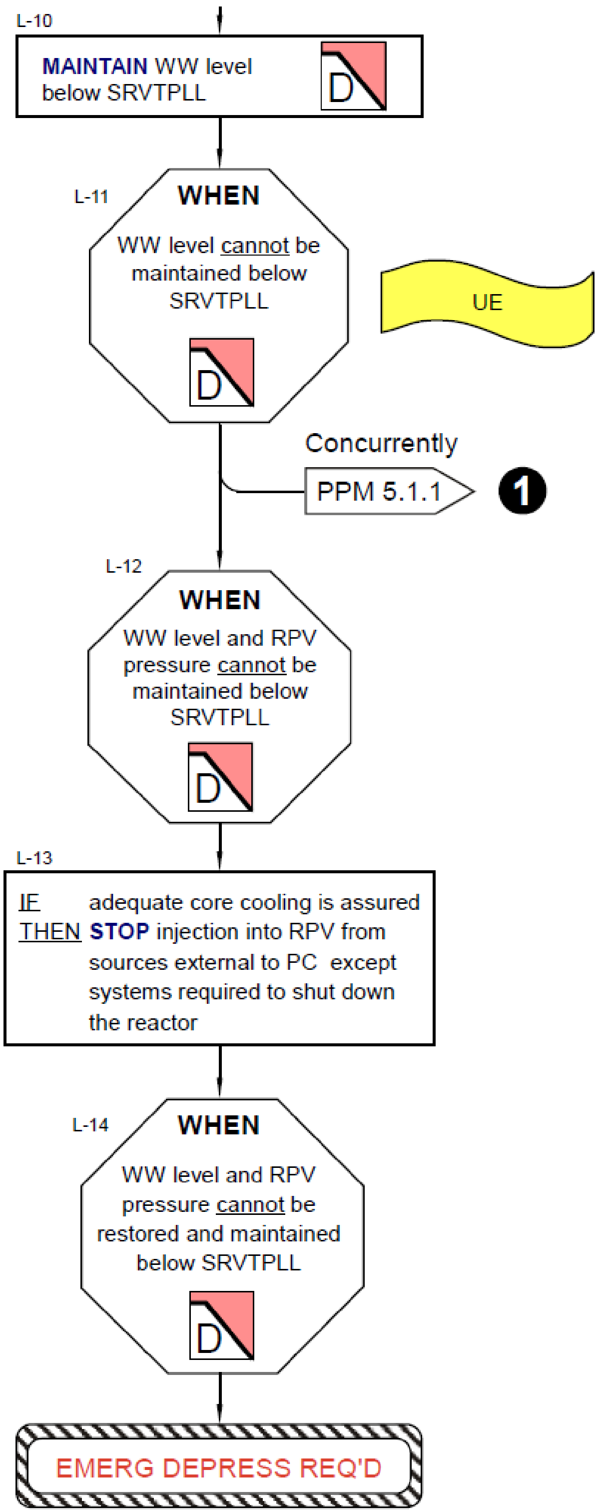
Comments / Reference: PPM 5.2.1, Primary Containment Control, Override P-4	Revision: 23



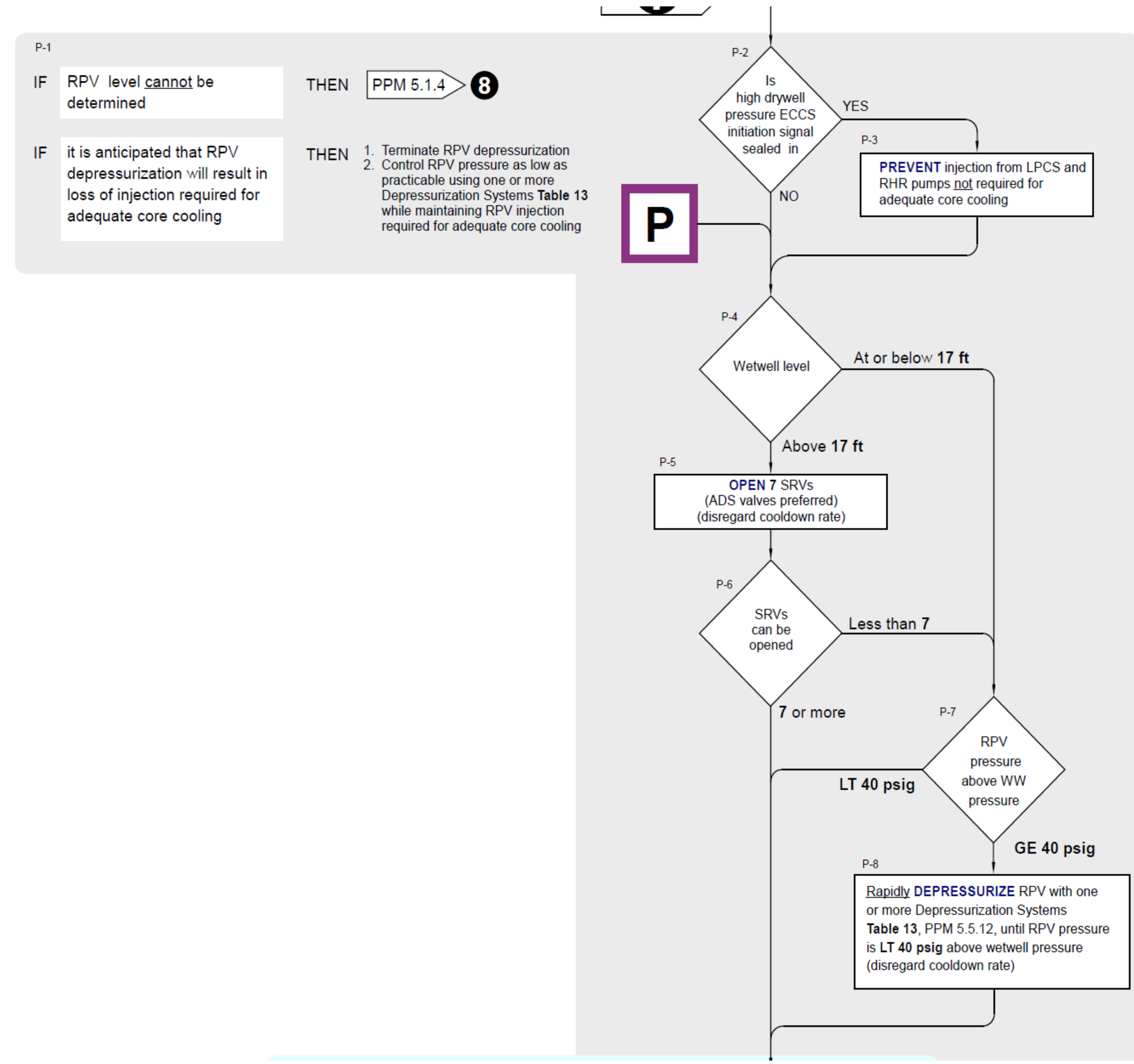
Comments / Reference: PPM 5.2.1, Primary Containment Control, PC Gas Leg	Revision: 23



Comments / Reference: PPM 5.2.1, Primary Containment Control, Wetwell Level High Leg	Revision: 23



Comments / Reference: PPM 5.1.3, Emergency RPV Depressurization	Revision: 20



Examination Outline Cross-reference:	Level	RO		SRO
Rev. 0 Date: 12/16/2016	Tier			1
	Group			2
	K/A	295033.2.2.37		
Level of Difficulty: 3	Importance Rating			4.6

High Secondary Containment Area Radiation Levels: Ability to determine operability and/or availability of safety related equipment.

Question # 85

CGS is operating in Mode 1.

- The crew enters PPM 5.3.1, Secondary Containment Control, due to an Area Radiation Monitor (ARM) above its alarm level.

What is the basis for the Max Safe Operating Value for Reactor Building ARMs?

The max safe value is set…

A. low enough to require emergency depressurization prior to exceeding the SITE AREA EMERGENCY (SAE) emergency action level.

B. high enough to require an emergency depressurization to be performed to prevent core damage.

C. low enough to perform mitigating actions without damaging safety related equipment due to high radiation exposure.

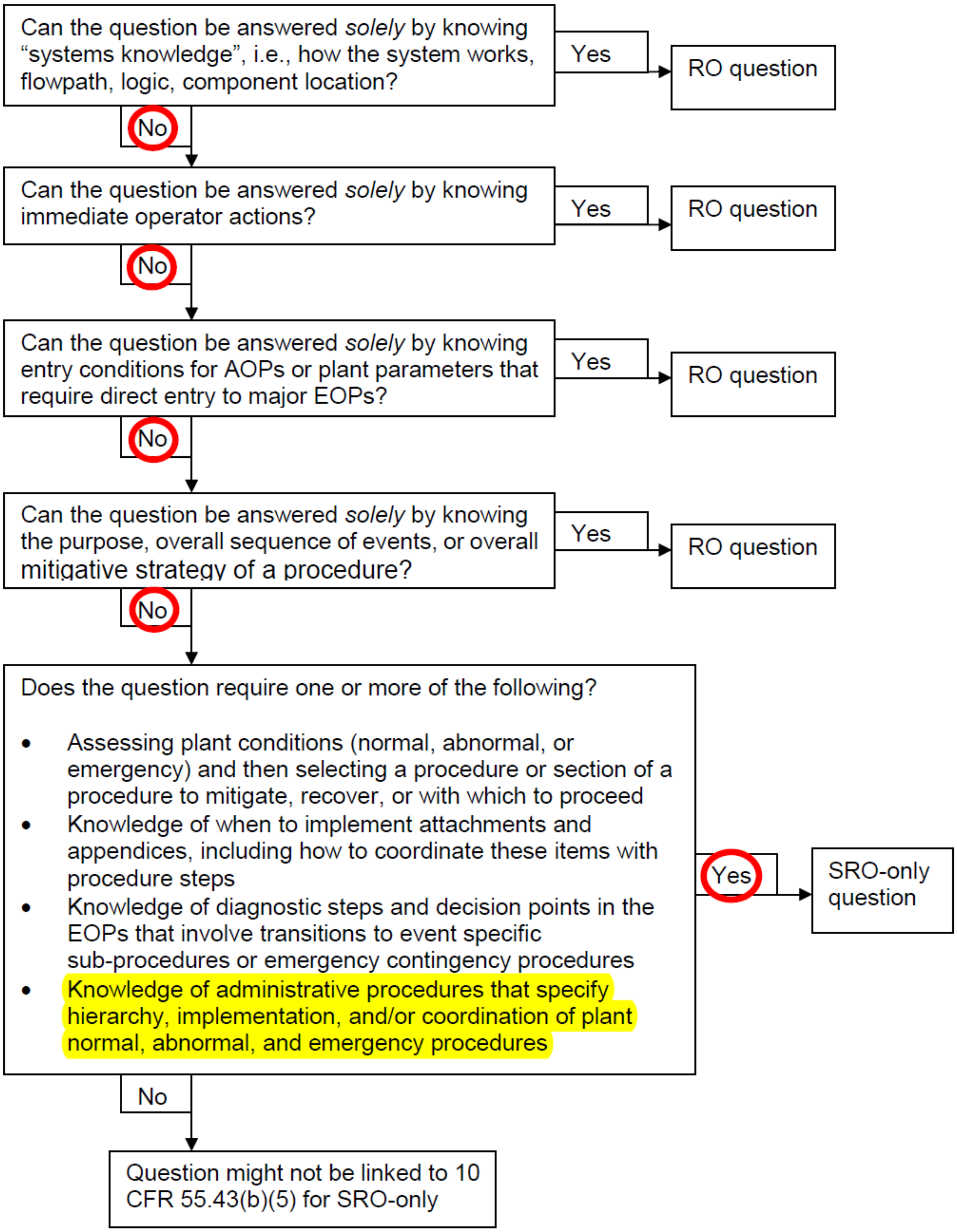
D. high enough to verify that a primary system is leaking into the secondary containment prior to a normal reactor shutdown.

Answer: C

K/A Match:
This question requires the candidate to demonstrate an understanding of how equipment operability is determined during a high secondary containment event.

SRO Only:
K/A is an “A2” statement and

Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5) (Assessment and selection of procedures)



Explanation:
A. Incorrect. Plausible since a RB ARM greater than the max safe operating value (MSOV) is a threshold for a Site Area Emergency EAL. However, this EAL will be called prior to emergency depressurization.
B. Incorrect. Plausible since emergency depressurization (ED) may be performed due to a RB ARM exceeding MSOV. However, RB ARMs exceeding MSOV is not the only requirement for ED. MSOV must be exceeded in two areas of the same parameter AND a primary system must be discharging into secondary containment prior to performing an ED.
C. Correct. The MSOV for RB ARMs is *“low enough to allow time for shutdown or isolation of a leak without exceeding the total integrated dose allowable for even the most sensitive safety related equipment”*.
D. Incorrect. Plausible since a normal reactor shutdown is performed if a parameter exceeds MSOV. However, it is only performed when a parameter exceeds MSOV in two areas AND a primary system IS NOT discharging into secondary containment.

Technical Reference(s)		Attached w/ Revision # See Comments / Reference
PPM 5.0.10, Flowchart Training Manual		
PPM 5.3.1, Secondary Containment Control		

Proposed references to be provided during examination: None

Learning Objective: 8456 - Define Maximum Safe Operating Value for the following secondary containment parameters: c.Area radiation levels (PPM 5.3.1)

Question Source: Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 ______
55.43 43.5

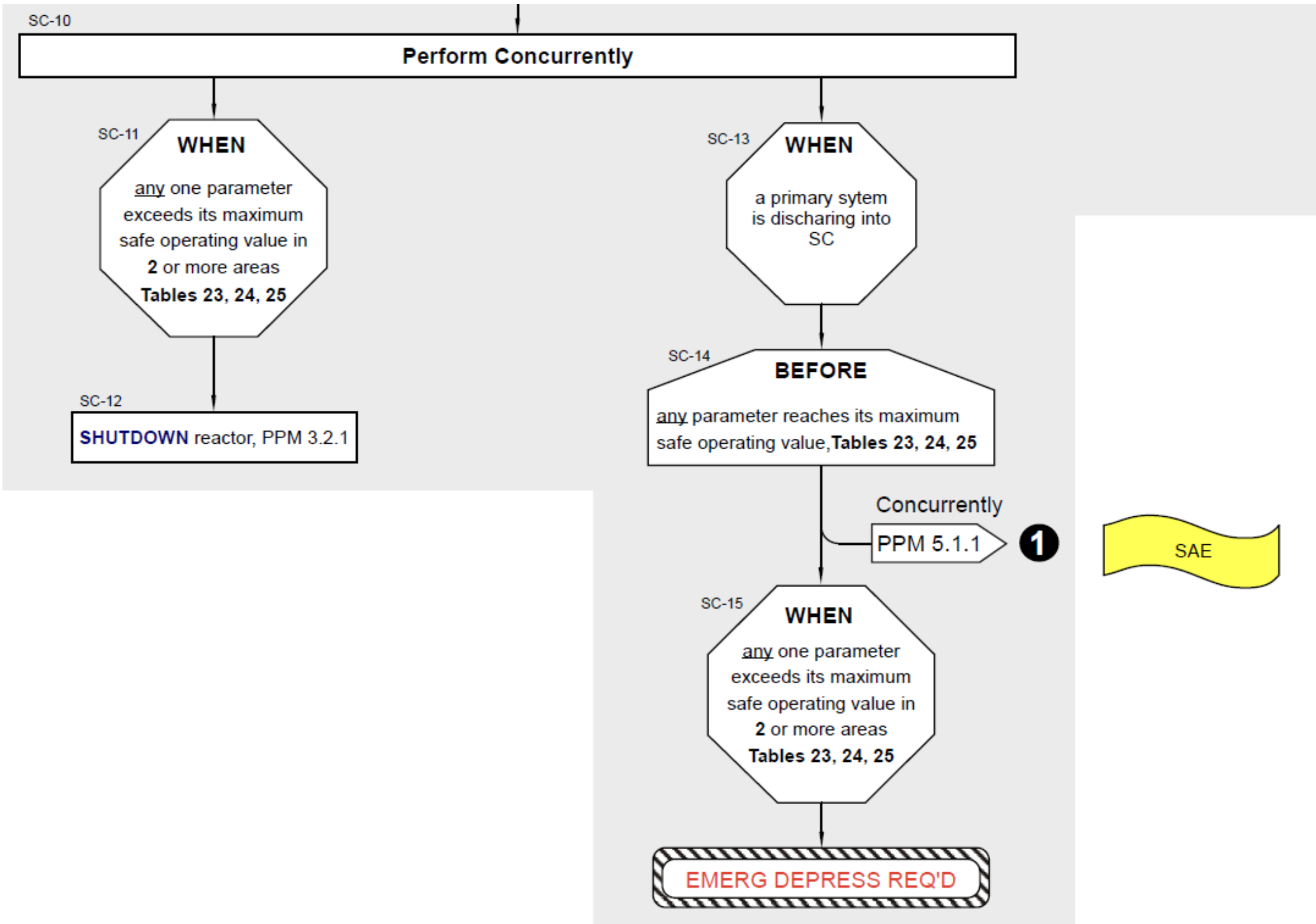
Comments / Reference: PPM 5.0.10	Revision: Major 21 Minor 001

Number: 5.0.10	Use Category: INFORMATION	Major Rev: 021 Minor Rev: 001 Page: 301 of 320
Title: Flowchart Training Manual		

10) An ARM alarm on any area radiation monitors in secondary containment except those on the refueling floor is defined to be an entry condition to this procedure. Since these setpoints vary due to plant operating mode and Health Physics radiation surveys, only nominal values which may not be accurate could be listed in Table 24 for the alarm level. A program is established to maintain the current setpoint values in PPM 4.602.A5 for annunciator window 3-1. Therefore, reference to the annunciator response procedure is identified in Table 24 for the entry condition value.

11) The maximum safe operating value is defined to be 10,000 Mr/hr in areas other than the refueling floor. This is the maximum indication on all but the high level instruments. This value is high enough to be indicative of substantial and immediate problems yet low enough to allow time for shutdown or isolation of a leak without exceeding the total integrated dose allowable for even the most sensitive safety related equipment. No area radiation levels are defined for the refueling floor because no primary systems are routed there.

Comments / Reference: PPM 5.3.1	Revision: 20



Examination Outline Cross-reference:	Level	RO		SRO
Rev. 1 Date: 12/16/2016	Tier			2
	Group			1
	K/A	205000 A2.06		
Level of Difficulty: 4	Importance Rating			3.5

Type the K/A System or Condition Here: Shutdown Cooling, Ability to (a) predict the impacts of the following on the SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE); and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: SDC/RHR pump trips.

Question # 86

CGS is operating in Mode 5.

- The reactor was shutdown 10 days ago for a refueling outage.
- Refueling operations are in progress. 15 fuel bundles have been removed from the core.
- RHR Loop A is in shutdown cooling (SDC).
- Reactor recirculation pumps are OOS for maintenance.
- The main condenser is OOS and open for tube cleaning.
- SW-P-1B has been placed OOS for corrective maintenance.

RHR-P-2A trips and cannot be restarted.

What actions should be taken?

The CRS should direct starting…

A. RHR-P-2B, with suction from the Spent Fuel Pool/FPC, and discharging through RHR-V-42B per SOP-FPC-ASSIST-ALT.

B. RHR-P-2C, with suction from the suppression pool and rejecting RPV water to the suppression pool per SOP-RHR-INJECTION.

C. RHR-P-2B, with suction from the Skimmer Surge Tanks and returning to the Fuel Pool Cooling system per SOP-FPC-ASSIST-ALT.

D. a condensate pump with suction from the condensate storage tank and rejecting RPV water to the suppression pool per ABN-RHR-SDC-ALT.

Answer: C

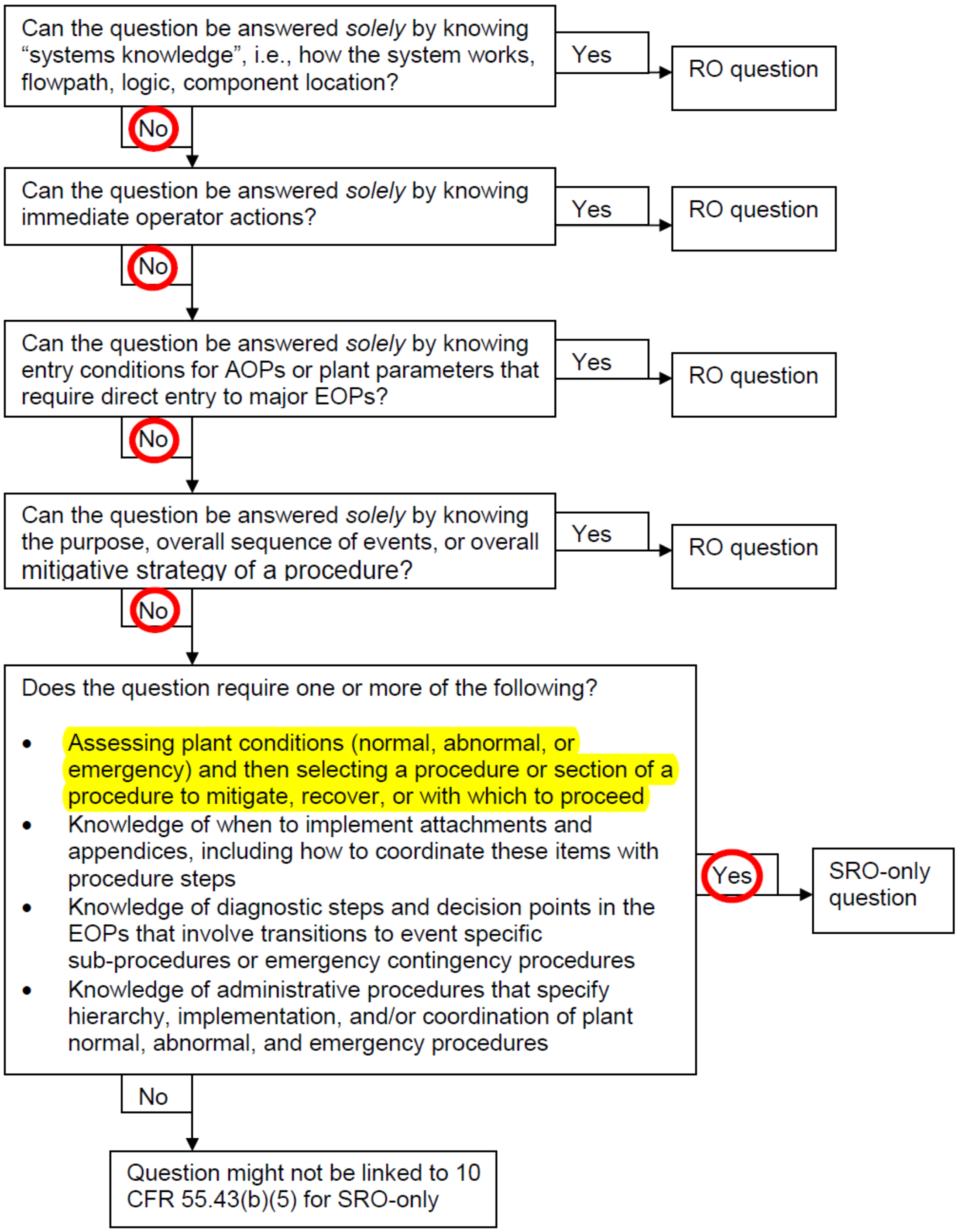
K/A Match:

This question requires the candidate to demonstrate an understanding of the impact on the shutdown cooling system on a loss of RHR heat exchanger cooling, and the method used to mitigate this loss by validating an alternate shutdown cooling method is available.

SRO Only:
K/A is an “A2” statement and

ES-401 **8** **Attachment 2**

**Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5)
(Assessment and selection of procedures)**



Explanation:
A. Incorrect. Plausible since this is an allowed method of alternate shutdown cooling listed in ABN-RHR-SDC-LOSS. However, this method can only be used if the reactor has been shutdown > 13 days and should not be used if fuel is removed from the core to preclude possible damage to core instrumentation. B. Incorrect. Plausible since this is an allowed method of alternate shutdown cooling listed in ABN-RHR-SDC-LOSS. However, this method requires a functional suppression pool cooling system in service, which is not available. C. Correct. With the given plant conditions, alternate shutdown cooling is provided by circulating water between the RPV and spent fuel pool via RHR-P-2B. D. Incorrect. Plausible since this is an allowed method of alternate shutdown cooling listed in ABN-RHR-SDC-LOSS. However, this method requires a functional suppression pool cooling system in service, which is not available.

Technical Reference(s)		Attached w/ Revision # See Comments / Reference
ABN-RHR-SDC-LOSS, Loss of Shutdown Cooling		

Proposed references to be provided during examination: None

Learning Objective: 11814 - Predict the impact of the following on the Residual Heat Removal System: b. Pump trips

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 ____
55.43 "A2" Statement

Comments / Reference: ABN-RHR-SDC-LOSS, section 4.8	Revision: 006

Number: ABN-RHR-SDC-LOSS	Use Category: CONTINUOUS	Major Rev: 006 Minor Rev: N/A Page: 7 of 14
Title: Loss of Shutdown Cooling		

CAUTION

Injection through RHR-V-42A(B)(C) is not the preferred flow path with any fuel removed from the core due to potential damage to in-core instrumentation.

4.8 IF Technical Specifications require availability of an alternate method of decay heat removal,
THEN **VERIFY** an alternate method of decay heat removal is available for each inoperable RHR SDC loop. N/A the methods not used.

- A functional, but inoperable SDC loop. _____
- Condensate system injection,
  AND RPV steam/water rejected to an operable Condenser or Suppression Pool, with a functional Suppression Pool Cooling system in service. _____
- LPCI A,B,C injection from Suppression Pool,
  AND RPV steam/water rejected to an operable Condenser or Suppression Pool, with a functional Suppression Pool Cooling system in service. _____
- RHR-P-2A(B) taking suction from the Spent Fuel Pool/FPC,
  AND discharging through RHR-V-53A(B) or RHR-V-42A(B),
  AND the reactor has been shut down for 13 days. _____
- RHR-P-2B taking suction from the Skimmer Surge Tanks,
  AND returning to the Fuel Pool Cavity per SOP-FPC-ASSIST-ALT, Sections 5.1 and 5.2. _____

Examination Outline Cross-reference:	Level	RO		SRO
Rev. 1 Date: 12/16/2016	Tier			2
	Group			1
	K/A	209001.2.4.9		
Level of Difficulty: 3	Importance Rating			4.2

LPCS: Knowledge of low power/shutdown implications in accident (e.g., loss of coolant accident or loss of residual heat removal) mitigation strategies.

Question # 87

CGS is operating in Mode 3.

- A LOCA occurred requiring entry into the EOPs.
- An Emergency Depressurization (ED) has been initiated due to low RPV level.

Current plant conditions:

- RPV level: -200 inches
- RPV pressure: 40 psig, down slow

How is adequate core cooling maintained?

A. Inject with RCIC to raise RPV level > -161 inches.

B. Establish RPV injection using LPCS at ≥ 6000 gpm.

C. Flood containment using all available injection sources.

D. Stabilize RPV pressure to allow Steam Cooling without injection.

Answer: B

K/A Match:
This question requires the candidate to demonstrate knowledge of the conditions when LPCS is required to mitigate effects of a Loss of Coolant Accident from a shutdown condition.

SRO Only:
K/A is a "G" statement and

ES-401 **8** **Attachment 2**

Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5) (Assessment and selection of procedures)

Can the question be answered *solely* by knowing "systems knowledge", i.e., how the system works, flowpath, logic, component location?

Yes → RO question

No

Can the question be answered *solely* by knowing immediate operator actions?

Yes → RO question

No

Can the question be answered *solely* by knowing entry conditions for AOPs or plant parameters that require direct entry to major EOPs?

Yes → RO question

No

Can the question be answered *solely* by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure?

Yes → RO question

No

Does the question require one or more of the following?

- Assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed
- Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps
- Knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific sub-procedures or emergency contingency procedures
- Knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures

Yes → SRO-only question

No

Question might not be linked to 10 CFR 55.43(b)(5) for SRO-only

Explanation:
A. Incorrect. Plausible since restoring RPV level above above -161 inches, which is Top of Active Fuel (TAF), will ensure adequate core cooling is maintained. However, with the given conditions in the question stem, there is insufficient steam pressure to operate the RCIC turbine. B. Correct. With RPV level below -198 inches but above -210 inches, injecting with HPCS OR LPCS with a flow rate ≥ 6000 gpm will maintain adequate core cooling. C. Incorrect. Plausible since flooding containment to provide core cooling is the strategy used in the Severe Accident Guidelines (SAGs). However, this strategy does not ensure adequate core cooling. Additionally, the SAGs are not entered until ALL EOP mitigation strategies are ineffective in maintaining core cooling. D. Incorrect. Plausible since Steam Cooling is an effective method for adequate core cooling. However, Steam Cooling is only effective when RPV level is ≥ -198 inches.

Technical Reference(s)		Attached w/ Revision # See Comments / Reference
PPM 5.0.10, Flowchart Training Manual		
PPM 5.1.1, RPV Control, RPV Level Leg		

Proposed references to be provided during examination: None

Learning Objective: LO8041 – List the four methods used to provide adequate core cooling in the EOPs.

Question Source: Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam None

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 ______
55.43 43.5

Comments / Reference: PPM 5.0.10, Flowchart Training Manual	Revision: 020

Number: 5.0.10	Use Category: INFORMATION	Major Rev: 020 Minor Rev: N/A Page: 17 of 321
Title: Flowchart Training Manual		

4.2 Definitions

4.2.1 The following words, acronyms and abbreviations are given specific definitions in order to ensure a correct and consistent interpretation of EOP flowchart terminology. A standard English language dictionary should be consulted for the definition of words and abbreviations not included in this list.

- ADD

  1) To combine; to place into.

- ADEQUATE CORE COOLING

  1) Heat removal from the reactor sufficient to prevent rupturing the fuel clad. Four viable mechanisms of adequate core cooling exist; in order of preference they are:

     - Core Submergence (-161 in.)
     - Steam cooling with injection of makeup water to the RPV (-186 in.)
     - Steam cooling without injection of makeup water to the RPV (-198 in.)
     - Spray cooling with HPCS or LPCS injecting at equal to or greater than 6000 gpm with RPV water level at or above 2/3 core height (-210 in.)

  2) Core submergence is the mechanism of core cooling whereby each fuel element is completely covered with water. Indicated RPV water level at or anywhere above the elevation corresponding to the top of active fuel (TAF) constitutes the principal means of confirming the adequacy of core cooling achieved through this mechanism. Assurance of continued adequate core cooling through core submergence is achieved when RPV water level can be maintained at or anywhere above TAF.

  3) Steam cooling is the mechanism of core cooling whereby steam updraft through the uncovered portion of the reactor core is sufficient to prevent the temperature of the hottest fuel rod from exceeding the appropriate limiting value, which is specific to the mode of steam cooling being employed. Two modes of steam cooling are employed in the EOP flowcharts with and without injection of makeup water to the RPV. For each mode, water in the covered portion of the reactor core and lower plenum is the source of the steam. A high fuel-to-steam differential temperature is required for the steam cooling methods of heat transfer to be effective.

Number: 5.0.10	Use Category: INFORMATION	Major Rev: 020 Minor Rev: N/A Page: 18 of 321
Title: Flowchart Training Manual		

4) Steam cooling with injection is employed in PPM 5.1.1, if RPV water level cannot be restored and maintained above the top of the active fuel; in PPM 5.1.6, during RPV flooding when the reactor may not be shutdown; and in PPM 5.1.2, if RPV water level is intentionally lowered below the top of the active fuel to reduce reactor power or if emergency RPV depressurization is required. When RPV water level cannot be restored and maintained above the top of the active fuel in PPM 5.1.1 and when RPV water level is intentionally lowered below the top of the active fuel in PPM 5.1.2, adequate steam flow is established by maintaining RPV water level above the Minimum Steam Cooling RPV Water Level. When the reactor may not be shutdown during RPV flooding in PPM 5.1.6 and when emergency RPV depressurization is required under failure-to-scram conditions in PPM 5.1.2, adequate steam flow exists as long as RPV pressure is above the Minimum Steam Cooling Pressure. In all cases, the peak clad temperature is limited to 1500°F, the threshold for fuel rod perforation.

5) With no injection into the RPV established, adequate core cooling exists only so long as the covered portion of the reactor core generates sufficient steam to preclude the peak clad temperature of the hottest fuel rod from exceeding 1800°F, the threshold temperature for significant metal-water reaction. This mechanism of core cooling is employed in PPM 5.1.1. Indicated RPV water level at or above the Minimum Zero-Injection RPV Water Level is the only means available for confirming the adequacy of core cooling achieved through this mechanism. The transient nature of this mechanism of adequate core cooling prevents being able to assure that it can be maintained.

6) Spray cooling with HPCS or LPCS injecting at or above rated flow (6000 gpm) into the RPV assures adequate core cooling when RPV water level is at or above 2/3 core height (-210 in.). This strategy implements Columbia design basis that ensures peak cladding temperature under design basis events is less than the maximum peak cladding temperature (2200°F) allowed per 10CFR50.46. This means of providing adequate core cooling is only relied upon in PPM 5.1.1, RPV Control.

Comments / Reference: PPM 5.7.1, RPV and Primary Containment Flooding SAG	Revision: Major 006 Minor 001

Number: 5.7.1	Use Category: INFORMATION	Major Rev: 006 Minor Rev: 001 Page: 4 of 148
Title: RPV AND PRIMARY CONTAINMENT FLOODING SAG		

1.0 INTRODUCTION

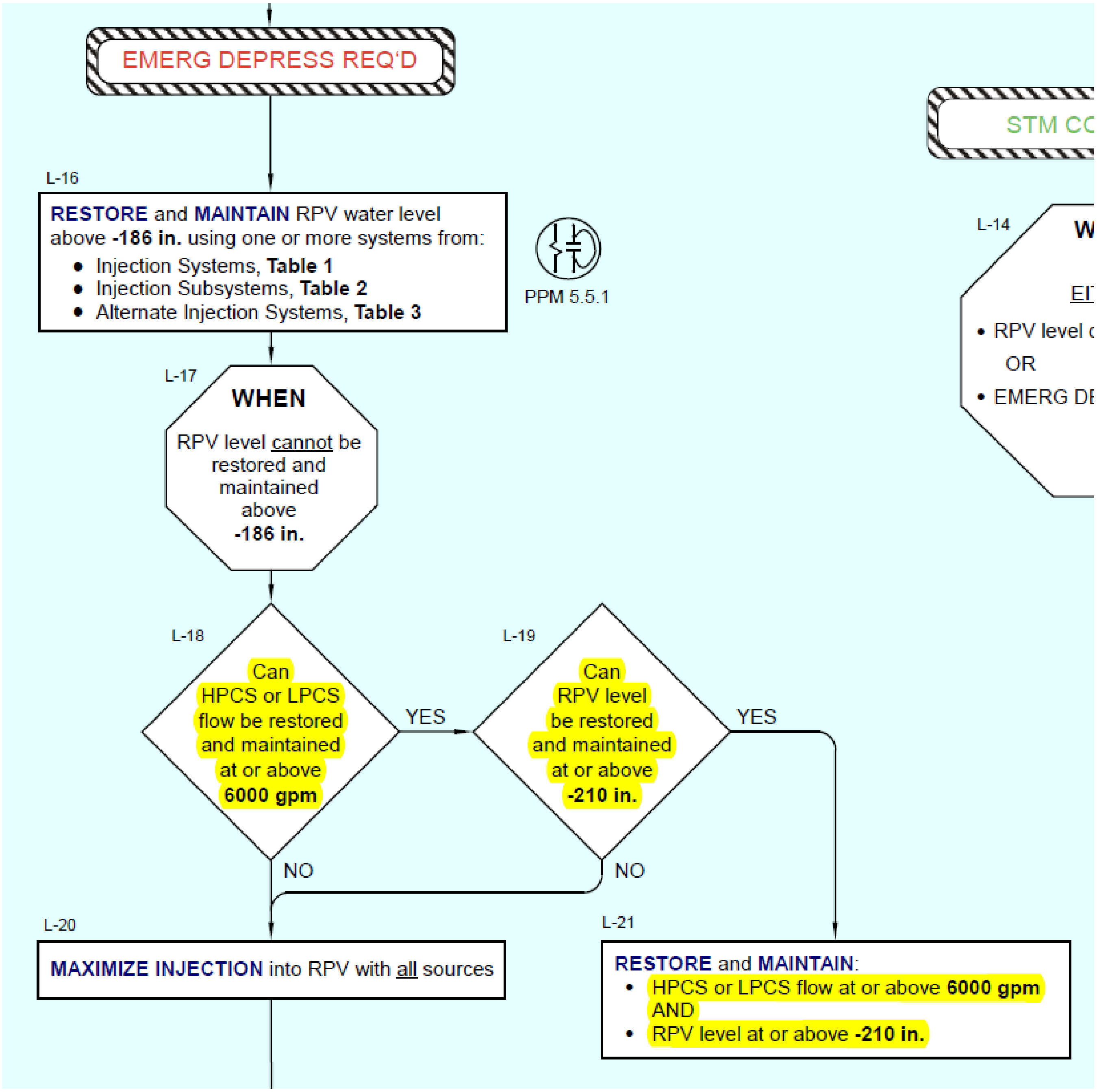
Severe accidents are generally defined to begin with the onset of core damage. The SAGs coordinate control of key plant parameters under severe accident conditions and provide guidance on flooding the primary containment if appropriate to submerge the core and core debris.

1.1 SAG Entry is Required When either:

- HPCS or LPCS flow at or above 6,000 gpm cannot restore and maintain RPV water level at or above -210 in.
- RPV water level cannot be restored and maintained above -183 in.
- If RPV water level cannot be determined, it is determined that substantial core damage is occurring due to loss of core cooling

Under these conditions, the core cannot be adequately cooled using all available RPV injection sources. When SAG entry is required, all EOP flowchart parameter control functions in PPM 5.1.1 through PPM 5.4.1 transfer to the SAGs—RPV control functions to SAG-1, RPV and Primary Containment Flooding, and all primary containment, secondary containment, and radioactivity release control functions to SAG-2, Containment and Radioactivity Release Control. Any additional EOP flowchart entry conditions may then be disregarded.

Comments / Reference: PPM 5.1.1, RPV Control, RPV Level Leg (Post Emergency Depressurization)	Revision: 21



Examination Outline Cross-reference:	Level	RO		SRO
Rev. 0 Date:	Tier			2
	Group			1
	K/A	215003.A2.05		
Level of Difficulty: 3	Importance Rating			3.5

IRM: Ability to (a) predict the impacts of the following on the INTERMEDIATE RANGE MONITOR (IRM) SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Faulty or erratic operation of detectors/system.

Question # 88

CGS is operating in Mode 2.

A reactor startup is in progress in accordance with PPM 3.1.2, Startup Flow Chart.

Current Conditions:

- IRM-B is bypassed due to a failed power supply. Repair parts are on order.
- All other IRM channels are on range 10

The crew is verifying IRM/APRM overlap when IRM channel D fails.

- Maintenance estimates 14 hours to repair IRM channel D.

Using the provided reference, what is the earliest action that will satisfy all technical specifications requirements?

A. Place the reactor mode switch in “Run” within 8 hours.

B. Place IRM-D mode switch in “Standby” within 10 hours.

C. Repair IRM-D and return it to operable status within 14 hours.

D. Place the reactor mode switch in “Shutdown” within 24 hours.

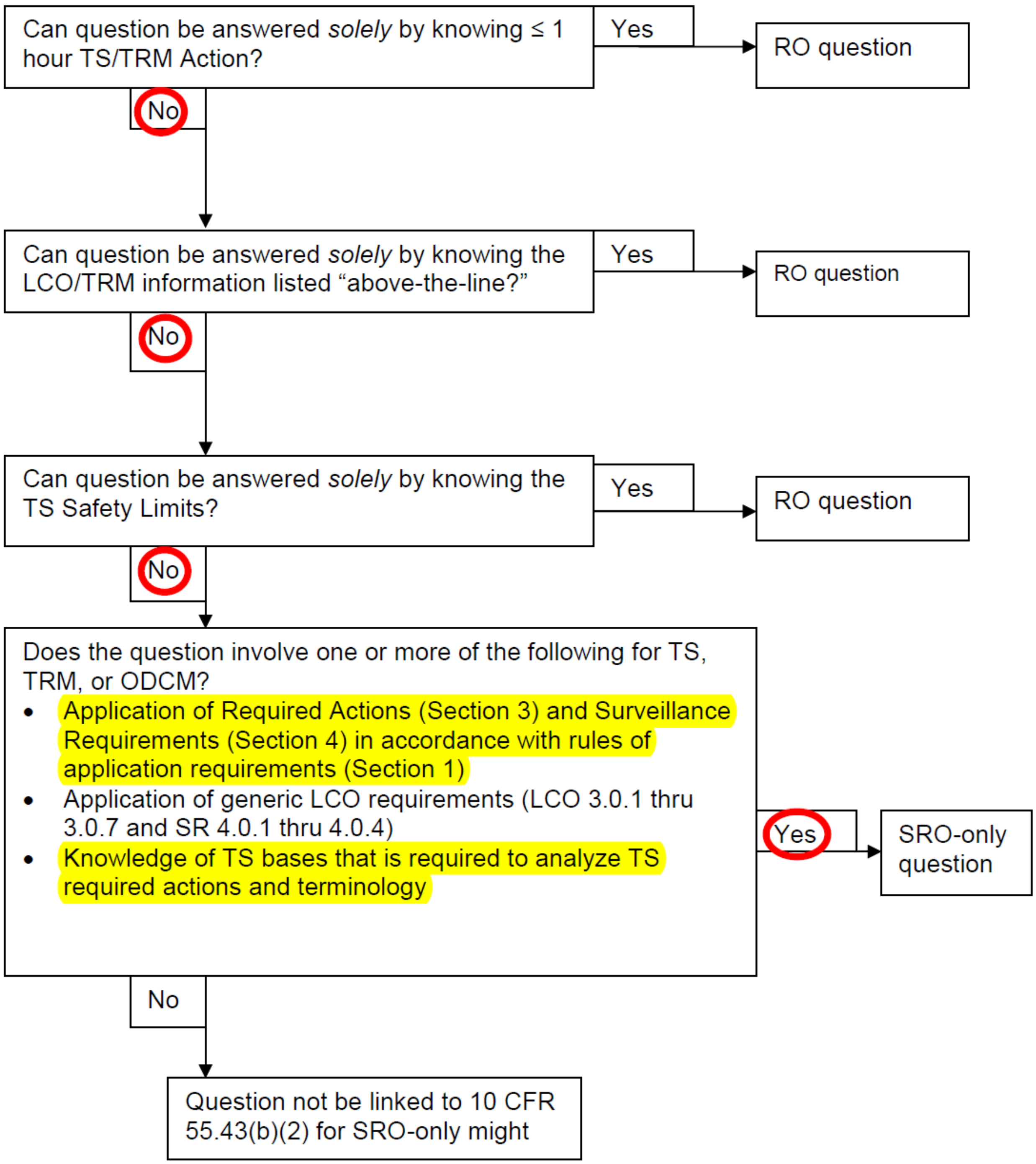
Answer: A

K/A Match:
The question requires the candidate to understand that the IRM failure cause entry into a technical specification Limiting Condition for Operation, and identify the actions required to meet technical specification requirements.

SRO Only:
K/A is an “A2” statement and

ES-401 5 Attachment 2

Figure 1: Screening for SRO-only linked to 10 CFR 55.43(b)(2) (Tech Specs)



Explanation:
A. Correct. Placing the reactor mode switch in “Run” puts the reactor in Mode 1, which is outside the mode of applicability of TS 3.3.1.1 for IRMs. Even though IRM/APRM overlap cannot be completed with two inoperable IRM channels on a single RPS system, this overlap is ONLY required to be met during entry into Mode 2 from Mode 1 (reactor shutdown). B. Incorrect. Plausible since placing the IRM mode switch in “Standby” inserts a trip signal and meets LCO 3.3.1.1, condition A requirements to trip the failed channel. However, it is not the earliest action listed as required by the question stem. C. Incorrect. Plausible since returning IRM-D to operable within 14 hours places the plant outside the mode of applicability for LCO 3.3.1.1 prior to a required plant shutdown per LCO 3.3.1.1, condition G. However, it is not the earliest action listed as required by the question stem. D. Incorrect. Placing the reactor mode switch in “Shutdown” puts the reactor in Mode 3, which is outside the mode of applicability of TS 3.3.1.1 for IRMs. However, it is not the earliest action listed as required by the question stem.

Technical Reference(s)		Attached w/ Revision # See Comments / Reference
CGS Technical Specifications		
CGS Technical Specifications Bases		

Proposed references to be provided during examination: TS 3.3.1.1,

Learning Objective: 7637 – Predict the effects that a failure of the IRM system will have on the following: RPS

Question Source: Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 Best match only
55.43 ______

Comments / Reference: CGS Technical Specifications (Proposed Reference)	Revision: 237

3.3 INSTRUMENTATION

3.3.1.1 Reactor Protection System (RPS) Instrumentation

LCO 3.3.1.1 The RPS instrumentation for each Function in Table 3.3.1.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.1.1-1 after implementation of Power Range Neutron Monitor (PRNM) upgrade.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required channels inoperable.	A.1 Place channel in trip.	12 hours
	<u>OR</u>	
	-----NOTE----- Not applicable for Functions 2.a, 2.b, 2.c, 2.d, or 2.f. -----	
	A.2 Place associated trip system in trip.	12 hours
-----NOTE----- Not applicable for Functions 2.a, 2.b, 2.c, 2.d, or 2.f. -----	B.1 Place channel in one trip system in trip.	6 hours
	<u>OR</u>	
B. One or more Functions with one or more required channels inoperable in both trip systems.	B.2 Place one trip system in trip.	6 hours

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. One or more Functions with RPS trip capability not maintained.	C.1	Restore RPS trip capability.	1 hour
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1	Enter the Condition referenced in Table 3.3.1.1-1 for the channel.	Immediately
E. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	E.1	Reduce THERMAL POWER to < 30% RTP.	4 hours
F. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	F.1	Be in MODE 2.	6 hours
G. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	G.1	Be in MODE 3.	12 hours
H. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	H.1	Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
I. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	I.1 Initiate alternate method to detect and suppress thermal hydraulic instability oscillations. AND ---------------NOTE------------- LCO 3.0.4 is not applicable. -------------------------------------	12 hours
	I.2 Restore required channels to OPERABLE	120 days
J. Required Action and associated Completion Time of Condition I not met.	J.1 Reduce THERMAL POWER to less than the value specified in the COLR.	4 hours

SURVEILLANCE REQUIREMENTS

--NOTES--

1. Refer to Table 3.3.1.1-1 to determine which SRs apply for each RPS Function.
2. When a 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 RPS trip capability.

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.1	Perform CHANNEL CHECK.	12 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.3.1.1.2	------------------------------NOTE------------------------------ Not required to be performed until 12 hours after THERMAL POWER ≥ 25% RTP. --- Verify the absolute difference between the average power range monitor (APRM) channels and the calculated power ≤ 2% RTP while operating at ≥ 25% RTP.	7 days
SR 3.3.1.1.3	------------------------------NOTE------------------------------ Not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. --- Perform CHANNEL FUNCTIONAL TEST.	7 days
SR 3.3.1.1.4	Perform CHANNEL FUNCTIONAL TEST.	7 days
SR 3.3.1.1.5	Verify the source range monitor (SRM) and intermediate range monitor (IRM) channels overlap.	Prior to withdrawing SRMs from the fully inserted position
SR 3.3.1.1.6	------------------------------NOTE------------------------------ Only required to be met during entry into MODE 2 from MODE 1. --- Verify the IRM and APRM channels overlap.	7 days
SR 3.3.1.1.7	Calibrate the local power range monitors.	1130 MWD/T average core exposure

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.8	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.1.1.9	Deleted.	
SR 3.3.1.1.10	-----------------------------NOTES----------------------------- 1. Neutron detectors are excluded. 2. For Function 1, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. 3. For Functions 2.b and 2.f, the recirculation flow transmitters that feed the APRMs are included. --- Perform CHANNEL CALIBRATION.	18 months for Functions 1, 3, 4, 6, 7, and 9 through 11 <u>AND</u> 24 months for Functions 2, 5, and 8
SR 3.3.1.1.11	Deleted.	
SR 3.3.1.1.12	Verify Turbine Throttle Valve - Closure, and Turbine Governor Valve Fast Closure Trip Oil Pressure - Low Functions are not bypassed when THERMAL POWER is $\geq$ 30% RTP.	18 months
SR 3.3.1.1.13	Perform CHANNEL FUNCTIONAL TEST.	24 months

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.14	Perform LOGIC SYSTEM FUNCTIONAL TEST.	24 months
SR 3.3.1.1.15	-----------------------------NOTES---------------------------- 1. Neutron detectors are excluded. 2. Channel sensors for Functions 3 and 4 are excluded. 3. For Function 5, "n" equals 4 channels for the purpose of determining the STAGGERED TEST BASIS Frequency. 4. For Function 2.e, "n" equals 8 channels for the purpose of determining the STAGGERED TEST BASIS Frequency. Testing of APRM and oscillation power range monitor (OPRM) outputs shall alternate. --- Verify the RPS RESPONSE TIME is within limits.	24 months on a STAGGERED TEST BASIS
SR 3.3.1.1.16	-----------------------------NOTES---------------------------- 1. For Function 2.a, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. 2. For Functions 2.b and 2.f, the CHANNEL FUNCTIONAL TEST includes the recirculation flow input processing, excluding the flow transmitters. --- Perform CHANNEL FUNCTIONAL TEST.	184 days
SR 3.3.1.1.17	Verify the OPRM is not bypassed when APRM Simulated Thermal Power is greater than or equal to the value specified in the COLR and recirculation drive flow is less than the value specified in the COLR.	24 months

Table 3.3.1.1-1 (page 1 of 4)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Intermediate Range Monitors					
a. Neutron Flux - High	2	3	G	SR 3.3.1.1.1 SR 3.3.1.1.3 SR 3.3.1.1.5 SR 3.3.1.1.6 SR 3.3.1.1.10 SR 3.3.1.1.14	≤ 122/125 divisions of full scale
	5[a]	3	H	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.10 SR 3.3.1.1.14	≤ 122/125 divisions of full scale
b. Inop	2	3	G	SR 3.3.1.1.3 SR 3.3.1.1.14	NA
	5[a]	3	H	SR 3.3.1.1.4 SR 3.3.1.1.14	NA
2. Average Power Range Monitors					
a. Neutron Flux - High (Setdown)	2	3[b]	G	SR 3.3.1.1.1 SR 3.3.1.1.6 SR 3.3.1.1.7 SR 3.3.1.1.10[d],[e] SR 3.3.1.1.16	≤ 20% RTP
b. Simulated Thermal Power - High	1	3[b]	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.7 SR 3.3.1.1.10[d],[e] SR 3.3.1.1.16	≤ 0.63W + 64.0% RTP and ≤ 114.9% RTP[c]

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

(b) Each APRM/OPRM channel provides inputs to both trip systems.

(c) ≤ 0.63W + 60.8% RTP and ≤ 114.9% RTP when reset for single loop operation per LCO 3.4.1, "Recirculation Loops Operating."

(d) If the as-found channel setpoint is outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

(e) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Limiting Trip Setpoint (LTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the LTSP are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the surveillance procedures (Nominal Trip Setpoint) to confirm channel performance. The LTSP and the methodologies used to determine the as-found and as-left tolerances are specified in the Licensee Controlled Specifications.

Table 3.3.1.1-1 (page 2 of 4)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. Average Power Range Monitors					
c. Neutron Flux - High	1	3(b)	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.7 SR 3.3.1.1.10(d),(e) SR 3.3.1.1.16	≤ 120% RTP
d. Inop	1,2	3(b)	G	SR 3.3.1.1.16	NA
e. 2-Out-of-4 Voter	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.14 SR 3.3.1.1.15 SR 3.3.1.1.16	NA
f. OPRM Upscale	(f)	3(b)	I	SR 3.3.1.1.1 SR 3.3.1.1.7 SR 3.3.1.1.10(d),(e) SR 3.3.1.1.16 SR 3.3.1.1.17	NA(g)

(b) Each APRM/OPRM channel provides inputs to both trip systems.

(d) If the as-found channel setpoint is outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

(e) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Limiting Trip Setpoint (LTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the LTSP are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the surveillance procedures (Nominal Trip Setpoint) to confirm channel performance. The LTSP and the methodologies used to determine the as-found and as-left tolerances are specified in the Licensee Controlled Specifications.

(f) THERMAL POWER greater than or equal to the value specified in the COLR.

(g) The OPRM Upscale does not have an Allowable Value. The Period Based Detection Algorithm (PBDA) trip setpoints are specified in the COLR.

Table 3.3.1.1-1 (page 3 of 4)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
3. Reactor Vessel Steam Dome Pressure - High	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.14 SR 3.3.1.1.15	≤ 1079 psig
4. Reactor Vessel Water Level - Low, Level 3	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.14 SR 3.3.1.1.15	≥ 9.5 inches
5. Main Steam Isolation Valve - Closure	1	8	F	SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.14 SR 3.3.1.1.15	≤ 12.5% closed
6. Primary Containment Pressure - High	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.14	≤ 1.88 psig
7. Scram Discharge Volume Water Level - High					
a. Transmitter/Trip Unit	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.14	≤ 529 ft 9 inches elevation
	5(a)	2	H	SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.14	≤ 529 ft 9 inches elevation
b. Float Switch	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.14	≤ 529 ft 9 inches elevation
	5(a)	2	H	SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.14	≤ 529 ft 9 inches elevation
8. Turbine Throttle Valve - Closure	≥ 30% RTP	4	E	SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.12 SR 3.3.1.1.14 SR 3.3.1.1.15	≤ 7% closed

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

Table 3.3.1.1-1 (page 4 of 4)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
9. Turbine Governor Valve Fast Closure, Trip Oil Pressure - Low	≥ 30% RTP	2	E	SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.12 SR 3.3.1.1.14 SR 3.3.1.1.15	≥ 1000 psig
10. Reactor Mode Switch - Shutdown Position	1,2	2	G	SR 3.3.1.1.13 SR 3.3.1.1.14	NA
	5(a)	2	H	SR 3.3.1.1.13 SR 3.3.1.1.14	NA
11. Manual Scram	1,2	2	G	SR 3.3.1.1.4 SR 3.3.1.1.14	NA
	5(a)	2	H	SR 3.3.1.1.4 SR 3.3.1.1.14	NA

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

Comments / Reference: CGS Technical Specifications Bases	Revision: 92

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The Intermediate Range Monitor Neutron Flux - High Function must be OPERABLE during MODE 2 when control rods may be withdrawn and the potential for criticality exists. In MODE 5, when a cell with fuel has its control rod withdrawn, the IRMs provide monitoring for and protection against unexpected reactivity excursions. In MODE 1, the APRM System, the RWM and Rod Block Monitor provide protection against control rod withdrawal error events and the IRMs are not required. The IRMs are automatically bypassed when the mode switch is in the run position.

<u>1.b. Intermediate Range Monitor (IRM) - Inop</u>

This trip signal provides assurance that a minimum number of IRMs are OPERABLE. Anytime an IRM mode switch is moved to any position other than "Operate," the detector voltage drops below a preset level, loss of the negative DC voltage, or a module is not plugged in, an inoperative trip signal will be received by the RPS unless the IRM is bypassed. Since only one IRM in each trip system may be bypassed, only one IRM in each RPS trip system may be inoperable without resulting in an RPS trip signal.

Examination Outline Cross-reference:	Level	RO		SRO
Rev. 0 Date: 12/16/2016	Tier			2
	Group			1
	K/A	218000 2.4.6		
Level of Difficulty: 3	Importance Rating			4.7

ADS: Knowledge of EOP mitigation strategies.

Question # 89

CGS is operating in Mode 1.

- A LOCA occurs requiring the crew to enter the EOPs.
- The reactor mode switch is in shutdown.
- Reactor power is 10% and steady.
- RPV level: +25 inches, down slow.
- MSIVs are open.

Why is ADS inhibited?

ADS is inhibited to prevent…

A. adding positive reactivity from LPCI injection.

B. loss of adequate core cooling by depressurizing the RPV.

C. a severe thermal transient from increased HPCS injection rate.

D. impacting RPV level recovery by degrading RCIC operation.

Answer: A

K/A Match:
The question requires the candidate to demonstrate knowledge of the reasons for inhibiting ADS in the EOPs, specifically PPM 5.1.2, RPV Control – ATWS.

SRO Only:
K/A is a "G" statement and

**Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5)
(Assessment and selection of procedures)**

Can the question be answered *solely* by knowing "systems knowledge", i.e., how the system works, flowpath, logic, component location?
Yes → RO question
No

Can the question be answered *solely* by knowing immediate operator actions?
Yes → RO question
No

Can the question be answered *solely* by knowing entry conditions for AOPs or plant parameters that require direct entry to major EOPs?
Yes → RO question
No

Can the question be answered *solely* by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure?
Yes → RO question
No

Does the question require one or more of the following?

- Assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed
- Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps
- Knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific sub-procedures or emergency contingency procedures
- Knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures

Yes → SRO-only question
No

Question might not be linked to 10 CFR 55.43(b)(5) for SRO-only

Explanation:
A. Correct. In an ATWS condition, ADS is inhibited to prevent positive reactivity addition via an uncontrolled addition of cold, unborated water from low pressure injection sources (LPCI or LPCS).
B. Incorrect. Plausible since depressurization with RPV level below Top of Active Fuel (TAF) could cause a loss of adequate core cooling if RCIC were the only available injection source. However, the conditions given in the stem do not support this.
C. Incorrect. Plausible since ADS may cause a severe thermal transient. However, the transient is not from an increase in HPCS injection, but rather from uncontrolled low pressure injection (LPCI/LPCS). Additionally, during an ATWS, HPCS is manually secured.
D. Incorrect. Plausible since RCIC operation is degraded if RPV pressure is reduced by ADS. However, in this situation, RPV level is intentionally lowered to assist in reducing reactor power.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
PPM 5.0.10, Flowchart Training Manual	
PPM 5.1.2, RPV Control - ATWS	

Proposed references to be provided during examination: None

Learning Objective: LO8089 – Given a list, identify the statement that describes the two reasons that ADS is inhibited during an ATWS

Question Source: Bank # ____
Modified Bank # LO00098 (Note changes or attach parent)
New ____

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 ____
55.43 43.5

Comments / Reference: PPM 5.0.10, Flowchart Training Manual	Revision: Major 21 Minor 001

i. The reactor power control section first attempts to insert control rods and shut down the reactor with those actions which can be quickly performed by the control room operator; such as, placing the mode switch to shutdown, and initiating ARI. (Reducing reactor recirculation flow also reduces reactor power but this is accomplished automatically when the RRC pumps trip on low RPV water level.) If these immediate actions fail in inserting sufficient negative reactivity to shut down the reactor, the operator is then directed to implement alternate rod insertion methods in accordance with PPM 5.5.11, Alternate Rod Insertion. Concurrently, before Wetwell temperature reaches 110°F or when periodic neutron flux oscillations greater than 25% commence and continue, boron is injected into the RPV with the SLC system or alternate boron injection with the RCIC system. If reactor power is elevated (above the APRM downscale trip setpoint) or cannot be determined, however, boron injection is initiated immediately in PPM 3.3.1 without the need to assess the status of Wetwell temperature or check for neutron flux oscillations. Boron injection is continued until reactor shutdown can be assured on control rod insertion.

8.3.3 Flowchart Technical Discussion

a. Entry Conditions:

1) PPM 5.2.1 is entered from PPM 5.1.1

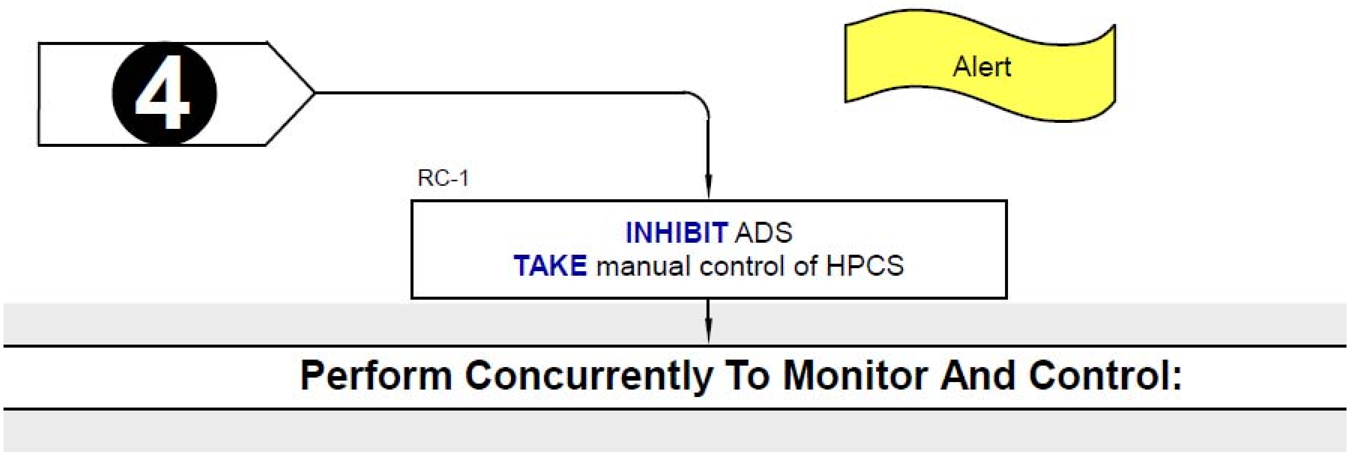
2) As an additional aid to the operator, a small table of PPM 5.1.1 entry conditions is provided in the upper left corner of PPM 5.1.2 to serve as a reminder for re-entry back into PPM 5.1.1 upon occurrence of an RPV Control entry condition.

b. Step RC-1:

1) Actions in the RPV level control flowpath may deliberately lower RPV water level below the automatic initiation setpoint of ADS. Actuation of ADS imposes a severe thermal transient on the RPV and complicates the efforts to maintain RPV water level within prescribed ranges Rapid and uncontrolled injection of relatively cold, unborated water from low pressure injection systems may occur as RPV pressure decreases below the shutoff heads of these pumps. This would quickly dilute boron concentration in the core and reduce reactor coolant temperature. When the reactor is not shutdown, or when the shutdown margin is small, sufficient positive reactivity might be added in this way to cause a reactor power excursion large enough to severely damage the core. Therefore, ADS initiation is purposely prevented as the first action of the failure-to-scram procedure.

2) When required to initiate ADS, explicit requirements to depressurize the RPV are provided throughout the EOP flowcharts, thereby negating any need to maintain the automatic initiation capability of ADS.

Comments / Reference: PPM 5.1.2, RPV Control – ATWS	Revision: 24



Comments / Reference: Bank Question LO00098	Revision: N/A

The first step in PPM 5.1.2, RPV Control - ATWS is to “Inhibit ADS”

Which of the following is the basis for this action?

A. This action prevents unwanted depressurization of the RPV which could result in uncontrolled injection from unborated water causing a power excursion.

B. The ADS system is not analyzed for ATWS conditions, more controlled pressure reductions are performed manually when conditions in the EOPs direct it.

C. This action prevents unwanted depressurization of the RPV which could result in uncontrolled reactivity addition due to addition of voids in the core.

D. The ADS system is not analyzed for ATWS conditions. Auto actuation at the ADS setpoint would be premature for ATWS conditions.

Correct Answer: A

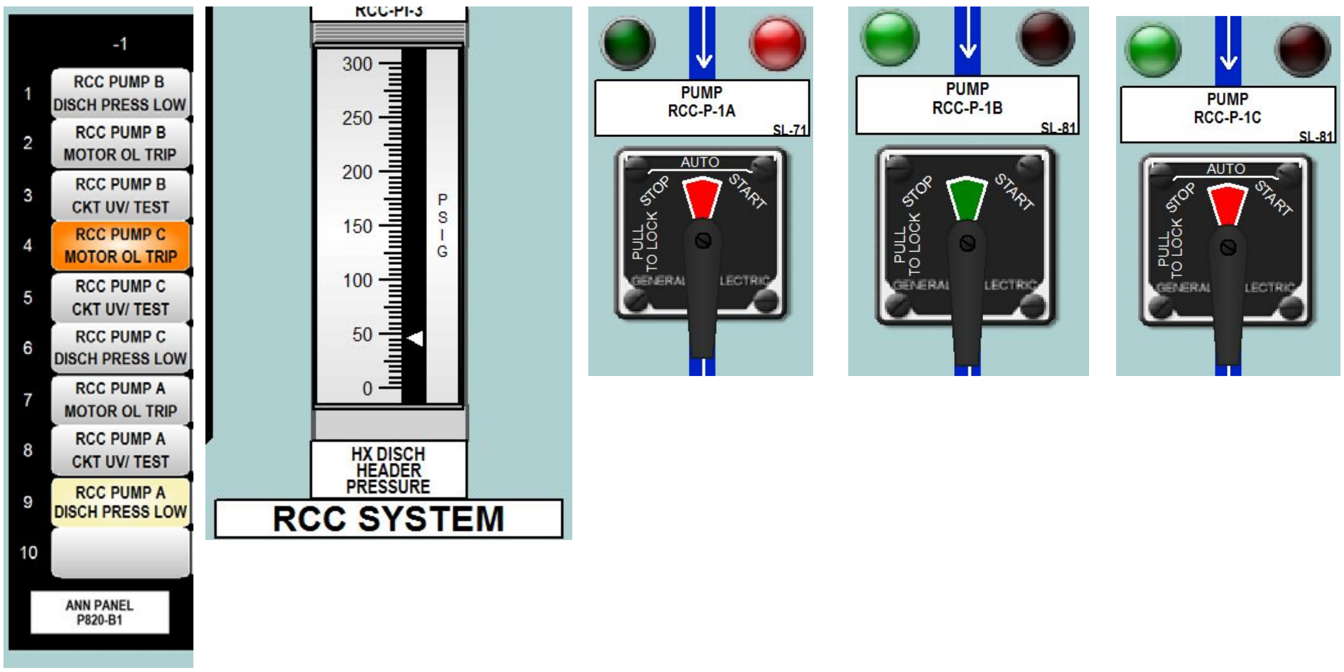
Examination Outline Cross-reference:	Level	RO		SRO
Rev. 0 Date: 12/16/2016	Tier			2
	Group			1
	K/A	400000.A2.01		
Level of Difficulty:	Importance Rating			3.4

<u>Component Cooling Water</u>: Ability to (a) predict the impacts of the following on the CCWS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation: Loss of CCW pump.	
Question # 90	

CGS is operating in Mode 1.

- RCC-P-1A and RCC-P-1C are running.
- RCC-P-1B is in standby

An event causes the following:



What actions should be taken?

The CRS should direct the crew to…

A. scram the reactor and start RCC-P-1B.

B. start RCC-P-1B and open RCC-V-6, Radwaste/Rx Bldg Supply.

C. scram the reactor and secure both Reactor Recirculation (RRC) pumps.

D. verify RCC-P-1A is running and place RCC-P-1B and RCC-P-1C in PTL.

Answer: B	

K/A Match:
This question requires the candidate to demonstrate knowledge that one running RCC pump tripped and that the standby RCC pump failed to automatically start, and the actions necessary to mitigate this event.

SRO Only:
K/A is an "A2" statement and

Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5) (Assessment and selection of procedures)

Can the question be answered *solely* by knowing "systems knowledge", i.e., how the system works, flowpath, logic, component location?
Yes → RO question
No ↓

Can the question be answered *solely* by knowing immediate operator actions?
Yes → RO question
No ↓

Can the question be answered *solely* by knowing entry conditions for AOPs or plant parameters that require direct entry to major EOPs?
Yes → RO question
No ↓

Can the question be answered *solely* by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure?
Yes → RO question
No ↓

Does the question require one or more of the following?

- Assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed
- Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps
- Knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific sub-procedures or emergency contingency procedures
- Knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures

Yes → SRO-only question
No ↓

Question might not be linked to 10 CFR 55.43(b)(5) for SRO-only

Explanation:
A. Incorrect. Plausible since a reactor scram is initiated on a complete loss of RCC where no pumps can be restarted and starting RCC-P-1B is the correct action since it should have started automatically. However, RCC-P-1A is still running and a reactor scram is not required. B. Correct. Two RCC pumps are normally running with the third pump in standby. The standby pump will automatically start when the breaker for one of the running pumps opens. In accordance with ABN-RCC, section 4.2, the operator will start the standby RCC pump if not already running and then reopen RCC-V-6 if two RCC pumps are running. C. Incorrect. Plausible since a scramming the reactor and securing RRC are the correct actions on a complete loss of RCC where no pumps can be restarted. However, RCC-P-1A is still running and these actions are not required. D. Incorrect. Plausible since these are the immediate actions for a loss of RCC-P-1B and RCC-P-1C due to a loss of bus SL-81 (see ABN-ELEC-SM3/SM8). However, when SL-81 is lost, the breakers for RCC-P-1B and RCC-P-1C remain closed and the Red indicating light for RCC-P-1C will remain lit. Additionally, “RCC PUMP C MOTOR OL” annunciator would not be lit.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
ABN-RCC, Loss of RCC	
ABN-ELEC-SM3/SM8, SM-3, SM-8, SM-85, SM-82, SL-81, SL-83 & SL-31 Distribution System Failures	
SD000196, CGS System Description, Volume 3, Chapter 1, Reactor Closed Cooling Water	

Proposed references to be provided during examination: None

Learning Objective: LO5706 - Explain the interlocks associated with the following components or system conditions, including setpoints: a. RCC pump auto start, b. RCC Pump trips.

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 ____
55.43 “A2” statement

Comments / Reference: ABN-RCC, Loss of RCC	Revision: Major 006 Minor 003

Number: ABN-RCC	Use Category: CONTINUOUS	Major Rev: 006 Minor Rev: 003 Page: 3 of 12
Title: Loss of RCC		

1.0 ENTRY CONDITIONS

Unplanned reduction in RCC cooling.

2.0 AUTOMATIC ACTIONS

2.1 The standby RCC pump starts when a running RCC pump trips.

2.2 RCC-V-6 **CLOSES** (RW/RB Supply) when LT two RCC pump motor breakers are **CLOSED**.

2.3 RWCU-V-4, RWCU (Suction Outboard Isolation) **CLOSES** on RWCU non-regenerative heat exchanger outlet temperature GT 140°F.

2.4 The following components will trip or isolate on an F (1.68 psig) or A (-50") signal:

- RCC-P-1A, Reactor Closed Cooling Pump
- RCC-P-1B, Reactor Closed Cooling Pump
- RCC-P-1C, Reactor Closed Cooling Pump
- RCC-V-104, Drywell Supply Outboard Isolation
- RCC-V-21, Drywell Return Outboard Isolation
- RCC-V-5, Drywell Supply Inboard Isolation
- RCC-V-40, Drywell Return Inboard Isolation

3.0 IMMEDIATE OPERATOR ACTIONS

None

4.0 SUBSEQUENT OPERATOR ACTIONS

NOTE: Step 4.1 provides direction for a **complete loss** of RCC.
Step 4.2 provides direction for a **partial loss** of RCC cooling capacity.

4.1 IF a complete loss of RCC flow occurs
OR a loss of RCC flow to the Drywell occurs,
THEN **PERFORM** the following:

4.1.1 **SCRAM the Reactor** per PPM 3.3.1. ____

4.1.2 **STOP** both Reactor Recirculation Pumps (RRC-P-1A and RRC-P-1B). ____

Number: ABN-RCC	Use Category: CONTINUOUS	Major Rev: 006 Minor Rev: 003 Page: 5 of 12
Title: Loss of RCC		

4.2 IF a partial loss of RCC flow occurs
THEN **PERFORM** the following:

NOTE: If RCC flow is lost to the Radwaste Building, OG-RF-20A(B)(C) may trip.

4.2.1 IF an RCC pump has tripped,
THEN **VERIFY** the standby RCC pump has started. _____

4.2.2 IF two RCC pumps are running,
THEN **VERIFY** RCC-V-6 is **OPEN**. _____

CAUTION

Closing RWCU-V-4 without throttling open RWCU-V-104 will result in RWCU-RV-3 lifting, if CRD seal purge is not isolated.

4.2.3 IF RCC-V-6 (RW/RB Supply) is closed,
THEN **PERFORM** the following.

a. **STOP** RWCU-P-1A(1B). _____

b. **THROTTLE OPEN** RWCU-V-104. _____

c. **CLOSE** RWCU-V-4 (RWCU Suction Outboard Isolation). _____

Comments / Reference: ABN-ELEC-SM3/SM8	Revision: 018

Number: ABN-ELEC-SM3/SM8	Use Category: CONTINUOUS	Major Rev: 018 Minor Rev: N/A Page: 14 of 27
Title: SM-3, SM-8, SM-85, SM-82, SL-81, SL-83 & SL-31 Distribution System Failures		

4.5 Loss of E-SL-81

4.5.1 **VERIFY** RCC-P-1A is operating,
THEN **PLACE** RCC-P-1B and RCC-P-1C in **PTL**, ____
AND **VERIFY** RCC-V-6 closes, ____
THEN **REFER** to ABN-RCC. ____

Comments / Reference: SD000196	Revision: 14

RCC flows through each fan unit's cooling coil. Each cooling coil has an RCC inlet isolation valve (RCC-V-71A, B, C and 72A and B) and a cooling coil outlet temperature control valve (RCC-TCV-71A, B, C and 72B). The RCC temperature control valves are deactivated in the open position and corresponding controllers have been removed. All valve, fan motor control switches, valve and position indications are mounted on Board H.

1. Drywell fan cooling coil inlet isolation valves:

   - RCC-V-71A/B/C (Serve Lower Drywell Coolers)

     Cooling coil inlet isolation valves for lower drywell cooling fan coils CRA-CC-1A/1B/1C

   - RCC-V-72A/72B (Serve Upper Drywell Coolers)

     Cooling coil inlet isolation for upper drywell cooling fan coils CRA-CC-2A/2B

V. CONTROL THEORY AND INTERLOCKS

LO-5706

A. Control Room Controls

1. Board B

a) PUMP RCC-P-1A (B, C)

Four-position switch: PTL, STOP, AUTO, START. Spring return to AUTO from STOP or START positions.

PTL	pump cannot be started and will not auto restart
STOP	pump stops
AUTO	the tripping or stopping of either of the running pumps, without an "F" or "A" signal present, causes any non-running pump in "AUTO" to start
START	pump starts

PUMP TRIPS:

- Overcurrent
- "F" signal, 1.68 psig drywell pressure
- "A" signal, Rx Wtr Lvl 2 (-50")

RCC pump breakers do not trip (i.e. remain closed) on a loss of voltage, the breaker indicating lights will correctly show the breaker closed, system parameters must be checked to determine pump status. When power is restored, the pumps will restart because the breakers are closed. RCC pumps will also auto restart on FA reset (if not placed in PTL).

Examination Outline Cross-reference:	Level	RO		SRO
Rev. 0 Date: 12/16/2016	Tier			2
	Group			2
	K/A	219000.A2.13		
Level of Difficulty: 4	Importance Rating			3.7

RHR/LPCI: Torus/Pool Cooling Mode: Ability to (a) predict the impacts of the following on the RHR/LPCI: TORUS/SUPPRESSION POOL COOLING MODE; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: High suppression pool temperature.

Question # 91

CGS is operating in Mode 4.

- Reactor coolant temperature is 165°F.
- RPV time to boil is 20 hours.
- RRC-P-1A is in operation.
- RHR Loop B is in shutdown cooling (SDC).
- RHR Loops A and C are in standby.
- Annunciator 601.A11.1-3, DRYWELL/SUPPRESSION POOL TEMPERATURE HIGH, is in alarm due to high suppression pool temperature.

Suppression pool temperatures:

- CMS-TR-5, point 220, Average Supp Pool Temp (Upper Level): 88.7 AVE °F
- CMS-TR-5, point A02, Average Supp Pool Temp (Lower Level): 82.2 AVE °F
- CMS-TR-6, point 220, Average Supp Pool Temp (Upper Level): 87.2 AVE °F
- CMS-TR-6, point A02, Average Supp Pool Temp (Lower Level): 81.5 AVE °F

What actions should be taken?

The CRS should direct the crew to place…

A. RHR Loop A in suppression pool cooling per SOP-RHR-SPC, Suppression Pool Cooling/Spray/Discharge/Mixing, and maintain RHR Loop B in SDC.

B. RHR Loop B in suppression pool cooling per SOP-RHR-SPC, Suppression Pool Cooling/Spray/Discharge/Mixing, and place RHR Loop A in SDC.

C. RHR Loop C in suppression pool mixing per SOP-RHR-SPC, Suppression Pool Cooling/Spray/Discharge/Mixing, and maintain RHR Loop B in SPC.

D. LPCS system in suppression pool mixing per SOP-LPCS-SP, LPCS Suppression Pool Mixing, and maintain RHR Loop B in SDC.

Answer: A

K/A Match:
This question requires the candidate to demonstrate an understanding of methods available to lower suppression pool temperature, including the impact that these methods have on RHR system operability, and select the correct method for given plant conditions.

SRO Only:
K/A is an “A2” statement and

ES-401 **8** **Attachment 2**

**Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5)
(Assessment and selection of procedures)**

Can the question be answered *solely* by knowing “systems knowledge”, i.e., how the system works, flowpath, logic, component location?

Yes → RO question

No

Can the question be answered *solely* by knowing immediate operator actions?

Yes → RO question

No

Can the question be answered *solely* by knowing entry conditions for AOPs or plant parameters that require direct entry to major EOPs?

Yes → RO question

No

Can the question be answered *solely* by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure?

Yes → RO question

No

Does the question require one or more of the following?

- Assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed
- Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps
- Knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific sub-procedures or emergency contingency procedures
- Knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures

Yes → SRO-only question

No

Question might not be linked to 10 CFR 55.43(b)(5) for SRO-only

Explanation:
A. Correct. Suppression pool bulk temperature is > 85°F. Therefore, suppression pool mixing will not lower bulk suppression pool temperature below the alarm setpoint. B. Incorrect. Plausible since RHR Loop B may be used to lower suppression pool temperature. However, RHR Loop A should not be placed in SDC with RRC-P-1A running. C. Incorrect. Plausible since placing RHR Loop C in suppression pool mixing could reduce suppression pool temperature by reducing stratification. However, suppression pool bulk temperature is > 85°F and suppression pool mixing should not be used. D. Incorrect. Plausible since placing LPCS in suppression pool mixing is allowed for the given plant conditions. However, suppression pool bulk temperature is > 85°F and suppression pool mixing should not be used.

Technical Reference(s)		Attached w/ Revision # See Comments / Reference
SOP-RHR-SPC, Suppression Pool Cooling/Spray/Discharge/Mixing		
SOP-RHR-SDC, RHR Shutdown Cooling		

Proposed references to be provided during examination: None

Learning Objective: 11808 Describe the effect that a loss or malfunction of the Suppression Pool Cooling mode of the Residual Heat Removal System will have on suppression pool temperature control.

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 ____
55.43 “A2” statement

Comments / Reference: SOP-RHR-SPC	Revision: Major 008 Minor 003

Number: SOP-RHR-SPC	Use Category: CONTINUOUS	Major Rev: 008 Minor Rev: 003 Page: 7 of 24
Title: Suppression Pool Cooling/Spray/Discharge/Mixing		

4.11 In Mode 3 or 4, two RHR Shutdown Cooling loops are required to be operable, and with no recirculation pump in operation, at least one RHR shutdown subsystem is required to be in operation. Refer to TS 3.4.9 and 3.4.10 for allowed outage times. Refer to TS Bases for definition of SDC operability.

4.12 Two RHR SDC loops are required to be operable and one loop of RHR is required to be in operation in Mode 5 with irradiated fuel in the RPV and with the water level LT 22' above the top of the RPV flange. One RHR SDC loop is required to be operable and in operation in Mode 5 with irradiated fuel in the RPV and with the water level GE 22' above the top of the RPV flange. Refer to TS 3.9.8 and 3.9.9.

4.13 Per NEI 99-02 and M-Rule, the unavailability of the RHR systems for SPC, when they are required to be operable, is required to be tracked in the Plant Logging System. The time when they are unavailable should be minimized.

4.14 The following methods may be used to minimize the amount of time SPC is in operation. {P-220039}

4.14.1 If any of the quarterly Operability Surveillances for RHRA/B/C, LPCS, or HPCS are due near the time when stirring or cooling of the S/P is needed then pull them forward. This can minimize the number of starts on RHR-P-2A&B, and reduce the amount of time in SPC, and efficiently utilize the affected systems.

4.14.2 When possible, maximize Suppression Pool Clean-up mode of the Fuel Pool Cooling (FPC).

4.14.3 When indicated temperature nears the HIGH alarm for Suppression Pool temperature, operate RHR-C in the Suppression Pool Mixing Mode. Monitoring Suppression Pool temperature will show that an apparent rapid cooling will occur as the pool is stirred and then the cooldown will cease. Approximately 15 minutes of pump operation should be all that is required. Since stirring the pool does not actually provide cooling, the bulk temperature will eventually rise and the frequency of mixing operations will rise, which will require more RHR-C pump starts. If the frequency of starts on RHR-P-2C rises to an unacceptable level or the bulk suppression pool temperature is above 85°F, then start SPC mode to cool the bulk Suppression Pool temperature to a reasonable value.

Comments / Reference: SOP-RHR-SDC	Revision: Major 025 Minor 002

Number: SOP-RHR-SDC	Use Category: CONTINUOUS	Major Rev: 025 Minor Rev: 002 Page: 8 of 81
Title: RHR Shutdown Cooling		

4.0 PRECAUTIONS AND LIMITATIONS

4.1 Technical Specifications require Reactor Vessel and Head Flange temperatures to be maintained GE 80°F when the vessel head flange bolts are under tension.

4.2 The cooldown rate of the Reactor Vessel shall not exceed 80° F per hour per administrative requirements (Technical Specification limit 100° F per hour). {R-5668}

4.3 Do not operate a Reactor Recirculation pump and an RHR pump in the Shutdown Cooling Mode in the same loop, except as described in PPM 8.3.422.

Examination Outline Cross-reference:	Level	RO		SRO
Rev. 1 Date: 12/16/2016	Tier			2
	Group			2
	K/A	234000 A3.02		
Level of Difficulty: 3	Importance Rating			3.7

Ability to monitor automatic operation of the FUEL HANDLING EQUIPMENT including: Interlock operation

Question # 92

CGS is in Mode 5.

- Core refueling is in progress.
- The refueling bridge crane main hoist grapple has been lowered to the grapple position for a fuel assembly in the core.
- The Grapple Engage/Release switch has been toggled to the ENGAGE position.
- The GRAPPLE ENGAGED light on the Right-Hand Controller is illuminated.

Prior to raising the fuel assembly, what action is required to verify that the grapple is properly engaged per SOP-REFUEL-OPS, Refueling Bridge Operation?

Verify the…

A. HOIST LOADED light on the Left-Hand Controller is illuminated.

B. back and forth rotation of mast is consistent with a seated fuel assembly.

C. SLACK CABLE light on the Left-Hand Controller is extinguished.

D. grapple is at the correct position by checking the tape markers on the main hoist cable.

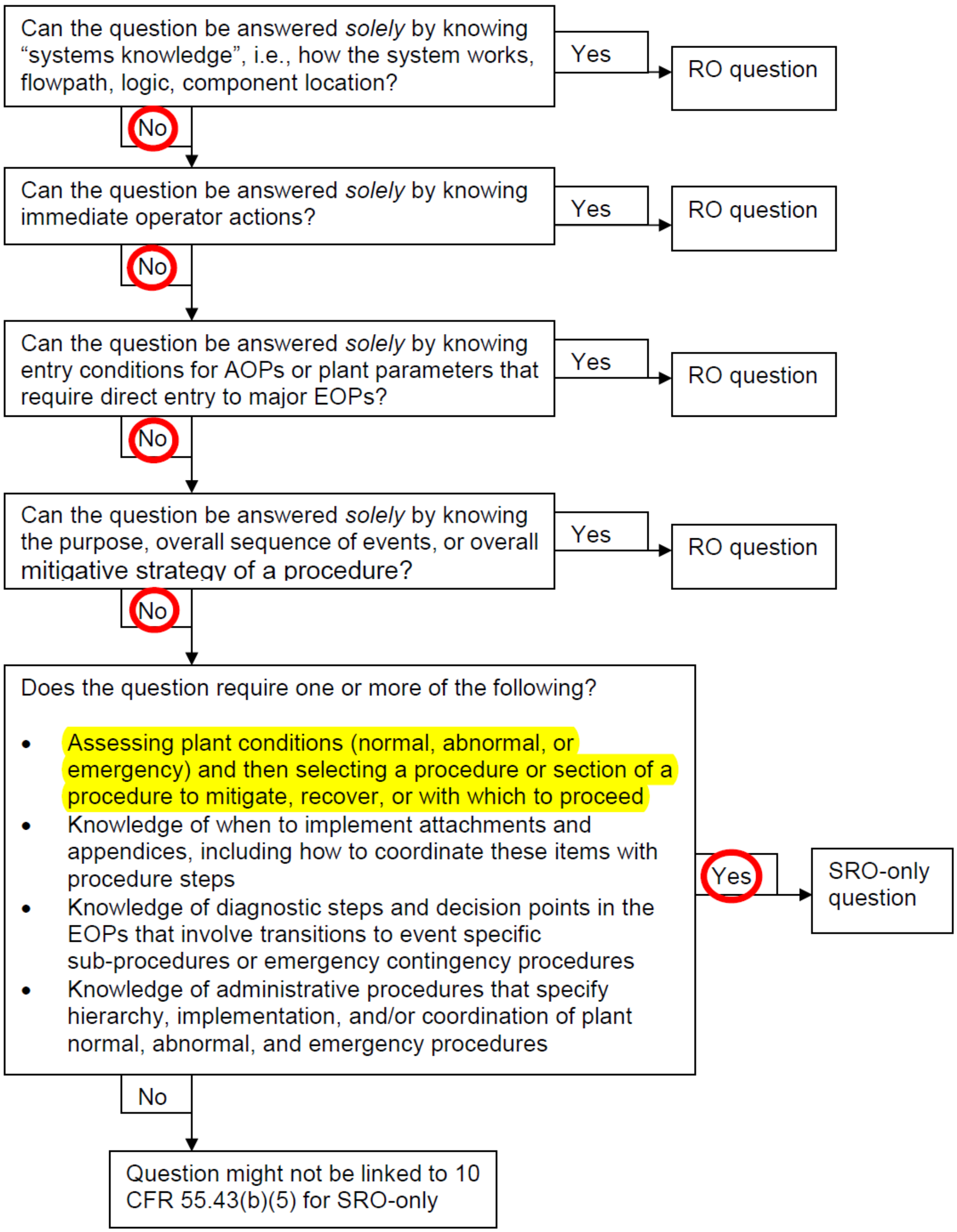
Answer: B

K/A Match:
The question requires the candidates to demonstrate knowledge of the procedure required to verify that the MAIN HOIST FUEL LOADED interlock will be satisfied after grappling a fuel bundle.

SRO Only:
K/A is a Tier 2, Group 2 selection related to fuel handling facilities and procedures and

ES-401 **8** **Attachment 2**

Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5)
(Assessment and selection of procedures)



Explanation:
A. Incorrect. Plausible since the HOIST LOADED light should be illuminated while removing the fuel from the core (step 5.2.14). However, the light will not illuminate when verifying that grapple is engaged while the grapple is resting on the fuel cell. B. Correct. In addition to verifying that the GRAPPLE ENGAGED light is illuminated OR that the grapple is fully engaged by visual inspection, operators are required to verify back and forth rotation of mast limited to that allowed by seated fuel assembly. C. Incorrect. Plausible since verifying the SLACK CABLE light is required to be verified prior to engaging the grapple (step 5.2.9). However, it is not an indication used to verify proper grapple engagement prior to lifting the fuel assembly. D. Incorrect. Plausible since tape markers may be used on the Frame Mounted Hoist and the Monorail Mounted Hoist. However, it is not acceptable to use tape on the cable of the Main Hoist.

Technical Reference(s)		
SD000207, Fuel Handling System Description		Attached w/ Revision # See Comments / Reference
SOP-REFUEL-OPS, Refueling Bridge Operation		

Proposed references to be provided during examination: None

Learning Objective: 8968 – Discuss the indications that should be verified to ensure a bundle is properly grappled.

Question Source: Bank # ____
Modified Bank # LO01485 (Note changes or attach parent)
New ____

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 ____
55.43 N/A

Comments / Reference: SD000207, Fuel Handling System Description	Revision: Major 013 Minor 002

(2) BACKUP HOIST LIMIT

This interlock is a backup to the normal grapple upper limit switch. The hoist is stopped by the backup limit 6" higher than the normal limit on grapple normal up switch failure.

(3) GRAPPLE FULL DOWN

This interlock stops the main hoist's downward travel 554" (46' 2") on the hoist readout.

(4) SLACK CABLE

This interlock prevents lowering of the main hoist when there is < 50# on the load cell indicating that the grapple is resting on something. It is activated by a hydraulic line pressure switch, which is set to release when the fuel grapple hoist load is less than 50 pounds.

(5) FUEL HOIST INTERLOCK

This interlock disables the fuel hoist whenever the platform is over the reactor vessel, a control rod is withdrawn and there is a fuel load on the main (FUEL) hoist. When the light is on the fuel hoist will be inoperative. (750# w/NF 500 mast). The load cell force switch, which provides this interlock, also feeds the Main Hoist Fuel Loaded interlock.

(6) MAIN HOIST FUEL LOADED

This interlock ensures that the grapple is fully closed when lifting a fuel bundle. If the grapple is not fully closed and there is a load on the main hoist, the hoist raise logic stops upward movement. (750# w/NF 500 mast)

Comments / Reference: SOP-REFUEL-OPS, step 5.2.14	Revision: 009

Number: SOP-REFUEL-OPS	Use Category: MULTIPLE	Major Rev: 009 Minor Rev: N/A Page: 21 of 66
Title: Refueling Bridge Operation		

5.2.14 WHEN fuel assembly is raised,
THEN **VERIFY** the following:

- Slack Cable light extinguished

CAUTION

Absence of the HOIST LOADED light in the following step may indicate the fuel is not grappled correctly and requires reseating the fuel, then regrappling the fuel.

- Hoist Loaded light illuminated (Left-hand Controller)
- Hoist load cell indicates GT 600 pounds

Comments / Reference: SOP-REFUEL-OPS, step 5.2.11	Revision: 009

Number: SOP-REFUEL-OPS	Use Category: MULTIPLE	Major Rev: 009 Minor Rev: N/A Page: 20 of 66
Title: Refueling Bridge Operation		

5.2.11 **VERIFY** the fuel assembly is properly engaged, as demonstrated by both of the following checks:

a. **VERIFY** the Grapple Engaged light illuminated (Right-hand Controller) OR grapple jaws fully engage by visual observation.

b. **VERIFY** back and forth rotation of mast limited to that allowed by seated fuel assembly. {P-102273}

Comments / Reference: SOP-REFUEL-OPS, step 5.2.9	Revision: 009

Number: SOP-REFUEL-OPS	Use Category: MULTIPLE	Major Rev: 009 Minor Rev: N/A Page: 19 of 66
Title: Refueling Bridge Operation		

CAUTION

Slow speed should be used when the fuel grapple is being lowered and raised near the following positions:

- 552″ - Grapple seated on fuel assembly in core
- 356″ - 4” section to the 3” section
- 201″ - Grapple seated on fuel assembly in MPC
- 186″ - Grapple seated on fuel assembly in spent fuel rack
- 178″ – 5” section to the 4” section

5.2.5 **LOWER** the mast, using the GRAPPLE variable speed controller on the right-hand controller, until the grapple is a few inches above the fuel assembly.

5.2.6 **ADJUST** the position of Bridge and Trolley as necessary to center grapple.

NOTE: The Grapple Jog Down pushbutton may be used for jogging the grapple to its final position.

5.2.7 **LOWER** the grapple to one of the following positions, as indicated on Flat Panel Display:

- Approximately 552″ (Core)
- Approximately 186″ (Fuel Pool)
- Approximately 171″ (Full-down Prep Machines)
- Approximately 201″ (MPC)

CAUTION

Use care when confirming proper grapple orientation on the bail handle by rotation of mast. Irradiated fuel that is not channeled is susceptible to damage from aggressive rotation.

5.2.8 **VERIFY** grapple properly situated over fuel bail by rotating mast gently in both directions. {P-102273}

5.2.9 **VERIFY** the Slack Cable light illuminated.

5.2.10 **CLOSE** the grapple jaws by toggling Grapple Engage/Release switch to **ENGAGE**.

Comments / Reference: SOP-REFUEL-OPS, Precautions	Revision: 009

Number: SOP-REFUEL-OPS	Use Category: MULTIPLE	Major Rev: 009 Minor Rev: N/A Page: 12 of 66
Title: Refueling Bridge Operation		

4.22 Operation of the hoist while in the Safety Travel Interlock Zone is only permitted by engaging the Travel Override pushbutton unless the boundary zone switch is in bypass.

4.23 When routine maintenance on the Refuel Bridge is performed inside the FMC zone, an FMC Log should be used to document control of tools and supplies. Movement of the Refuel Bridge in and out of the FMC zone for major repairs or major PMs requires a Work Order Task (WOT) that has specific sign-off steps for exit of the zone and entry back into the zone and for logging of the bridge in the FMC logbook per PPM 10.1.13. {P-140132}, {P-158894}

4.24 Due to physical length of the double blade guides, the Main Hoist override button is required to be used to raise the mast to approximately -6.0". This will clear the other double blade guides seated in the Spent Fuel Pool and will clear the Cattle Chute. HP is not required to monitor this evolution. {P-141162}

4.25 The Hoist Motor Overspeed Trip will prevent movement of the mast without providing any abnormal indication on either the hand controllers or the display panel. It may be necessary to reset the Hoist Motor Overspeed Trip brass pushbutton (extreme northern end of the main hoist motor train) to allow the mast to operate.

4.26 Avoid using Spent Fuel Pool racks E4, E5, E6, and E7 due to radiation streaming. Avoid using Spent Fuel Pool rack E3 and E-18 due to previous damage. {AR-347326}{P-157414}

4.27 Prior to commencing Mode 5, Refueling Operations or Mode 4, Situation Interlock Testing, ensure that no temporary modifications to Refuel Bridge Logic are installed (Sections 5.18 and 5.19).

4.28 When any tool is attached to the Auxiliary or Monorail hoist cable and that tool is in use, then a daily inspection of the cable-to-tool threaded connections should be performed. HP assistance is required if the tool has to be removed from the Fuel Pool for this inspection. {P-171515}

4.29 It is acceptable to use one or two wraps of tape to mark positions on the cable of the Frame Mounted Hoist and the Monorail Mounted Hoist. It is not acceptable to use tape on the cable of the Main Hoist.

4.30 Mast test weights are normally installed in the Cask Pit to support surveillances on the main refuel mast prior to fuel movement. Mast test weights are normally removed from the Cask Pit and stored on RB 606 between refuel outages.

4.31 Prior to use, ensure correct direction of rotation for auxiliary platform hoists and drive by verifying correct response to pendant controls. Notify the Refueling Engineer if directional response is opposite of pendant command.

Comments / Reference: Bank Question LO01485,	Revision: Original

Procedurally, which of the following provides adequate demonstration that a fuel assembly is properly engaged?

A. Grapple elevation reading of 552"
B. Slack Cable light is illuminated
C. The grapple Engage Switch is in the 'ENGAGE' position
D. Grapple Engaged light is illuminated
E. Visual observation of the grapple jaws fully engaged
F. Back and forth rotation of the mast

A. D and F

B. C and D

C. A, C, and F

D. B, D, and E

Answer: A

Examination Outline Cross-reference:	Level	RO		SRO
Rev. 0 Date: 12/16/2016	Tier			2
	Group			2
	K/A	256000.2.4.47		
Level of Difficulty: 3	Importance Rating			4.2

Reactor Condensate: Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material.

Question # 93

CGS is operating in Mode 1. Reactor power is 50%.

- Primary chemistry parameters for the past 2 days are as follows:

Date	Time	Reactor Water Conductivity (µS/cm)	Reactor Water Sulfate (ppb)	Feedwater Conductivity (µS/cm)	Condensate Demineralizer Inlet (CDI) Conductivity (µS/cm)
1/1/2017	1200	0.10	4	.055	0.03
1/1/2017	1800	0.23	4	.056	0.04
1/2/2017	0000	0.33	4	.058	0.08
1/2/2017	0600	0.37	7	.060	0.12
1/2/2017	1200	0.40	9	.055	0.25
1/2/2017	1800	0.52	11	.057	0.65
1/3/2017	0000	0.57	12	.059	0.75
1/3/2017	0600	0.68	14	.062	0.88
1/3/2017	1200	0.70	14	.059	0.92

Using the reference provided, and based on the trend information, what is the <u>next</u> Primary Chemistry Action Level (PCAL) to be entered?

Enter PCAL __(1)__ based on __(2)__ .

A. (1) 2
(2) reactor water conductivity

B. (1) 1
(2) feedwater conductivity

C. (1) 2
(2) reactor water sulfate

D. (1) 3
(2) condensate demineralizer inlet conductivity

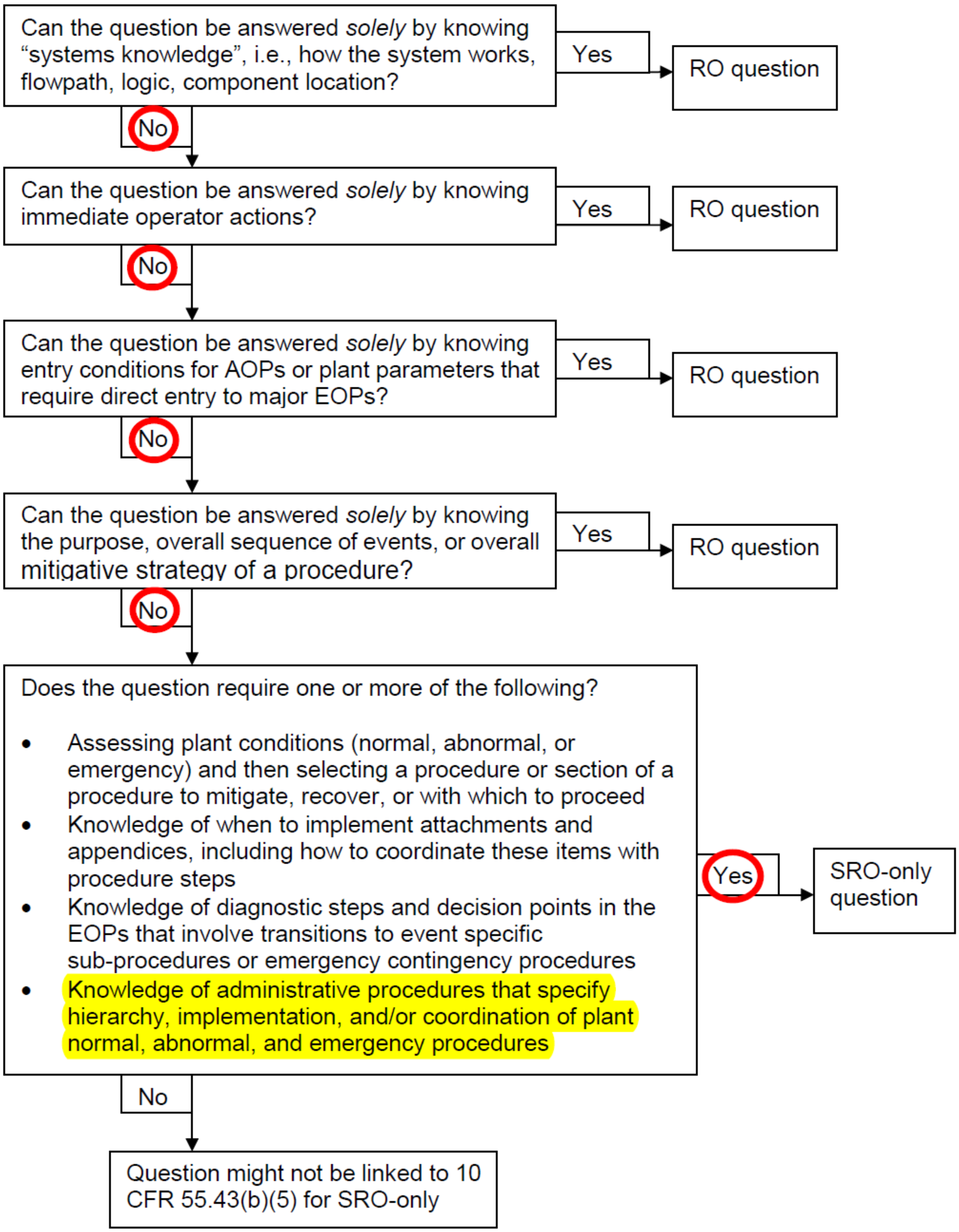
Answer: D

K/A Match:
This question requires the candidate to evaluate conductivity trends and determine required actions.

SRO Only:
K/A is a “G” statement and

ES-401 **8** **Attachment 2**

Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5)
(Assessment and selection of procedures)



Explanation:
A. Incorrect. Plausible since Primary Chemistry Action Level (PCAL) 2 for reactor water conductivity is > 1.0 μS/cm and . However, trend data shows that Condensate Demineralizer Inlet (CDI) conductivity will reach PCAL 2 first. B. Incorrect. Plausible since feedwater conductivity is close to its PCAL 1 limit of .065 μS/cm. However, trend data shows that Condensate Demineralizer Inlet (CDI) conductivity will reach PCAL 2 first. C. Incorrect. Plausible since PCAL 2 for reactor water sulfate is >20 ppb and this parameter will reach PCAL 2 in about 30 hours. However, trend data shows that Condensate Demineralizer Inlet (CDI) conductivity will reach PCAL 2 in less than 24 hours. D. Correct. Trend data shows that Condensate Demineralizer Inlet (CDI) conductivity will reach PCAL 3 in less than 24 hours, while the other parameters will take at least 24 hours to reach their next PCAL.

Technical Reference(s)		Attached w/ Revision # See Comments / Reference
SWP-CHE-02, Chemical Process Management and Control		

Proposed references to be provided during examination: SWP-CHE-02, tables 6.1.13, 6.1.15

Learning Objective: ________

Question Source:
- Bank # ________
- Modified Bank # ________ (Note changes or attach parent)
- New X

Question History: Last NRC Exam N/A

Question Cognitive Level:
- Memory or Fundamental Knowledge ____
- Comprehension or Analysis X

10 CFR Part 55 Content:
- 55.41 ________
- 55.43 ________

Comments / Reference: SWP-CHE-02	Revision: 026

Number: SWP-CHE-02	Use Category: INFORMATION	Major Rev: 026 Minor Rev: N/A Page: 22 of 62
Title: Chemical Process Management and Control		

6.1.13 Table 6.1.13, Reactor Water Control Parameters – Power Operation, >10% Power

Category	Chemistry Regime	Value	Frequency of Measurement (Needed)	Additional Guidance
Conductivity				
Good Practice	All (with zinc injection)	≤0.15 μS/cm	Continuous	1
	All (without zinc injection)	≤0.08 μS/cm	Continuous	
	All (corrected with zinc injection)	≤0.08 μS/cm	Quarterly	
Primary Chemistry Action Level 1	All	>0.30 μS/cm	Continuous	2, 3
Primary Chemistry Action Level 2	All	>1.0 μS/cm	Continuous	4
Primary Chemistry Action Level 3	All	>5 μS/cm	Continuous	
Chloride				
Good Practice	All	≤1 ppb	Daily	5
Primary Chemistry Action Level 1	All	≥5 ppb	8 hours	
Primary Chemistry Action Level 2	All	>20 ppb	4 hours	4, 6
Primary Chemistry Action Level 3	All	>100 ppb	4 hours	4, 6
Sulfate				
Good Practice	All	≤2 ppb	Daily	5
Primary Chemistry Action Level 1	All	>5 ppb	8 hours	
Primary Chemistry Action Level 2	All	>20 ppb	4 hours	4, 6
Primary Chemistry Action Level 3	All	>100 ppb	4 hours	4, 6

Number: SWP-CHE-02	Use Category: INFORMATION	Major Rev: 026 Minor Rev: N/A Page: 23 of 62
Title: Chemical Process Management and Control		

Table 6.1.13 Additional Guidance: Reactor Water Control Parameters, Power Operation, >10% Power:

1. Good Practice Conductivity Values
   The Good Practice continuous reactor water conductivity value of ≤0.15 µS/cm for plants with zinc injection is achievable with Good Practice levels of chloride and sulfate and up to approximately 20 ppb soluble zinc; the Good Practice continuous conductivity value of ≤0.08 µS/cm is for plants without zinc injection. The Good Practice quarterly conductivity value of ≤0.08 µS/cm is the value after correction for contributions from soluble iron and soluble zinc; monitoring is performed quarterly based on grab samples. Guidance for performing conductivity balances and corrections is provided in BWRVIP-190 Volume 2 Appendix C.

2. Conductivity
   An ion-conductivity balance should be performed to estimate the concentration of unidentified and potentially corrosive ions. Conductivity may be corrected for contributions from soluble iron and zinc to confirm Action Level entry and for crack growth rate calculations. Guidance for performing conductivity balances and corrections is provided in BWRVIP-190 Volume 2 Appendix C.

3. Action Level 1 Conductivity Value during OLNC Injection
   During noble metal injection for OLNC, the conductivity may exceed the Action Level 1 value due to increases in sodium from the platinum compound, Na2Pt(OH)6 (see BWRVIP-190 Volume 2 Chapter 1). Short duration conductivity increases from this known source of sodium can be predicted in advance and are not a concern for accelerated IGSCC or fuel cladding corrosion. Reactor water sampling and analysis results during the OLNC application should be reviewed to confirm that the increased conductivity is not due to aggressive anions or ionic impurities from other sources. If confirmed, no additional corrective actions or Action Level 1 responses are required and the time in excess of 24 hours that reactor water conductivity exceeds 0.30 µS/cm does not have to be considered when calculating Hydrogen Availability (see BWRVIP-190 Volume 2 Appendix E (Mitigation Performance Indicator) for the definition of Hydrogen Availability).

4. Action Level 2 and Action Level 3
   A plant-specific analysis should be available to determine whether plant shutdown or continued Power Operation is the most prudent approach with regard to IGSCC and fuel damage (see BWRVIP-190 Volume 2 Appendix B). When the Action Level 3 value is exceeded, and such an analysis is not available and cannot be conducted within 4 hours following the excursion, orderly shutdown of the plant is the most prudent approach.

5. Chloride and Sulfate Sampling and Analysis
   Recognizing that chloride and sulfate have associated near-term operational actions for off-normal conditions, Daily sampling frequency is Needed and should not be relaxed. For additional information see BWRVIP-190 Volume 2 Chapter 7.

6. Action Level 2 and Action Level 3 Values for OLNC+HWC Evaluation of the effects of a chemistry transient that exceeds Action Level 2 or Action Level 3 should be based on welds or components of concern as defined by the utility and on the mitigation status as provided in BWRVIP- 62 Revision 1 (Chapter 4 and Table 4-1). Welds and components of concern and their mitigation status may be defined in the plant-specific Strategic Water Chemistry Plan; see BWRVIP-190 Volume 2 Appendix A (Guidance on Developing a BWR Strategic Water Chemistry Plan).

Number: SWP-CHE-02	Use Category: INFORMATION	Major Rev: 026 Minor Rev: N/A Page: 28 of 62
Title: Chemical Process Management and Control		

6.1.15 Table 6.1.15, Reactor Feedwater/Condensate Control Parameters – Power Operation, >10% Power

Category	Chemistry Regime	Value	Frequency of Measurement (Needed)	Additional Guidance
Feedwater Conductivity				
Good Practice	All	≤0.060 µS/cm	Continuous	
Primary Chemistry Action Level 1	All	>0.065 µS/cm	Continuous	1
Feedwater Dissolved Hydrogen				
Needed	OLNC+HWC	0.21 ppm or 10 SCFM Hydrogen Flow Rate	Continuous	2, 3
Feedwater Dissolved Oxygen				
Good Practice	All	40 - 100 ppb	Continuous	4,5
Primary Chemistry Action Level 1	All	<30 ppb >200 ppb	Continuous	
CDI Conductivity				
Primary Chemistry Action Level 1	All	>0.10 µS/cm	Continuous	
Primary Chemistry Action Level 3	All	>1.0 µS/cm	Continuous	6
Condensate Dissolved Oxygen				
Good Practice	All	40 - 100 ppb	Continuous	6,7
Primary Chemistry Action Level 1	All	<30 ppb >200 ppb	Continuous	
Feedwater Total Iron				
Good Practice (Cycle average)	All	≤1.0 ppb	Integrated	8,9
Needed (Quarterly average)	All	≤3.0 ppb Fe with FW Cu ≤0.05 ppb and FW Zn ≤0.5 ppb ≤2.0 ppb Fe with FW Cu ≤0.05 ppb and FW Zn >0.5 ppb ≤2.0 ppb Fe with FW Cu >0.05ppb and FW Cu+Zn >0.5 ppb or Cycle-Specific Values established by the Cross Discipline Review Team	Integrated	9, 10, 11
Primary Chemistry Action Level 1	All	>5.0 ppb	Integrated 96 hr	9, 10

Number: SWP-CHE-02	Use Category: INFORMATION	Major Rev: 026 Minor Rev: N/A Page: 29 of 62
Title: Chemical Process Management and Control		

Category	Chemistry Regime	Value	Frequency of Measurement (Needed)	Additional Guidance
Feedwater Total Zinc				
Needed (Quarterly or cycle average)	All	≤0.5 ppb quarterly average ≤0.4 ppb cycle average or Cycle-Specific Values established by the Cross Discipline Review Team	Integrated	9, 11
Feedwater Total Copper				
Good Practice (Cycle average)	All	≤0.05 ppb	Integrated	9, 12
Needed (Quarterly average)	All	≤0.10 ppb	Integrated	9, 12, 13
Primary Chemistry Action Level 1	All	>0.2 ppb	Integrated 96 hr	9, 13

Table 6.1.15 Additional Guidance: Reactor Feedwater/Condensate Control Parameters – Power Operation, >10% Power:

1. Feedwater Conductivity
   Plant-specific Action Levels should be adopted if elevated conductivity results from the feedwater iron injection program or increased soluble iron due to the catalytic effects of noble metal deposits in the sample line.

2. Feedwater Hydrogen
   When the feedwater hydrogen concentration (or calculated concentration based on the feedwater hydrogen injection rate and feedwater flow rate) decreases below the plant-specific Needed value established for IGSCC mitigation, hydrogen injection should be restored to the required rate as soon as possible to minimize the impact on HWC Availability.

3. OLNC+HWC Plant-Specific Feedwater Hydrogen With NMCA/OLNC+HWC, the plant-specific hydrogen injection rate is determined by:

   – Performing an HWC benchmark test to determine the hydrogen injection rate that provides an H2:O2 molar ratio of 2, as described in BWRVIP-245.

   – Running BWRVIA at the plant conditions at the time of the HWC benchmark test and determining a scaling factor relating BWRVIA predictions to the plant-specific measured hydrogen injection rate as described in BWRVIP-245.

   – Applying the BWRVIA model to determine the hydrogen injection rate required for a molar ratio ≥3 at the upper downcomer location at beginning of cycle (BOC),middle of cycle (MOC) and end of cycle full power (EOC) conditions.

Number: SWP-CHE-02	Use Category: INFORMATION	Major Rev: 026 Minor Rev: N/A Page: 30 of 62
Title: Chemical Process Management and Control		

4. Feedwater Dissolved Oxygen
   If noble metal contamination or other causes of oxygen depletion in the sample line leads to observations of lower than expected oxygen concentrations in the feedwater or when feedwater dissolved oxygen monitoring is unavailable, condensate oxygen may be used to assure sufficient dissolved oxygen to control FAC. If oxygen is injected into the condensate, the alternate sample point should be located downstream of the oxygen injection point

5. Feedwater and Condensate Dissolved Oxygen
   The technical bases for the low dissolved oxygen Action Level and Good Practice values for FAC control are given in BWRVIP-190 Volume 2 Chapter 3. A feedwater dissolved oxygen Good Practice value <40 ppb but ≥30 ppb may be adopted based on supporting operating experience. A plant-specific upper limit may have to be adopted to ensure consistency with ECP or hydrogen:oxidants molar ratio requirements under OLNC+HWC. The BWRVIA radiolysis model should be run if dissolved oxygen in the feedwater exceeds the plant-specific upper limit to determine whether an increase in the hydrogen injection rate is required. In addition, the licensing basis for plant life extension should be reviewed to determine if a commitment to limit the maximum feedwater oxygen has been made in the analysis of the fatigue life of feedwater piping (see BWRVIP-190 Volume 2 Chapter 2).

6. Condensate (CDI) Conductivity
   An evaluation is required to establish a plant-specific limit, which depends on the condensate polishing system design and cooling water chemistry.

7. Condensate Dissolved Oxygen
   The sample may be obtained from either the condensate pump discharge or condensate polisher outlet.

8. Feedwater Total Iron, Feedwater Total Zinc and Feedwater Total Copper (Action Level 1, Quarterly and Cycle Averages)
   Power and time weighting may be used over the integrated sample collection period. Details are given in BWRVIP-190 Volume 2 Chapter 7. Corrective actions for exceeding Action Level 1 for feedwater iron and feedwater zinc apply after 96 hours.

9. Feedwater Total Iron (Effects of Noble Metal in Sample Lines)
   If noble metals deposit in the Feedwater sample lines, the measured FW Fe concentration may be elevated due to catalytic effects in the sample line (see BWRVIP-190 Volume 2 Chapter 7). In this case, an alternate Feedwater sample point (further upstream) or the condensate polishing system effluent sample point can be used. Plants with forward-pumped drains must also take into account the Fe contribution from these drains to the final Feedwater.

Number: SWP-CHE-02	Use Category: INFORMATION	Major Rev: 026 Minor Rev: N/A Page: 31 of 62
Title: Chemical Process Management and Control		

10. Feedwater Total Iron (Quarterly Average)
The following values apply, based on quarterly averages for feedwater iron, copper and zinc, unless cycle-specific values based on an approved fuel risk assessment or change management assessment are established:

– FW Fe ≤3.0 ppb with FW Cu ≤0.05 ppb and FW Zn ≤0.5 ppb

– FW Fe ≤2.0 ppb with FW Cu ≤0.05 ppb and FW Zn >0.5 ppb

– FW Fe ≤2.0 ppb with FW Cu >0.05 ppb and FW Cu+Zn >0.5 ppb

If any of the cycle-specific values, or the above values if cycle-specific values are not developed, are exceeded, then notify Fuels that criteria for performing a fuel risk assessment in accordance with the Fuel Reliability Guidelines: BWR Fuel Cladding Crud and Corrosion, Revision 1 [2-8] and change management assessment in accordance with Fuel Reliability Guidelines: Fuel Surveillance and Inspection Revision 2 may have been met.

11. Feedwater Total Zinc (Quarterly Average and Cycle Average)
The following values apply unless cycle-specific values based on an approved fuel risk assessment or change management assessments are established:

– FW Zn ≤0.5 ppb quarterly average

– FW Zn ≤0.4 ppb cycle average

If any of the cycle-specific values, or the above values if cycle-specific values are not developed, are exceeded, then notify Fuels that criteria for performing a fuel risk assessment in accordance with the Fuel Reliability Guidelines: BWR Fuel Cladding Crud and Corrosion, Revision 1 [2-8] and change management assessment in accordance with Fuel Reliability Guidelines: Fuel Surveillance and Inspection Revision 2 may have been met.

12. Feedwater Total Copper (Sampling Frequency and Good Practice Value)
Frequency of measurement may be reduced for plants without copper-alloy condenser tubes and for which long-term trending has shown copper levels ≤0.05 ppb.

13. Feedwater Total Copper (Quarterly Average)
The following value applies unless a cycle-specific value based on an approved fuel risk assessment or change management assessment is established:

– FW Cu ≤0.1 ppb quarterly average

If the cycle-specific value is exceeded, then notify Fuels that the criterion for performing a fuel risk assessment in accordance with Fuel Reliability Guidelines: BWR Fuel Cladding Crud and Corrosion, Revision 1 and change management assessment in accordance with Fuel Reliability Guidelines: Fuel Surveillance and Inspection Revision 2 may have been met.

Examination Outline Cross-reference:	Level	RO		SRO
Rev. 2 Date: 12/16/2016	Tier			3
	Group			1
	K/A	2.1.9		
Level of Difficulty: 2	Importance Rating			4.5

Ability to direct personnel activities inside the control room.

Question # 94

When may the CRS direct control room operators to perform actions that deviate from technical specifications?

Deviations from technical specifications and licensing conditions are permitted per 10CRF50.54X…

A. if failing to deviate from technical specifications will result in unnecessary emergency equipment damage.

B. upon approval of a license amendment by the NRC branch chief that delineates the specific actions to be taken that are outside technical specifications.

C. after the operations manager and NRC resident inspector agree that such actions are needed to protect the health and safety of the public.

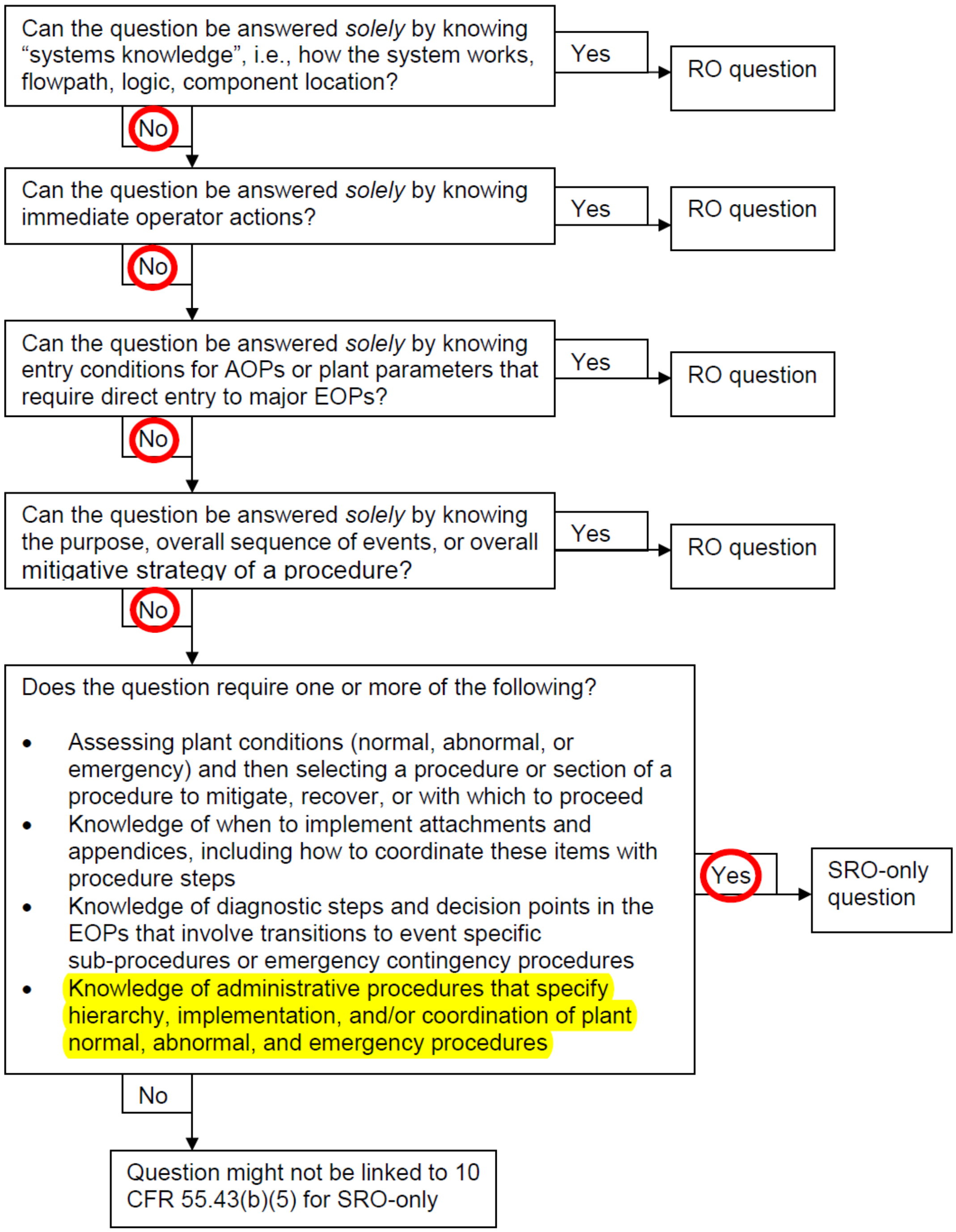
D. if no adequate means is apparent of protecting the health and safety of the public consistent with tech specs during an emergency requiring immediate action.

Answer: D

K/A Match:
This question requires the candidate to demonstrate when the CRS may direct control room operators during conditions requiring actions per 10 CFR 50.54X.

SRO Only:
K/A is a “G” statement and

Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5) (Assessment and selection of procedures)



Explanation:
A. Incorrect. Plausible since actions should be taken to prevent equipment damage, such as a reactor scram. However, this is not a reason to take actions outside technical specifications per 10 CFR 50.54X. B. Incorrect. Plausible since license amendments are used to modify technical specifications for specific instances, such as upgrading equipment. However, the amendment process is not covered in 10 CFR 50.54X. C. Incorrect. Plausible since notification of entry into 10 CFR 50.54X is required to be made to the Operations Manager, NRC Resident, and NRC Operations Center as soon as possible after entering 10 CFR 50.54X (within 1 hour). However, approval from these is not required prior to taking actions. D. Correct. Though considered the exception, reasonable action that departs from Technical Specifications and licensing conditions are permitted per 10 CFR 50.54(x) provided An emergency exists and such action is immediately needed to protect the health and safety of the public when no adequate or equivalent means of protection consistent with Technical Specifications or License Conditions are apparent.

Technical Reference(s)		Attached w/ Revision # See Comments / Reference
PPM 1.3.1, Operating Policies, Programs and Practices		

Proposed references to be provided during examination: None

Learning Objective: 6073 - State when deviations from Technical Specifications and Licensing conditions are permitted per 10CFR50.54X. [PPM 1.3.1]

Question Source: Bank # LX00897
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 ______
55.43 Tier 2 “G” Statement

Comments / Reference: PPM 1.3.1, section 4.3	Revision: Major 120 Minor 005

Number: 1.3.1	Use Category: INFORMATION	Major Rev: 120 Minor Rev: 005 Page: 19 of 98
Title: Operating Policies, Programs and Practices		

4.3 Operation Outside Plant Technical Specifications or License Conditions (10 CFR 50.54x) {R-306}, {R-1743}

> NOTE: While not specifically required to invoke 10 CFR 50.54(x), if time and circumstances permit, additional Technical and Senior Management concurrence is highly desirable prior to actions taken based on the invocation of 10 CFR 50.54(x).

4.3.1 Though considered the exception, reasonable action that departs from Technical Specifications and licensing conditions are permitted per 10 CFR 50.54(x) provided:

a. An emergency exists and such action is immediately needed to protect the health and safety of the public when no adequate or equivalent means of protection consistent with Technical Specifications or License Conditions are apparent and

b. The deviation is evaluated and approved by the CRS/Shift Manager on a case-by-case basis. {R-1744}

c. The NRC resident, NRC Operations Center and Operations Department Manager are notified of the deviation prior to the action if time permits and if not, as soon as possible but in all cases within one hour.

Comments / Reference: PPM 1.3.1, section 4.8.3	Revision: Major 120 Minor 005

Number: 1.3.1	Use Category: INFORMATION	Major Rev: 120 Minor Rev: 005 Page: 26 of 98
Title: Operating Policies, Programs and Practices		

4.8.3 Procedure Usage During an Emergency

a. The EOPs direct the operator to execute a particular section or an entire Volume 2, 3 or 4 Procedure. All required concurrent execution of any Volume 2, 3, or 4 Procedure are specifically called out by the EOPs.

This does not mean that a Volume 4, Abnormal Procedure, cannot otherwise be concurrently executed with the EOPs so long as its specified actions do not conflict with the direction given by the EOPs.

b. Once EPIPs have been entered (Emergency Classification occurs), recovery actions not specifically authorized by plant procedures which have a potential for radioactive release to the environment require Emergency Director concurrence (this includes items listed in FAZ Procedures).

c. If it becomes necessary to take actions other than those prescribed by the License Basis Documents or by plant procedures; (to protect the immediate health and safety of the public) those actions shall be implemented using 10 CFR 50.54(x) per Section 4.3.

d. If a transient is in progress or an emergency condition dictates, take immediate corrective action up to and including power reduction or SCRAM to stabilize the situation. Then, as soon as possible, refer to appropriate procedures. Appropriate times to exercise this option include:

- To maintain the safety of the public
- To protect the safety of individuals (e.g., deenergize equipment that is causing an electrocution or stop a leak that is endangering an individual)
- To preclude unnecessary equipment damage (e.g., stop a piece of equipment that is quickly degrading or is in imminent danger)
- SCRAM prior to exceeding an RPS limit

Examination Outline Cross-reference:	Level	RO		SRO
Rev. 0 Date: 12/16/2016	Tier			3
	Group			1
	K/A	2.1.23		
Level of Difficulty: 3	Importance Rating			4.4

Ability to perform specific system and integrated plant procedures during all modes of plant operation.

Question # 95

CGS is operating in Mode 3.

- A normal plant shutdown is in progress per PPM 3.2.1, Normal Plant Shutdown.
- RPV pressure is 355 psig.
- DEH is lowering RPV pressure 5 psig/min.
- The crew notes evidence of level "notching" on RFW-LI-606A, Narrow Range Reactor Water Level.
- The other 3 RPV narrow range level instruments are tracking normally.

How should the CRS control the plant cooldown?

A. Reduce the RPV depressurization rate per PPM 3.2.1, section 5.6, RPV Depressurization, to allow the condensing pot to refill the reference leg.

B. Maintain the RPV depressurization rate and lineup for reference leg backfill per PPM 10.27.64, Continuous Backfill System Operation.

C. Stop the RPV depressurization perform an operability evaluation per PPM 3.2.1, attachment 7.1, RPV Depressurization with Evidence of Notching.

D. Stop the RPV depressurization and declare the affected water level trip functions inoperable per TS 3.3.1.1, Reactor Protection System Instrumentation.

Answer: C

K/A Match:

This questions requires the candidate to understand how to proceed with an integrated plant procedure (normal plant shutdown) with an abnormal condition (notching) that requires the use of an attachment.

SRO Only:
K/A is a "G" statement tied to 10CFR.55.43 and

Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5)
(Assessment and selection of procedures)

Can the question be answered *solely* by knowing "systems knowledge", i.e., how the system works, flowpath, logic, component location?
Yes → RO question
No ↓

Can the question be answered *solely* by knowing immediate operator actions?
Yes → RO question
No ↓

Can the question be answered *solely* by knowing entry conditions for AOPs or plant parameters that require direct entry to major EOPs?
Yes → RO question
No ↓

Can the question be answered *solely* by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure?
Yes → RO question
No ↓

Does the question require one or more of the following?

- Assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed
- Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps
- Knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific sub-procedures or emergency contingency procedures
- Knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures

Yes → SRO-only question
No ↓

Question might not be linked to 10 CFR 55.43(b)(5) for SRO-only

Explanation:
A. Incorrect. Plausible since slowing the RPV depressurization rate would assist in refilling the reference leg. However, attachment 7.1, step 7.3.1 requires operators to stop RPV depressurization and evaluate instrument operability.
B. Incorrect. Plausible since lining up continuous backfill is an approved method to recover the affected instrument. However, attachment 7.1, step 7.3.1 requires operators to stop RPV depressurization and evaluate instrument operability.
C. Correct. With evidence of level instrument notching, attachment 7.1, step 7.3.1 requires operators to stop RPV depressurization and evaluate instrument operability.
D. Incorrect. Plausible since attachment 7.1, step 7.3.1 requires operators to stop RPV depressurization and evaluate instrument operability with evidence of level notching. However, TS 3.3.1.1 does not apply in the stated mode of operation.

Technical Reference(s)		Attached w/ Revision # See Comments / Reference
PPM 3.2.1, Normal Plant Shutdown		
CGS Technical Specifications, LCO 3.3.1.1, Reactor Protection System Instrumentation.		

Proposed references to be provided during examination: None

Learning Objective: 9875 Given appropriate conditions, indications and copies of plant procedures, determine the operator actions necessary to mitigate a plant transient from an analysis of plant conditions. [Plant Shutdown]

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 ____
55.43 43.5

Comments / Reference: PPM 3.2.1, attachment 7.1, RPV Depressurization with Evidence of Notching	Revision: Major 083 Minor 001

Number: 3.2.1	Use Category: CONTINUOUS	Major Rev: 083 Minor Rev: 001 Page: 43 of 48
Title: Normal Plant Shutdown		

RPV DEPRESSURIZATION WITH EVIDENCE OF NOTCHING

CAUTION

During RPV depressurization at pressures below 450 psig, non-condensable gasses may come out of solution in the RPV water level instrument reference legs. Changes in the instrument reference legs could cause indicated RPV level to be higher than actual RPV level. {C-2.8}

NOTE: Notching can be recognized as a positive level "pulse" (i.e., a level rise for a short time, then a return to normal), typically lasting one minute or less. {C-2.8}

NOTE: De-gassing can be recognized as erratic peak to peak noise like pulses that can result in an erroneously raised level indication.

NOTE: Notching or de-gassing which occurs at RPV pressures GE 60 psig indicates that significant amounts of dissolved gasses may be present in the reference leg. Any evidence of notching or de-gassing is to be evaluated to determine instrument operability. {C-2.8}

NOTE: An unexplained level change of GT 6" lasting GT 2 minutes, or as determined by the Shift Manager, requires the affected instrument(s) to be declared inoperable and the appropriate Tech Spec actions taken.

7.3.1 IF evidence of notching or de-gassing is observed,
THEN **STOP** depressurization (if possible) until operability of the affected channel can be evaluated. _____

7.3.2 WHEN RPV pressure approaches 450 psig,
THEN **PERFORM** the following: {P-2.31} _____

NOTE: If the plant process computer is not available, N/A the following computer-related steps.

a. **START** a real time Computer and TDAS plot of file PPM NOTCHING at 20 samples per second consisting of the following signals.

- B033
- X341
- X151
- B027
- B028
- X327

_____
STA

Attachment 7.1, RPV Depressurization With Evidence of Notching

Number: 3.2.1	Use Category: CONTINUOUS	Major Rev: 083
Title: Normal Plant Shutdown		Minor Rev: 001 Page: 44 of 48

b. **START** a PPCRS display of the following narrow range instruments. (A span of approximately 20"-50" and a time interval of 300 seconds should be used):

- RFW-LI-606A (B033) (MS-LIS-24A)
- RFW-LI-606B (B027) (MS-LIS-24D)
- RFW-LI-606C (B028) (MS-LIS-24C)

_____ STA

> NOTE: If available, the tv monitor should be used to monitor MS-LIS-24B to minimized personal exposure.

c. **STATION** a gauge observer at MS-LIS-24B (IR-P026, RB 522) to monitor for indications of notching or de-gassing during cooldown. _____

d. **STATION** a gauge observer at MS-LIS-24D (IR-P027, RB 522) to monitor for indication of notching or de-gassing during Cooldown. Mark this step N/A if computer monitoring is available. _____

7.3.3 IF the Shift Manager determines that unacceptable notching or degassing has occurred,
THEN **DECLARE** the affected reactor water level trip functions inoperable. _____

a. **ENTER** the following applicable LCOs for the affected channels:

- RPS, Table 3.3.1.1-1, Function 4 (Modes 1 & 2) _____
- Isolation group 5, Table 3.3.6.1-1, Function 2a (Modes 1, 2 & 3) _____
- Isolation group 6, Table 3.3.6.1-1, trip function 5d (Modes 3, 4 & 5) _____

> NOTE: Restoration of an inoperable trip channel requires that its indication return to a level consistent with operable water level channels and maintain consistent water level indication for at least five minutes. Recovery may occur either through normal refill of the reference leg from the condensing chamber, use of the Continuous Backfill System or by plant personnel backfilling the reference legs manually (PPM 10.27.58A,(B,C,D).

7.3.4 IF the trip of an inoperable channel is required to comply with Tech Specs, THEN **PLACE** the **appropriate** test switches in test (or fuses removed) per the following list. _____

Attachment 7.1, RPV Depressurization With Evidence of Notching

Number: 3.2.1	Use Category: CONTINUOUS	Major Rev: 083 Minor Rev: 001 Page: 45 of 48
Title: Normal Plant Shutdown		

7.3.5 IF placing the switches in test causes an isolation,
THEN **ALIGN** the systems to their isolation position prior to switch actuation.
REFER to PPM 3.3.1 for valve groups. _____

LEVEL INSTRUMENT SHOWING NOTCHING	NARROW RANGE CH.	TRIP INSTRUMENT	TRIP CHANNEL	ISOLATION LOGIC/GRP	SWITCH PANEL	TEST SWITCH	FUSE
RFW-LI-606A	A	MS-LIS-24A	A	OUTBD / 5	H13-P609	S19A	H13-P609 C72A-F06A TBAA, F07
				OUTBD / 6	H13-P609	S82A	
RFW-LI-606B	B	MS-LIS-24D	D	INBD / 5	H13-P611	S19D	H13-P611 C72A-F06B TBMM, F07
				INBD / 6	H13-P611	S82D	
RFW-LI-606C	C	MS-LIS-24C	C	INBD / 5	H13-P609	S19C	H13-P609 C72A-F06C TBAA, F07
				INBD / 6	H13-P609	S82C	
NONE	D	MS-LIS-24B	B	INBD / 5	H13-P611	S19B	H13-P611 C72A-F06D TBMM, F07
				INBD / 6	H13-P611	S82B	

7.3.6 WHEN any reactor water level channel has been restored to normal,
THEN the Shift Manager may declare it operable and direct its trip functions be restored to normal. _____

a. **EXIT** the appropriate LCO. _____

b. **RETURN** the applicable test switches to **NORMAL** (reinstall fuses). _____

7.3.7 WHEN Mode 4 is entered,
THEN **SECURE** the MS-LIS-24B and MS-LIS-24D watch. _____

7.3.8 IF any RPV water level channel has not yet been restored to operable status,
THEN **RESTORE** their test switches/fuses to **NORMAL** at the direction of the Shift Manager. _____

END

Attachment 7.1, RPV Depressurization With Evidence of Notching

Comments / Reference: TS LCO 3.3.1.1, Applicability	Revision: 21

Table 3.3.1.1-1 (page 3 of 4)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
3. Reactor Vessel Steam Dome Pressure - High	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.14 SR 3.3.1.1.15	≤ 1079 psig
4. Reactor Vessel Water Level - Low, Level 3	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.14 SR 3.3.1.1.15	≥ 9.5 inches
5. Main Steam Isolation Valve - Closure	1	8	F	SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.14 SR 3.3.1.1.15	≤ 12.5% closed
6. Primary Containment Pressure - High	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.14	≤ 1.88 psig
7. Scram Discharge Volume Water Level - High					
a. Transmitter/Trip Unit	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.14	≤ 529 ft 9 inches elevation
	5[(a)]	2	H	SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.14	≤ 529 ft 9 inches elevation
b. Float Switch	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.14	≤ 529 ft 9 inches elevation
	5[(a)]	2	H	SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.14	≤ 529 ft 9 inches elevation
8. Turbine Throttle Valve - Closure	≥ 30% RTP	4	E	SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.12 SR 3.3.1.1.14 SR 3.3.1.1.15	≤ 7% closed

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

Examination Outline Cross-reference:	Level	RO		SRO
Rev. 0 Date: 12/16/2016	Tier			3
	Group			
	K/A	2.2.11		
Level of Difficulty: 4	Importance Rating			3.3

Knowledge of the process for controlling temporary design changes.

Question # 96

A Temporary Modification (TM) has been approved for installation.

- The TM includes jumper installation in the control room.
- Maintenance personnel are in the control room, ready to install the jumper.

What is the CRS responsible for during TM installation in accordance with PPM 1.3.9, Temporary Modifications?

Prior to installing the jumper, the CRS should…

A. verify or place the original Determ/Reterm Data Sheets in the Work Order.

B. verify that affected operations procedures and top-tier drawings have been updated.

C. verify independent verification or peer check of the jumper installation is performed.

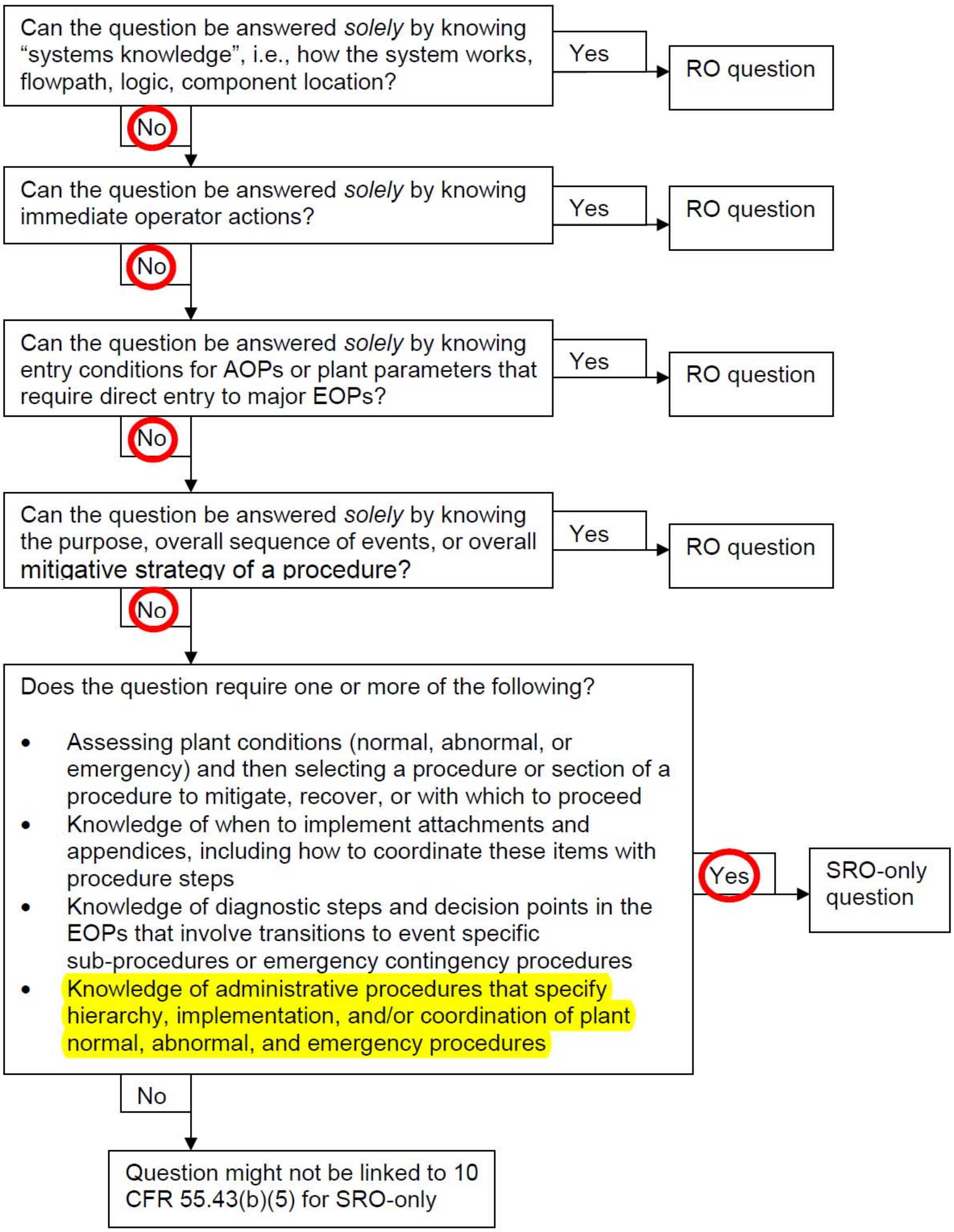
D. perform a walkdown with the personnel installing the jumper emphasizing positive jumper control.

Answer: D

K/A Match:
The question requires the candidate to demonstrate knowledge of SRO responsibilities when installing a temporary modification.

SRO Only:
K/A is a “G” statement tied to 10CFR.55.43 and

Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5)
(Assessment and selection of procedures)



Explanation:
A. Incorrect. Plausible since this step is required by the procedure. However, it is not the responsibility of the CRS.
B. Incorrect. Plausible since this step is required by the procedure. However, it is not the responsibility of the CRS, and it is not performed until the modification installation is complete.
C. Incorrect. Plausible since the CRS should verify that provisions have been made for jumper verification. However, simultaneous verification is required for jumper installation.
D. Correct. See PPM 1.3.9, Temporary Modifications, step 4.2.1.c.

Technical Reference(s)		Attached w/ Revision # See Comments / Reference
A	PPM 1.3.9, Temporary Modifications	

Proposed references to be provided during examination: None

Learning Objective: 11251 – Knowledge of the process for making field changes

Question Source: Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 ______
55.43 43.3

Comments / Reference: PPM 1.3.9, Temporary Modifications	Revision: Major 53, Minor N/A

Number: 1.3.9	Use Category: REFERENCE	Major Rev: 053 Minor Rev: N/A Page: 10 of 40
Title: Temporary Modifications		

4.0 PROCEDURE

4.1 Temporary Modification Initiation, Review and Approval

NOTE: Anyone can be an initiator.

4.1.1 **PROCESS** an AR EVAL in accordance with DES-2-9 (if not done already), unless the TM is urgent.

4.1.2 IF the TM is emergent, THEN **INITIATE** an EC directly per DES-2-16, and the AR EVAL may be processed concurrently or later per DES-2-9.

4.2 Installation of Temporary Modification via EC TCHG/TMOD

NOTE: The Temporary Modification is prepared, reviewed and approved as an Engineering Change (EC) Type TCHG/TMOD per DES-2-16. The Temporary Modification is installed in accordance with PPM 1.3.68 and PPM 10.1.27.

4.2.1 **APPROVE** the installation of a Temporary Modification as follows (CRS/Shift Manager):

a. **THOROUGHLY REVIEW** the approved EC (available on Asset Suite EC EDMS).

b. **VERIFY** the Temporary Modification installation is compatible with existing plant conditions, reactor power limitations, and assessment of potential hazards to personnel. {R-2557}

c. **PRIOR** to issuing jumpers for installation of jumpers in a Main Control Room cabinet or panel, **PERFORM** a walkdown and briefing with the personnel performing the installation and an SRO regarding safe use of the jumper. This brief and walkdown should focus on risk for inadvertent contact with the adjacent metal cabinets or other devices during jumper installation and what specific actions/technique will be used to ensure positive jumper control at all times.

{AR-256570}

NOTE: When significant radiation exposure is likely as a result of this verification, the CRS/Shift Manager may waive this requirement. This should be noted on the EC Milestone Notes.

NOTE: As a guideline, an estimated exposure of 10 mrem for a specific verification task is considered a significant radiation exposure. {P-105765}

Number: 1.3.9	Use Category: REFERENCE	Major Rev: 053 Minor Rev: N/A Page: 11 of 40
Title: Temporary Modifications		

d. **VERIFY** provisions have been made to simultaneously verify all lifted leads and/or jumpers. Refer to Error Prevention Tools (EPT) Standards STANDARD 01. {R-2513}, {R-2517}

e. **VERIFY** on-shift Operations personnel are notified of the Temporary Modification installation and its potential effects.

f. **VERIFY** the Work Order number is provided on the EC Cross References panel. {R-2518}, {R-3595}, {R-3600}

g. **NAVIGATE** to the E100 – Engineering Change panel.

h. **CLICK** on the 'Milestones' button.

i. **ENTER** the approval date into the '340-I TCHG OPS' milestone on the EC Milestones panel. (This milestone sign-off reflects Installation of the Temporary Change modification has received Operations Approval.) {R-6470}

   1) IF a waiver was provided per step 4.2.1d,
      THEN **PROVIDE** an explanation in the '340 - I TCHG OPS' Milestone Notes.

      - **CLICK** the 'Text Update' button
      - **SAVE** and **CLOSE**.

j. **UPDATE** the TM Log Index by utilizing electronic tracking via Asset Suite or by utilizing the Temp Mods Index Report (crystal report). See attachment 8.6 for instructions on how to obtain the Temp Mods Index Report (crystal report). Otherwise manually **UPDATE** the TM Log Index to include at a minimum the TM number (i.e. EC number), description, and "Installation Approval Date" (Attachment 8.2).

Examination Outline Cross-reference:	Level	RO		SRO
Rev. 0 Date: 12/16/2016	Tier			3
	Group			
	K/A	2.2.20		
Level of Difficulty:	Importance Rating			3.8

Knowledge of the process for managing troubleshooting activities.

Question # 97

When should the Outage Control Center (OCC) be activated to assist in troubleshooting an equipment fault in accordance with SWP-MAI-03, Emergent Issue Management?

The OCC should be activated …

A. when the fault causes entry into a 72 hour LCO.

B. after any reactor power reduction of 5%.

C. on an emergency classification of ‘Alert’ or higher.

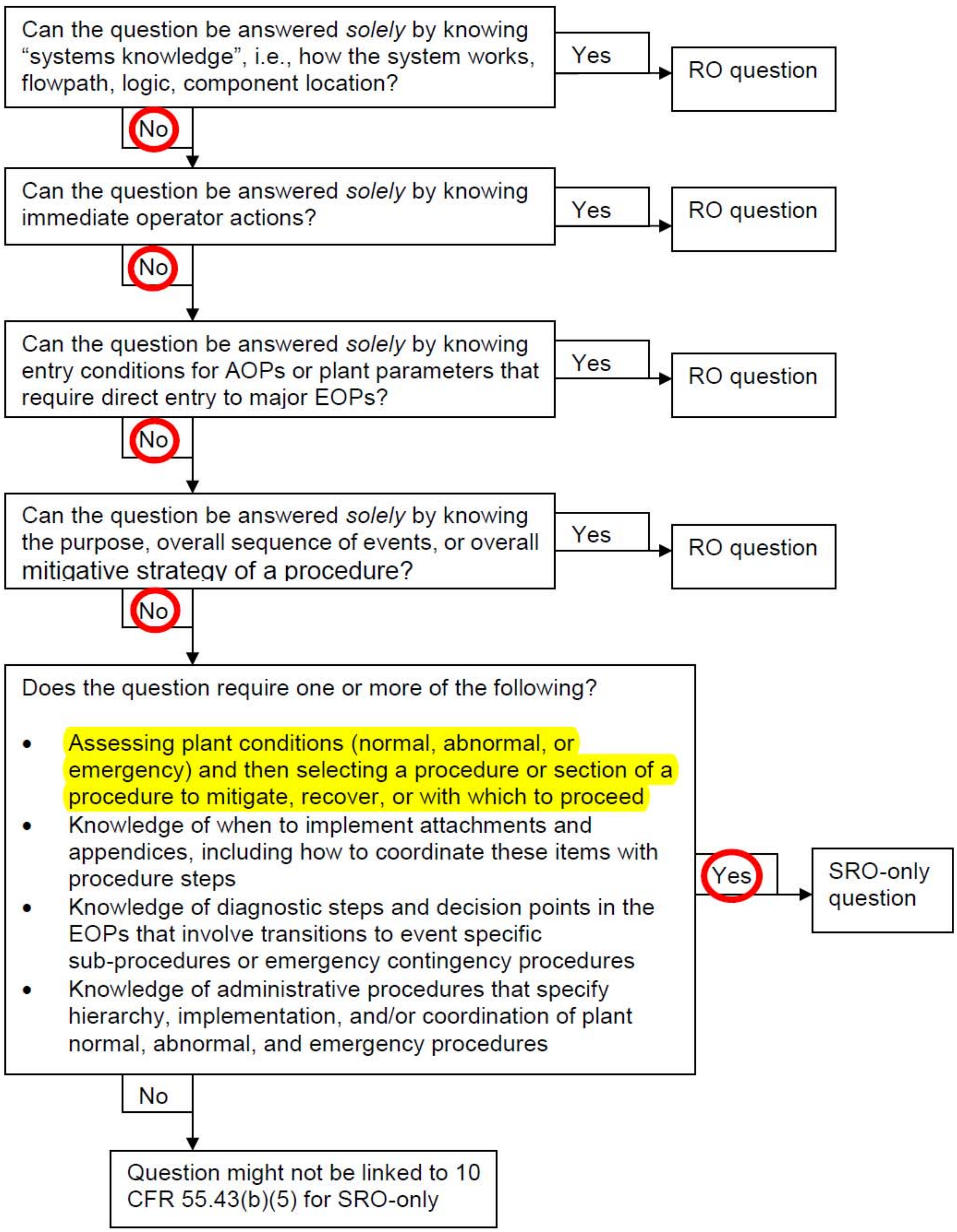
D. when the Engineering VP determines that it is warranted.

Answer: A

K/A Match:
The question requires the candidate to demonstrate knowledge of criteria for activating additional resources for troubleshooting activities.

SRO Only:
K/A is a “G” statement tied to 10CFR.55.43 and

Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5) (Assessment and selection of procedures)



Explanation:
A. Correct. In accordance with SWP-MAI-03, the OCC is activated when a LCO action statement of ≤ 72 hours is entered. B. Incorrect. Plausible since the OCC is activated for significant unplanned power derates. However, SWP-MAI-03 defines significant derates as > 10% power. C. Incorrect. Plausible since the OCC may be activated on an emergency classification level of "Unusual Event". It is not activated on a higher classification since the Emergency Response Organization will be activated instead. D. Incorrect. Plausible since the OCC may be activated when it is determined necessary by the shift manager, work week manager, or duty manager.

Technical Reference(s)		Attached w/ Revision # See Comments / Reference
SWP-MAI-03, Emergent Issue Management		

Proposed references to be provided during examination: None

Learning Objective: LO6188 – Identify concurrences needed for troubleshooting activities that are high risk to plant operations.

Question Source: Bank # ___
Modified Bank # ___ (Note changes or attach parent)
New X

Question History: Last NRC Exam ___

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ___

10 CFR Part 55 Content: 55.41 ___
55.43 Tier 3 Generic K/A

Comments / Reference: SWP-MAI-03, emergent Issue Management	Revision: 013

Number: SWP-MAI-03	Use Category: REFERENCE	Major Rev: 013 Minor Rev: N/A Page: 4 of 13
Title: Emergent Issue Management		

1.0 PURPOSE

> NOTE: Outage Command Center (OCC) activation requires Plant General Manager approval.

1.1 This procedure provides guidance to site personnel for responding to: {AR-315073}

1.1.1 Significant unplanned derate (>10% power reduction) or entry into a forced outage.

1.1.2 Unplanned entry into a short duration Limiting Condition for Operation (LCO of 72 hours or less) without a clear plan for exiting LCO within the LCO period, or planned LCO work that experiences expanded scope or failed post maintenance testing activities without a clear plan for exiting LCO within LCO period.

1.1.3 Unplanned unavailability of High Pressure Core Spray (HPCS), Reactor Core Isolation Cooling (RCIC), Residual Heat Removal (RHR), Service Water (SW), or Diesel Generator (DG) systems with no clear success path.

1.1.4 Significant event that the Shift Manager, Work Week Manager (WWM), or Duty Manager determines warrants implementation of this procedure.

1.2 The procedure provides a consistent and systematic response by plant organizations to unplanned power reductions, forced outages, or significant events. The procedure can be used for issues that occur on-line, in an outage, or as needed to resolve a parallel emergent issue that arises during an existing event.

1.3 This procedure does not take precedence over Emergency Procedures and is not used for emergency classifications higher than an Unusual Event.

Examination Outline Cross-reference:	Level	RO		SRO
Rev. 0 Date: 12/16/2016	Tier			3
	Group			3
	K/A	2.3.11		
Level of Difficulty: 2	Importance Rating			4.3

Ability to control radiation release.

Question # 98

PPM 5.4.1 Radioactivity Release Control requires Emergency Depressurization if the exclusion area boundary release rate approaches or exceeds the General Emergency limit.

Which of the following describes the reason for this requirement?

Emergency Depressurization…

A. allows the containment vent path (MSIVs) to open and vent the primary system to the condenser.

B. allows low pressure systems to inject into the reactor, limiting the release to the environment.

C. reduces the flow of primary systems that are unisolated and discharging outside of containment.

D. slows the rate of fuel damage in the reactor core and heat addition to the primary system.

Answer: C

K/A Match:
The question requires that the candidate understand when an emergency depressurization is used to mitigate an off site radiation release based on emergency action levels.

SRO Only:
K/A is a "G" statement tied to 10CFR.55.43 and

Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5)
(Assessment and selection of procedures)

Can the question be answered *solely* by knowing "systems knowledge", i.e., how the system works, flowpath, logic, component location?
- Yes → RO question
- No ↓

Can the question be answered *solely* by knowing immediate operator actions?
- Yes → RO question
- No ↓

Can the question be answered *solely* by knowing entry conditions for AOPs or plant parameters that require direct entry to major EOPs?
- Yes → RO question
- No ↓

Can the question be answered *solely* by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure?
- Yes → RO question
- No ↓

Does the question require one or more of the following?

- Assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed
- Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps
- Knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific sub-procedures or emergency contingency procedures
- Knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures

- Yes → SRO-only question
- No ↓

Question might not be linked to 10 CFR 55.43(b)(5) for SRO-only

Explanation:
A. Incorrect. Plausible since venting the primary to the condenser would be desirable to reduce radiation release. However, emergency depressurization is not a prerequisite to reopening MSIVs. B. Incorrect. Plausible since this is the purpose of performing an emergency depressurization for a low RPV level. However, this does not necessarily reduce off site radiation release. C. Correct. RPV depressurization places the primary system in the lowest possible energy state and reduces the driving head and flow of primary systems that are unisolated and discharging outside of containment.. D. Incorrect. Plausible since emergency depressurization may slow the rate of fuel damage if the RPV is refilled. However, this will not necessarily reduce off site radiation release.

Technical Reference(s)		Attached w/ Revision # See Comments / Reference
PPM 5.0.10, Flowchart Training Manual		
PPM 5.4.1, Secondary Containment Control		

Proposed references to be provided during examination: None

Learning Objective: 8481 - Identify the statement that describes the two reasons for emergency depressurizing the RPV if radioactivity release rates approach or exceed the General Emergency level and a primary system is discharging reactor coolant outside primary and secondary containment. (PPM 5.4.1)

Question Source: Bank # LO00181
Modified Bank # ____ (Note changes or attach parent)
New ____

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ____

10 CFR Part 55 Content: 55.41 11
55.43 4

Comments / Reference: PPM 5.0.10, Flowchart Training Manual	Revision: Major 21 Minor 001

d. Step R-3:

1) An offsite radioactivity release rate above the General Emergency action level represents a substantial increase in the severity of the offsite radioactivity release, relative to the entry condition, and accordingly presents a more immediate threat to the continued health and safety of the public. Before the release rate reaches the General Emergency level, emergency RPV depressurization is performed to reduce the radioactivity release rate.

2) If a primary system is discharging into an area outside the primary and secondary containments, then either:

- A primary system discharge cannot be isolated because system operation is required to assure adequate core cooling or shutdown the reactor (Step R-2), or
- No isolation valves exist upstream of a primary system break, or if isolation valves do exist, they cannot be closed because of some mechanical/ electrical/pneumatic failure, or
- The source of the discharge cannot be determined.

3) Entry or re-entry to PPM 5.1.1, RPV Control, requires initiation of a reactor scram if one has not already been initiated. This action reduces to decay heat levels the energy that the RPV may be discharging outside of primary and secondary containments. RPV depressurization places the primary system in the lowest possible energy state and reduces the driving head and flow of primary systems that are unisolated and discharging outside of containment.

4) Action to scram the reactor and depressurize the RPV is not appropriate (and is not required by the wording of this step) if no primary system is discharging outside the primary and secondary containment. Shutting down the reactor in accordance with normal operating procedures is not precluded, however, should the CRS conclude that such action is prudent for the current status of the plant.

5) The wording of this step permits, but does not require, emergency depressurization "before" the offsite radioactivity release rate reaches the General Emergency action level. An earlier blowdown could reduce onsite and offsite doses, but may not be advisable in all events. A judgment concerning the timing of the blowdown should consider such factors as:

- The availability of low-pressure injection systems.
- Whether it has been determined that the reactor will remain shutdown under all conditions without boron.
- The location, rate, and effects of the primary system discharge.

Examination Outline Cross-reference:	Level	RO		SRO
Rev. 0 Date: 12/16/2016	Tier			3
	Group			
	K/A	2.4.29		
Level of Difficulty: 3	Importance Rating			4.4

Knowledge of the emergency plan.

Question # 99

CGS is operating in Mode 1.

- The shift manager declared an "Unusual Event" at 1525 due to a casualty.
- The casualty was upgraded to an "Alert" at 1535, prior to initiating off site notifications.

Which of the following correctly describes when the initial off site notifications are required to be completed?

Notifications to state and local agencies must be completed by __(1)__. The NRC must be notified by __(2)__.

A. (1) 1550
(2) 1635

B. (1) 1540
(2) 1625

C. (1) 1540
(2) 1640

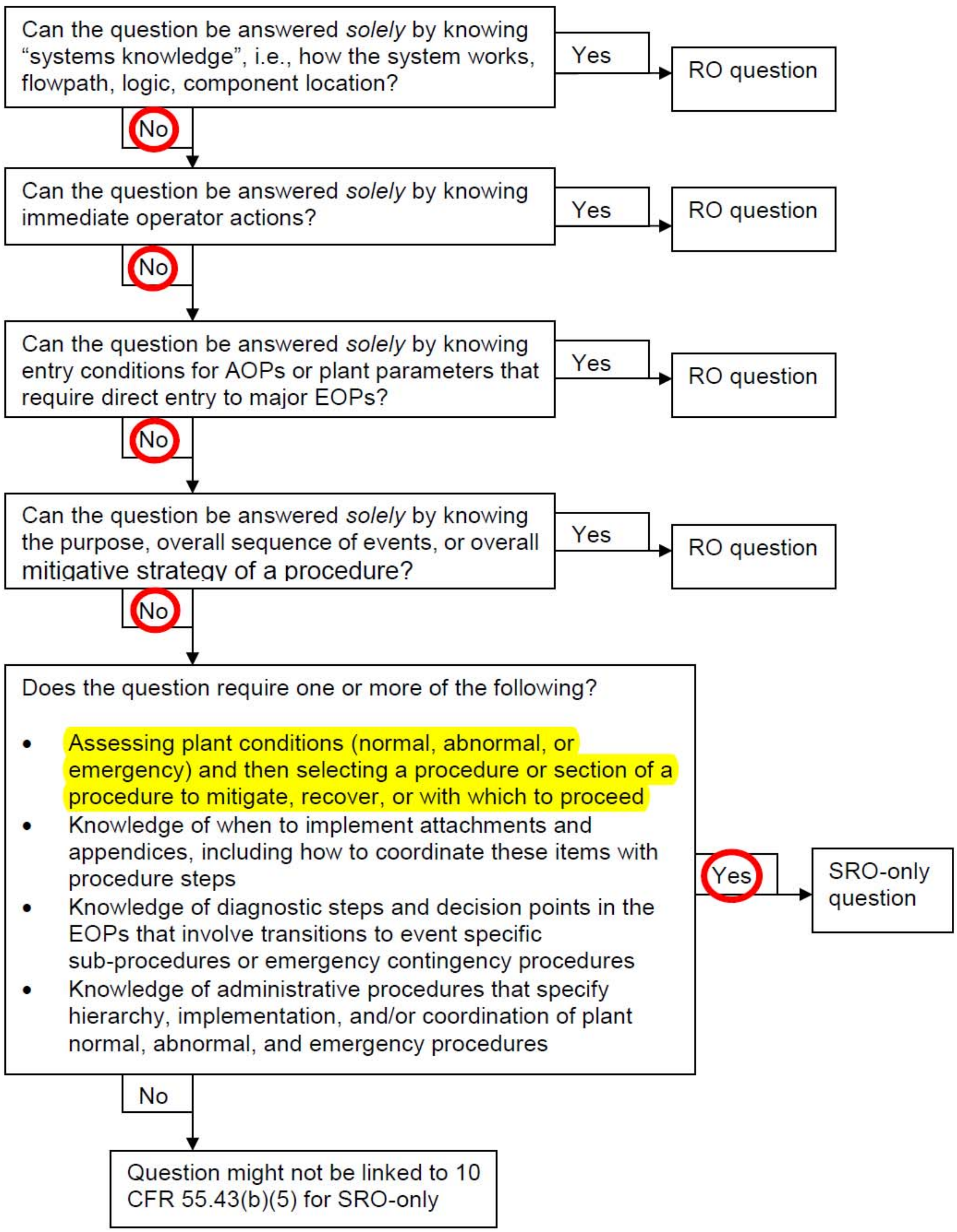
D. (1) 1550
(2) 1650

Answer: B

K/A Match:
The question requires SRO only knowledge of the emergency plan; specifically, time requirements for off site reporting of emergency declarations.

SRO Only:
K/A is a “G” statement tied to 10CFR.55.43 and

Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5)
(Assessment and selection of procedures)



Explanation:

A. Incorrect. Plausible since these are the required times for the second emergency declaration. However, required off site notifications are timed from the initial emergency declaration.

B. Correct. In accordance with PPM 13.4.1, Emergency Notifications, *"Initial notification of Washington State and local authorities must be made within 15 minutes following declaration of the emergency event", and "Initial notification of the NRC via the Emergency Notification System (ENS) should be made immediately after notification of the appropriate state and local authorities, and must be made not later than one (1) hour after emergency event declaration."* The time for the initial notifications starts from the initial emergency declaration.

C. Incorrect. Plausible since the time for local notification is correct. However, time to notify NRC starts from the initial notification not from the local notification.

D. Incorrect. Plausible since time for local notification is from the second emergency declaration. However, required off site notifications are timed from the initial emergency declaration, and time to notify NRC starts from the initial notification not from the local notification.

Technical Reference(s)		Attached w/ Revision # See Comments / Reference
PPM 13.4.1, Emergency Notifications		

Proposed references to be provided during examination: None

Learning Objective: 10222 – Identify the methods used to make emergency response notifications.

Question Source: Bank # ___
Modified Bank # LR01288 (Note changes or attach parent)
New ___

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge ___
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 Best match only
55.43 ___

Comments / Reference: Original Bank Question LR01288	Revision: N/A

Columbia was operating at full power when a series of events occurred that caused a reactor scram. Due to low RPV water level the Shift Manager has declared an Alert at 1525.

Which of the following correctly describes when notification to the State and local authorities must be made?

Notifications must be made no later than:

A. 1540

B. 1555

C. 1610

D. 1625

Comments / Reference: PPM 13.4.1, Emergency Notifications	Revision: Major 43, Minor 02

Number: 13.4.1	Use Category: REFERENCE	Major Rev: 043 Minor Rev: 002 Page: 6 of 33
Title: EMERGENCY NOTIFICATIONS		

2.2.3 During severe weather, affected security measures may be suspended when immediately needed to protect the personal health and safety of security force personnel and no other immediately apparent action consistent with the license conditions and technical specifications can provide adequate or equivalent protection. This suspension of security measures must be approved, as a minimum, by a licensed senior operator, with input from the security supervisor or manager, before taking this action. {R-20125}

2.2.4 The temporary suspension must be approved by the Emergency Director prior to taking the action unless a life-threatening situation exists and there is insufficient time to obtain prior approval. {R-20124}

2.2.5 If security measures are temporarily suspended, the NRC Operations Center should be notified as soon as practicable, and in all cases, within one hour of the occurrence. Upon restoration of security measures, the NRC Operations Center shall be notified as soon as practicable. {R-20127}

2.2.6 Suspended security measures must be restored as soon as practical. {R-20126}

2.3 Precautions and Limitations

2.3.1 State and local authorities are required to receive emergency event notifications within 15 minutes of event classification, a change in event classification, or changes in Protective Action Recommendations (PARs).

NOTE: The time of classification entered on the new CNF should be the time of the new classification or Protective Action Recommendation.

2.3.2 If after beginning to fill out a Classification Notification Form (CNF), but before the event is communicated to anyone offsite, event conditions change which make it necessary to reclassify the event or change PARs, discontinue completing the first CNF and begin filling out a new one. If the hard copy CNF is used, mark the discontinued CNF void and include it with the After Action Report per PPM 13.13.4. The initial 15 minute notification requirement is not waived and the new CNF must be completed within 15 minutes of declaring the previous classification.

2.3.3 If event conditions change which make it necessary to reclassify the event or change PARs and offsite notifications are in progress, the current 15 minute notification requirement is not waived. Notifications in progress for the lower level classification or PARs must be completed. Inform the offsite agencies on the Crash phone that classification or PARs will be upgraded and another notification will be forthcoming shortly.

Number: 13.4.1	Use Category: REFERENCE	Major Rev: 043 Minor Rev: 002 Page: 5 of 33
Title: EMERGENCY NOTIFICATIONS		

1.0 PURPOSE

This procedure provides instructions for notification of Federal, State and County organizations should a classified emergency provided for in PPM 13.1.1 be declared, upgraded, terminated, or a Protective Action Recommendation (PAR) be made or modified. It also provides instruction for notification, acknowledgment, and response actions by Energy Northwest emergency response personnel. {R-1586}, {R-1587}, {R-1588} {R-1589}, {R1590}

2.0 DISCUSSION

2.1 General

Initial notification of Washington State and local authorities must be made within 15 minutes following declaration of the emergency event. For Energy Northwest, local authorities are defined as Benton County, Franklin County, Washington State, and the Department of Energy, Richland. Initial notification of the NRC via the Emergency Notification System (ENS) should be made immediately after notification of the appropriate state and local authorities, and must be made not later than one (1) hour after emergency event declaration. Immediate notifications are outlined in Attachment 7.1, Emergency Notification List, Part A – Immediate Notification List. Notification of other offsite agencies is outlined in Attachment 7.3, Emergency Notification List, Part C - Offsite Agency Notification List. {R-5731}

Notification of selected non-ERO supervisory staff (Construction and Project Management) is

Examination Outline Cross-reference:	Level	RO		SRO
Rev. 1 Date: 12/16/2016	Tier			3
	Group			1
	K/A	2.4.31		
Level of Difficulty: 3	Importance Rating			4.1

Knowledge of annunciator alarms, indications, or response procedures.

Question # 100

CGS is operating in Mode 1.

- A control room annunciator is cycling at the setpoint due to setpoint shift:

What action must be taken prior to disabling the annunciator?

A. Notify the Crew Assistant Operations Manager.

B. Place a Caution Tag on the annunciator.

C. Write a work request to adjust the annunciator setpoint.

D. Perform a 50.59/72.48 evaluation to validate FSAR compliance.

Answer: D

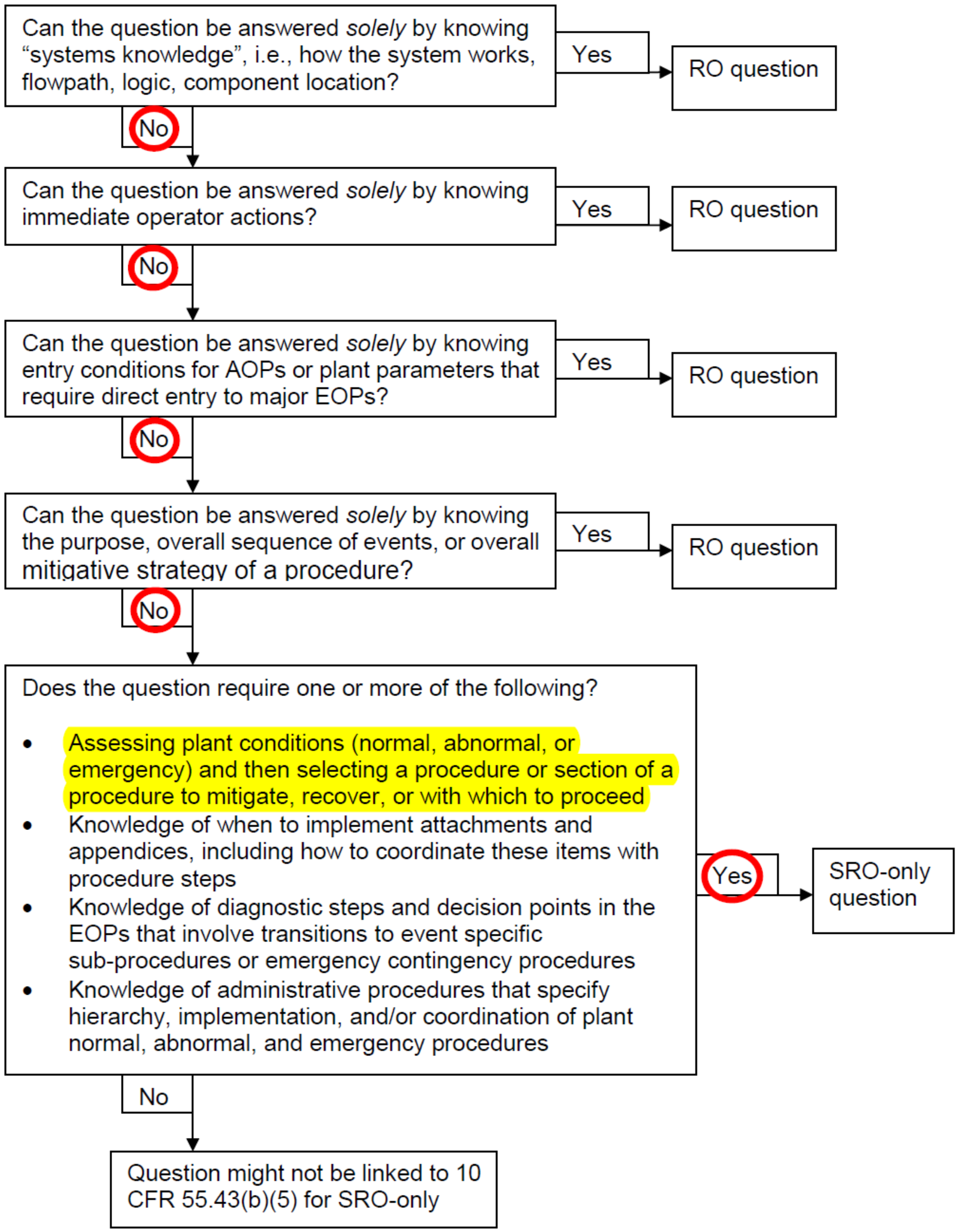
K/A Match:

This question requires the candidate to demonstrate knowledge of requirements to disable a valid annunciator.

SRO Only:
K/A is a “G” statement and

ES-401 **8** **Attachment 2**

Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5) (Assessment and selection of procedures)



Explanation:
A. Incorrect. Plausible since the Crew Assistant Operations Manager should be informed if an annunciator remains disabled for greater than one shift. However, this is not required prior to disabling the annunciator. B. Incorrect. Plausible since a caution clearance should be intiated if an annunciator remains disabled for greater than one shift. However, this is not required prior to disabling the annunciator. C. Incorrect. Plausible since a work request should be written if an annunciator is disabled for greater than one shift due to inoperable source inputs. However, this is not required prior to disabling the annunciator due to setpoint shift. D. Correct. Prior to disabling an annunciator that is cycling at its setpoint (not due to active maintenance), a 50.59/72.48 evaluation should be performed.

Technical Reference(s)		Attached w/ Revision # See Comments / Reference
PPM 1.3.1, Operating Policies, Programs and Practices		

Proposed references to be provided during examination: None

Learning Objective: 11270 Knowledge of annunciators alarms and indications / and use of the response instructions.

Question Source:
Bank #
Modified Bank # (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level:
Memory or Fundamental Knowledge X
Comprehension or Analysis

10 CFR Part 55 Content:
55.41
55.43 Tier 2 “G” Statement

Comments / Reference: PPM 1.3.1, Operating Policies, Programs and Practices	Revision: Major 120, Minor 003

Number: 1.3.1	Use Category: INFORMATION	Major Rev: 120 Minor Rev: 003 Page: 51 of 98
Title: Operating Policies, Programs and Practices		

4.13.7 Disabling a valid annunciator, cycling at setpoint

a. Check redundant indication.

CAUTION

If an annunciator is a valid alarm, cycling at or near setpoint, then a separate 50.59/72.48 is required to be performed prior to disabling the annunciator.

b. IF the annunciator is being disabled for other than active maintenance, THEN perform a 50.59/72.48 to determine that no other FSAR described SSCs will be affected by disabling of the annunciator.

c. IF the annunciator is Fire Protection related, THEN evaluate the need to initiate a fire impairment.

d. IF applicable, THEN revise the applicable annunciator procedure to reference the 50.59 and specific conditions for disabling the annunciator.

e. Obtain permission from the Shift Manager or above, prior to disabling the annunciator. {AR-45835}

f. Disable the annunciator.

g. Log the disabled annunciator in the Control Room Log. {AR-45835}

h. Establish a temporary log to monitor the redundant indication at a frequency determined by the CRS.

i. Initiate a Condition Report.

j. IF the annunciator is disabled GT one shift, THEN perform the following:

1) Notify the Assistant Operations Manager of the disabled annunciator.

2) Place a Caution Clearance containing the following information on the disabled annunciator: {R-3595}, {R-3600}, {AR-45835}

- The annunciator taken out of service,
- Why the annunciator was taken out of service,
- What actions or plant conditions are required to recover the annunciator and any compensatory actions being taken,
- When the annunciator is expected to be returned to service

Number: 1.3.1	Use Category: INFORMATION	Major Rev: 120 Minor Rev: 003 Page: 52 of 98
Title: Operating Policies, Programs and Practices		

4.13.8 Annunciators disabled due to inoperable source inputs

NOTE: An annunciator may be disabled if all the inputs to the annunciator have failed, causing the annunciator to alarm. If some of the inputs are still operable, then alternate methods should be used to disable the failed input(s).

a. Obtain permission from the Shift Manager or above, prior to disabling the annunciator. {AR-45835}

b. Disable the annunciator.

c. Log the disabled annunciator and reason disabled in the Control Room Log. {AR-45835}

d. Establish a temporary log to monitor the redundant indication at a frequency determined by the CRS.

e. IF the annunciator is disabled GT one shift,
THEN perform the following:

1) Initiate a Condition Report.

2) Initiate a Work Request.