Answer Sheet

Exam Title: 2014 ILT SRO NRC Written Exam

Name KEY

Date 12/8/2015

1	a	26	a	51	a	76	c
2	a	27	a	52	c	77	c
3	d	28	d	53	a	78	a
4	d	29	c	54	d	79	d
5	a	30	b	55	d	80	c
6	d	31	d	56	a	81	d
7	c	32	a	57	c	82	b
8	c	33	b	58	d	83	a
9	d	34	c	59	a	84	b
10	d	35	d	60	a	85	d
11	c	36	c	61	d	86	b
12	b	37	d	62	a	87	c
13	d	38	a	63	a	88	b
14	c	39	a	64	a	89	d
15	d	40	c	65	b	90	c
16	b	41	b	66	b	91	c
17	c	42	b	67	d	92	b
18	c	43	c	68	b	93	a
19	c	44	c	69	d	94	b
20	c	45	a	70	c	95	c
21	b	46	b	71	a	96	c
22	a	47	c	72	a	97	d
23	c	48	c	73	d	98	d
24	d	49	a	74	a	99	b
25	a	50	b	75	c	100	c

U.S. Nuclear Regulatory Commission

Site-Specific RO Written Examination

Applicant Information

☐ Name:

Date:	Facility/Unit: Cooper Nuclear Station
Region: I ☐ II ☐ III ☐ IV ■	Reactor Type: W ☐ CE ☐ BW ☐ GE ■
Start Time:	Finish Time:

Instructions

Use the answer sheets provided to document your answers. Staple this cover sheet on top of the answer sheets. To pass the examination, you must achieve a final grade of at least 80.00 percent. Examination papers will be collected 6 hours after the examination begins.

Applicant Certification

All work done on this examination is my own. I have neither given nor received aid.

____
Applicant's Signature

Results

Examination Value	75	Points
Applicant's Score	________	Points
Applicant's Grade	________	Percent

Examination Outline Cross-Reference	Level	RO
Revised to determine applicable TSs required to be entered. Corrected references, added “are” to stem and remove "AP" from LHGR in D.	Tier#	1
	Group#	1
	K/A #	295001G2.2.36
	Rating	3.1
295001 Partial or Complete Loss of Forced Core Flow Circulation. G 2.2.36 Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operations. (CFR: 41.10 / 43.2 / 45.13)		

Question 1

The plant is in Mode 2 during startup with routine relay surveillances in progress on 4160 VAC Bus 1C.

- RPV pressure is at 360 psig.
- 4160V Bus 1C trips due to an inadvertent bus lockout.

Plant conditions have stabilized with Reactor Recirculation flow and neutron flux indications stable.

Which of the following Limiting Conditions for Operation (LCOs) is/are required to be entered?

TS LCO 3.4.1 (Recirculation Loops Operating)...

A. ONLY.

B. and LCO 3.2.1 Average Planar Linear Heat Generation Rate (APLHGR).

C. and LCO 3.2.2 Minimum Critical Power Ratio (MCPR).

D. and LCO 3.2.3 Linear Heat Generation Rate (LHGR).

Answer:
A. ONLY.
Explanation: When bus 1C 4Kv bus locks out, the A RR MG set will trip. This places the unit in

single loop operation in mode 2. Technical Specifications 3.4.1 for Reactor Recirculating Loops operating are applicable in Mode 1 and 2. TS 3.4.1 requires both loops in service with flows matched or one recirculation loop shall be in operation outside of the Stability Exclusion Region of the power/flow map specified in the COLR with the following limits 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; **ONLY applicable with THERMAL POWER ≥ 25% RTP**
b. LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," single loop operation limits specified in the COLR; **ONLY applicable with THERMAL POWER ≥ 25% RTP**
c. LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)," single loop operation limits specified in the COLR; **ONLY applicable with THERMAL POWER ≥ 25% RTP**
d. LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," Function 2.b (Average Power Range Monitor Neutron Flux- High (Flow Biased)), Allowable Value of Table 3.3.1.1-1 is reset for single loop operation. **ONLY applicable in MODE 1.**

Candidates which do not recall ALL the applicable single loop power distribution specifications to TS 3.4.1 makes each distractor plausible.

Distracters:

B. This answer is incorrect, but plausible if the candidate does not realize that TS 3.2.1 is applicable with Thermal Power ≥ 25%.
C. This answer is incorrect, but plausible if the candidate does not realize that TS 3.2.2 is applicable with Thermal Power ≥ 25%.
D. This answer is incorrect, but plausible if the candidate does not realize that TS 3.2.3 is applicable in MODE 1.

Technical References:
Technical Specifications 3.4.1. 3.2.1, 3.2.2, and 3.2.3.

References to be provided to applicants during exam: None

Learning Objective:
INT0320131W0W0100 Given plant condition(s), and the applicable Abnormal/Emergency Procedure, determine the correct subsequent actions required to mitigate the event(s).

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b) 10	

Level of Difficulty:	3	
SRO Only Justification:	N/A	

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 outside of the Stability Exclusion Region of the power/flow map specified in the COLR.

OR

One recirculation loop shall be in operation outside of the Stability Exclusion Region of the power/flow map specified in the COLR with the following limits 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 Monitor Neutron Flux—High (Flow Biased)), Allowable Value of Table 3.3.1.1-1 is reset for single loop operation.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or two recirculation loops in operation with core flow as a function of core THERMAL POWER in the Stability Exclusion Region of the power/flow map.	A.1 Initiate action to exit the Stability Exclusion Region.	Immediately

(continued)

APLHGR
3.2.1

3.2 POWER DISTRIBUTION LIMITS

3.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)

LCO 3.2.1 All APLHGRs shall be less than or equal to the limits specified in the COLR.

APPLICABILITY: THERMAL POWER ≥ 25% RTP.

ACTIONS

MCPR
3.2.2

3.2 POWER DISTRIBUTION LIMITS

3.2.2 MINIMUM CRITICAL POWER RATIO (MCPR)

LCO 3.2.2 All MCPRs shall be greater than or equal to the MCPR operating limits specified in the COLR.

APPLICABILITY: THERMAL POWER ≥ 25% RTP.

ACTIONS

LHGR
3.2.3

3.2 POWER DISTRIBUTION LIMITS

3.2.3 LINEAR HEAT GENERATION RATE (LHGR)

LCO 3.2.3 All LHGRs shall be less than or equal to the limits specified in the COLR.

APPLICABILITY: THERMAL POWER ≥ 25% RTP.

ACTIONS

Examination Outline Cross-Reference	**Level**	RO
	Tier#	1
	Group#	1
	K/A #	295003 AK1.01
	Rating	2.7

295003 Partial or Complete Loss of AC / 6

AK1. Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER : (CFR: 41.8 to 41.10)

AK1.01 Effect of battery discharge rate on capacity

Question 2

5.3SBO (Station Blackout) has been entered by the operating crew.

Which one of the following completes the statements below regarding the coping time of the 250 VDC Batteries and the impact on battery capacity if actions required for the HPCI system are not taken IAW 5.3SBO (Station Blackout)?

The MINIMUM battery coping time during a SBO is ____(1)____ hours.

In order to reduce battery discharge rate to ensure battery capacity will meet the coping time, operators are required to manually secure HPCI after ____(2)____ of operation.

A. (1) 4
(2) one cycle

B. (1) 4
(2) two cycles

C. (1) 8
(2) one cycle

D. (1) 8
(2) two cycles

Answer:
A. (1) 4

(2) one cycle

Explanation:
This K/A, the operational implications of the effect of battery discharge rate on capacity during a loss of AC power, can be related to operator actions for reducing battery loading during a Station Blackout. The SBO analysis provides for battery capacity to cope for 4 hours following the loss of power event. CNS has committed to secure HPCI following one cycle of operation (~10 minutes) to extend battery capacity. One cycle is auto initiation on low RPV water level followed by trip on high level. If HPCI trips on high level the Auxiliary Oil pump (AOP – DC powered) will continue to run until auto start on low level (raising battery discharge rate). If HPCI is not secured, battery discharge rate will rise – reducing the battery capacity.

Distracters:

B. This answer is incorrect due to HPCI being secured one cycle of operation. RCIC is credited to automatically cycle to maintain RPV water level during the SBO. This choice is plausible due to the number of HPCI cycles being easily confused. The unprepared candidate might not remember that HPCI Aux Oil Pump is required to be placed in P-T-L vs. placing system back in service. The candidate that correctly identifies the coping time and confuses how many cycles HPCI is allowed to be operated during a SBO would select this answer.

C. This answer is incorrect due to coping time being 4 hours. This choice is plausible due to 8 hours being the capacity for station emergency lighting. The candidate that confuses the 250 VDC Battery coping time with emergency lighting and correctly identifies HPCI system operation raising the discharge rate would select this answer.

D. This answer is incorrect due to coping time being 4 hours and HPCI being secured within 10 minutes. This choice is plausible due to 8 hours being the capacity for station emergency lighting and the number of HPCI cycles being easily confused, not remembering that HPCI AOP is required to be placed in P-T-L vs. placing system back in service. The candidate that confuses the 250 VDC Battery coping time with emergency lighting and how many cycles HPCI is allowed to be operated during a SBO would select this answer.

Technical References:
Procedure 5.3SBO (Station Blackout), Rev. 35

References to be provided to applicants during exam: NONE

Learning Objective:

INT0060119001150D Describe each of the following special events evaluated in the CNS USAR that could challenge the integrity of the radioactive material barriers: Station Blackout

INT00601190012200 Given a specific USAR analyzed Special Event, describe the initial plant condition assumed in the analysis.

INT00601190012500 Given a specific analyzed Anticipated Operational Transient or Special Event, state the appropriate safety actions necessary to prevent exceeding their safety design bases.

INT0320131W0W0100 Given plant condition(s), and the applicable Abnormal/Emergency Procedure, determine the correct subsequent actions required to mitigate the event(s).		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b) 8	
Level of Difficulty:	3	
SRO Only Justification:	N/A	

4. SUBSEQUENT OPERATOR ACTIONS

4.1 Record current time and date. Time/Date: ______/______

4.2 Steps 4.2.1 through 4.2.6 may be performed concurrently and in any order:

4.2.1 Enter and direct Doniphan Control Center to perform Energy Control Center Instruction, Cooper Nuclear Station - Black Plant Procedure (located in Switching Order Book).Ⓟ[3]

4.2.2 Enter Procedure 2.4FPC.Ⓟ[4]

4.2.3 Enter Procedure 5.3ALT-STRATEGY.

<u>NOTE</u> – If Supplemental Diesel Generator will be placed in service inside of the 4 hour coping period of a station black out, then requirements of 10CFR50.54(x) must be addressed.

4.2.4 Place Supplemental Diesel Generator in service per Procedure 2.2.99.

4.2.5 Ensure Maintenance notified Severe Accident Management Guideline (SAMG) Diesel Generator installation required per Procedure 2.2.100.

4.2.6 Perform following Attachments concurrently:

RPV AND CONTAINMENT GUIDELINE	Attachment 1	Page 5
BALANCE OF PLANT GUIDELINE	Attachment 2	Page 14
ELECTRICAL SYSTEMS GUIDELINE	Attachment 3	Page 16

ATTACHMENT 4 INFORMATION SHEET

1. DISCUSSION

1.1 This procedure provides operator guidance for a loss of all AC power (on and off-site). Entry into this procedure is not required for the momentary transient while the diesel generators are starting and loading.

1.2 Although all attachments are carried out concurrently, the following priorities should be considered:

1.2.1 Reactor is shut down.

1.2.2 Reactor critical parameters are monitored and controlled.

1.2.3 Primary containment parameters are monitored.

1.2.4 Restoration of AC power.

1.2.5 Reduction of DC loads.Ⓟ[7]

1.2.6 SAMG DG installation.

1.3 After initial attempts to restore AC power from Panel C unsuccessful, determining which 4160V Bus should be energized with Supplemental Diesel should be considered.

1.3.1 When powered from the Supplemental Diesel, transferring 4160V Bus 1F or 4160 Bus 1G to Emergency Transformer or Emergency Diesel Generator will require de-energizing the bus.

1.3.2 The Supplemental Diesel is not considered an emergency power supply and is not considered in classifications involving loss of power. The applicable EALs in Procedure 5.7.1 still apply and CNS remains in the Station Blackout Special Plant Event until a critical bus is energized from an Emergency Diesel or an off-site power supply.

1.4 The only reactor water level indications left in the Control Room are the three narrow range instruments, associated recorder, and NBI-LT-59D on RVLC/RFPT HMI. All other wide range and fuel zone indications must be monitored locally in the Reactor Building at the instrument racks and/or ASD Room.

1.5 This procedure assumes that RPV water level and pressure is initially controlled by HPCI. Per Reference Step 2.3.4, CNS has committed to secure HPCI after one cycle of operation in order to extend station battery life during station blackout. Per Reference Step 2.3.3, one cycle of HPCI is ~ 10 minutes. SBO analysis assumes RCIC is operable and maintains RPV level and pressure until on-site or off-site electrical power can be restored. If RCIC is unable to perform this function, compensatory actions must be taken to ensure adequate core cooling.

auto started and is supplying power to Panel [illegible].

NOTE – Emergency lights have eight (8) hour capacity.

4.7 Obtain portable lighting and additional batteries. Pre-staged lighting (flashlights) can be obtained from:

4.7.1 Control Room Emergency Locker.

4.7.2 ASD Locker.

4.8 Obtain following reviews:

- Control Room Supervisor Signature/Date: ______________ / ________
- Shift Manager Signature/Date: ______________ / ________
- AOM-Shift Signature/Date: ______________ / ________

4.9 Forward completed procedure to OSG Supervisor

Examination Outline Cross-Reference	Level	RO
	Tier#	1
	Group#	1
	K/A #	295004 AK2.01
	Rating	3.1

295004 Partial or Total Loss of DC Pwr / 6

AK2. Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF D.C. POWER and the following: (CFR: 41.7 / 45.8)

AK2.01 Battery charger

Question 3

The plant is operating at rated power when the AC INPUT BREAKER on 24 VDC Charger B2 trips open.

What is the impact on the Div. II 24 VDC bus?

A. Power will immediately be lost to the Div. II 24 VDC bus.

B. Div. II 24 VDC Bus remains energized by the B1 Battery Charger ONLY.

C. Div. II 24 VDC Bus remains energized by the B2 battery and the B1 Battery Charger, which maintains B2 battery fully charged.

D. Div. II 24 VDC Bus remains energized by the B2 battery and the B1 Battery Charger until the B2 battery discharges.

Answer:

D. Div. II 24 VDC Bus remains energized by the B2 battery and the B1 Battery Charger until the B2 battery discharges.

Explanation:

Bus B2 will initially remain energized by the capacity left in the B2 battery. Once the battery capacity has diminished, voltage will no longer be sufficient to maintain operation of the Div. II 24 VDC Bus.

Distracters:

A. This answer is incorrect due to the Div. II 24 VDC bus remaining energized until the B2 Battery discharges. This choice is plausible due to confusing the DC lineup which provides parallel charger power output to both the battery and the bus (loss of charger output causes loss of power to the positive bus). The candidate that confuses the Charger alignment would select this answer.

B. This answer is incorrect due to the Div. II 24 VDC bus remaining energized until the B2 Battery discharges. This choice is plausible due to confusing the DC lineup (confuses negative charger is sufficient to keep bus energized) . The candidate that confuses the Charger alignment would select this answer.

C. This answer is incorrect due to the Div. II 24 VDC bus remaining energized until the B2 Battery discharges. This choice is plausible due to confusing the DC lineup which provides parallel charger power output to both the battery and the bus (confuses B1 charger providing power to B2 battery). The candidate that confuses the Charger alignment would select this answer.

Technical References:
B&R Drawing 3058, Rev. 64
Procedure 2.2.26 (24 VDC Electrical System), Rev. 24

References to be provided to applicants during exam: None

Learning Objective: 8.k, 8.l
Given a specific DC Electrical Distribution system malfunction, determine the effect on any of the following: IRMs, SRMs.

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b) 7	
Level of Difficulty:	3	
SRO Only Justification:	N/A	

ATTACHMENT 1 INFORMATION SHEET

1. DISCUSSION

1.1 FUNCTION

1.1.1 The system provides uninterruptable DC power to Neutron Monitoring and Process Radiation Monitoring equipment.

1.2 OPERATING CHARACTERISTICS

1.2.1 The 24 VDC Power System supplies the source and intermediate range neutron monitors and their trip auxiliaries, selected process radiation monitors, and process radiation monitor trip auxiliaries. The chargers are supplied from the critical distribution panels which receive their power from 460V critical motor control centers. Loss of one of the two 24 VDC Systems will not affect plant safety since redundant instrumentation will continue to be supplied by the other system. However, total loss of the 24 VDC System will result in actuation of the intermediate range neutron monitor trips. During normal operation, the load requirements are supplied from the battery chargers. Upon failure of the charger to supply power, the DC loads are supplied from the batteries.

2. REFERENCES

2.1 UPDATED SAFETY ANALYSIS REPORT

2.1.1 Section VIII-7.0, 24 Volt D-C Power Systems.

2.2 DRAWINGS

2.2.1 B&R Drawing 3002, Sheet 1, Auxiliary One Line Diagram.

2.2.2 B&R Drawing 3006, Sheet 5, Auxiliary One Line Diagram.

2.2.3 B&R Drawing 3058, DC One Line Diagram.

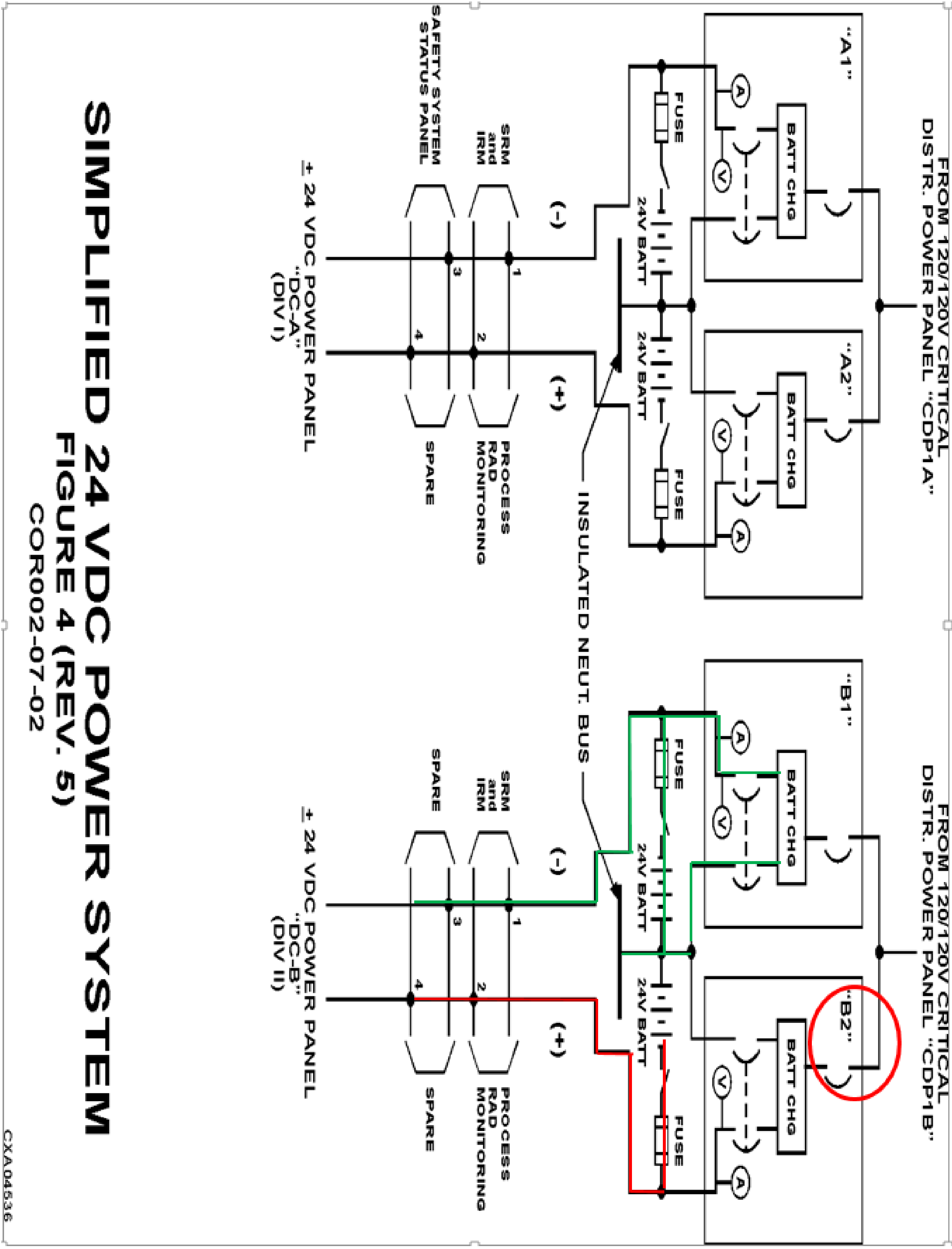
2.3 VENDOR MANUALS

2.3.1 CNS Number 0228, Station Batteries, Battery Chargers, and Inverters.

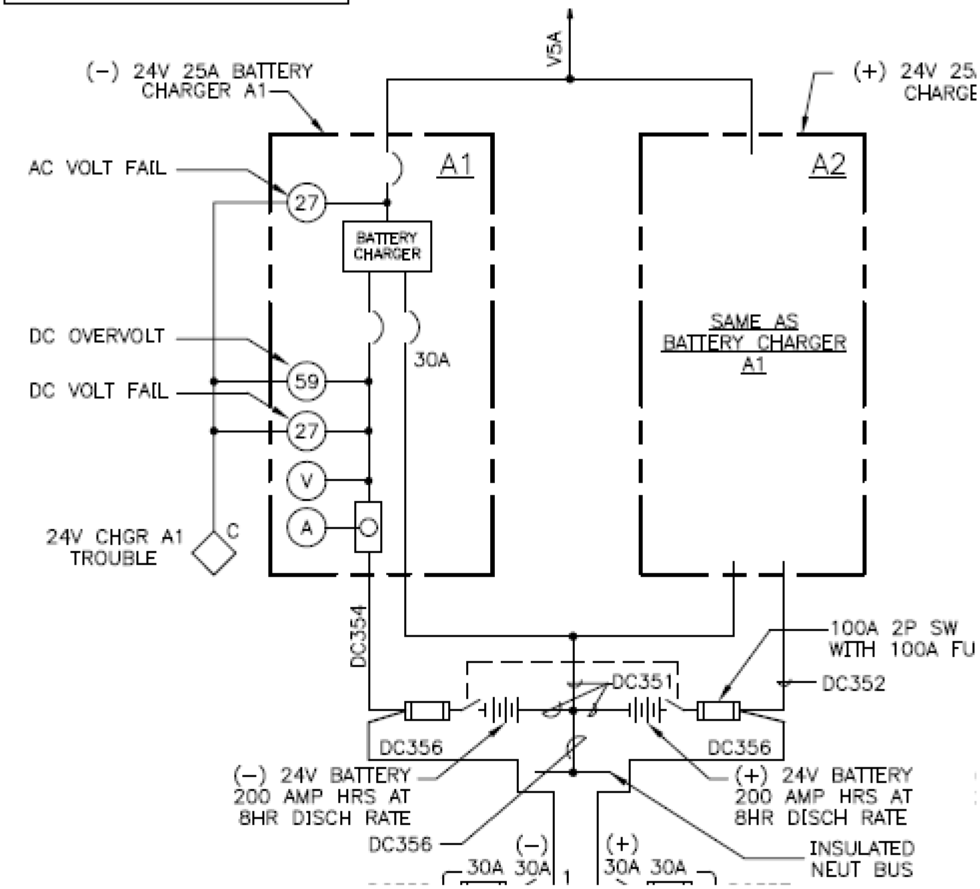
2.4 PROCEDURES

2.4.1 System Operating Procedure 2.2A_24DC.DIV1, 24 VDC Power Checklist (Div 1).

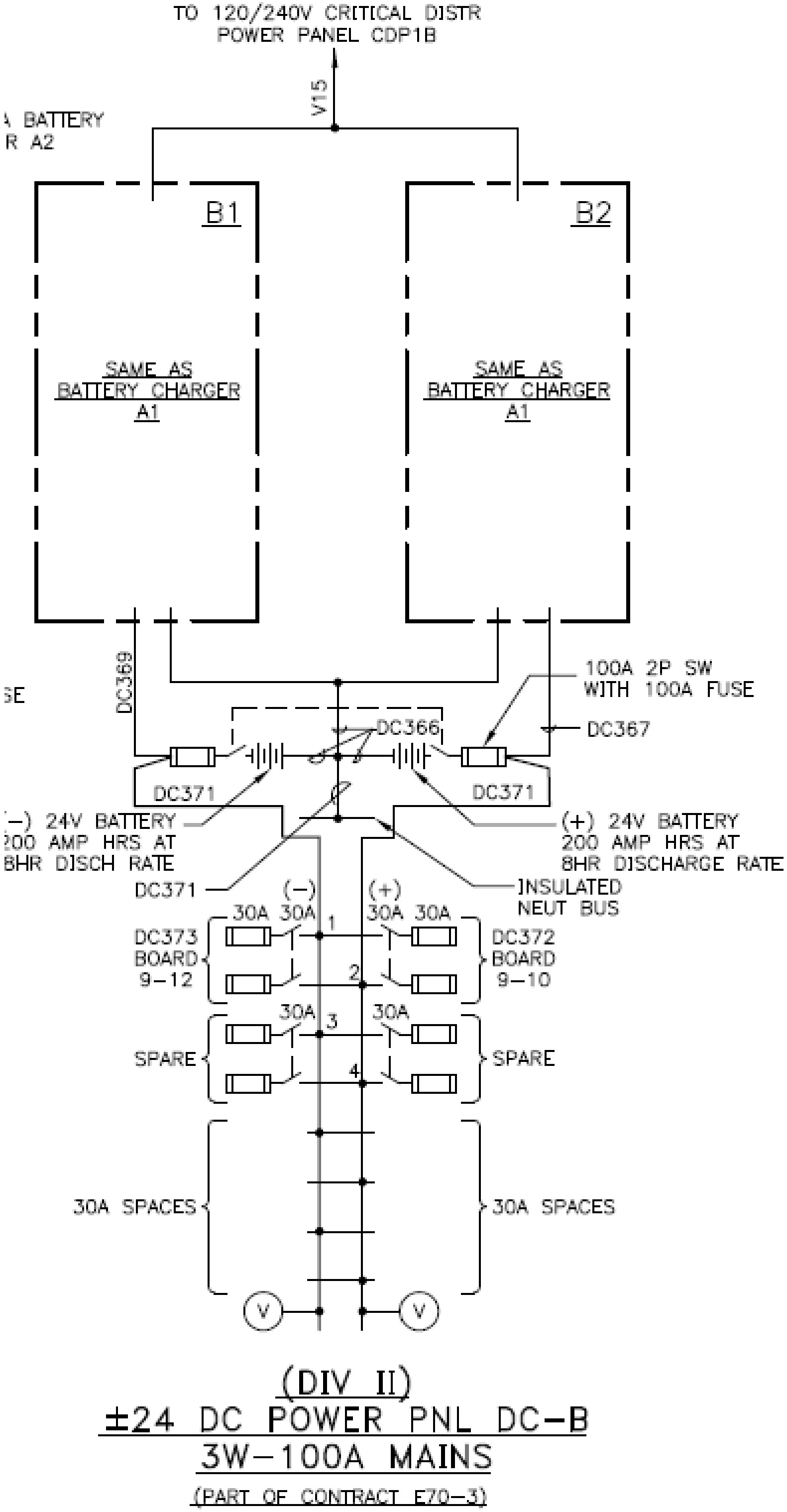
2.4.2 System Operating Procedure 2.2A_24DC.DIV2, 24 VDC Power Checklist (Div 2).


FROM 120/120V CRITICAL DISTR. POWER PANEL "CDP1A"
"A1"
"A2"
BATT CHG
BATT CHG
FUSE
FUSE
24V BATT
24V BATT
(-)
(+)
INSULATED NEUT. BUS
SRM and IRM
SAFETY SYSTEM STATUS PANEL
PROCESS RAD MONITORING
SPARE
± 24 VDC POWER PANEL "DC-A" (DIV I)
FROM 120/120V CRITICAL DISTR. POWER PANEL "CDP1B"
"B1"
"B2"
BATT CHG
BATT CHG
FUSE
FUSE
24V BATT
24V BATT
(-)
(+)
SRM and IRM
SPARE
PROCESS RAD MONITORING
SPARE
± 24 VDC POWER PANEL "DC-B" (DIV II)
SIMPLIFIED 24 VDC POWER SYSTEM
FIGURE 4 (REV. 5)
COR002-07-02
CXA04536

Portion of
Burns & Roe 3058



Portion of
Burns & Roe 3058 R64



Examination Outline Cross-Reference	Level	RO
Candidate is required to analyze MT trip while operating at 90% impact on BPV operation. Determining whether BPVs fast open or throttle open based upon the current power level and why is higher cognitive.	Tier#	1
	Group#	1
	K/A #	295005 AK3.07
	Rating	3.8

295005 Main Turbine Generator Trip / 3

AK3. Knowledge of the reasons for the following responses as they apply to MAIN TURBINE GENERATOR TRIP: (CFR: 41.5 / 45.6)

AK3.07 Bypass valve operation

Question 4

The plant is operating at 90% power with DEH in Mode 4 when the Main Turbine trips.

Which one of the following completes the statement below regarding how the Bypass Valves (BPVs) respond and the reason for this response?

All BPVs ____(1)____ to control pressure in order to ____(2)____ following a Main Turbine trip.

A. (1) throttle open
(2) prevent SRVs from opening

B. (1) throttle open
(2) reduce the peak RPV pressure during the transient

C. (1) fast open for 5 seconds and then throttle
(2) prevent SRVs from opening

D. (1) fast open for 5 seconds and then throttle
(2) reduce the peak RPV pressure during the transient

Answer:
D. (1) fast open for 5 seconds and then throttle (2) reduce the peak RPV pressure during the transient
Explanation: BPV fast opening does NOT prevent SRVs from opening.

On a Main Turbine trip the BPVs in AUTO will fully open for ~ 5 seconds prior to throttling, as needed, to control pressure. This fast opening is provided to limit peak RPV pressure reached during MT trips above 25% power (total capacity of BPVs) to ensure MCPR Safety limit is not exceeded.
BPVs are required to be operable with reactor power ≥ 25%
The "above the line" required for TS 3.7.7 states the actions which must be met if the MT Bypass system is not OPERABLE ≥25% RTP. These actions are directly related to MCPR limits. The bypass capacity of the system is 25% of the Nuclear Steam Supply System rated steam flow. With initial power level >25%, SRVs may open along with BPVs to protect the RPV.
MT Bypass system fast opening along with SRV opening limits peak RPV pressure if the MT Trips from 90% power

Distracters:

A. This answer is incorrect due to BPVs fast opening for 5 seconds and then throttling to maintain pressure to reduce the peak pressure on a MT trip. This choice is plausible if reactor power were less than 25%, the BPVs throttle (no fast open permissive) and preventing SRV operation being desired. The candidate that confuses how BPVs operate at 90% power vs. <25% and the reason would select this answer.
B. This answer is incorrect due to BPVs fast opening for 5 seconds and then throttling to maintain pressure to reduce the peak pressure on a MT trip. This choice is plausible if reactor power were less than 25%, the BPVs throttle (no fast open permissive). The candidate that confuses how BPVs operate at 90% power vs. <25% and correctly identifies the reason would select this answer.
C. This answer is incorrect due to BPVs fast opening for 5 seconds and then throttling to maintain pressure to reduce the peak pressure on a MT trip. This choice is plausible due to preventing SRV operation being desired. The candidate that recognizes how BPVs operate at 90% power and confuses the reason would select this answer.

Technical References:
TS 3.7.7 Amendment 178
Procedure 2.2.77.1 {Digital Electro-Hydraulic (DEH) Control System } Rev. 37
Procedure 2.1.4.1 (Rapid Shutdown), Rev. 38

References to be provided to applicants during exam: None

Learning Objective: COR0020902001040L Describe how the DEH control system operates to control the following: Bypass valve position

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X

Question Cognitive Level:	Memory/Fundamental	

	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b) 5	
Level of Difficulty:	2	
SRO Only Justification:	N/A	

Main Turbine Bypass System
3.7.7

3.7 PLANT SYSTEMS

3.7.7 The Main Turbine Bypass System

LCO 3.7.7 The Main Turbine Bypass System shall be OPERABLE.

OR

LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," limits for one inoperable main turbine bypass valve, as specified in the COLR, are made applicable.

APPLICABILITY: THERMAL POWER ≥ 25% RTP.

ACTIONS

LCO The Main Turbine Bypass System is required to be OPERABLE to limit peak pressure in the main steam lines and maintain reactor pressure within acceptable limits during events that cause rapid pressurization, so that the Safety Limit MCPR is not exceeded. With one Main Turbine Bypass Valve inoperable, modifications to the MCPR operating limits (LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)") may be

Cooper B 3.7-31 11/25/12

4.22 Inform all Control Room Operators that Group 2 isolation likely to occur due to level shrink caused by scram.

Initials/Time/Date: ________/________/________

4.23 IF Annunciator B-1/E-1, BPV FAST-OPEN PERMISSIVE INHIBIT, is clear, THEN inform all Control Room Operators that on main turbine trip:

4.23.1 BPVs in AUTO will fully open for ~ 5 seconds prior to throttling, as needed, to control pressure.

4.23.2 BPVs in MANUAL will open for 3 seconds, then transfer bypass control back to MANUAL at the pre-trip setting.

Initials/Time/Date: ________/________/________

4.24 All steps in Section 4 have been reviewed.

SM/Time/Date: ________/________/________

4.25 Scram Reactor per Procedure 2.1.5.

5. RECORDS

5.1 Entire procedure is sent to Operations Department Clerk (quality record upon completion of procedure).

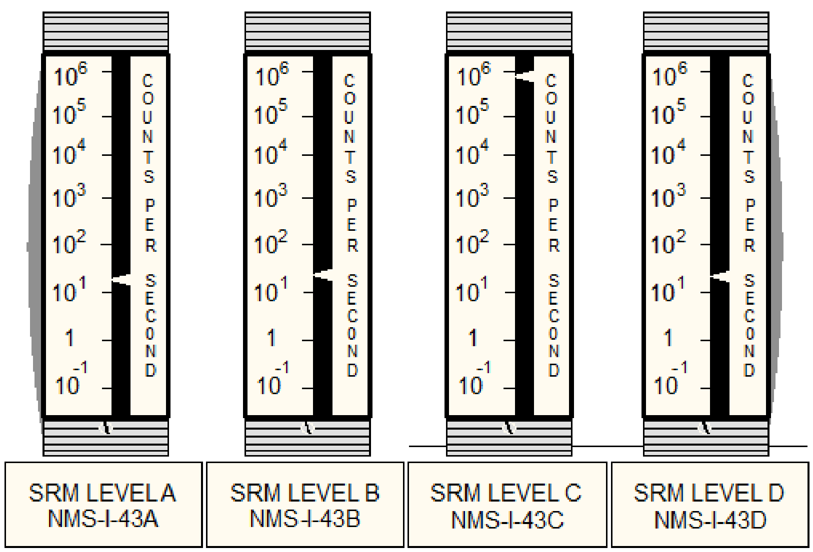
Examination Outline Cross-Reference	Level	RO
	Tier#	1
	Group#	1
	K/A #	295006 AA1.01
	Rating	4.2
295006 SCRAM / 1 AA1. Ability to operate and/or monitor the following as they apply to SCRAM : (CFR: 41.7 / 45.6) AA1.01 RPS		

Question 5

The plant is in a refueling outage with the following conditions:

- Reactor Mode Switch position is REFUEL
- All four SHORTING LINK SWITCH positions are OPEN

An SRM detector failure results in the following indications:



What is the status of RPS Logic Channels?

RPS Channel A ____ (1) ____ de-energized.

RPS Channel B ____ (2) ____ de-energized.

A. (1) is

(2) is

B. (1) is
(2) is NOT

C. (1) is NOT
(2) is

D. (1) is NOT
(2) is NOT

Answer:		
A. (1) is (2) is		
Explanation: The SRM UPSCALE TRIP (High-High Scram) is only utilized during certain refuel and/or special operations. With the C SRM exceeding the Upscale Trip setpoint (≤1 x 10E5 cps) and the shorting link switches in OPEN, a High-High channel trip on any one neutron monitoring channel will cause a FULL reactor Scram.		
Distracters: B. This answer is incorrect due to both RPS channels being de-energized. This choice is plausible due to shorting link switches normally in the CLOSED position signal will be generated, but not a Scram signal. The candidate that confuses RPS logic would select this answer. C. This answer is incorrect due to both RPS channels being de-energized. This choice is plausible due to shorting link switches normally in the CLOSED position signal will not generate a Scram signal. The candidate that confuses RPS logic would select this answer. D. This answer is incorrect due to both RPS channels being de-energized. This choice is plausible due to shorting link switches normally in the CLOSED position signal will not generate a Scram signal. The candidate that confuses Shorting Link switch positions would select this answer.		
Technical References: Procedure 4.1.1 (SRM System), Rev 23		
References to be provided to applicants during exam: NONE		
Learning Objective:		
COR002-21-02, 8. Given a specific RPS malfunction, determine the effect on any of the following: f. RPS logic channels		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	

	New	X
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b) 7	
Level of Difficulty:	2	
SRO Only Justification:	N/A	

1. PURPOSE

1.1 This procedure provides instructions for Operations personnel to operate Source Range Monitoring (SRM) System.

2. PRECAUTIONS AND LIMITATIONS

2.1 During periods when all four RPS shorting link switches are open, a High-High channel trip on any one neutron monitoring channel will cause a reactor scram.

3. REQUIREMENTS

3.1 System Component Checklist, Procedure 4.1.1A, is complete to support system operation.

4. INSERTING SRM DETECTORS

NOTE – When a detector is fully inserted, IN light on applicable IN/OUT DISPLAY switch will be turned on.

4.1 IF required, THEN momentarily press and release SRM/IRM DETECTOR POS display (POWER ON) button and check DETECTOR POSITION light turns on.

PROCEDURE 4.1.1	REVISION 23	PAGE 1 OF 9

Examination Outline Cross-Reference	**Level**	RO
Enlarged meters and removed Steam Table operator aide. Provided actual meter pictures with pressures indicating 1100. 1000, 900, & 800 psig per recommendation. Steam tables are provided as a standard reference and should not be included in the total reference count.	**Tier#**	1
	Group#	1
	K/A #	295016 AA2.06
	Rating	3.3

295016 Control Room Abandonment

AA2. Ability to determine and/or interpret the following as they apply to CONTROL ROOM ABANDONMENT : (CFR: 41.10 / 43.5 / 45.13)

AA2.06 Cooldown rate

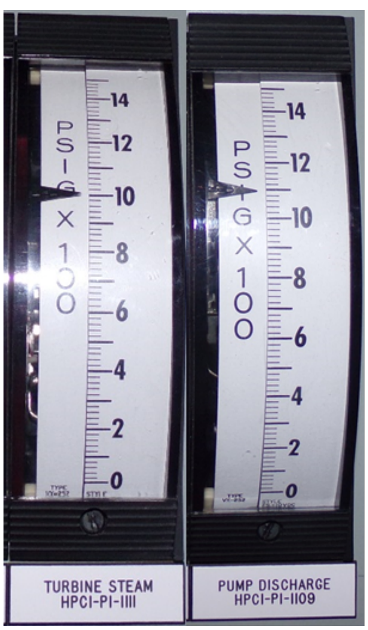
Question 6

Reference Provided

5.1ASD (Alternate Shutdown) has been entered with RPV level and pressure control established with HPCI.

The following indications exist at the times indicated:

TIME
1600



TIME
1615

What is the current RPV cooldown rate?

A. 24°F/hr

B. 26°F/hr

C. 96°F/hr

D. 104°F/hr

Answer:
D. 104°F/hr
Explanation: During operations at the ASD Panel, HPCI Steam Supply pressure is utilized to monitor RPV pressure and cooldown rate. There is an operator aide posted next to the HPCI Steam Supply pressure gauge which allows for gross interpretation of pressure (psig) to temperature (°F). However steam tables & calculators are contained in each operator's backpack which is obtained prior to reporting assigned locations. PSIA = PSIG + 14.7 15 min/60 min = .25 hr Steam Supply Pressure at 1600 = 1000 psig + 14.7 = 1014.7 psia -> 546°F Steam Supply Pressure at 1615 = 800 psig + 14.7 = 814.7 psia -> 520°F 546°F – 520°F = 26°F/.25 hr = 104°F/hr
Distracters: A. This answer is incorrect due to cooldown rate currently being 104°F/hr. This choice is plausible due confusing HPCI discharge pressure with Steam Supply pressure and not converting 15 minutes to an hour. The candidate that confuses HPCI discharge pressure for reactor pressure and does not convert minutes to hour would select this answer. Discharge Pressure at 1600 = 1100 psig + 14.7 = 1114.7 psia -> 558°F Discharge Pressure at 1615 = 900 psig + 14.7 = 914.7 psia -> 534°F 558°F – 534°F = 24°F/.25 hr = 96°F/hr B. This answer is incorrect due to cooldown rate currently being 104°F/hr. This choice is plausible due not converting 15 minutes to an hour. The candidate that does not convert minutes to hour would select this answer. C. This answer is incorrect due to cooldown rate currently being 104°F/hr. This choice is plausible due confusing HPCI discharge pressure with Steam Supply pressure. The candidate that confuses HPCI discharge pressure for reactor pressure and correctly converts minutes to hour would select this answer.
Technical References: 5.4FIRE-S/D (Fire Induced Shutdown From Outside Control Room Rev. 64 5.1ASD (Alternate Shutdown), Rev. 17 Steam Tables
References to be provided to applicants during exam: Steam Tables

Learning Objective:		
INT0320134H0H0100 Given plant condition(s), and the applicable Abnormal/Emergency Procedure, determine the correct subsequent actions required to mitigate the event(s).		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b) 10	
Level of Difficulty:	3	
SRO Only Justification:	N/A	

ATTACHMENT 1 ASD OPERATOR

1. Establish RPV level/pressure indications at HPCI ASD Panel by performing following:
   - 1.1 Place following ISOLATION switches in ISOL:
     - 1.1.1 LEVEL INDICATORS.
     - 1.1.2 HPCI CONTROL AND INDICATION.
     - 1.1.3 MO-15.
     - 1.1.4 MO-16.
2. Direct Control Building Operator to place RCIC ISOLATION SWITCH to ISOL per Attachment 2, Step 1.
3. IF subsequent Operator Actions in Step 4.3 of procedure body were completed in Control Room, THEN go to Step 7.
4. IF subsequent Operator Actions in Step 4.3 of procedure body were <u>not</u> completed in Control Room, THEN perform Steps 5 and 6 concurrently with Step 7.

<u>NOTE</u> – Steps 5 and 6 must be performed in the sequence listed. Inform ASD Operator immediately when each task is complete.

5. Direct following personnel to perform specified tasks:
   - 5.1 Control Building Operator scram reactor per Attachment 2, Step 2.
   - 5.2 Reactor Building Operator check all HCU scram inlet and outlet valves open per Attachment 5, Step 1.
6. WHEN Step 5 actions completed, THEN direct following personnel to perform specified task in order given:
   - 6.1 Turbine Building Operator trip Main Turbine per Attachment 3, Step 1.
   - 6.2 IF two RFPs running, THEN direct Diesel Generator Operator to trip per Attachment 4, Step 1.1, RFP not controlling level.
   - 6.3 Turbine Building Operator leave one condensate booster pump running (A or B preferred) and trip remaining condensate booster pump(s) per Attachment 3, Step 3.
   - 6.4 Turbine Building Operator leave one condensate pump running (A or B preferred) and trip remaining condensate pump(s) per Attachment 3, Step 4.
   - 6.5 Control Building Operator trip running RRMG Set(s) per Attachment 2, Step 5.

ATTACHMENT 14 ASD TOOLS

NOTE – Equipment in the ASD Locker is also used for Procedures 5.4POST-FIRE-CONTROL, 5.4POST-FIRE-TURBINE, 5.4POST-FIRE-REACTOR, 5.3ALT-STRATEGY, and 5.1ASD. ASD Locker, tool totes, and backpacks may contain equipment not used for 5.4FIRE-S/D actions.

1. The ASD Operator Backpack contains the following:

1.1 Procedure 5.4FIRE-S/D.

1.2 Procedure 5.1ASD.

1.3 Steam Tables.

1.4 Calculator.

1.5 One flashlight.

1.6 Battery backup for radio.

1.7 Twenty AA batteries for radio battery backup (12) and hard hat flashlights (8).

1.8 Three AAA batteries for hard hat head lamp.

1.9 The following keys:Ⓟ4

1.9.1 A432 Key (NLNPO key ring or Key 24, 43, or 55 in CR SM Key Locker), Elect. Breaker Locks.

1.9.2 Key CAT 45 (SO key ring), MOV LASP and DC starters.

1.9.3 Grand Master Key (NLNPO key ring or Key 12 or 25 in CR SM Key Locker), BLDG Grand Master.

1.9.4 P233 Key (Key 42 in CR SM Key Locker), N_2 Panel and SOV Bypass (Attachment 12).

2. Each designated Bldg Operator or DG (Supplemental) Operator Backpack contains the following:

2.1 Procedure 5.4FIRE-S/D.

2.2 Procedure 5.1ASD.

2.3 One flashlight.

2.4 Battery backup for radio.

2.5 Twenty AA batteries for radio battery backup (12) and hard hat flashlights (8).

Examination Outline Cross-Reference	Level	RO
Modified stem to "What REC load(s) is/are manually isolated" and spelled out RWCU Non-Regen HX Inlet vs. Reactor Water Cleanup.	Tier#	1
	Group#	1
	K/A #	295018G2.1.23
	Rating	4.3
295018 Partial or Total Loss of CCW / 8 G2.1.23 Ability to perform specific system and integrated plant procedures during all modes of plant operation. (CFR: 41.10 / 43.5 / 45.2 / 45.6)		

Question 7

The plant is operating at power when REC system pressure lowers to 50 psig with all available pumps operating.

What REC load(s) is/are manually isolated IAW 5.2REC (Loss of REC) Immediate Operator Actions?

A. Augmented Radwaste ONLY.

B. Control Rod Drive pumps ONLY.

C. Augmented Radwaste and RWCU Non-Regen HX Inlet.

D. Control Rod Drive pumps and RWCU Non-Regen HX Inlet.

Answer:
C. Augmented Radwaste and RWCU Non-Regen HX Inlet.
Explanation: Requires knowledge of 5.2REC operator immediate actions during a loss of CCW. REC system pressure < 62 psig with all available REC pumps in operation requires isolating REC to RWCU and Augment RW.
Distracters: A. This answer is incorrect due to RWCU also being required to be isolated. This choice is plausible due to Augmented RW being required to be isolated. The candidate that forgets RWCU also requires isolation would select this answer. B. This answer is incorrect due to Augmented RW & RWCU being required to be

isolated. This choice is plausible due to CRD being required to be isolated as part of the Supplemental actions. The candidate that confuses procedure supplemental with immediate actions would select this answer.

D. This answer is incorrect due to Augmented RW & RWCU being required to be isolated. This choice is plausible due to CRD being required to be isolated as part of the Supplemental actions. The candidate that confuses procedure supplemental with immediate actions would select this answer.

Technical References:
5.2REC (Loss of REC), Rev. 16

References to be provided to applicants during exam: NONE

Learning Objective: INT0320126N0N0100 Given plant condition(s), determine from memory all immediate operator actions required to mitigate the event(s).

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b) 10	
Level of Difficulty:	3	
SRO Only Justification:	N/A	

3. IMMEDIATE OPERATOR ACTIONS

Scram Actions

3.1 IF REC HEADER PRESSURE ≤ 62 psig, THEN start available REC pumps.

3.2 IF REC HEADER PRESSURE not restored, THEN close following valves:

3.2.1 REC-AO-710, RWCU NON-REGEN HX INLET.

3.2.2 REC-MO-1329, AUGMENTED RADWASTE SUPPLY.

4. SUBSEQUENT OPERATOR ACTIONS

4.1 Record current time and date. Time/Date: _______/_________

PROCEDURE 5.2REC	REVISION 16	PAGE 1 OF 17

4.2 IF low pressure isolation occurred and initiating condition cleared, THEN restore system from isolation per Attachment 4 (Page 8).

4.3 IF REC HEADER PRESSURE not restored after completing Immediate Operator Actions, THEN perform following:

4.3.1 **SCRAM and enter Procedure 2.1.5**.

4.3.2 Stop both Reactor Recirc pumps and enter Procedure 2.4RR.

4.3.3 Stop running CRD pump.

NOTE – Securing all AC lube oil pumps first will cause DC lube oil pumps to start unless DC oil pump control switches are first taken to STOP and allowed to spring return to their normal positions.

4.3.4 WHEN Recirc MG Sets have stopped, THEN perform following:

Examination Outline Cross-Reference	Level	RO
Comment incorporated – spelled out filter demineralizer.	Tier#	1
	Group#	1
	K/A #	295019 AK2.04
	Rating	2.8

295019 Partial or Total Loss of Inst. Air / 8

AK2. Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR and the following: (CFR: 41.7 / 45.8)

AK2.04 Reactor water cleanup

Question 8

The plant is operating at rated power.

Which one of the following completes the statement below regarding the impact a complete loss of Instrument Air has on the RWCU system?

The RWCU Filter Demineralizer Effluent Flow Control Valves (RWCU-FCV-15A & B) fail ____(1)____ and the RWCU pump ____(2)____.

A. (1) OPEN
(2) trips

B. (1) OPEN
(2) remains running

C. (1) CLOSED
(2) trips

D. (1) CLOSED
(2) remains running

Answer:
C. (1) CLOSED
(2) trips

Explanation:
With the plant operating at rated power, RWCU system is in service with 2 F/Ds and 2 pump operating. As instrument air pressure lowers, the F/D Effluent FCVs fail closed. As system flow lowers below 50 gpm, the RWCU pumps trip.

Distracters:

A. This answer is incorrect due to the FCVs failing closed. This choice is plausible due to being easily confused with Condensate F/D effluent FCVs which do not fail closed. The candidate that is unsure of FCV fail position and correctly identifies RWCU pump trips would select this answer.

B. This answer is incorrect due to the FCVs failing closed and RWCU pumps tripping. This choice is plausible due to being easily confused with Condensate F/D effluent FCVs which do not fail closed and not recognizing the low flow condition. The candidate that is unsure of FCV fail position and does not identify RWCU pump trips would select this answer.

D. This answer is incorrect due to RWCU pumps tripping. This choice is plausible due to not recognizing the low flow condition. The candidate that is correctly identifies FCV fail position and does not identify RWCU pump trips would select this answer.

Technical References:
Procedure 2.2.66 (Reactor Water Cleanup), Rev. 106
Procedure 5.2AIR (Loss Of Instrument Air), Rev. 21

References to be provided to applicants during exam: NONE

Learning Objective: COR0011702001030J Describe the interrelationships between the Plant Air system and the following: Reactor Water Cleanup

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b) 7	
Level of Difficulty:	3	
SRO Only Justification:	N/A	

therefore, was not further pursued.

2. INTERLOCKS AND SETPOINTS

2.1 Any of following will cause RWCU pumps to trip:

2.1.1 RWCU-MO-15 or RWCU-MO-18 not full open.

2.1.2 High pump bearing cooling water (REC) temperature - 140°F.

2.1.3 Low RWCU flow rate - 50 gpm.

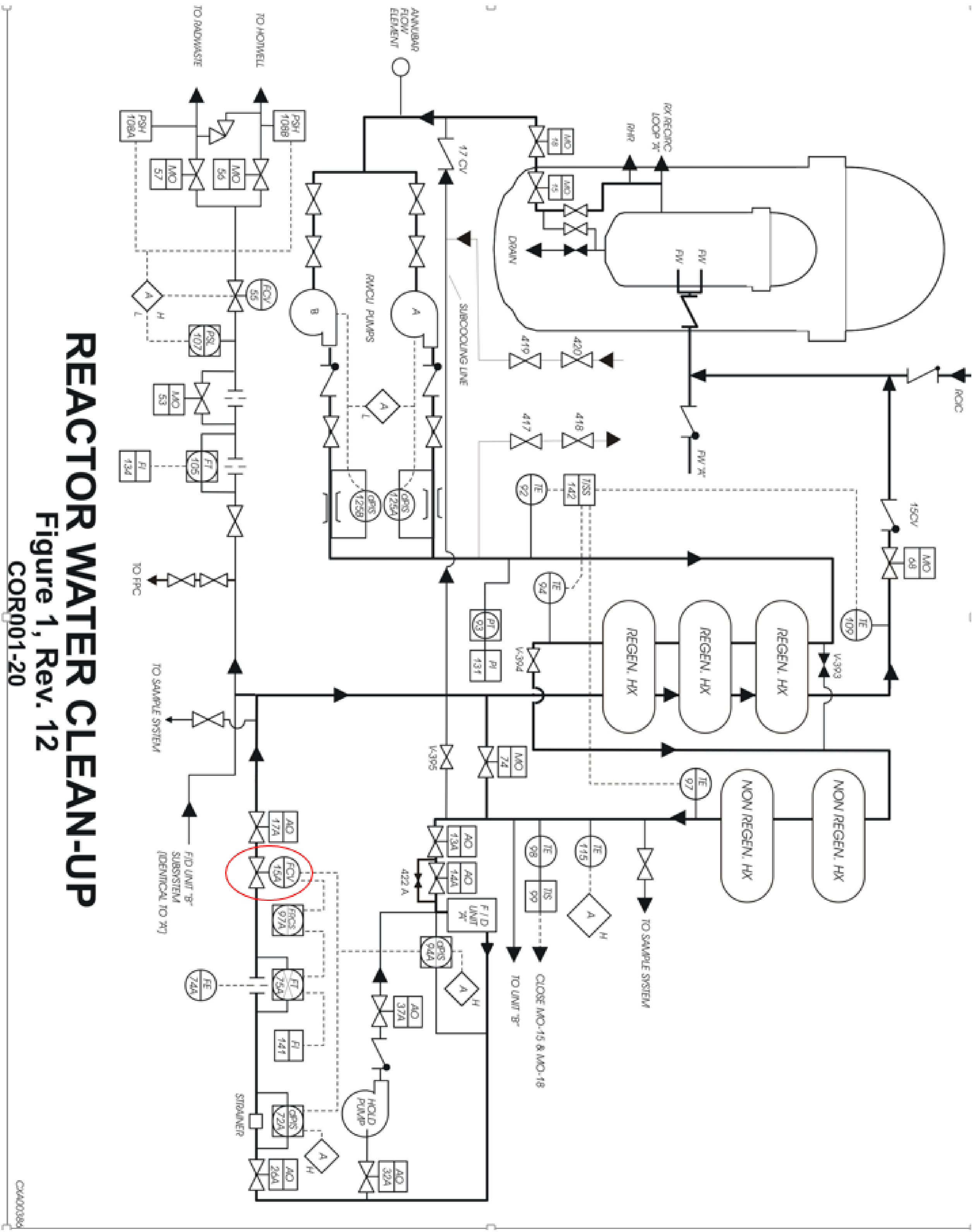
2.2 When starting a pump, switch must be held to START until system flow is > 50 gpm or pump will trip.

PROCEDURE 2.2.66	REVISION 106	PAGE 70 OF 74

ATTACHMENT 1 AFFECTED LOADS

LOADS	NOTES
PC-AO-243 and PC-AO-244, TORUS VAC RELIEF VLVs	Fail Open. Air loss does not affect operation of associated mechanical vacuum breakers (check valves).
REC-AOV-701AV, MG SET OIL HX INLET	Fail Open.
REC-AOV-710AV, NON-REGEN HX INLET	Fail Open.
REC-AOV-905AV, SAC 1C REC RETURN ISOLATION	Fail Closed.
REC-AOV-903AV, SAC 1B REC RETURN ISOLATION	Fail Open.
REC-AOV-901AV, SAC 1A REC RETURN ISOLATION	Fail Open.
REC-AOV-904AV, SAC 1C REC SUPPLY ISOLATION	Fail Closed.
REC-AOV-902AV, SAC 1B REC SUPPLY ISOLATION	Fail Open.
REC-AOV-900AV, SAC 1A REC SUPPLY ISOLATION	Fail Open.
RF-AOV-FCV11AA, RFP A STARTUP FLOW CONTROL VALVE	Fail As-Is.
RF-AOV-FCV11BB, RFP B STARTUP FLOW CONTROL VALVE	Fail As-Is.
RF-AOV-FCV11A, RFP A MINIMUM FLOW VALVE	Fail Open.
RF-AOV-FCV11B, RFP B MINIMUM FLOW VALVE	Fail Open.
RWCU-FCV-55	Fails closed, terminating normal RPV blowdown capabilities.
RWCU-AO-FCV15A	Fail closed and RWCU pump trips on low flow.
RWCU-AO-FCV15B	Fail closed and RWCU pump trips on low flow.
Secondary Containment Isolation Valves	Extended IA loss may cause valves inoperable. Accumulator tested to maintain pressure for 1 hour.
SGT-DPCV-546A, SGT A FLOW/RX BLDG DP CONTROL	Fail Open.
SGT-DPCV-546B, SGT B FLOW/RX BLDG DP CONTROL	Fail Open.
All Air Operated SGT Valves	Fail Open.
SLC Level Indication	Fails Low.
SW-AOV-451A/B, REC TCVs	Fail Open.
TEC-AOV-20AV, SAC A TEC SUPPLY ISO	Fail Closed.

PROCEDURE 5.2AIR REVISION 21 PAGE 7 OF 17



REACTOR WATER CLEAN-UP
Figure 1, Rev. 12
COR001-20

Examination Outline Cross-Reference	Level	RO
	Tier#	1
	Group#	1
	K/A #	295021 AK3.04
	Rating	3.3
295021 Loss of Shutdown Cooling / 4 AK3. Knowledge of the reasons for the following responses as they apply to LOSS OF SHUTDOWN COOLING : (CFR: 41.5 / 45.6) AK3.04 Maximizing reactor water cleanup flow		

Question 9

The Plant is in Mode 4 with RCS temperature at 175°F.

- RHR Loop B is in service in SDC with RHR Pump B in service.
- RHR Loop A is out of service for maintenance.
- Both Reactor Recirculation Pumps are out of service for maintenance.
- RWCU is in service with 1 pump and 2 filters.

RHR Pump B trips due to a motor failure.

Why is the RWCU System flow raised IAW Procedure 2.4SDC (Shutdown Cooling Abnormal) and 2.2.66 (Reactor Water Cleanup)?

A. To prevent flashing in the RWCU suction.

B. To raise core forced circulation inside the shroud.

C. To raise the reactor heat removal rate in the regenerative heat exchanger.

D. To raise the reactor heat removal rate in the non-regenerative heat exchanger.

Answer:
D. To raise the reactor heat removal rate in the non-regenerative heat exchanger.
Explanation: During a loss of shutdown cooling, with the vessel head on, the RWCU system is placed in service as the next preferred method of shutdown cooling. This lineup

bypasses the RWCU flow around the regenerative heat exchangers, then through MO-74 (Demin Suction Bypass) back through the regenerative heat exchanger to the reactor. Maximizing this flow will provide the greatest amount of cooling for the reactor water.

Distracters:

A. This answer is incorrect due to raising flow to raise heat removal rate within the NRHX. This choice is plausible due to the common misconception of RWCU losing NPSH while the reactor is depressurized. The candidate that confuses raising flow to prevent suction flashing would select this answer.

B. This answer is incorrect due to raising flow to raise heat removal rate within the NRHX. This choice is plausible due to the common misconception of RWCU having sufficient flow to provide forced circulation within the shroud. The candidate that confuses raising flow to provide forced circulation within the shroud would select this answer.

C. This answer is incorrect due to raising flow to raise heat removal rate within the NRHX. This choice is plausible due to the Regenerative HX being easily confused with the NRHX and heat removal rate within the Regenerative HX remaining the same. The candidate that confuses the Regenerative HX with the NRHX would select this answer.

Technical References:
Procedure 2.4SDC (Shutdown Cooling Abnormal), Rev. 14

References to be provided to applicants during exam: NONE

Learning Objective: COR001-20-01 OPS Reactor Water Cleanup
5. Briefly describe RWCU operation under the following conditions:
c. Loss of Shutdown Cooling

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b) 5	
Level of Difficulty:	2	
SRO Only Justification:	N/A	

1. CONTINGENCY ACTIONS FOR COMPLETE LOSS OF SDC

1.1 Commence monitoring plant heatup rate per Procedure 6.RCS.601. Ⓟ[8]

NOTE – Preferred level indication is NBI-LI-86, SHUTDOWN LVL. RFC-LI-94A, RFC-LI-94B, or RFC-LI-94C, RX NR LEVEL, may indicate up to 9" higher than actual during cold conditions.

1.2 Control RPV level > 48" to aid in thermal convection flow.

1.3 IF blade guides in RPV or fuel bundle removed from around core instrumentation, THEN Step 1.4 is N/A. Ⓟ[2]

1.4 Place or maintain one available RR pump in service per Procedure 2.2.68.

1.5 Place RWCU System in service per alternate heat removal section of Procedure 2.2.66. Ⓟ[8]

1.6 Review Attachment 5 (Page 15) using time to core boiling/uncovery figure for existing reactor cavity water level.

1.7 Monitor following temperatures and pressures frequently and log in Control Room Log every 4 hours: Ⓟ[4]

1.7.1 IF a RR pump is in service, THEN monitor RR-TI-151A(B), SUCT TEMP (PNL 9-4).

1.7.2 IF a RR pump is not in service, THEN monitor RPV metal temperatures on NBI-TR-89, REACTOR VESSEL METAL TEMPERATURE RECORDER (PNL 9-21), for approach to boiling.

1.7.3 IF RWCU is in service, THEN monitor inlet temperature on RWCU-TI-137, TEMP IND, using Point 3 on TEMP POINT SELECTOR (PNL 9-4).

1.7.4 Monitor following reactor pressure PMIS Points for indication of pressurization:

1.7.4.1 B025.

1.7.4.2 N013.

1.7.4.3 N014.

1.8 IF RPV head is off, THEN go to Step 1.12.

Examination Outline Cross-Reference	Level	RO
Added PMIS alarm point for low Fuel Pool level.	Tier#	1
	Group#	1
	K/A #	295023 AA1.02
	Rating	2.9

295023 Refueling Acc / 8

AA1. Ability to operate and/or monitor the following as they apply to REFUELING ACCIDENTS : (CFR: 41.7 / 45.6)

AA1.02 Fuel pool cooling and cleanup system

Question 10

While performing a full core off load in Mode 5, a seismic event cracks the Spent Fuel Pool (SFP) liner.

SFP level is lowering.

The following annunciators are received:

Annunciator	Panel/Window
FUEL POOL COOLING TROUBLE	PANEL/WINDOW: 9-4-2/A-3
SKIMMER TANK LOW LEVEL	PANEL/WINDOW: 9-4-2/C-3

The following PMIS point is in alarm:

- (1956) FUEL POOL LOW LEVEL

Which one of the following completes the statements below regarding the status of FPC pumps and the action(s) required?

The FPC pumps trip directly due to LOW ____(1)____ level.

Entry into ____(2)____ is/are required.

A. (1) Spent Fuel Pool
(2) 2.4FPC (Fuel Pool Cooling Trouble) ONLY

B. (1) Spent Fuel Pool

(2) 2.4FPC (Fuel Pool Cooling Trouble) **AND** 5.1RAD (Building Radiation Trouble)

C. (1) Skimmer Surge Tank
(2) 2.4FPC (Fuel Pool Cooling Trouble) ONLY

D. (1) Skimmer Surge Tank
(2) 2.4FPC (Fuel Pool Cooling Trouble) **AND** 5.1RAD (Building Radiation Trouble)

Answer:
D. (1) Skimmer Surge Tank (2) 2.4FPC (Fuel Pool Cooling Trouble) **AND** 5.1RAD (Building Radiation Trouble)
Explanation: Uncontrolled lowering of SFP level during refueling will cause FPC Skimmer Surge tank level to lower. When skimmer surge tank level reaches 50 ft^3 all FPC pumps trip. Loss of FPC pumps requires entry into 2.4FPC and abnormal (lowering) SFP level requires entry into 5.1RAD.
Distracters: A. This answer is incorrect due to FPC pumps tripping directly from Skimmer Surge Tank low level and both 2.4FPC & 5.1RAD required to be entered. This choice is plausible due to SFP level is commonly confused with Skimmer Surge tank level and lowering SFP level not being recognized as an entry condition to 5.1RAD. The candidate that confuses SFP vs. Skimmer Surge tank and does not recognize the 5.1RAD entry condition would select this answer. B. This answer is incorrect due to FPC pumps tripping directly from Skimmer Surge Tank low level. This choice is plausible due to SFP level is commonly confused with Skimmer Surge tank level. The candidate that confuses SFP vs. Skimmer Surge tank and correctly recognizes the 5.1RAD entry condition would select this answer. C. This answer is incorrect due to 5.1RAD required to be entered. This choice is plausible due to lowering SFP level not being recognized as an entry condition to 5.1RAD. The candidate that correctly identifies Skimmer Surge tank and does not recognize the 5.1RAD entry condition would select this answer.
Technical References: Procedure 2.3_9-4-2 (Panel 9-4 - Annunciator 9-4-2), Rev. 20 Procedure 2.4FPC (Fuel Pool Cooling Trouble), Rev. 32 Procedure 5.1RAD (Building Radiation Trouble), Rev. 17
References to be provided to applicants during exam: NONE
Learning Objective: COR0010602001090C Describe the FPC design features and/or interlocks that provide for the following: Pump protection

INT0320126L0L0100 Given plant condition(s), determine from memory the appropriate Abnormal/Emergency Procedure(s) to be utilized to mitigate the event(s).		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b) 7	
Level of Difficulty:	2	
SRO Only Justification:	N/A	

FUEL POOL COOLING TROUBLE	PANEL/WINDOW: 9-4-2/A-3

1. OPERATOR OBSERVATION AND ACTION

1.1 Check Annunciator Panel 25-15 for alarms.

1.1.1 IF alarm is due to FUEL POOL GATE SEAL LEAKAGE ALARM, THEN close FPC-36, FUEL STORAGE POOL GATE FIS-63 SHUTOFF (R-958-FPC HX Room), until fuel pool gates are properly seated.

1.2 IF Alarm (1998) FUEL POOL COOLING PUMP C DISCH PRESS LOW or (1999) FUEL POOL COOLING PUMP D DISCH PRESS LOW at 100 psig, THEN check Panel 25-16-3 or 25-16-4 for cause of loss of pump.

1.3 IF power lost from LPR1F, Breaker 28, locally monitor following due to loss of FPC-LS-60 and FPC-LS-65, THEN perform following:Ⓟ[1]

1.3.1 Fuel pool level.

1.3.2 Skimmer surge tank level.

1.4 IF FPC cooling or flow lost, THEN enter Procedure 2.4FPC.

1.5 IF Abnormal fuel pool water level exists, THEN enter Procedure 5.1RAD.

1.6 IF fuel pool gates are removed AND reactor cavity level is lowering unexpectedly, THEN perform following:Ⓒ[1]

1.6.1 Maintain reactor cavity level with CRD, Condensate, or CSCS per applicable operating procedure.

NOTE – If cause of unexpected level drop cannot be corrected, source of makeup water may be depleted and result in loss of fuel pool and cavity level. To ensure that irradiated fuel bundles remain at least partially covered, any fuel being handled needs to be as low as possible in fuel pool or reactor cavity.

1.6.2 IF cause of unexpected level drop cannot be corrected, THEN perform following:

1.6.2.1 IF refuel floor radiation levels permit, THEN ensure that fuel prep machine(s) containing a fuel bundle are in the fully lowered position.

1.6.2.2 IF refuel floor radiation levels permit, THEN ensure that any irradiated fuel bundles being moved are lowered as low as possible in reactor vessel or fuel pool as soon as possible.

(continued)

SETPOINT	CIC	9-4-2/A-3
1. (1950) FUEL POOL REFUELING BELLOWS LEAKAGE HIGH at 5 gpm	1. FPC-FIS-61	
2. (1951) FUEL POOL GATE SEAL LEAKAGE ALARM at 5 gpm	2. FPC-FIS-63	
3. (1952) FUEL POOL RX WELL BELLOWS OR LINER LEAKAGE at 5 gpm	3. FPC-FIS-64	
4. (1953) FUEL POOL COOLING SYSTEM PRESSURE HIGH at 145 psig	4. FPC-PS-73	
5. (1954) FUEL POOL COOLING PUMP B DISCH PRESS LOW at 100 psig	5. FPC-PIS-69B	
6. (1955) FUEL POOL COOLING PUMP A DISCH PRESS LOW at 100 psig	6. FPC-PIS-69A	
7. (1956) FUEL POOL LOW LEVEL at 4" below normal	7. FPC-LS-60B	
8. (1957) FUEL POOL HIGH LEVEL at 3" above normal	8. FPC-LS-60A	
9. (1998) FUEL POOL COOLING PUMP C DISCH PRESS LOW at 100 psig	9. FPC-PIS-69C	
10. (1999) FUEL POOL COOLING PUMP D DISCH PRESS LOW at 100 psig	10. FPC-PIS-69D	

SKIMMER TANK LOW LEVEL	PANEL/WINDOW: 9-4-2/C-3

1. AUTOMATIC ACTIONS

   1.1 Fuel pool pumps trips on either of following:

      1.1.1 50 ft^3 in skimmer surge tank.

      1.1.2 Low pump suction pressure -10' H_2O.

2. OPERATOR OBSERVATION AND ACTION

   2.1 Check FPC-LI-70, SKIMMER SURGE TANK LEVEL GAUGE, to determine rate of level loss (R-976-E).

   2.2 Fill skimmer surge tank per Procedure 2.2.32.

   2.3 IF fuel pool cooling pump tripped on low skimmer surge tank level, THEN perform following:

   CAUTION – Water from FPC-227 is radioactively contaminated.

      2.3.1 Open FPC-227, LI-70 CALIBRATION, until air free water flows (R-976-E).

      2.3.2 Contact Rad Protection for any water spills.

      2.3.3 Return FPC to service per Procedure 2.2.32.

   2.4 IF unable to restore FPC flow or maintain skimmer surge tank level with normal makeup, THEN enter Procedure 2.4FPC.

CNS OPERATIONS MANUAL ABNORMAL PROCEDURE 2.4FPC FUEL POOL COOLING TROUBLE	USE: CONTINUOUS QUALITY: QAPD RELATED EFFECTIVE: 12/22/14 APPROVAL: ITR-RDM OWNER: OSG SUPV DEPARTMENT: OPS

1. ENTRY CONDITIONS

1.1 Fuel pool water temperature ≥ 125°F.

1.2 Loss of all FPC pumps.

1.3 Loss of cooling flow to FPC heat exchangers.

2. AUTOMATIC ACTIONS

2.1 Fuel Pool Cooling Pumps A through D trip on either of following:

2.1.1 Low suction pressure of -10' H_2O.

2.1.2 Surge tank low level of 50 cu ft.

2.2 Filter demineralizer goes into hold when flow < 120 gpm.

3. IMMEDIATE OPERATOR ACTIONS

3.1 None.

4. SUBSEQUENT OPERATOR ACTIONS

4.1 Record current time and date. Time/Date: ________ / __________

CAUTION – Fuel Pool - Emergency Makeup may require dispatching personnel through or into potentially high radiation areas.

4.2 Refer to Attachment 7 when dispatching personnel to perform actions outside Control Room.Ⓟ[6]

4.3 Concurrently perform applicable Attachments:

LOSS OF FUEL POOL COOLING FLOW	ATTACHMENT 1	Page 3
LOSS OF COOLING TO FPC HX	ATTACHMENT 2	Page 9
FUEL POOL - EMERGENCY MAKEUP	ATTACHMENT 3	Page 13

4.4 IF Secondary Containment integrity not established, THEN initiate actions to restore Secondary Containment integrity prior to exceeding time-to-200°F.Ⓟ[2]

CNS OPERATIONS MANUAL EMERGENCY PROCEDURE 5.1RAD BUILDING RADIATION TROUBLE	USE: CONTINUOUS QUALITY: QAPD RELATED EFFECTIVE: 5/13/15 APPROVAL: ITR-RDM OWNER: OSG SUPV DEPARTMENT: OPS

1. ENTRY CONDITIONS℗[3]

1.1 High or unusual readings on any ARM or ARM recorder.

1.2 Local area radiation alarms.

1.3 Portable radiation monitoring units indicate abnormally high readings.

1.4 High or unusual readings on the Reactor Building Ventilation monitors/recorders.

1.5 High or unusual readings on any building Kamans.

1.6 Local Continuous Air Monitor alarms.

1.7 Abnormal fuel pool water level.

2. AUTOMATIC ACTIONS

2.1 None.

3. IMMEDIATE OPERATOR ACTIONS

3.1 None.

4. SUBSEQUENT OPERATOR ACTIONS

4.1 Record current time and date. Time/Date: ________/__________

NOTE 1 – Following scram, affected areas for RMA-RA-9, CRD HYDRAULIC EQUIP AREA (NORTH), include R-903-N, R-903-S, and R-931-SW.

NOTE 2 – Area Radiation Monitor alarms on refueling floor may be received when removing a Transfer Cask (TC) from spent fuel pool and Continuous Air Monitor alarms on refuel floor area may be received during vacuum drying of a Dry Shielded Canister (DSC).

4.2 Notify Plant personnel to clear affected area via gaitronics.

4.3 IF high radiation on refueling floor, THEN perform Attachment 1 (Page 3).℗[3]

4.4 IF radiation due to known system leakage outside Secondary Containment, THEN enter Procedure 5.1BREAK.

4.5 Close all possible doors or barriers.

Examination Outline Cross-Reference	Level	RO
Revised question to identify the LOWEST DW pressure which exceeds PCPL and the component of concern if PCPL is exceeded to maintain K/A focus. Revised question to identify the LOWEST DW pressure which exceeds PCPL-A and the component of concern if PCPL-A is exceeded being SRV operability. Requires determining the lowest DW pressure which exceeds PCPL with torus level in the normal range and determining the component of concern if limit is exceeded. Comparing provided values to the actual setpoint for normal & high torus level requires recall of acquired knowledge and associating two or more pieces of data which is HCL.	Tier#	1
	Group#	1
	K/A #	295024 EK1.01
	Rating	4.1

295024 High Drywell Pressure / 5

EK1. Knowledge of the operational implications of the following concepts as they apply to HIGH DRYWELL PRESSURE : (CFR: 41.8 to 41.10)

EK1.01 Drywell integrity: Plant-Specific.

Question 11

Primary Containment water level is rising during an accident.

Which one of the following identifies:

(1) the LOWEST Drywell pressure which exceeds Primary Containment Pressure Limit (PCPL-A)

AND

(2) the component of concern if this limit is exceeded?

A. (1) 63 psig
(2) SRV operability

B. (1) 63 psig
(2) RPV Vent valve operability

C. (1) 57 psig
(2) SRV operability

D. (1) 57 psig
(2) RPV Vent valve operability

Answer:
C. (1) 57 psig

(2) SRV operability

Explanation:
Primary Containment pressure is limited to 62.7 psig to protect containment integrity with normal torus level. With PC water level rising, PCPL-A becomes limiting at 56.8 psig which ensures SRV operability (SRVs open and remain open) with PC water level below 100 feet.

Distracters:

A. This answer is incorrect due to 57 psig exceeding Primary Containment Pressure limit A. This choice is plausible due to 62.7 psig being the limit with normal torus water level. The candidate that confuses the drywell pressure at normal vs. high torus level and correctly identifies the impacted component would select this answer.
B. This answer is incorrect due to 57 psig exceeding Primary Containment Pressure limit A and SRV operability being challenged. This choice is plausible due to 62.7 psig being the limit with normal torus water level and RPV vent valve operability being considered for the limit but not impacting the limit due to the valves being motor operated. The candidate that confuses the drywell pressure at normal vs. high torus level and confuses the impacted component would select this answer.
D. This answer is incorrect due to SRV operability being challenged. This choice is plausible due to RPV vent valve operability being considered for the limit but not impacting the limit due to the valves being motor operated. The candidate that correctly identifies the limiting pressure and confuses the impact would select this answer.

Technical References:
AMP-TBD00 (CNS PSTG/SATG Appendix B Technical Bases), Rev. 8
Procedure 5.8 Attachment 2 (EOP and SAG Graphs), Rev. 15

References to be provided to applicants during exam: NONE

Learning Objective: COR0020302003 State the design bases for the Containment system as described in the associated Student Text.
a. Primary Containment
5. Describe the interrelationship between the Primary Containment system and the following:
b. Drywell isolation/integrity

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b) 9	

Level of Difficulty:	3
SRO Only Justification:	N/A

16.25 Primary Containment Pressure Limits A, B and C

Primary Containment Pressure Limits A, B and C (PCPL-A, PCPL-B, and PCPL-C) are the lesser of:

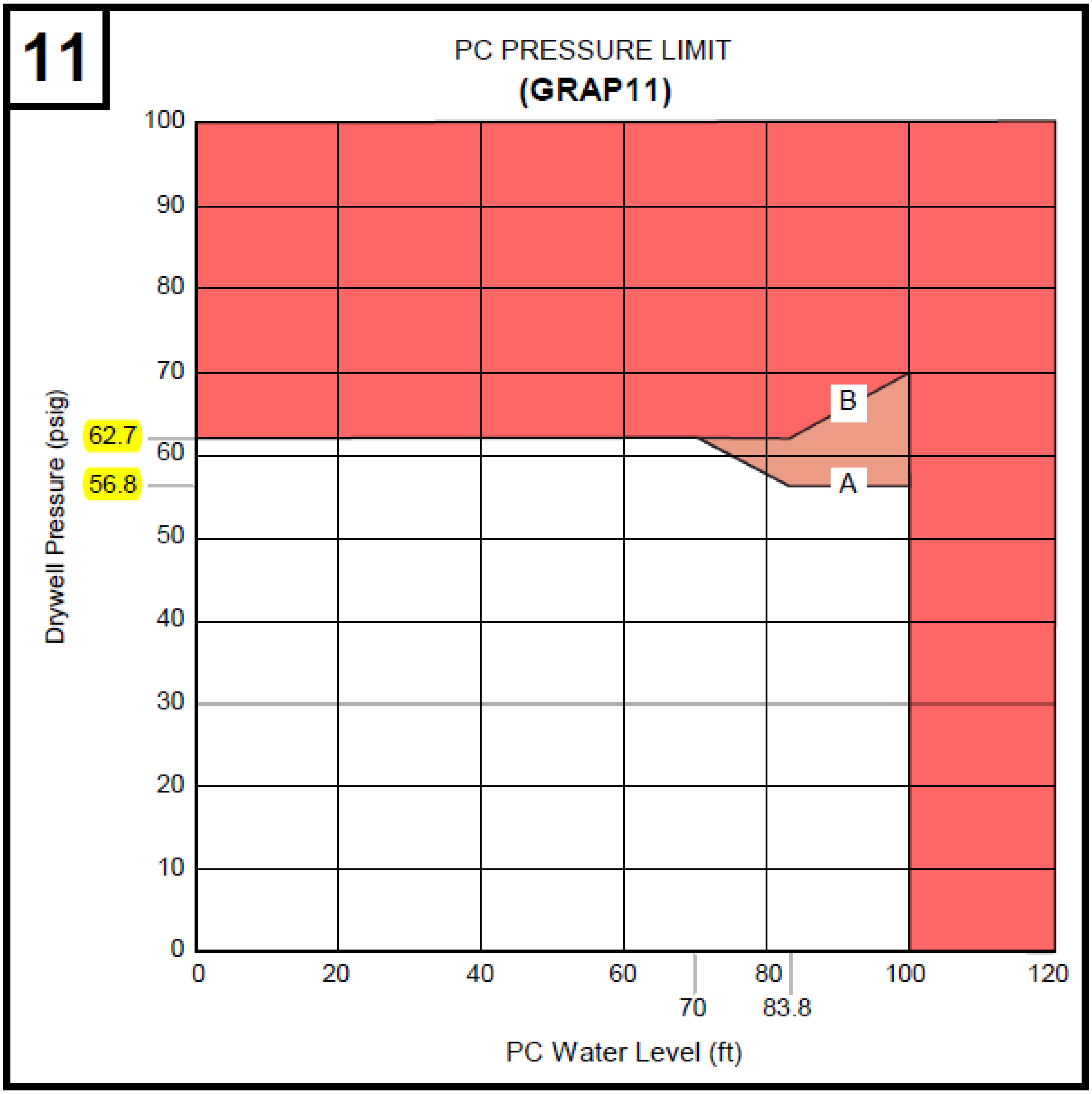
- The pressure capability of the primary containment.
- The maximum primary containment pressure at which vent valves sized to reject all decay heat from the containment can be opened and closed.
- For Primary Containment Pressure Limit A, the maximum primary containment pressure at which SRVs can be opened and will remain open.
- For Primary Containment Pressure Limits A and B, the maximum primary containment pressure at which RPV vent valves can be opened and closed.

Each PCPL is a function of primary containment water level and primary containment temperature. The limits are utilized to avoid challenges to primary containment vent valve operability, SRV operability, RPV vent valve operability, and primary containment integrity. At CNS, RPV vent valve operability is not a concern in derivation of the PCPL because RPV venting can be accomplished using the motor operated main steam line drain valves. Operability of these valves are not affected by containment atmospheric pressure.

The derivation of the PCPL is shown graphically in Figures B-16-21 and B-16-22. Figure B-16-23 plots limiting suppression chamber airspace pressures against primary containment water level:

- Lines "DW hd bolts" and "Torus mid" are the pressure capabilities of limiting primary containment components.
- Line "PCvent" is the maximum suppression chamber pressure at which the containment vent valves can be operated.
- Line "SRV" is the maximum suppression chamber pressure at which the SRVs can be opened and will remain open.
- The vertical line at 103.42 ft is the elevation of the highest containment vent capable of removing all decay heat.

The primary containment vent is slightly more limiting than SRV operability at low primary containment water levels. At higher water levels, SRV operability is limiting.


11
PC PRESSURE LIMIT
(GRAP11)
Drywell Pressure (psig)
100
90
80
70
62.7
60
56.8
50
40
30
20
10
0
B
A
0
20
40
60
80
100
120
70
83.8
PC Water Level (ft)

Each airspace pressure limit in Figure B-16-21 is constant while primary containment water level is below the elevation of the respective component. Once a component is submerged, however, hydrostatic head adds to the pressure exerted on the component (Line "H2O head") and the allowable airspace pressure must be derated as water level increases. The elevation of each component thus defines the break point on its associated curve. Line PCvent does not show a break point since the vent penetration is at the maximum elevation considered.

Figure B-16-22 illustrates the indicated pressure curves obtained from the airspace pressure curves of Figure B-16-21. When water level is at or above the elevation of the suppression chamber pressure instrument tap, indicated pressure senses hydrostatic head as well as the airspace pressure; thus, the envelope defined by Line "PCvent" and Line "SRV" is constant until the pressure tap is covered, and increases until the elevation of the highest primary containment vent capable of removing decay heat is reached. (Higher primary containment water levels are not permitted in the PSTG/SATG.)

PCPL-A, PCPL-B and PCPL-C are determined assuming:

1. The decay heat rejected is that which is generated ten minutes after shutdown from rated power.
2. The temperature in the drywell may vary between 100°F and 545°F.
3. The temperature in the suppression chamber may vary between 70°F and 350°F.

CNS input data required to calculate the PCPLs are:

1. The maximum containment pressure at which SRVs can be opened and will remain opened (PCPL-A only).
2. Elevations, limiting pressures and associated temperatures, and limiting materials of all potentially limiting locations in the containment (calculated stresses not exceeding ASME Level D allowable stresses).
3. Elevation of the suppression chamber pressure instrument tap.
4. Elevation of the highest containment vent capable of rejecting all decay heat.
5. Elevation of the lowest SRV pneumatic solenoid (for PCPL-A only).
6. Minimum temperature of the suppression pool when the containment is flooded.

Examination Outline Cross-Reference	Level	RO
Is a BANK question	Tier#	1
	Group#	1
	K/A #	295025 EK2.09
	Rating	3.9

295025 High Reactor Pressure / 3

EK2. Knowledge of the interrelations between HIGH REACTOR PRESSURE and the following: (CFR: 41.7 / 45.8)

EK2.09 Reactor power

Question 12

The plant is operating at rated power when an outboard MSIV disc-stem separation occurs.

Which one of the following is the FIRST to automatically Scram the reactor due to this failure, as documented in actual CNS operating experience?

A. MSIV closure.

B. APRM High Flux.

C. Low RPV water level.

D. High Reactor Pressure.

Answer:
B. APRM High Flux Scram.
Explanation: MSIV stem disc separation results in the MSIV disc rapidly closing causing a high pressure transient collapsing core steam bubbles and a spike in reactor power. This power spike is seen by all the APRMS and a reactor Scram on high flux results. The outboard MSIV's are located in the steam tunnel further down the steam lines than the inboard MSIVs so the pressure perturbation is less than an inboard MSIV stem disc separation.
Distracters: A. This answer is incorrect because APRM High Flux signal automatically Scrams the

reactor. The steam flow through the other steam lines does not appreciably rise due to the Scram. This answer is plausible due to closing one MSIV at rated power causing steam flow in the remaining steam lines to rise. The candidate that believes the rise in flow through the other Main Steam Lines will cause a Group 1 isolation would select this answer.

C. This answer is incorrect because the APRM High Flux signal automatically Scrams the reactor. The RPV water level does not lower enough to cause a low level Scram signal to be generated. This answer is plausible because RPV low level is normally received following a reactor Scram The candidate confuses the sequence of events would select this answer..

D. This answer is incorrect because the APRM High Flux signal automatically Scrams the reactor. The initial pressure spike does not the Scram setpoint (1034 psig). This answer is plausible because rising pressure results from an MSIV stem disc separation. The candidate that confuses the sequence of events would select this answer.

Technical References:
LER 89-001, Unplanned automatic Scram due to APRM high flux resulting from separation of an MSIV disc from its stem

References to be provided to applicants during exam: NONE

Learning Objective:
COR0021402001060D, Given a specific Main Steam system malfunction, determine the effect on any of the following: Reactor power

Question Source:	Bank # X	
(note changes; attach parent)	Modified Bank #	
	New	
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b) 7	
Level of Difficulty:	3	
SRO Only Justification:	N/A	

295025 High Reactor Pressure

EK2. Knowledge of the interrelations between HIGH REACTOR PRESSURE and the following:

EK2.09 Reactor Power

Question: 12

The plant is operating at 100% power when an MSIV disc-stem separation occurs on an outboard MSIV.

How does the plant respond in the next 30 seconds?

a. Group 1 isolation.

b. APRM High Flux scram.

c. Low RPV water level scram.

d. SRV opens on spring pressure.

Answer:

b. APRM High Flux scram.

NON-PUBLIC?: N
ACCESSION #: 8903030463
LICENSEE EVENT REPORT (LER)

FACILITY NAME: Cooper Nuclear Station PAGE: 1 OF 5

DOCKET NUMBER: 05000298

TITLE: Unplanned Automatic Scram Due to APRM High Flux Resulting from Separation Of An MSIV Disc From Its Stem
EVENT DATE: 01/25/89 LER #: 89-001-00 REPORT DATE: 02/24/89

OPERATING MODE: N POWER LEVEL: 100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
NAME: Donald L. Reeves, Jr. TELEPHONE: 402-825-3811

COMPONENT FAILURE DESCRIPTION:
CAUSE: X SYSTEM: SB COMPONENT: ISV MANUFACTURER: R344
REPORTABLE TO NPRDS: Y

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

On January 25, 1989, at 6:53 A.M., an automatic reactor scram due to high neutron flux occurred while at 100 percent power under normal steady state conditions. The ensuing Reactor Pressure Vessel water level transient resulted in actuation of Groups 2, 3, and 6 Isolations (Primary Containment, Reactor Water Cleanup, and Secondary Containment including Standby Gas Treatment System initiation). Water level was immediately restored and maintained by the Condensate/Feedwater System; no automatic or manual ECCS System actuations were required.

It was initially theorized, based on available plant data, that the neutron flux transient was due either to electronic noise in the Neutron Monitoring System or a pressure spike as a result of a main turbine pressure control system malfunction. Subsequently, problems were experienced when attempting to equalize through the inboard Main Steam Isolation Valve (MSIV) on the "A" Main Steam Line. Upon disassembly of the inboard valve, it was determined that the stem disc had separated from the stem during operation, and that the main disc seated, causing a pressure spike and the resulting flux transient.

Examination Outline Cross-Reference	**Level**	RO
Revised question to eliminate reference and asked when ED is required and why if HCTL is exceeded. Requires determining when ED is required in relation to HCTL and why. Comparing various other ED requirements which are required if a parameter is exceeded, prior to exceeding or cannot restored and maintained below along with eliciting a mental demand that requires a "why" response such that the examinee must derive the correct explanation tests at the comprehension level. Both aspects of question are required RO knowledge IAW INT0080613 (OPS EOP Flowchart 3A - Primary Containment Control).	**Tier#**	1
	Group#	1
	K/A #	295026 EK3.01
	Rating	3.8

295026 Suppression Pool High Water Temp. / 5

EK3. Knowledge of the reasons for the following responses as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE: (CFR: 41.5 / 45.6)

EK3.01 Emergency/normal depressurization

Question 13

Which one of the following completes the statement below regarding when and why Emergency Depressurization of the RPV is required?

The reactor is required to be Emergency Depressurized ____(1)____ HCTL is exceeded to ensure the reactor is depressurized prior to exceeding ____(2)____.

A. (1) before
(2) the Torus design temperature

B. (1) before
(2) the Primary Containment Pressure Limit

C. (1) ONLY when
(2) the Torus design temperature

D. (1) ONLY when
(2) the Primary Containment Pressure Limit

Answer:
D. (1) ONLY when (2) the Primary Containment Pressure Limit
Explanation:

When average Torus temperature and RPV Pressure cannot be maintained within the limits of the HCTL Graph – ED is required. The Heat Capacity Temperature Limit (HCTL) is the highest suppression pool temperature from which emergency RPV depressurization will not raise:

- Suppression chamber temperature above the maximum temperature capability of the suppression chamber, or
- Suppression chamber pressure above Primary Containment Pressure Limit A, while the rate of energy transfer from the RPV to the containment is greater than the capacity of the containment vent.

Distracters:

A. This answer is incorrect due to ED required only if HCTL is exceeded and to prevent exceeding PCPL. This choice is plausible due to other EOP steps providing guidance to ED prior to reaching a value (before offsite gaseous radioactivity release rate reaches that which requires a General Emergency) and the torus design temperature being easily confused with the temperature capability of the torus (which is variable based upon RPV pressure, Torus level, and initial Torus temperature). The candidate that confuses when ED is required and the reason for ED would select this answer.

B. This answer is incorrect due to ED required only if HCTL is exceeded. This choice is plausible due to other EOP steps providing guidance to ED prior to reaching a value (before offsite gaseous radioactivity release rate reaches that which requires a General Emergency). The candidate that confuses when ED is required and correctly identifies the reason for ED would select this answer.

C. This answer is incorrect due to ED required to prevent exceeding PCPL. This choice is plausible due to the torus design temperature being easily confused with the temperature capability of the torus (which is variable bases upon RPV pressure, Torus level, and initial Torus temperature). The candidate that correctly identifies when ED is required and confuses the reason for ED would select this answer.

Technical References:
AMP-TBD00 (CNS PSTG/SATG Appendix B Technical Bases), Rev. 8
EOP 3A (PCCP), Rev. 15
HCTL (Graph 7)

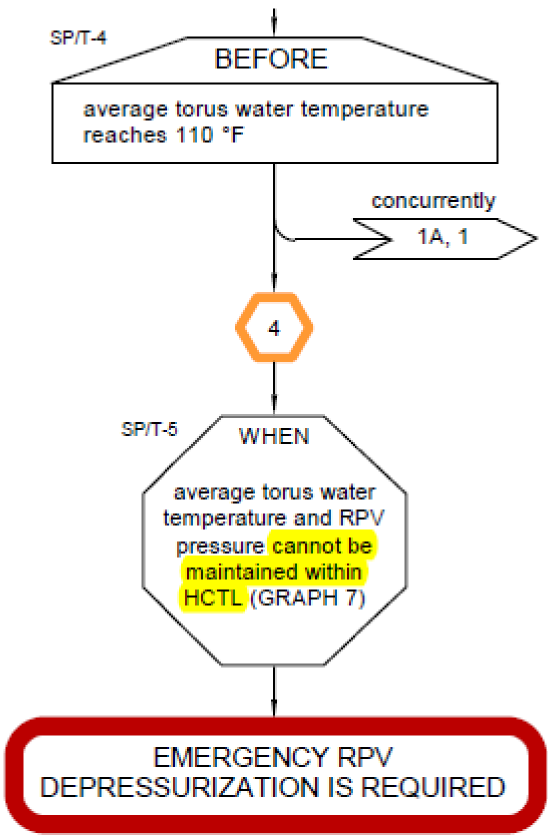
References to be provided to applicants during exam: NONE

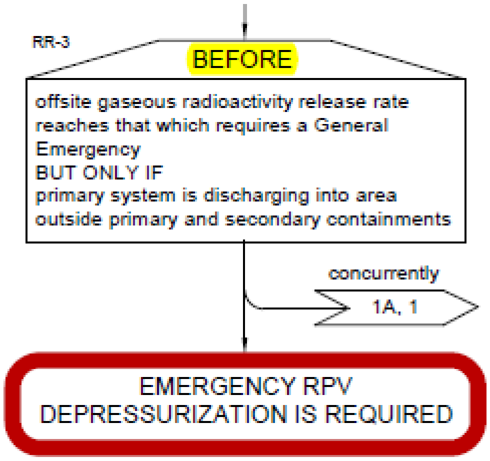
Learning Objective:

INT00806180010300 Given plant conditions and the EOP and SAG Graphs Flowchart, determine if operation is within the allowed region of a graph.
INT00806180020400 Using the Cautions provided in the EOP and SAG Flowcharts, explain the bases behind each of the Cautions.
INT0080613001040B State the basis for primary containment control actions as they apply to the following: Primary Containment Control Systems
INT0080613001040C State the basis for primary containment control actions as they apply to the following: Graphs reference on Flowchart 3A

Question Source:	Bank #	

(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b) 5	
Level of Difficulty:	3	
SRO Only Justification:	N/A	




RR-3
BEFORE
offsite gaseous radioactivity release rate reaches that which requires a General Emergency
BUT ONLY IF
primary system is discharging into area outside primary and secondary containments
concurrently
1A, 1
EMERGENCY RPV DEPRESSURIZATION IS REQUIRED

PSTG/SATG Step

SP/T-3 When average suppression pool temperature and RPV pressure cannot be maintained below the Heat Capacity Temperature Limit, EMERGENCY RPV DEPRESSURIZATION IS REQUIRED.

Discussion

The CNS Heat Capacity Temperature Limit (HCTL) is the highest suppression pool temperature from which emergency RPV depressurization will not raise:

- Suppression chamber temperature above maximum temperature capability of the suppression chamber, or
- Suppression chamber pressure above Primary Containment Pressure Limit A,

while the rate of energy transfer from the RPV to the containment is greater than the capacity of the containment vent. Depressurizing the RPV when suppression pool temperature and RPV pressure cannot be maintained below the HCTL thus avoids failure of the containment and equipment necessary for the safe shutdown of the plant. Refer to Section 16 of this appendix for a detailed discussion of the HCTL.

Control of RPV pressure relative to the HCTL is directed in the RPV Control guideline, entry to which is specified in Step SP/T-2. Therefore, the structure of the PSTG provides for controlling both suppression pool temperature and RPV pressure before reaching a requirement to initiate emergency RPV depressurization based on plant conditions relative to the HCTL.

Examination Outline Cross-Reference	Level	RO
Comment incorporated – changed second part to can CAN be used for trending, only if above Minimum Indicated Level (MIL) and CANNOT be used for trending, regardless of indicated level.	Tier#	1
	Group#	1
	K/A #	295028 EA2.03
	Rating	3.7

295028 High Drywell Temperature / 5

EA2. Ability to determine and/or interpret the following as they apply to HIGH DRYWELL TEMPERATURE : (CFR: 41.10 / 43.5 / 45.13)

EA2.03 Reactor water level

Question 14

A LOCA has occurred.

Which one of the following completes the statements below regarding Wide Range RPV water level indication if DW Temperature is in the UNSAFE region of the RPV Saturation Temperature curve?

(Assume NO indications of instrument leg boiling has been observed.)

High Drywell temperature causes INDICATED RPV Water level to be ____(1)____ than ACTUAL level.

Wide Range RPV water level indications ____(2)____.

A. (1) lower
(2) CAN be used for trending, only if above Minimum Indicated Level (MIL).

B. (1) lower
(2) CANNOT be used for trending, regardless of indicated level.

C. (1) higher
(2) CAN be used for trending, only if above Minimum Indicated Level (MIL).

D. (1) higher
(2) CANNOT be used for trending, regardless of indicated level.

Answer:
C. (1) higher (2) CAN be used for trending, only if above Minimum Indicated Level (MIL).

Explanation:
If Average DW Temperature is in the Unsafe region of the RPV Saturation Temperature curve, reference leg boiling may occur. Although unsafe, level instruments remain available for trending as long as Minimum Indicated level is within the safe region. With no indication provided of reference leg flashing (erratic indication) indications are still available for trending.

Distracters:

A. This answer is incorrect due to indicated level being higher than actual level. This choice is plausible due to commonly confusing reference & variable leg impacts from high drywell temperature. The candidate that confuses indicated level and correctly determines WR indication can be used for trending when above MIL would select this answer.

B. This answer is incorrect due to indicated level being higher than actual level and WR indication being available for trending above MIL. This choice is plausible due to commonly confusing reference & variable leg impacts from high drywell temperature and misunderstanding being in the unsafe region of the RPV Saturation Temperature curve of reference leg boiling occurring making a level instrument unavailable for trending. The candidate that confuses indicated level being lower and incorrectly determines trending is unavailable would select this answer.

D. This answer is incorrect due to WR indication being available for trending above MIL. This choice is plausible due to the common misunderstanding of being in the unsafe region of the RPV Saturation Temperature curve making a level instrument unavailable for trending The candidate that correctly identifies indicated being higher than actual level and misinterprets being unsafe on the RPV Saturation Temperature curve is boiling occurring in the reference leg making WR indication unavailable for trending would select this answer.

Technical References:
AMP-TBD00 (CNS PSTG/SATG Appendix B Technical Bases), Rev. 8
EOP Caution 1 Graphs

References to be provided to applicants during exam: NONE

Learning Objective:

INT00806090010200 Given plant conditions, determine if any limitations exist on RPV water level indicators by Maximum RUN Temperature (MRT) and Minimum Indicated Level (MIL).

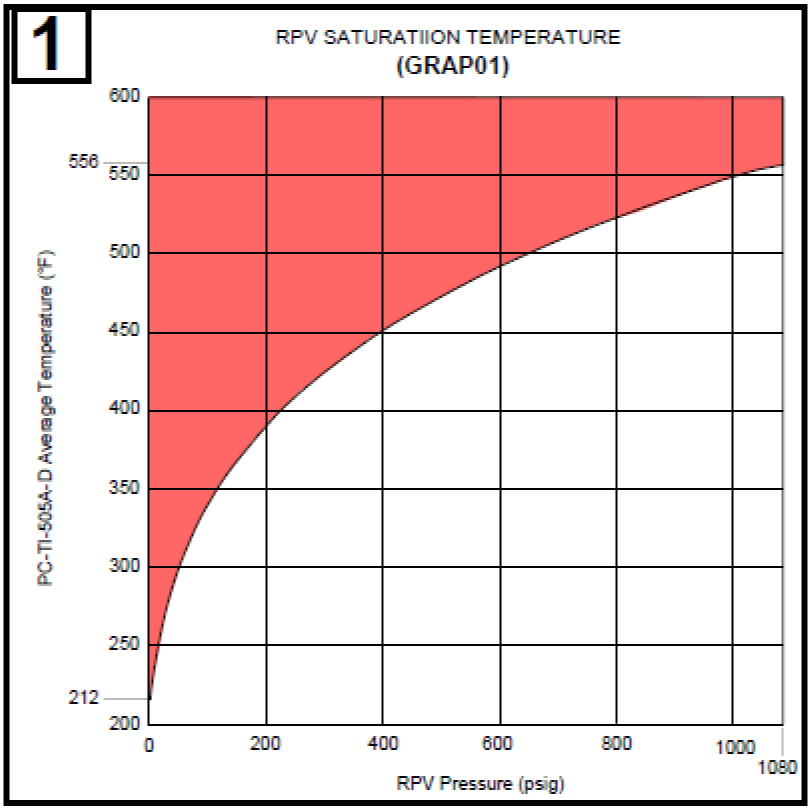
INT00806180010200 For each graph used in the flowcharts, identify the action(s) required if the parameters associated indicate operation in the restricted or prohibited area.

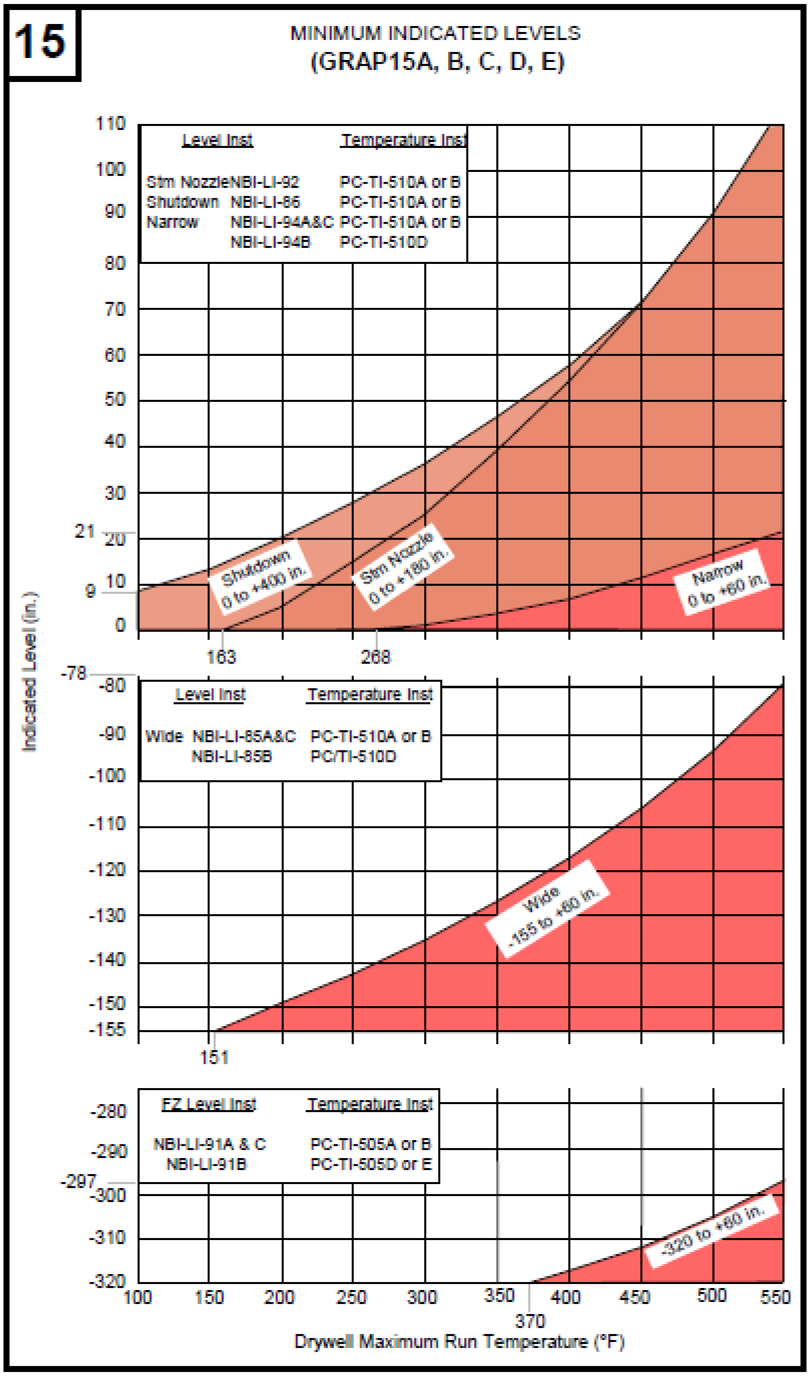
INT00806090011300 Given plant conditions, assess if RPV water level can be determined or not.

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X

Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b) 10	
Level of Difficulty:	4	
SRO Only Justification:	N/A	

1	RPV water level indications are affected by instrument run temperatures and RPV pressure: • If PC-TI-505A-D temperatures near any instrument run are above RPV Saturation Temperature (GRAPH 1), the instrument may be unreliable due to boiling in the run • Water level indication must read above Minimum Indicated Level for instrument (GRAPH 15) to be usable (apply level correction (GRAPH 14) for RPV pressure at or above 200 psig)
2	Elevated suppression chamber pressure may trip RCIC turbine on high exhaust pressure
3	Operation of HPCI, RCIC, CS, or RHR with suction from suppression pool and pump flow above NPSH or Vortex Limit (GRAPHs 3, 4, 5, 16, 18) may result in equipment damage
4	Operation of HPCI or RCIC turbines with suction temperatures above 140°F may result in equipment damage


1
RPV SATURATIION TEMPERATURE
(GRAP01)
PC-TI-505A-D Average Temperature (°F)
600
556
550
500
450
400
350
300
250
212
200
0
200
400
600
800
1000
1080
RPV Pressure (psig)


15
MINIMUM INDICATED LEVELS
(GRAP15A, B, C, D, E)
Level Inst
Temperature Inst
Stm Nozzle NBI-LI-92 PC-TI-510A or B
Shutdown NBI-LI-86 PC-TI-510A or B
Narrow NBI-LI-94A&C PC-TI-510A or B
NBI-LI-94B PC-TI-510D
Shutdown 0 to +400 in.
Stm Nozzle 0 to +180 in.
Narrow 0 to +60 in.
Wide NBI-LI-85A&C PC-TI-510A or B
NBI-LI-85B PC/TI-510D
Wide -155 to +60 in.
FZ Level Inst
NBI-LI-91A & C PC-TI-505A or B
NBI-LI-91B PC-TI-505D or E
-320 to +60 in.
Indicated Level (in.)
Drywell Maximum Run Temperature (°F)
163
268
151
370
-78
-297
21
9

16.9 Maximum Run Temperature

The Maximum Run Temperature (MRT) for an RPV water level instrument is the instrument run temperature that would produce an on-scale indication with actual water level at the variable leg RPV tap. The MRT is utilized in establishing the conditions under which an RPV water level instrument may be used to determine RPV water level. Separate temperatures are provided for each RPV water level instrument.

CNS RPV water level instruments sense level by measuring the differential pressure (dP) between a variable leg water column and a reference leg water column (Figure B-16-8). The reference leg is kept full of water by a condensing pot; the variable leg height depends upon RPV water level. When the actual RPV water level decreases, the variable leg height also decreases, causing the sensed dP to increase. The higher dP results in a lower indicated level. Conversely, when the actual RPV water level rises, the variable leg height increases, the dP decreases, and the indicated level increases.

The level instruments are calibrated to provide accurate indication under expected operating conditions. The indicated level will be inaccurate if primary containment temperature, secondary containment temperature, or RPV pressure varies from its assumed value. An elevated primary containment temperature, for example, will decrease the density of water in the instrument runs and cause a corresponding change in indicated level. The direction and magnitude of the change depend upon the arrangement of the instrument runs. If the variable leg vertical run in an area is longer than the reference leg vertical run, the indicated level will tend to decrease as temperature in the area rises. Conversely, if the reference leg vertical run is longer, indicated level will tend to increase. The greater the relative difference in the vertical runs, the greater the change in indicated level. CNS RPV water level instruments are not affected by variation in secondary containment temperatures near the instrument runs because the vertical length of variable leg run and reference leg run are the same.

Changes in primary containment temperatures can produce on-scale readings on some instruments even when the actual level is below their variable leg taps. Since dP is not affected by level changes below the variable leg tap, the indicated level would then no longer reflect changes in actual level. Not only would the indicated level be inaccurate, but the instrument could not even be used to determine the level trend. Figure B-16-9 illustrates the variation of indicated level with instrument run temperature for an instrument with a reference leg vertical run longer than the variable leg vertical run. Line 1 plots indicated level as a function of instrument run temperature with the actual level at the variable leg tap. The MRT is the run temperature that causes indicated level

Can/Cannot be determined The current value or status of an identified parameter relative to that specified in the procedure can/cannot be ascertained using all available indications (direct and indirect, singly or in combination). An offscale indication does not itself mean that a parameter value cannot be determined provided the instrument is believed to be functioning properly and the offscale indication is consistent with plant conditions. For example, if all available RPV water level instruments are observed to trend downward following indication of a large primary system break with limited makeup capability, an eventual downscale indication would be consistent with plant conditions and could be considered valid, subject to the instrumentation limits addressed in Caution #1 (refer to the discussion of Caution #1 in Section 4 of this appendix). Whether a parameter value can be determined within the context of PSTG/SATG strategies depends upon whether identified action levels and decisions can be evaluated, not whether the precise value of the parameter is known. If Fuel Zone RPV water level instruments are indicating downscale and the downscale indications are believed to be valid, RPV water level relative to the top of the fuel and the Minimum Steam Cooling RPV Water Level can be determined even though level is below the indicating range. Similarly, if RPV water level indications are driven offscale high and the upscale indications are believed to be valid, level can be determined to be above the top of the fuel. Whether an offscale indication can be considered valid, however, and the length of time an offscale indication can be relied upon, requires a judgment based on the nature of the event, plant conditions, and the instrument characteristics.

B - 2-11 Rev. 8

Part 1 of Caution #1 identifies the limiting conditions beyond which boiling of the water in the instrument legs may occur.

The RPV Saturation Temperature is a plot of the saturation temperature of water as a function of pressure. If the temperature of the water in an RPV water level instrument run exceeds this temperature, the water may start to boil, resulting in unreliable level indication.

Boiling is of concern in all instrument runs—horizontal and vertical, reference and variable. Boiloff from the reference leg reduces the height of water in the leg. This decreases the pressure on the reference leg side of the differential pressure transmitter and increases the indicated level. Boiling in the variable leg increases the pressure on the variable leg side of the transmitter, likewise increasing the indicated level (Figure B-4-2). Continued boiling would produce an increasing level trend.

Boiling and loss of valid level indication may not occur immediately when drywell temperature exceeds the RPV Saturation Temperature. As drywell temperatures increase, the temperature in the instrument runs will lag the temperature in the surrounding area. The temperature of the water in the instrument run may therefore be lower than the indicated air temperatures.

The wording of Caution #1 permits continued use of a level instrument until boiling is actually observed, avoiding premature transfers to RPV flooding. While the onset of

boiling is difficult to predict, it is expected that vapor formation sufficient to affect the validity of the level indication will also be observable. Indications of boiling may vary, depending on the temperature of the water and the extent of the condition. Bulk boiling will likely result in extremely erratic indication accompanied by loss of inventory from the reference leg. Localized vapor formation may result in instrument mismatches and a notching effect similar to degassing.

The RPV Saturation Temperature is plotted from 0 psig to the pressure setpoint of the lowest-lifting SRV. It is expressed in terms of average temperature near the instrument runs as indicated on PC-TI-505A-D since CNS is not equipped with sensors that directly monitor instrument run temperatures.

Examination Outline Cross-Reference	Level	RO
HPCI is secured as an EOP step if SP level cannot be maintained above 11 feet due to turbine exhaust directly to SP air space. Vortex limit is for the HPCI pump which is never reached due to being secured due to turbine exhaust. Added HPCI step to references. No change to question.	Tier#	1
	Group#	1
	K/A #	295030G2.4.20
	Rating	3.8
295030 Low Suppression Pool Wtr Lvl / 5 G2.4.20 Knowledge of the operational implications of EOP warnings, cautions, and notes. (CFR: 41.10 / 43.5 / 45.13)		

Question 15

Reference Provided

The plant has experienced an earthquake resulting in the following conditions:

- RPV level is +10 inches and stable.
- RPV Pressure is 200 psig and stable (Controlled by HPCI).
- Torus level is 12 feet and lowering fast.
- RHR Pumps A & D flows are 6500 gpm each.
- CS Pump A flow is 6000 gpm.
- HPCI flow is 4000 gpm in pressure control.
- RCIC flow is 400 gpm.

IAW Caution 3, which system will reach its vortex limit FIRST as Torus level continues to lower?

A. HPCI

B. RCIC

C. RHR

D. CS

Answer:
D. CS
Explanation: Caution 3 reminds the operator of potential equipment damage when operating above

NPSH & Vortex limits. The vortex limits are defined to be the lowest suppression pool water level above which air entrainment is not expected to occur in pumps taking suction on the pool. These levels are functions of pump flow. Exceeding the limits can lead to air entrainment at the pump suction strainers. Since Core Spray is operating at 6000 gpm, its vortex limit would be reached first at ~ 8.5 feet. RCIC is reached at 6 feet. RHR would be reached at 5.5 feet. HPCI is not allow to be operated below 11 feet due to direct steam exhaust to the torus air space.

Distracters:

A. This answer is incorrect due to CS reaching the vortex limit first. This choice is plausible due to not recognizing HPCI being required to be secured below 11' torus level due to exhaust steam discharging directly to the torus air space. The candidate that does not recognize the requirement to secure HPCI or considers 9.1' (specifically identified next to HPCI) would select this answer.
B. This answer is incorrect due to CS reaching the vortex limit first. This choice is plausible due to not recognizing RCIC vortex limit of 6'. The candidate that does not recognize RCICs vortex limit of 6' would select this answer.
C. This answer is incorrect due to CS reaching the vortex limit first. This choice is plausible due to CS & RHR vortex limits being easily confused. The candidate that confuses CS & RHR vortex limits would select this answer.

Technical References:

AMP-TBD00 Tech. Basis – App. B (CNS PSTG/SATG Appendix B Technical Bases), Rev. 8
EOP 3A (Primary Containment Control), Rev. 15
Emergency Operating Procedure 5.8 Attachment 2 (EOP and SAG Graphs), Rev. 15

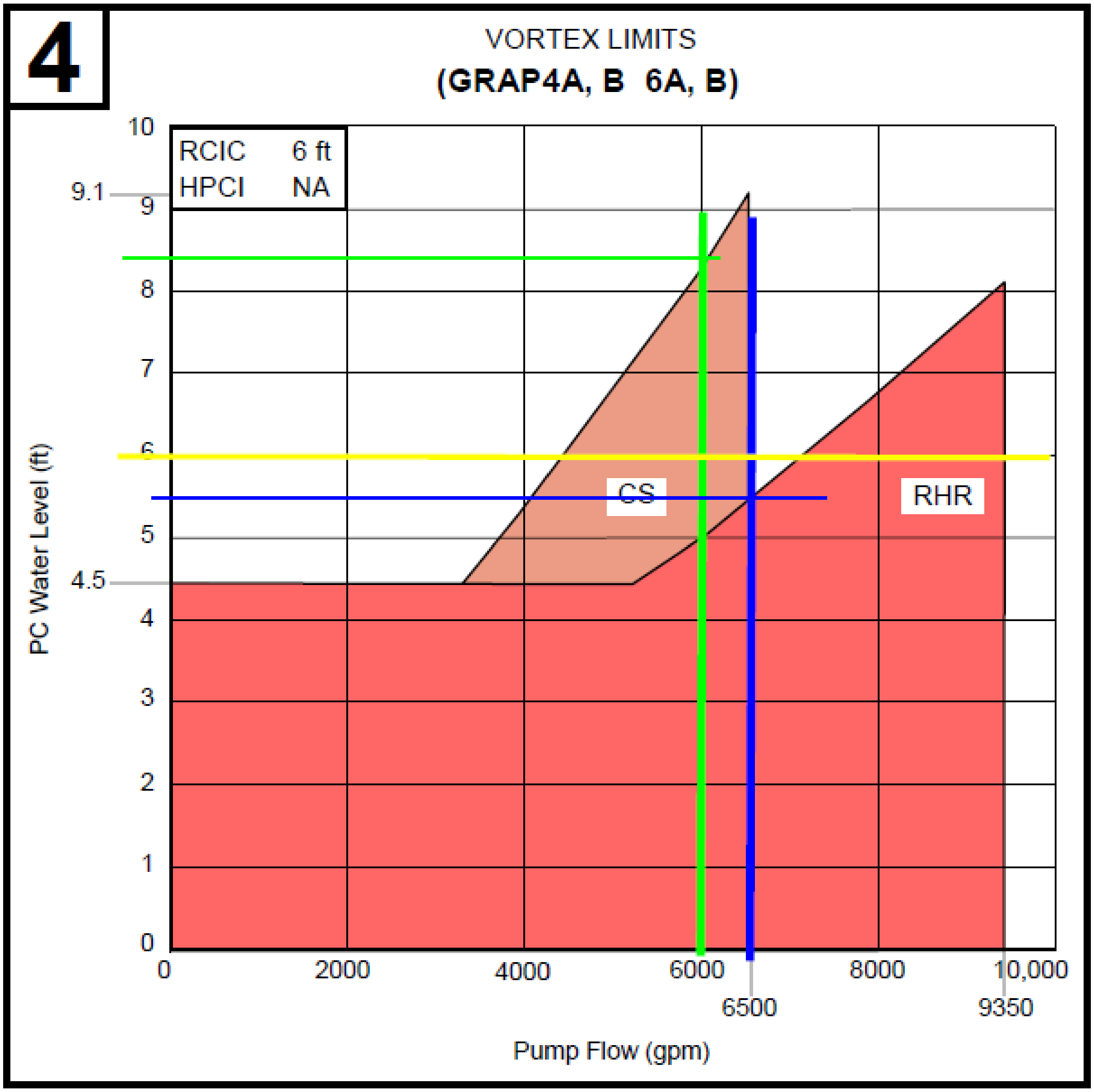
References to be provided to applicants during exam: EOP Vortex Limits (Graphs 4A,B 6A,B)

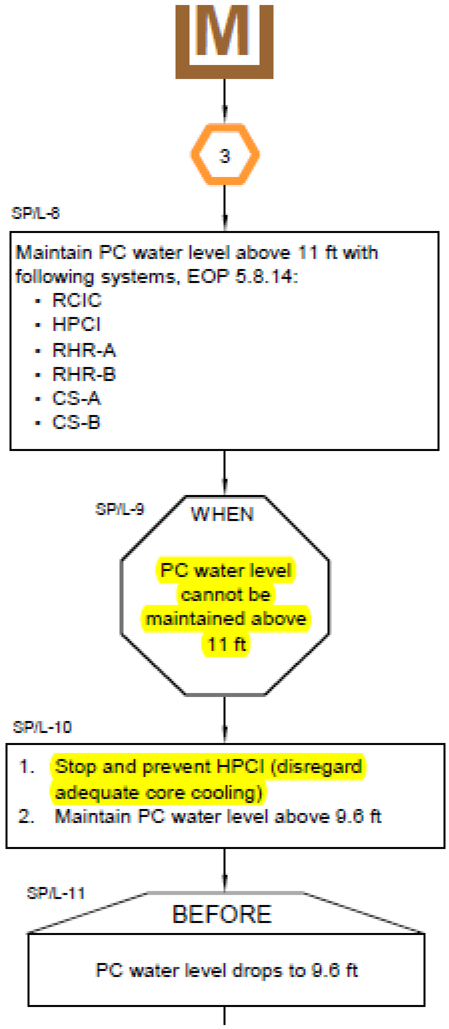
Learning Objective:

INT00806180010300 Given plant conditions and the EOP and SAG Graphs Flowchart, determine if operation is within the allowed region of a graph.

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b) 10	
Level of Difficulty:	4	
SRO Only Justification:	N/A	

3 Operation of HPCI, RCIC, Core Spray, or RHR with suction from suppression pool and pump flow above NPSH or Vortex Limit (GRAPHs 3, 4, 5, 16, 18) may result in equipment damage




M
3
SP/L-8
Maintain PC water level above 11 ft with following systems, EOP 5.8.14:
• RCIC
• HPCI
• RHR-A
• RHR-B
• CS-A
• CS-B
SP/L-9
WHEN
PC water level cannot be maintained above 11 ft
SP/L-10
1. Stop and prevent HPCI (disregard adequate core cooling)
2. Maintain PC water level above 9.6 ft
SP/L-11
BEFORE
PC water level drops to 9.6 ft

PSTG/SATG Step

❸

> Operation of HPCI, RCIC, Core Spray, or RHR with suction from the suppression pool and pump flow above the NPSH or vortex limit may result in equipment damage.

Discussion

The NPSH (Net Positive Suction Head) limits are defined to be the highest suppression pool temperature which provides adequate net positive suction head for pumps taking suction on the pool. The NPSH Limits are functions of pump flow and suppression chamber overpressure (airspace pressure plus the hydrostatic head of water over the pump suction). It is utilized to preclude pump damage from cavitation.

The vortex limits are defined to be the lowest suppression pool water level above which air entrainment is not expected to occur in pumps taking suction on the pool. These levels are functions of pump flow. Exceeding the limits can lead to air entrainment at the pump suction strainers.

Refer to Section 16 of this appendix for detailed discussions of the NPSH and vortex limits.

The NPSH and vortex limits are addressed through a caution for the following reasons:

- It is difficult to define in advance exactly when the limits should be observed and when pumps should be operated outside the limits.
- Pumps to which the limits apply are used in more than one parameter control path. RHR pumps, for example, may be used in PSTG Steps RC/L, SP/T, DW/T, and PC/P. HPCI and RCIC may be used in both PSTG Steps RC/L and RC/P. Authorizing operation of the pumps outside NPSH and vortex limits in one path may conflict with instructions in another path where flow would normally be controlled below the limits.
- Pump characteristics, and the shape of the NPSH and vortex limit curves, vary from system to system. If a limit is relatively flat, throttling pump flow will be of little benefit; the operator can only choose whether or not to operate the pump.

Examination Outline Cross-Reference	Level	RO
Natural Circulation is used – deleted thermal convection flow from the stem and explanation.	Tier#	1
	Group#	1
	K/A #	295031 EK1.02
	Rating	3.8

295031 Reactor Low Water Level / 2

EK1. Knowledge of the operational implications of the following concepts as they apply to REACTOR LOW WATER LEVEL : (CFR: 41.8 to 41.10)

EK1.02 Natural circulation: Plant Specific

Question 16

The plant is in MODE 4 when a complete loss of Shutdown Cooling (SDC) occurs.

No Reactor Recirculation Pumps are in service.

Which one of the following identifies the LOWEST RPV Water Level that will support natural circulation IAW 2.4SDC (Shutdown Cooling Abnormal)?

A. 44 inches

B. 49 inches

C. 54 inches

D. 58 inches

Answer:
B. 49 inches
Explanation: Minimum RPV water level to establish natural circulation is > 48 inches.
Distracters: A. This answer is incorrect due to the lowest level required to establish and maintain natural circulation being > 48 inches. This choice is plausible due to 44 inches being the lowest level provided. The candidate that selects the lowest level provided would choose this answer.

C. This answer is incorrect due to the lowest level required to establish and maintain natural circulation being > 48 inches. This choice is plausible due to 54 inches being the TS high level trip setpoint for HPCI, RCIC, and the Main Turbine. The candidate that confuses high level trip level would choose this answer.

D. This answer is incorrect due to the lowest level required to establish and maintain natural circulation being > 48 inches. This choice is plausible due to 58 inches being the lowest level to allow venting through RHR Subsystem high point vents. The candidate that confuses the lowest level required for venting the RHR system would choose this answer.

Technical References:
Procedure 2.4SDC (Shutdown Cooling Abnormal), Rev. 14
Procedure 2.2.69.2 (RHR System Shutdown Operations), Rev. 90

References to be provided to applicants during exam: NONE

Learning Objective: COR0022302001090D Explain the significance of the following as they apply to a loss of Shutdown Cooling: Natural circulation

Question Source:	Bank # 2366	
(note changes; attach parent)	Modified Bank #	
	New	
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b) 10	
Level of Difficulty:	2	
SRO Only Justification:	N/A	

QUESTION: 9, 2366 (1 point(s))

Given the following plant conditions:

% A plant cooldown is in progress.
% Reactor temperatures is 198EF

For the above plant conditions, according to procedure 2.1.4 (Normal Shutdown), which of the following is the minimum required reactor vessel level to aid in thermal convection?

a. 44 inches.

b. 48 inches.

c. 52 inches.

d. 56 inches.

ANSWER: 9, 2366

b. 48 inches.

REFERENCE: Procedure 2.1.4 Page 26:

Maintain reactor water level ≥ 48".

ATTACHMENT 2 CONTINGENCY ACTIONS FOR COMPLETE LOSS OF SDC℗[10]

1. CONTINGENCY ACTIONS FOR COMPLETE LOSS OF SDC

1.1 Commence monitoring plant heatup rate per Procedure 6.RCS.601.℗[8]

NOTE – Preferred level indication is NBI-LI-86, SHUTDOWN LVL. RFC-LI-94A, RFC-LI-94B, or RFC-LI-94C, RX NR LEVEL, may indicate up to 9" higher than actual during cold conditions.

1.2 Control RPV level > 48" to aid in thermal convection flow.

1.3 IF blade guides in RPV or fuel bundle removed from around core instrumentation, THEN Step 1.4 is N/A.℗[2]

1.4 Place or maintain one available RR pump in service per Procedure 2.2.68.

1.5 Place RWCU System in service per alternate heat removal section of Procedure 2.2.66.℗[8]

1.6 Review Attachment 5 (Page 15) using time to core boiling/uncovery figure for existing reactor cavity water level.

1.7 Monitor following temperatures and pressures frequently and log in Control Room Log every 4 hours:℗[4]

1.7.1 IF a RR pump is in service, THEN monitor RR-TI-151A(B), SUCT TEMP (PNL 9-4).

1.7.2 IF a RR pump is not in service, THEN monitor RPV metal temperatures on NBI-TR-89, REACTOR VESSEL METAL TEMPERATURE RECORDER (PNL 9-21), for approach to boiling.

1.7.3 IF RWCU is in service, THEN monitor inlet temperature on RWCU-TI-137, TEMP IND, using Point 3 on TEMP POINT SELECTOR (PNL 9-4).

1.7.4 Monitor following reactor pressure PMIS Points for indication of pressurization:

1.7.4.1 B025.

1.7.4.2 N013.

1.7.4.3 N014.

1.8 IF RPV head is off, THEN go to Step 1.12.

NOTE – If RHR Subsystem B SDC was removed from service for a short period of time, no air was introduced into it and flow path remained open from RPV to RHR-MO-27B, Steps 7.8 through 7.26 are N/A.

7.8 IF RPV is depressurized, THEN ensure RPV level is ≥ 58" by NR level indication to allow venting through RHR Subsystem B high point vents.

7.9 Ensure RHR-MO-66B, HX BYPASS VLV, is open.

7.10 Open RHR-302, RHR HEAT EXCHANGER 1B INLET VENT SHUTOFF VALVE (RHR-931-HX B Room).

CAUTION – Water from vent should be treated as contaminated. Care should be taken to prevent spreading contamination.

7.11 Throttle open RHR-125, RHR HEAT EXCHANGER 1B INLET VENT ROOT VALVE (RHR-931-HX B Room), until air free water flows.

7.12 (Independent Verification) Close RHR-125. Performed By: ________

Verified By: ________

7.13 (Independent Verification) Close RHR-302. Performed By: ________

Verified By: ________

7.14 Open RHR-455, RHR-B STEAM CONDENSING VENT (R-931-RHR HX B Room).

7.15 Throttle open RHR-454, RHR-B STEAM CONDENSING VENT SHUTOFF (R-931-RHR HX B Room), until air free water flows.

7.16 (Independent Verification) Close RHR-454. Performed By: ________

Verified By: ________

Table 3.3.5.1-1 (page 3 of 6)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. LPCI System (continued)					
g. Low Pressure Coolant Injection Pump Discharge Flow - Low (Bypass)	1,2,3, 4(a), 5(a)	1 per subsystem	E	SR 3.3.5.1.2 SR 3.3.5.1.4(c)(d) SR 3.3.5.1.5	≥ 2107 gpm
3. High Pressure Coolant Injection (HPCI) System					
a. Reactor Vessel Water Level - Low Low (Level 2)	1, 2(f), 3(f)	4	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.4(c)(d) SR 3.3.5.1.5	≥ -42 inches
b. Drywell Pressure - High	1, 2(f), 3(f)	4	B	SR 3.3.5.1.2 SR 3.3.5.1.4(c)(d) SR 3.3.5.1.5	≤ 1.84 psig
c. Reactor Vessel Water Level - High (Level 8)	1, 2(f), 3(f)	2	C	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.5	≤ 54 inches
d. Emergency Condensate Storage Tank (ECST) Level - Low	1, 2(f), 3(f)	2	D	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5	≥ 23 inches
e. Suppression Pool Water Level - High	1, 2(f), 3(f)	2	D	SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.5	≤ 4 inches

(continued)

(a) When the associated ECCS subsystem(s) are required to be OPERABLE per LCO 3.5.2, ECCS - Shutdown.

(c) 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.

(d) 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 Limiting Trip Setpoint and the methodologies used to determine the as-found and the as-left tolerances are specified in the Technical Requirements Manual.

(f) With reactor steam dome pressure > 150 psig.

Examination Outline Cross-Reference	Level	RO
	Tier#	1
	Group#	1
	K/A #	295037 EK2.03
	Rating	4.1

295037 SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown / 1

EK2. Knowledge of the interrelations between SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN and the following: (CFR: 41.7 / 45.8)

EK2.03 ARI/RPT/ATWS: Plant Specific

Question 17

A DEH failure results in the following conditions while operating at rated power:

- Reactor Power is 24%.
- RPV Water Level lowered to +5 inches and is currently stable at 35 inches.
- RPV Pressure peaked at 1080 psig and is currently stable at 1000 psig.

Which one of the following identifies the current status of RPS and ARI logics?
(Assume RPS and ARI logics function as designed.)

	RPS	ARI
A.	Energized	Energized
B.	Energized	De-energized
C.	De-energized	Energized
D.	De-energized	De-energized

Answer:
C. De-energized Energized

Explanation:
RPV pressure peaking at 1080 psig causes a reactor Scram (1034 psig), ARI and

ATWS RPT (1060 psig). RPS logic will be de-energized and ARI logic will be energized. RPS is de-energized, resulting in partial control rod insertion, as evidenced by power reducing from 100% to 24%.

Distracters:

A. This answer is incorrect due to RPS being de-energized. This choice is plausible due to having reactor power at 24% being indicative of not having a Scram and ARI logic commonly confused as being normally energized. The candidate that does not recognize ATWS conditions being present and confuses ARI logic status would choose this answer.

B. This answer is incorrect due to RPS being de-energized and ARI being energized. This choice is plausible due to having reactor power at 24% being indicative of not having a Scram or ARI logic initiation. The candidate that does not recognize ATWS conditions being met would choose this answer.

D. This answer is incorrect due to ARI logic being energized. This choice is plausible due to ARI logic commonly confused as being normally energized. The candidate that recognizes Scram conditions being met and confuses ARI logic status would choose this answer.

Technical References:
Procedure 4.5 (Reactor Protection/AlternateRod Insertion Systems), Rev. 31

References to be provided to applicants during exam: NONE

Learning Objective: COR0023302001090A Given plant conditions, determine if the following has occurred: Automatic ARI initiation.

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b) 7	
Level of Difficulty:	2	
SRO Only Justification:	N/A	

ATTACHMENT 3 INFORMATION SHEET

1.1.7 The RPS is a fail-safe system, composed of two independent trip systems (A and B), each made up of two auto scram channels and a manual scram channel. Trip System A consists of Reactor Auto Scram Channels A1 and A2, and Reactor Manual Scram Channel A3. Trip System B consists of Reactor Auto Scram Channels B1 and B2, and Reactor Manual Scram Channel B3. Each auto scram channel receives an input from at least one independent sensor monitoring each of the critical parameters. A trip occurring in any trip channel(s) of Trip System A, together with a trip occurring in any trip channel(s) of Trip System B, generates a reactor scram. Note that a trip of one trip system, with the other system not tripped, does not cause a reactor scram.

1.1.8 In order to scram the control rods, air operated scram valves in Trip Systems A and B must open. Instrument air normally flows through the backup scram valves, test valve, and scram header pilot valves to the scram header vent valves and scram discharge volume (SDV) drain valve. Instrument air also flows through the backup scram valves to both sets of scram valves (137) in Trip System A and (137) in Trip System B. Normally, both sets of pilot scram valve solenoids are energized and instrument air holds the scram valves closed. If a scram trip setpoint is exceeded in Trip System A (only in A), the scram valve solenoids in Trip System A de-energize, but instrument air still flows through the scram valve solenoids in Trip System B to hold the scram valves closed. If a scram trip setpoint is exceeded in Trip System B (only in B), the scram valve solenoids in Trip System B de-energize and instrument air holds the scram valves closed through the Trip System A scram valve solenoids. When a scram trip setpoint is exceeded in both Trip Systems A and B, both sets of scram valve solenoids are de-energized and instrument air is vented from the scram valves. This permits the scram valves to open, causing all control rods to be rapidly driven in (scrammed).

1.1.9 Pilot scram valve solenoids are the 117 SOVs for Trip System A and 118 SOVs for Trip System B.

1.1.10 When a trip occurs in both trip systems, both scram header pilot valve solenoids are de-energized, the flow of instrument air is blocked past this point, and the header is vented, causing the scram header vent valves (two valves) and SDV drain valves to close. Moreover, whenever a reactor scram occurs, both backup scram valve solenoids (normally de-energized) are energized and instrument air is blocked and vented at this point. This backup action, by itself, would cause the insertion of the control rods and closure of the scram header vent valves and SDV drain valve. However, the backup scram valves take longer to bleed air from the header. Thus, scram times could be exceeded, if only backup scram valves caused the scram.

1.2.4 REACTOR VESSEL LOW WATER LEVEL

1.2.4.1 The reactor vessel low water level scram is set at ≥ +3" indicated level. The level indicating switches which initiate the reactor scram signals are NBI-LIS-101A, NBI-LIS-101B, NBI-LIS-101C, and NBI-LIS-101D. This is a one out of two taken twice logic.

1.2.5 MAIN STEAM LINE ISOLATION VALVE CLOSURE

1.2.5.1 There are four main steam lines with two isolation valves per line, one inside and one outside the drywell. These valves are numbered 80A through 80D inside the drywell and 86A through 86D outside the drywell. The logic is set up on these valves so one valve < 10% closure will never cause a scram signal. Two (two lines isolated) MSIVs < 10% closure gives a half scram, three lines with valves < 10% closure always gives a full scram. The main steam line isolation valve closure scram is bypassed anytime the REACTOR MODE switch is in START & HOT STBY, SHUT DOWN, or REFUEL.

1.2.6 REACTOR VESSEL HIGH PRESSURE

1.2.6.1 A rise in reactor pressure could result in the collapse of core steam voids depending on core void content. This would result in a power rise and could cause fuel damage. Pressure Sensors NBI-PS-55A through NBI-PS-55D de-energize relays in the RPS should reactor pressure rise to ≤ 1050 psig causing a scram. This logic is set up on a one out of two taken twice for full scram.

1.2.7 DRYWELL HIGH PRESSURE

1.2.7.1 The drywell is maintained inerted with N_2. A rise in drywell pressure could indicate a steam or water leak from the Primary Coolant System. Pressure Switches PC-PS-12A through PC-PS-12D provide inputs to the RPS when drywell pressure reaches ≤ 1.84 psig to scram the reactor. This logic is set up on a one out of two taken twice for full scram.

1.2.8 NEUTRON MONITORING SYSTEM - APRM

1.2.8.1 Three APRM channels provide outputs to RPS A and RPS B. One APRM channel in each logic (A or B) may be bypassed with an interlocked switch (Panel 9-5). Signals which generate a scram signal from the APRM are:

a. APRM high flux trip (Flow Bias) ≤ 0.75W + 62.0% - 0.75ΔW.

b. With the MODE switch in any position except RUN, the APRM high flux trip is set at ≤ 14.5%.

1.3 ALTERNATE ROD INSERTION

1.3.1 The Alternate Rod Insertion (ARI) System provides means of reactor shutdown that is as diverse as possible and independent from the Reactor Protection System (RPS). The ARI System causes the reactor to scram by exhausting the scram air header through the ARI vent valves if reactor pressure reaches or exceeds 1072 psig or if reactor water level reaches or drops below -42". The ARI System can also be manually initiated from Control Room Panel 9-5 by placing both ARI CHANNEL MANUAL switches to the ARMED position and pressing them simultaneously. When ARI is manually initiated, the red MAN INIT light on Panel 9-5 will be on while the switches are depressed.

1.3.2 A scram initiated by ARI also causes the air to be bled off of the SDV vent and drain valves, and the valves close to isolate the SDV.

PROCEDURE 4.5	REVISION 31	PAGE 17 OF 20

ATTACHMENT 3	INFORMATION SHEET

1.3.3 The ARI scram air header exhaust valves are arranged in series so that both channels of ARI must initiate to cause a reactor scram. Additionally, the ARI vent valves are energized to operate so that no scram occurs on a loss of power to the ARI System.

1.3.4 Once the ARI logic is initiated, the scram signal is held for 37.5 seconds to allow a full scram to take place. When this time delay is timed out, the white RESET light on Panel 9-5 will turn on. After the time delay has timed out <u>and</u> the level and pressure signals are clear, the logic may be reset, by pressing the ARI RESET pushbutton on Panel 9-5. This resets the logic and de-energizes the ARI vent valves. Specific instructions for resetting ARI are contained in Procedure 2.1.5.©[1]

Examination Outline Cross-Reference	Level	RO
	Tier#	1
	Group#	1
	K/A #	295038 EK3.03
	Rating	3.7

295038 High Off-site Release Rate / 9

EK3. Knowledge of the reasons for the following responses as they apply to HIGH OFF SITE RELEASE RATE: (CFR: 41.5 / 45.6)

EK3.03 Control room ventilation isolation: Plant Specific.

Question 18

Which one of the following completes the statements below regarding Control Room Ventilation isolation during a Fuel Handling Accident (FHA) causing high Off Site release rates?

Control Room ventilation automatically isolates due to ____(1)____ to protect Control Room operators by maintaining the Control Room Envelope (CRE) at a ____(2)____ pressure.

A. (1) Control Room High Radiation
(2) positive

B. (1) Control Room High Radiation
(2) negative

C. (1) Reactor Building Exhaust Plenum High-High Radiation
(2) positive

D. (1) Reactor Building Exhaust Plenum High-High Radiation
(2) negative

Answer:
C. (1) Reactor Building Exhaust Plenum High-High Radiation
(2) positive

Explanation:
Control Room ventilation automatically isolates due to a Group 6 isolation signal to maintain the CRE pressure positive to support Control Room operator habitability.

Group 6 isolation occurs if Reactor Building Exhaust Plenum High-High Radiation setpoint (10mr/hr) is reached.		
Distracters: A. This answer is incorrect due Control Room ventilation isolates due to RB Exh H-Hi Rad. This choice is plausible due to Control Room Hi Rad requiring entry into procedure 5.1RAD which provides guidance to manually align Control Room ventilation and CR Hi Rad providing isolation signals at other BWRs. The candidate that confuses CR HVAC auto isolation signals and recognizes the CR pressure is maintained positive would choose this answer. B. This answer is incorrect due Control Room ventilation isolates due to RB Exh H-Hi Rad and the pressure is maintained positive This choice is plausible due to Control Room Hi Rad requiring entry into procedure 5.1RAD which provides guidance to manually align Control Room ventilation and CR Hi Rad providing isolation signals at other BWRs and the RB is required to maintained at a negative pressure . The candidate that confuses CR HVAC auto isolation signals and required building pressure would choose this answer. D. This answer is incorrect due to pressure maintained positive This choice is plausible due to the RB is required to maintained at a negative pressure . The candidate that correctly identifies the CR HVAC auto isolation signal and confuses the required building pressure would choose this answer.		
Technical References: Procedure 2.3_9-3-1 (Panel 9-3 - Annunciator 9-3-1), Rev. 34 Procedure 2.3_9-4-1 (Panel 9-4 - Annunciator 9-4-1), Rev. 53 Procedure 2.1.22 (Recovering From A Group Isolation), Rev. 60 Procedure 2.2.84 (HVAC Main Control Room And Cable Spreading Room), Rev. 34		
References to be provided to applicants during exam: NONE		
Learning Objective: COR0010802001140A Briefly describe the following concepts as they apply to Control Room HVAC: Airborne contamination (e.g., radiological,toxic gas, smoke) control		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b) 5	
Level of Difficulty:	2	
SRO Only Justification:	N/A	

9.3.4.10 HV-MO-268, MG SET-1B OUTLET VLV.

9.3.5 Control Room Emergency Filter System (CREFS) starts (VBD-R):

9.3.5.1 BF-C-1A, EMER BSTR FAN, starts.

9.3.5.2 EF-C-1B, TOILET EXH FAN, stops.

9.3.5.3 HV-270AV, CONTROL ROOM HVAC INLET VALVE, closes.

9.3.5.4 HV-271AV, CONTROL ROOM HVAC EMERGENCY BYPASS SYSTEM INLET VALVE, opens.

9.3.5.5 HV-272AV, CONTROL ROOM PANTRY EXHAUST FAN ISOLATION VALVE, closes.

9.3.6 Following REC and SW valves are open (VBD-M):

9.3.6.1 REC-MO-711, NORTH CRITICAL LOOP SUPPLY VLV.

9.3.6.2 SW-MO-650, REC HX A SERVICE WATER OUTLET (intermediate open if valve was closed before isolation occurred).

9.3.6.3 REC-MO-714, SOUTH CRITICAL LOOP SUPPLY VLV.

9.3.6.4 SW-MO-651, REC HX B SERVICE WATER OUTLET (intermediate open if valve was closed before isolation occurred).

9.4 Align SGT per Procedure 2.2.73 within 1 hour of receiving Group 6.©[1]

9.5 Monitor RRMG Set temperatures closely and take action per Procedure 2.4HVAC.

9.6 Determine isolation cause:

ISOLATION	ALLOWABLE VALUE	COMMENTS
Low Reactor Water Level	≥ -42"	Ensure Group 3 Isolation.
High Drywell Pressure	≤ 1.84 psig	Ensure Group 2 Isolation.
Reactor Building H&V Exhaust Plenum High Radiation	≤ 49 mR/hr	
RPS Power Supply Failure	Loss of power	

ATTACHMENT 1 INFORMATION SHEET

ATTACHMENT 1 INFORMATION SHEET

1. DISCUSSION

1.1 FUNCTION

1.1.1 The system provides HVAC to the Control Room and Cable Spreading Room for personnel comfort and optimum equipment performance.

1.2 OPERATING CHARACTERISTICS

1.2.1 Main Control Room and Cable Spreading Room HVAC is supplied by Air Conditioning Unit AC-C-1A. Air is drawn in through the dampers both from the outside and from the Booster Fan BF-C-1B outlet through the recirculation damper.

1.2.2 Control Room pantry and toilet have a common Exhaust System Fan EF-C-1B and isolation valve. EF-C-1B has a control switch on Board R and a local starter near the Control Room bathroom door. If EF-C-1B is secured from the local starter, the isolation valve will not close and the fan will indicate that it is running on Board R.

1.2.3 Air Conditioning Unit AC-C-1A is comprised of a filter, 3 DX cooling coils, heating oil, and two Supply Fans SF-C-1A and SF-C-1B.

1.2.4 Fire/Smoke Dampers HV-AD-AD1544, HV-AD-AD1545, HV-AD-AD1546, HV-AD-AD1547, HV-AD-AD1581, and HV-AD-AD1582 automatically close when fire or smoke is detected locally at the damper or when smoke is detected in the Cable Spreading Room to prevent smoke from spreading to the Control Room when there is a fire in the Cable Spreading Room.

1.2.5 The Control Room System has an Emergency Bypass System consisting of a Pre-Filter PF-C-1A, High Efficiency Filter HEF-C-1A, Carbon Filter CF-C-1A, and Emergency Booster Fan BF-C-1A which can be supplied from either MCC-LX or MCC-TX via a manual transfer switch in the Auxiliary Relay Room. Upon a Group 6 Isolation signal, this Bypass System is energized and allows outside air to pass through it to the AC unit. During Bypass System operation, one AC unit supply fan is required to run in order to maintain positive Control Room pressure. Additionally, the exhaust booster fan is required to run to provide backpressure which prevents inlet air flow rates from exceeding the Tech Spec limit. This system is designed to maintain the Control Room environment for 200 man-days under the above conditions. Testing has shown that, for maximum radiological protection during a radiological event, one of two system lineups is recommended. The first is the Bypass System lineup with the emergency booster fan, exhaust booster fan, and one supply fan operating. This is the designed emergency operating configuration. The second lineup is a lineup where no fans operate. This configuration is assured during a loss of power only if all fans are aligned to the same divisional power source.

CONTROL ROOM HIGH RAD	PANEL/WINDOW: 9-3-1/B-10

1. OPERATOR OBSERVATION AND ACTION

1.1 IF associated indicator on Panel 9-11 is below alarm setpoint, THEN reset alarm.

1.2 IF associated indicator on Panel 9-11 remains above alarm setpoint, THEN perform following:

1.2.1 Clear Control Room of all unnecessary personnel.

1.2.2 Notify Plant personnel to stay clear of Control Room via gaitronics.

1.2.3 Notify Radiation Protection to survey area.

1.2.4 Concurrently enter Procedure 5.1RAD.

1.3 IF conditions require Control Room evacuation, THEN enter Procedure 5.1ASD.

RX BLDG VENT HI-HI RAD	PANEL/WINDOW: 9-4-1/E-4

1. AUTOMATIC ACTIONS

1.1 Group 6 Isolation.

2. OPERATOR OBSERVATION AND ACTION

2.1 Notify Radiation Protection.

2.2 Make gaitronics announcement for all personnel to evacuate Reactor Building.

2.3 Concurrently enter Procedures 2.1.22 and 5.1RAD.

Examination Outline Cross-Reference	Level	RO
Remote is away from the CO2 bottles which are located inside DG rooms. Revised to reflect more plausible location. Added DG room maps to references. Comments incorporated. Added either & OR to B & D for answer clarity.	Tier#	1
	Group#	1
	K/A #	600000 AA1.08
	Rating	2.6

600000 Plant Fire On Site / 8

AA1 Ability to operate and / or monitor the following as they apply to PLANT FIRE ON SITE:

AA1.08 Fire fighting equipment used on each class of fire

Question 19

DG1 lube oil has caught on fire.

Which one of the following completes the statements below regarding the DG1 Room Fire Suppression system utilized for this fire and action required if the system fails to automatically initiate?

DG1 Room fire suppression uses ____(1)____ pressure CO_2.

The remote pneumatic release bottles located outside DG1 room (TB 903' entrance) actuates suppression INTO ____(2)____.

A. (1) LOW
   (2) DG1 room ONLY

B. (1) LOW
   (2) either DG1 OR DG2 room

C. (1) HIGH
   (2) DG1 room ONLY

D. (1) HIGH
   (2) either DG1 OR DG2 room

Answer:
C. (1) HIGH (2) DG1 room ONLY

Explanation:
The DG rooms utilize HIGH pressure CO2 for fire suppression.
The DG1 CO2 system is actuated by actuating either manual release bottle located outside DG1 room (TB 903' entrance) or one within DG2 room (West wall).
The 2 release bottles located outside the applicable DG room allows for the CO2 in that DG room to be discharged into the room (closest to the door - DG1 CO2 discharge to DG1 room) or opposite DG CO2 discharge into the room (furthest from the door – DG2 CO2 discharge to DG1 room) which is commonly confused.

Distracters:

A. This answer is incorrect due to high pressure CO_2 providing suppression. This choice is plausible due to low pressure CO_2 being provided to the turbine bearings. The candidate that confuses the CO2 pressure and knows the location to manually initiate would select this answer.

B. This answer is incorrect due to high pressure CO2 providing suppression and the pneumatic actuators outside DG1 room supplying DG1 room only. This choice is plausible due to there being 2 actuators, which both dispense CO2 into DG1 from either the bottles located in DG1 room itself, or the bottles located in DG2 room. The candidate that confuses the CO2 pressure and what room the remote manual release bottles supply would select this answer.

D. This answer is incorrect due to the pneumatic actuators outside DG1 room supplying DG1 room only. This choice is plausible due to there being 2 actuators, which both dispense CO2 into DG1 from either the bottles located in DG1 room itself, or the bottles located in DG2 room. The candidate that confuses what room the remote manual release bottles supply would select this answer.

Technical References:
Procedure 2.2.2 Carbon Dioxide Systems

References to be provided to applicants during exam: NONE

Learning Objective:
COR0010502001080E Describe the Fire Protection system design features and/or interlocks that provide for the following: Manual initiation

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b) 7	
Level of Difficulty:	3	
SRO Only Justification:	N/A	

14. MANUAL INITIATION OF DG-1 CO_2 SYSTEM

NOTE 1 – Following an emergency start of DG1 or placement of IS/DG1A in ISOLATE, manual initiation of DG-1 CO_2 System will result in CO_2 being exhausted to atmosphere due to DG HVAC System interlock with CO_2 System being bypassed.

NOTE 2 – Manual initiation results in immediate discharge of CO_2 into affected room.

NOTE 3 – Pneumatic release bottles are located near DG-1 Room entrance outside of room, in Boiler Room near entrance to DG Building, and on west wall of each DG Room near double doors.

14.1 Ensure all personnel are evacuated from affected room.

14.2 Actuate pneumatic release bottles per posted instructions at bottles.

14.3 Ensuring CO_2-PS-CO1, DG 1 H&V TRIP/RESET (outside room near Security door in Boiler Room near entrance to DG Building), has actuated (plunger extended).

PROCEDURE 2.2.2	REVISION 39	PAGE 8 OF 20

1.2.4 The high pressure systems may by manually discharged by pneumatic release bottles near the Boiler Room entrance to the DG Building, near the DG-1 Room Security door, and on the west wall inside each Diesel Generator Room. There are three manual pneumatic release bottles associated with a given Diesel Generator Room. Actuation of the pneumatic release bottle nearest the DG-1 Room Security door releases the CO_2 bottles in the DG-1 Room into the DG-1 Room. Actuation of the other pneumatic release bottle near the DG-1 Room Security door releases the DG-2 Room CO_2 bottles into the DG-1 Room. Actuation of the pneumatic release bottle on the west wall inside the DG-1 Room releases the CO_2 bottles in the DG-2 Room into the DG-2 Room. Actuation of the pneumatic release bottle nearest the Boiler Room entrance to the DG Building releases the CO_2 bottles in the DG-2 Room into the DG-2 Room. Actuation of the other pneumatic release bottle near the Boiler Room entrance to the DG Building releases the DG-1 Room CO_2 bottles into the DG-2 Room. Actuation of the pneumatic release bottle on the west wall inside the DG-2 Room releases the CO_2 bottles in the DG-1 Room into the DG-1 Room.

1.2.5 During normal plant operations, upon initiation of the DG CO_2 System, the DG H&V fans will be isolated via Pressure Switches CO_2-PS-CO1(CO2). During an emergency start of the DG, Pressure Switches CO_2-PS-CO1(CO2) will be bypassed and the DG H&V fans will continue to run even upon a CO_2 actuation. Furthermore, the Diesel Isolation Switches IS/DG1A(DG2A) provide a bypass of Pressure Switches CO_2-PS-CO1(CO2). During the event of the isolation switches in the ISOLATE position, the CO_2 pressure switches will be bypassed and the DG H&V fans will continue to run even upon a CO_2 actuation.

PROCEDURE 2.2.2	REVISION 39	PAGE 17 OF 20

ATTACHMENT 2 INFORMATION SHEET

ATTACHMENT 2 INFORMATION SHEET

1. DISCUSSION

1.1 FUNCTION

1.1.1 The turbine bearing CO_2 System, upon manual or automatic actuation, causes a discharge of liquid and gaseous CO_2 into the Turbine Bearing 1, 2, and 3 areas. Upon manual actuation of any CO_2 hose station, the hose stations in the Reactor Building MG Set Room, Control Building Cable Spreading Room, Control Room Entrance, and Non-Critical Switchgear Room are charged with CO_2. This system will also provide CO_2 to purge the main generator.

1.1.2 The high pressure CO_2 System, upon manual or automatic actuation, causes a discharge of liquid and gaseous CO_2 to totally flood a Diesel Generator Room and the associated Diesel Fuel Oil Day Tank Room.

1.2 OPERATING CHARACTERISTICS

1.2.1 The low pressure CO_2 System consists of a liquid CO_2 storage tank located in the basement of the Turbine Generator Building. The CO_2 is piped to the turbine bearings and four manual hose stations. The turbine bearing CO_2 System is actuated by temperature switches or by either of two MANUAL pushbuttons located on the northwest corner of the turbine shield wall or near the entrance to the front standard on the turbine shield wall. The bearings are sprayed for 50 seconds after a 30 second time delay following initiation. Repeated timed sprays are initiated by pressing the MANUAL pushbuttons or the RESET button in the Control Room if the local bearing temperature is > 400°F. If the bearing area temperatures are < 400°F, the Control Room RESET button will reset the system for normal operation. The manual hose stations are charged automatically by removing the hose nozzle from its storage rack.

1.2.1.1 Due to tank design, 40% tank volume equates to 43% indicated level. Additionally, the installed float valve will shut off at 47% indicated tank level. This is to ensure the 40% tank volume required to ensure 120% of minimum amount of CO_2 required for assumed fire-fighting activities is protected.

Examination Outline Cross-Reference	**Level**	RO
Agree – this is a Bank question.	**Tier#**	1
	Group#	1
	K/A #	700000 AA2.04
	Rating	3.6

700000 Generator Voltage and Electric Grid Disturbances / 6

AA2. Ability to determine and/or interpret the following as they apply to GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES: (CFR: 41.5 and 43.5 / 45.5, 45.7, and 45.8)

AA2.04 VARs outside capability curve

Question 20

Reference Provided

The plant is operating at rated power when grid disturbances require the crew to enter 5.3GRID (Degraded Grid Voltage).

The following generator conditions are present:

- Generator MW is 810 MW.
- Generator hydrogen pressure is 60 psig.

The load dispatcher requests Cooper to maximize MVAR output and maintain current load to stabilize the grid.

What is the maximum MVAR output allowed during this grid disturbance IAW 5.3GRID?

A. +475 MVARs

B. +500 MVARs

C. +525 MVARs

D. +560 MVARs

Answer:
C. +525 MVARs

Explanation:
Since the crew has entered 5.3GRID and high MVAR output is required to support the grid the power factor limits of the generator capability curve are allowed to be exceeded. The gas pressure limits however still require compliance. Therefore it is allowable to go to +525 MVAR. This exceeds the power factor limit but is on the line with regards to the gas pressure limit.

Distracters:
A. This answer is incorrect due to Max VARs being +525 MVARs. This choice is plausible due to 475 MVARs being the limit if pf of .866 is considered limiting. The candidate that interprets pf vs.H2 pressure as limiting would select this answer.
B. This answer is incorrect due to Max VARs being +525 MVARs. This choice is plausible due to 500 MVARs being the limit if pf of .85 is considered limiting. The candidate that interprets pf vs.H2 pressure as limiting would select this answer.
D. This answer is incorrect due to Max VARs being +525 MVARs. This choice is plausible due to 560 MVARs being the limit if the curve for pf of .800 is confused with 800MW. The candidate that .800 pf with 800 MW would select this answer.

Technical References:
Procedure 5.3GRID (Degraded Grid Voltage), Rev. 42

References to be provided to applicants during exam: 5.3GRID, Attachment 2 (Main Generator Capability Curve)

Learning Objective:
COR0011302001110B Concerning the Main Generator capability curves: Given generator conditions and the Main Generator capability curve, determine if operation is within the acceptable region of the curve.

Question Source:	Bank # 12581	
(note changes; attach parent)	Modified Bank #	
	New	
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b) 5	
Level of Difficulty:	3	
SRO Only Justification:	N/A	

QUESTION: 10 12581 (1 point(s))
The plant is operating at near rated power when grid disturbances require the crew to enter **5.3GRID.**

The following generator conditions are present:

- Generator MW is 810 MW.
- Generator hydrogen pressure is 60 psig for power uprate test.

The load dispatcher requests that in order to stabilize the grid Cooper maximize MVAR output and maintain current load

What is the maximum MVAR output allowed during **this grid disturbance**?

a. +450 MVARs

b. +475 MVARs

c. +500 MVARs

d. +525 MVARs

ANSWER: 10 12581
d. +525 MVARs

Explanation:

Since the crew has entered 5.3GRID and high MVAR output is required to support the grid the power factor limits of the generator capability cure are allowed to be exceeded. The gas pressure limits however still require compliance. Therefore it is allowable to go to +525 MVAR. This exceeds the power factor limit but is on the line with regards to the gas pressure limit.

4. SUBSEQUENT OPERATOR ACTIONS

4.1 Record current time and date. Time/Date: ______/________

NOTE 1 – Operation outside Power Factor (PF) limits, but within appropriate H_2 curve(s), of Attachment 2, MAIN GENERATOR CAPABILITY CURVE (PMIS01), is allowed during abnormal operation to support grid.

NOTE 2 – Normal operation must be maintained within Attachment 2 and the following:

- Appropriate H_2 curve(s).
- 0.85 LAGGING PF line (positive or over-excited).
- 0.95 LEADING PF line (negative or under-excited).

4.2 Stop and/or suspend all activities that may reduce generating capability or cause a unit trip, or impacts critical electrical distribution.

4.3 IF an off-site power source becomes inoperable, degraded, or the risk of loss is significantly raised due to plant or environmental activity during power operation, THEN the Diesel Generators, HPCI, RCIC, and Critical DC buses should be maintained or returned to an available status as soon as practical.

4.4 IF procedure entry is from a Security Analysis violation, THEN validate with DCC it is a 4160V violation.

4.4.1 IF not a 4160V violation, THEN exit this procedure.

4.5 IF procedure entry is from any of following, THEN **concurrently** enter Attachment 1 (Page 7):

4.5.1 DCC notification entry condition.

4.5.2 Procedure 2.1.12.

4.6 Observe PANEL C 345 KV NEB CITY LINE VOLTS, 161 KV LINE VOLTS, and 69 KV LINE VOLTS indication.

4.7 IF voltage is degraded or oscillating on 161 kV and/or 69 kV line, THEN contact DCC System Operator for line status and predicted reliability.

NOTE – PMIS Points E002 and E003 are the preferred indication for Main Generator GROSS WATTS and GROSS VARS. PANEL C MAIN GENERATOR GROSS MEGAWATTS and MAIN GENERATOR MVAR meters are alternate indication.

4.8 IF 345 kV voltage rising coincident with Gen Exciter AMPS rising, THEN perform following:

4.8.1 Place GEN VOLTAGE REGULATOR switch to OFF.

4.8.1.1 Notify DCC System Operator that GEN VOLTAGE REGULATOR is off.

ATTACHMENT 2 MAIN GENERATOR CAPABILITY CURVE (PMIS01)

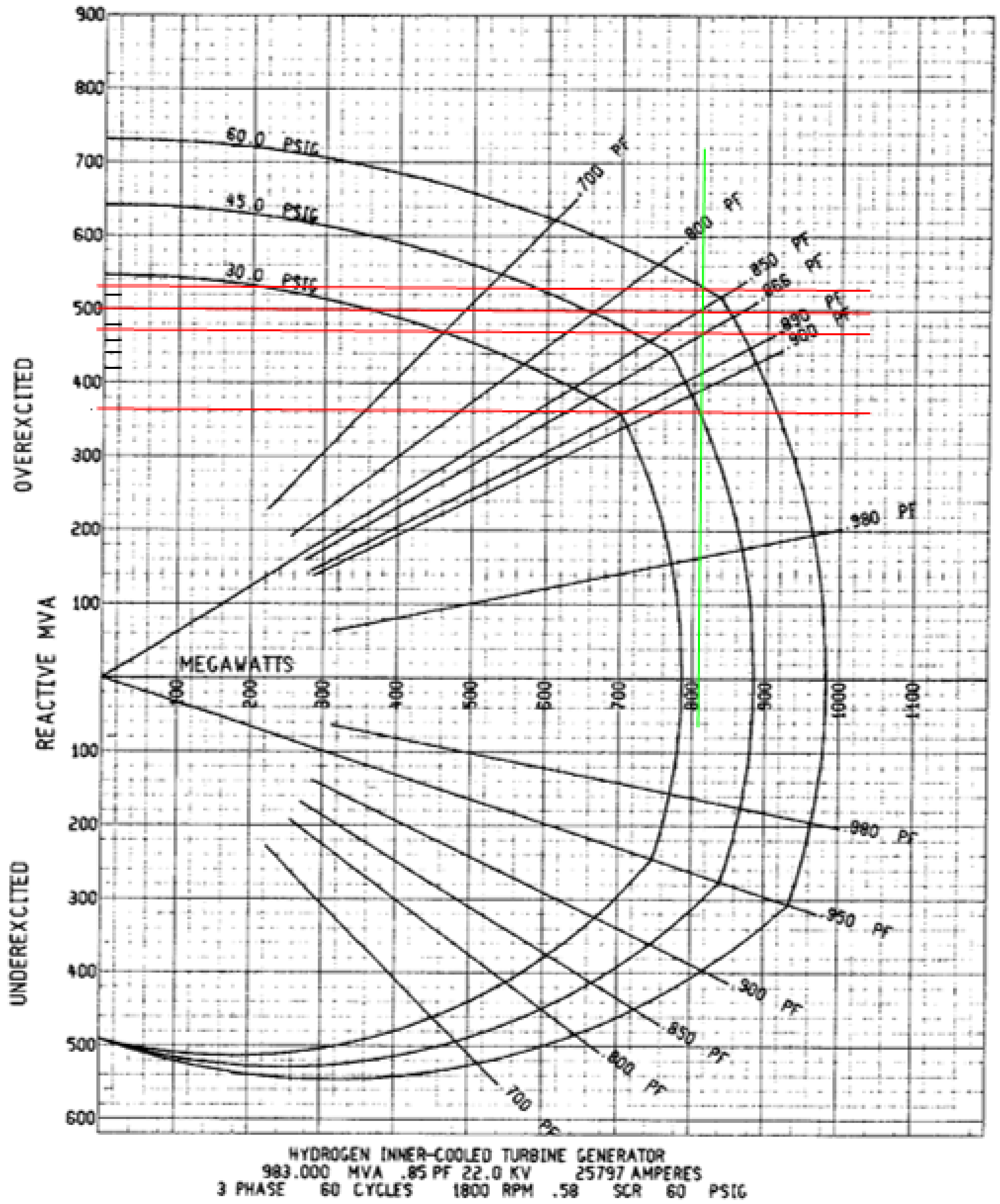


Figure 1

Examination Outline Cross-Reference	Level	RO
Procedure OI-8 provides SRV D pressure band of 855 psig to 992 psig – changed first part of Answers A & B to 992 psig to be consistent with actual setpoints.	Tier#	1
	Group#	2
	K/A #	295007 AK3.04
	Rating	4.0

295007 High Reactor Pressure / 3

AK3. Knowledge of the reasons for the following responses as they apply to HIGH REACTOR PRESSURE : (CFR: 41.5 / 45.6)

AK3.04 Safety/relief valve operation: Plant Specific

Question 21

The reactor is at rated power when reactor pressure rises to 1080 psig.

Which one of the following completes the statement below describing the expected response of the Safety Relief Valves (SRVs) and the reason for having the Low-Low Set response?

Following initial SRV opening, SRV D cycles to control reactor pressure between 855 and ____(1)____ psig, in order to ____(2)____.

A. (1) 992
(2) minimize heat addition to the torus

B. (1) 992
(2) reduce the thrust loading on SRV discharge piping

C. (1) 1015
(2) minimize heat addition to the torus

D. (1) 1015
(2) reduce the thrust loading on SRV discharge piping

Answer:

B. (1) 992
(2) reduce the thrust loading on the SRV discharge piping

Explanation:

LLS arms at < 1050 psig (TS) and the Rx auto Scrams. After the initial opening of both SRVs, SRV "D" will cycle between 855 and 992 psig (TS ≥966.5 & ≤1010 psig OPEN and ≥835 & ≤875.5 CLOSE) to reduce the thrust loading on the SRV discharge piping.

Distracters:

A This answer is incorrect due to the purpose of LLS logic is to reduce thrust loading on the SRV discharge piping. This choice is plausible due to a single SRV cycling could be interpreted as minimizing the heat addition to the torus. The candidate that recognizes the correct SRV opening setpoint and confuses the reason for having LLS logic would select this answer.

C. This answer is incorrect due to the opening setpoint being 992 psig (<1010 psig TS) and the purpose of LLS logic is to reduce thrust loading on the SRV discharge piping. This choice is plausible due to 1015 psig being the previous OPEN setpoint and a single SRV cycling could be interpreted as minimizing the heat addition to the torus. The candidate that does not remember the correct SRV opening setpoint and confuses the reason for having LLS logic would select this answer.

D. This answer is incorrect due to the opening setpoint being 992 psig (<1010 psig TS). This choice is plausible due to 1015 psig being the previous OPEN setpoint. The candidate that does not remember the correct SRV opening setpoint and correctly identifies the reason for having LLS logic would select this answer.

Technical References:

TS 3.3.8.3 Low-Low Set (LLS) Instrumentation

Operations Instruction #8 (Guideline For Successful Transient Mitigation) Rev. 14

References to be provided to applicants during exam: NONE

Learning Objective:

COR0021602001030J Describe the interrelationships between the Nuclear Pressure Relief system and the following: RPS (low-low set initiation)

COR0021602001050C Describe the Nuclear Pressure Relief system design features and/or interlocks that provide for the following: Minimizes SRV cyclic operation

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank # 19078	
	New	
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b) 5	
Level of Difficulty:	3	
SRO Only Justification:	N/A	

QUESTION: 100 19078 (1 point(s))

The reactor is at 100% power when a transient causes reactor pressure to initially increase to 1080 psig.

Which of the following describes the expected response of the safety relief valves including the reason for this response?

Both SRV "D" & "F" open initially...

a. then both SRVs will cycle to control reactor pressure between 885 and 1025 psig to reduce the thrust loading on the SRV discharge piping.

b. then both SRVs will cycle to control reactor pressure between 875 and 1015 psig to minimize heat addition to the torus.

c. then SRV "D" will cycle to control reactor pressure between 875 and 1015 psig to reduce the thrust loading on the SRV discharge piping.

d. then SRV "D" will cycle to control reactor pressure between 885 and 1025 psig to minimize heat addition to the torus.

ANSWER: 100 19078

c. then SRV "D" will cycle to control reactor pressure between 875 and 1015 psig to reduce the thrust loading on the SRV discharge piping.

LLS Instrumentation
3.3.6.3

Table 3.3.6.3-1 (page 1 of 1)
Low-Low Set Instrumentation

FUNCTION	REQUIRED CHANNELS PER FUNCTION	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Reactor Pressure - High	1 per LLS valve	SR 3.3.6.3.3 SR 3.3.6.3.4 SR 3.3.6.3.5	≤ 1050 psig
2. Low-Low Set Pressure Setpoints	2 per LLS valve	SR 3.3.6.3.3 SR 3.3.6.3.4 SR 3.3.6.3.5	Low: Open ≥ 986.5 psig and ≤ 1010 psig Close ≥ 835 psig and ≤ 875.5 psig High: Open ≥ 996.5 psig and ≤ 1040 psig Close ≥ 835 psig and ≤ 875.5 psig
3. Discharge Line Pressure Switch	1 per SRV	SR 3.3.6.3.1 SR 3.3.6.3.2 SR 3.3.6.3.4 SR 3.3.6.3.5	≥ 25 psig and ≤ 55 psig

Operations Instruction #8 GUIDELINE FOR SUCCESSFUL TRANSIENT MITIGATION	Class: Information Use Effective: 06/10/15

REACTOR PRESSURE	AUTO ACTION
1240 ± 3% psig	Safety Valves lift setpoint.
1100 ± 3% psig 1090 ± 3% psig 1080 ± 3% psig	Safety Relief Valve's safety function lift setpoint.
~ 1060 psig (Tech Spec ≤ 1072 psig)	ARI and ATWS RPT.
~ 1034 psig (Tech Spec ≤ 1050 psig)	Reactor Scram. Low - Low Set is activated by Reactor High pressure with a SRV discharge line pressure switch.
High Open ~ 1022 psig Low Open ~ 992 psig Close ~ 855 psig (Tech Spec High Open ≥ 996.5 and ≤ 1040 psig Low Open ≥ 966.5 and ≤ 1010 psig Close ≥ 835 and ≤ 875.5 psig)	Low - Low Set SRV operating setpoints.
~ 1025 psig	Reactor high pressure alarm.
≤1020 psig	Technical Specification Reactor Pressure Limit
~ 848 psig (Tech Spec ≥ 835 psig)	Group 1 Main Steam Line Low Pressure with Reactor Mode Switch in RUN.
~ 400 psig (Tech Spec ≥ 291 and ≤ 436 psig)	ECCS Injection Valve permissive.
~ 210 psig (Tech Spec ≥ 199 and ≤ 246 psig)	Recirc Discharge Valve Permissive. RR-MO-53A(B) goes closed with LPCI Injection signal present.
~ 67 psig (Tech Spec ≤ 72 psig)	RHR Shutdown Cooling Isolation.

OPERATIONS INSTRUCTION #8	REVISION 14	PAGE 15 OF 15

ATTACHMENT 1 INFORMATION SHEET

1. DISCUSSION

1.1 FUNCTION

1.1.1 ADS logic and instrumentation is designed to lower the reactor pressure during postulated conditions so that reflooding of the core can take place by the low pressure CSC Systems.

1.1.2 The LLS logic and instrumentation is designed to mitigate the effects of postulated thrust loads on the safety/relief valve (SRV) discharge lines by preventing subsequent actuations with an elevated water leg in the SRV discharge line. It also mitigates the effects of postulated pressure loads on suppression chamber structural components by preventing multiple actuations in rapid succession of the SRVs subsequent to their initial actuation.

1.1.3 The safety/relief valves and safety valves provide over pressure relief protection and over pressure safety protection by opening (self-actuated) at a pre-determined pressure in the main steam line.

1.1.4 The relief valves may be manually opened by positioning switches in the Control Room when the reactor pressure is > 50 psig.

1.2 OPERATING CHARACTERISTICS

1.2.1 The ADS serves as a backup to the HPCI System under Loss Of Coolant Accident conditions. If the water level lowers to the initiation setpoint level and does not recover, a 109 second time delay relay energizes and starts timing. At the end of the time delay, if a low pressure CSCS pump is developing sufficient discharge pressure (AC interlock) to inject into the reactor vessel, relief valves MS-RV-71A, MS-RV-71B, MS-RV-71C, MS-RV-71E, MS-RV-71G, and MS-RV-71H open. This vents reactor steam to the suppression chamber; thereby, lowering the reactor pressure where the CS or RHR pumps are able to inject water.

1.2.2 In the event of a Group 4 isolation where the HPCI System is not available, the RCIC System can be used to restore water level. However, the RCIC System does not have sufficient capacity to restore the water level in the time period allowed and actuation of the ADS valves will occur unless manually defeated. The manual INHIBIT switches allow the Operator to prevent ADS actuation if the RCIC System is in operation and restoring vessel water level or if directed by the EOPs.

Examination Outline Cross-Reference	Level	R0
Removed SF & FF values and modified chart indication to reflect a constant SF/FF mismatch below 80% power. Explanation corrected.	Tier#	1
	Group#	2
	K/A #	295009 AA2.02
	Rating	3.6

295009 Low Reactor Water Level / 2

AA2. Ability to determine and/or interpret the following as they apply to LOW REACTOR WATER LEVEL : (CFR: 41.10 / 43.5 / 45.13)

AA2.02 Steam flow/feed flow mismatch

Examination Outline Cross-Reference	Level	R0
Removed SF & FF values and modified chart indication to reflect a constant SF/FF mismatch below 80% power. Explanation corrected.	Tier#	1
	Group#	2
	K/A #	295009 AA2.02
	Rating	3.6

295009 Low Reactor Water Level / 2

AA2. Ability to determine and/or interpret the following as they apply to LOW REACTOR WATER LEVEL : (CFR: 41.10 / 43.5 / 45.13)

AA2.02 Steam flow/feed flow mismatch

Question 22

The plant is operating at power with the following valid indications:

What is the impact on RPV Water Level if these trends continue?
(Consider the effect of Steam Flow – Feed Flow mismatch only)

A. RPV Water Level lowers to the Scram setpoint.

B. RPV Water Level rises to the Turbine Trip setpoint.

C. RPV Water Level rises to a stable level higher than the level control setpoint.

D. RPV Water Level lowers to a stable level lower than the level control setpoint.

Answer:		
A. RPV Water Level lowers to the Scram setpoint.		
Explanation: The indications provide for a final steam flow/feed flow mismatch in which more inventory is being taken away from the reactor than feed water can makeup. Since feedwater flow is lower than steam flow RPV water level lowers. RPV water level will continue to lower to +3 inches which causes a reactor Scram.		
Distracters: B. This answer is incorrect due RPV Water Level lowering to the Scram setpoint. This choice is plausible due to confusing FF and SF indications. The candidate that confuses SF FF indications would select this answer. C. This answer is incorrect due RPV Water Level lowering to the Scram setpoint. This choice is plausible due to confusing FF and SF indications with SF stabilizing at a lower value. The candidate that confuses SF FF indications and incorrectly assumes level will remain stable at a higher level would select this answer. D. This answer is incorrect due RPV Water Level lowering to the Scram setpoint. This choice is plausible due to common misconception of the impact of this SF – FF mismatch. The candidate that confuses SF-FF mismatch impact on RPV level would select this answer.		
Technical References: Procedure 4.4.1 (Reactor Vessel Level Control System), Rev. 7		
References to be provided to applicants during exam: NONE		
Learning Objective:		
COR0023202001050B Briefly describe the following concepts as they apply to the RVLC system: Steam flow/Feed Flow Mismatch		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b) 10	

Level of Difficulty:	2
SRO Only Justification:	N/A

2. INTERLOCKS AND SETPOINTS

2.1 If both reactor recirculation pumps are running and not locked out, RR pumps run back towards 45% speed if any of the following conditions are met:

2.1.1 Total steam flow > 8.25 Mlbm/hr concurrent with a condensate pump low discharge header pressure < 110 psig and a condensate pump tripped.

2.1.2 Total steam flow > 8.25 Mlbm/hr concurrent with a condensate booster pump low discharge header pressure < 450 psig and a condensate booster pump tripped.

2.1.3 Total steam flow > 8.25 Mlbm/hr and RFP suction pressure < 350 psig.

2.1.4 Total steam flow > 9 Mlbm/hr with at least 1 RFP tripped/flow < 1 Mlbm/hr and selected reactor water level < 27.5".

2.2 The following conditions will automatically shift from 3 element to 1 element:

2.2.1 MASTER LEVEL controller taken to MAN.

2.2.2 Wide FW flow transmitter is invalid when associated RFP is in AUTO.

2.2.3 Three steam flow elements are invalid.

2.2.4 Turbine 1st stage flow invalid with less than four steam flow elements.

2.2.5 Both individual RFP controls not in AUTO.

2.2.6 Total steam flow < 1 Mlbm/hr.

2.2.7 All Reactor vessel water level elements are invalid.

Examination Outline Cross-Reference	Level	RO
Added "IAW Procedure 2.2.8" in stem to eliminate SR conflict. Corrected distractor explanations. The accumulator is operable otherwise scram insertion times would not be assured.	Tier#	1
	Group#	2
	K/A #	295022 AK1.01
	Rating	3.3

295022 Loss of CRD Pumps / 1

AK1. Knowledge of the operational implications of the following concepts as they apply to LOSS OF CRD PUMPS: (CFR: 41.8 to 41.10)

AK1.01 Reactor pressure vs. rod insertion capability

Question 23

Which one of the following identifies the LOWEST reactor pressure which will fully insert a control rod within the TS Allowable Control Rod Scram Time with NO CRD pumps available IAW Procedure 2.2.8 (Control Rod Drive Hydraulic System)?

A. 400 psig

B. 600 psig

C. 900 psig

D. 960 psig

Answer:

C. 900 psig

Explanation:

With one or more control rod Scram accumulators inoperable and the reactor steam dome pressure < 900 psig, the pressure supplied to the charging water header must be adequate to ensure that accumulators remain charged. With the reactor steam dome pressure < 900 psig, the function of the accumulators in providing the Scram force becomes much more important since the Scram function could become severely degraded during a depressurization event or at low reactor pressures.

Distracters:

A. This answer is incorrect due to the lowest pressure being 900 psig. This choice is plausible due to 400 psig being the Low Pressure ECCS injection valve open permissive pressure. The candidate that confuses ECCS injection valve opening pressure vs. reactor pressure would select this answer.

B. This answer is incorrect due to the lowest pressure being 900 psig. This choice is plausible due to 600 psig being the maximum pressure a CRD accumulator can be charged to. The candidate that confuses CRD accumulator charge pressure vs. reactor pressure would select this answer.

D. This answer is incorrect due to the lowest pressure being 900 psig. This choice is plausible due to 960 psig being CRD accumulator low pressure alarm setpoint. The candidate that confuses CRD accumulator low pressure alarm vs. reactor pressure would select this answer.

Technical References:
Procedure 2.2.8 (Control Rod Drive Hydraulic System), Rev. 91

References to be provided to applicants during exam: NONE

Learning Objective:
COR002040213. Describe the interrelationships between the Control Rod Drive Hydraulic system (CRDH) and the following:
g. Control Rod Drive Mechanisms

Question Source:	Bank # 926	
(note changes; attach parent)	Modified Bank #	
	New	
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b) 8	
Level of Difficulty:	3	
SRO Only Justification:	N/A	

QUESTION: 133 926 (1 point(s))

What is the minimum reactor pressure required (per Tech Specs) in order to meet minimum control rod scram insertion times even without HCU accumulators?

a. 700 psig

b. 800 psig

c. 900 psig

d. 960 psig

ANSWER: 133 926
c. 900 psig

1.2.5.1 Failure of the CRD pumps stops cooling water flow to the drives which shortens seal life. Failure of the CRD pumps also allows the CRD hydraulic accumulators to slowly discharge through inlet scram valve and/or charging water header check valve leakage. This is of special concern during the initial stages of plant startup when reactor pressure is insufficient to accomplish a scram. Below ~ 900 psig reactor pressure, the hydraulic accumulators are required to ensure any withdrawn control rods are fully scrammed within the required time.

1.2.5.2 Elevated cooling water differential pressure relative to reactor pressure may result in inadvertent control rod movement. Elevated cooling water differential pressure above 60 psid can cause control rod drive insertion. Excessive differential pressure above 35 psid can interfere with control rod settle and latch function. Subsequently, with the rod not properly latched, control rod drive withdrawal has occurred due to direction control valve failure and during isolation of the associated hydraulic control unit (HCU). The elevated cooling water differential pressure can be a result of isolating large numbers of hydraulic control units without relieving or eliminating the pressure developed by the control rod drive hydraulic system pump.Ⓟ[7]

8.24 Fully open CRD-111(XX-XX).

CAUTION – Exceeding 600 psig on PI-131 could result in internal CRDM damage in the event of a reactor scram.

8.25 Slowly open charging hose rig charging valve to obtain a pressurization rate ≤ 100 psig per minute allowing for pressure on PI-131 to stabilize at or just below line on Attachment 1 and not to exceed 600 psig on PI-131.Ⓟ[4]

8.25.1 If necessary, adjust regulator to obtain desired charging rate.

8.26 IF N_2 bottle requires change out, THEN perform following:

8.26.1 Close charging hose rig charging valve.

8.26.2 Close N_2 bottle supply valve.

8.26.3 Close N_2 bottle to manifold shutoff opened in Step 8.8.

8.26.4 Adjust N_2 regulator handle fully counter-clockwise.

8.26.5 Disconnect regulator from bottle.

8.26.6 Replace bottle.

8.26.7 Connect regulator to bottle.

8.26.8 Open N_2 bottle supply valve.

8.26.9 Adjust N_2 bottle regulator to obtain ~ 600 psig on N_2 bottle regulator gauge.

8.26.10 Open desired N_2 bottle to manifold shutoff valve.

8.26.10.1 N_2-22 (North).

8.26.10.2 N_2-23 (North).

8.26.10.3 N_2-24 (North).

8.26.10.4 N_2-16 (South).

8.26.10.5 N_2-17 (South).

8.26.10.6 N_2-18 (South).

8.26.11 Return to Step 8.25.

8.27 WHEN pressure on PI-131 is stable at or just below line on Attachment 1 and ≤ 600 psig, THEN close charging hose rig charging valve.

8.28 Close CRD-111(XX-XX).

Operations Instruction #8 GUIDELINE FOR SUCCESSFUL TRANSIENT MITIGATION	Class: Information Use Effective: 06/10/15

REACTOR PRESSURE	AUTO ACTION
1240 ± 3% psig	Safety Valves lift setpoint.
1100 ± 3% psig 1090 ± 3% psig 1080 ± 3% psig	Safety Relief Valve's safety function lift setpoint.
~ 1060 psig (Tech Spec ≤ 1072 psig)	ARI and ATWS RPT.
~ 1034 psig (Tech Spec ≤ 1050 psig)	Reactor Scram. Low - Low Set is activated by Reactor High pressure with a SRV discharge line pressure switch.
High Open ~ 1022 psig Low Open ~ 992 psig Close ~ 855 psig (Tech Spec High Open ≥ 996.5 and ≤ 1040 psig Low Open ≥ 966.5 and ≤ 1010 psig Close ≥ 835 and ≤ 875.5 psig)	Low - Low Set SRV operating setpoints.
~ 1025 psig	Reactor high pressure alarm.
≤1020 psig	Technical Specification Reactor Pressure Limit
~ 848 psig (Tech Spec ≥ 835 psig)	Group 1 Main Steam Line Low Pressure with Reactor Mode Switch in RUN.
~ 400 psig (Tech Spec ≥ 291 and ≤ 436 psig)	ECCS Injection Valve permissive.
~ 210 psig (Tech Spec ≥ 199 and ≤ 246 psig)	Recirc Discharge Valve Permissive. RR-MO-53A(B) goes closed with LPCI Injection signal present.
~ 67 psig (Tech Spec ≤ 72 psig)	RHR Shutdown Cooling Isolation.

CRD ACCUM LOW PRESS OR HIGH LEVEL	PANEL/WINDOW: 9-5-2/G-6

1. OPERATOR OBSERVATION AND ACTION

   1.1 Enter applicable LCO 3.1.5, Conditions and Required Actions.

   1.2 Correct cause per Procedure 2.2.8.

SETPOINT	CIC	9-5-2/G-6
1. (2756) CRD ACCUM LOW PRESSURE at 960 psig	1. CRD-PS-130(XX-YY)	
2. (2756) CRD ACCUM HIGH LEVEL at 37 cc	2. CRD-LS-129(XX-YY)	

PROBABLE CAUSES

- Instrument block seal leakage.

REFERENCES

- Technical Specifications LCO 3.1.5, Control Rod Scram Accumulators.
- System Operating Procedure 2.2.8, Control Rod Drive Hydraulic System.

BASES

SURVEILLANCE REQUIREMENTS

The four SRs of this LCO are modified by a Note stating that during a single control rod scram time surveillance, the CRD pumps shall be isolated from the associated scram accumulator. With the CRD pump isolated, (i.e., charging valve closed) the influence of the CRD pump head does not affect the single control rod scram times. During a full core scram, the CRD pump head would be seen by all control rods and would have a negligible effect on the scram insertion times.

SR 3.1.4.1

The scram reactivity used in DBA and transient analyses is based on an assumed control rod scram time. Measurement of the scram times with reactor steam dome pressure ≥ 800 psig demonstrates acceptable scram times for the transients analyzed in References 3 and 4.

Maximum scram insertion times occur at a reactor steam dome pressure of approximately 800 psig because of the competing effects of reactor steam dome pressure and stored accumulator energy. Therefore, demonstration of adequate scram times at reactor steam dome pressure ≥ 800 psig ensures that the measured scram times will be within the specified limits at higher pressures. Limits are specified as a function of reactor pressure to account for the sensitivity of the scram insertion times with pressure and to allow a range of pressures over which scram time testing can be performed. To ensure that scram time testing is performed within a reasonable time following a shutdown duration of ≥ 120 days, control rods are required to be tested before exceeding 40% RTP following the shutdown. This Frequency is acceptable considering the additional surveillances performed for control rod OPERABILITY, the frequent verification of adequate accumulator pressure, and the required testing of control rods affected by fuel movement within the associated core cell and by work on control rods or the CRD System.

Examination Outline Cross-Reference	Level	RO
Revised question to test when and why DW sprays are secured due to high PC level. Deleted “s” from Sprays.	Tier#	1
	Group#	2
	K/A #	295029 EK1.01
	Rating	3.4

295029 High Suppression Pool Wtr Lvl / 5

EK1. Knowledge of the operational implications of the following concepts as they apply to HIGH SUPPRESSION POOL WATER LEVEL : (CFR: 41.8 to 41.10)

EK1.01 Containment integrity

Question 24

Which one of the following completes the statement below regarding when and why Drywell Spray is secured due to high Primary Containment water level?

Drywell Spray is required to be secured if PC water level cannot be maintained below a MINIMUM of ____(1)____ feet to ensure the ____(2)____ vacuum breakers function as designed.

A. (1) 16.0
(2) Reactor Building to Suppression Chamber

B. (1) 16.0
(2) Suppression Chamber to Drywell

C. (1) 16.5
(2) Reactor Building to Suppression Chamber

D. (1) 16.5
(2) Suppression Chamber to Drywell

Answer:
D. (1) 16.5 (2) Suppression Chamber to Drywell
Explanation:
Drywell spray is not allowed and is secured if PC level reaches 16.5 feet to ensure the function of the Suppression Chamber to Drywell vacuum breakers is maintained. If

submerged, the vacuum breakers cannot function as designed to relieve noncondensibles into the drywell and equalize drywell and suppression chamber pressures. Spray initiation is therefore permitted only when suppression pool water level is below the bottom of the vacuum breaker openings to protect primary containment integrity.

Distracters:

A. This answer is incorrect due to PC level above 16.5 feet prevents operation of the Suppression chamber to Drywell vacuum breakers. This choice is plausible due to 16 feet being the level requiring ED and securing sources of injection taking suction outside of PC and the response of Suppression chamber spray being easily confused with Drywell spray. With a high PC water level and high PC pressures, spray operation could reduce pressure more rapidly causing vacuum breaker operation. The candidate that confuses PC level requiring ED and RB to Suppression Chamber vs. Suppression Chamber to Drywell vacuum breaker operation would select this answer.

B This answer is incorrect due to PC level above 16.5 feet prevents operation of the Suppression chamber to Drywell vacuum breakers. This choice is plausible due to 16 feet being the level requiring ED and securing sources of injection taking suction outside of PC The candidate that confuses PC level requiring ED and correctly identifies the reason being Suppression Chamber to Drywell vacuum breaker operation would select this answer.

C. This answer is incorrect due to PC level above 16.5 feet prevents operation of the Suppression chamber to Drywell vacuum breakers. This choice is plausible due to the response of Suppression chamber spray being easily confused with Drywell spray. With a high PC water level and high PC pressures, spray operation could reduce pressure more rapidly causing vacuum breaker operation. The candidate that correctly identifies the PC level and confuses RB to Suppression Chamber vs. Suppression Chamber to Drywell vacuum breaker operation would select this answer.

Technical References:

CNS PSTG/SATG Appendix B Technical Bases, Rev. 8
EOP 3A (PCCP), Rev.15

References to be provided to applicants during exam: NONE

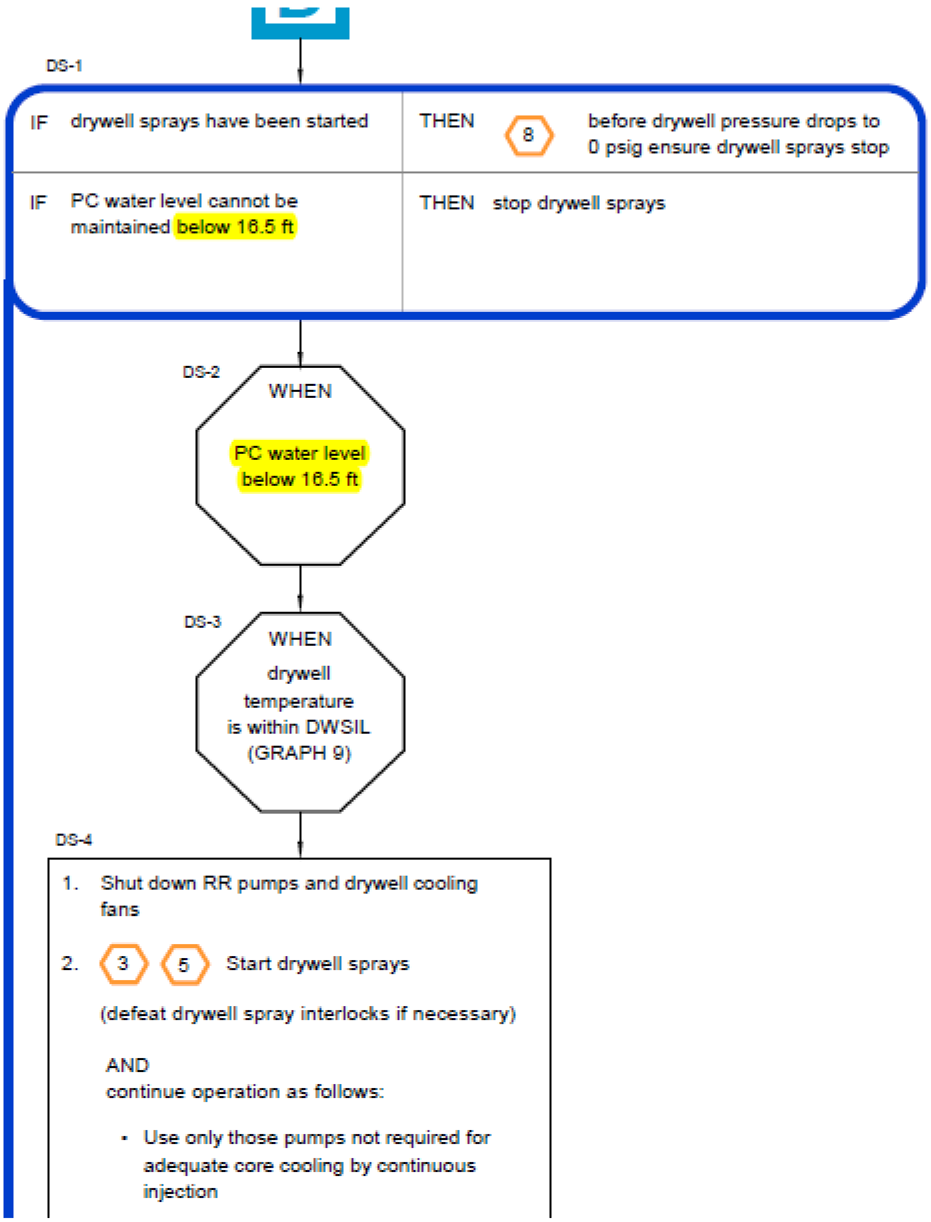
Learning Objective:

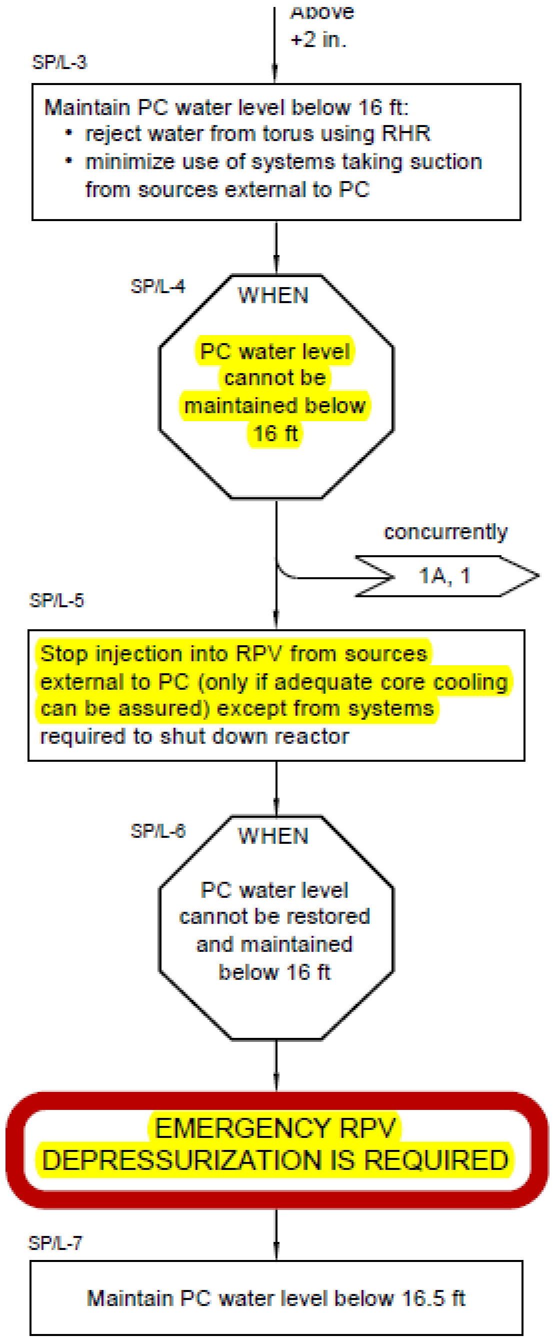
INT00806130011200 Given plant conditions and EOP flowchart 3A, PRIMARY CONTAINMENT CONTROL, state the reasons for the actions contained in the steps.

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X

Question Cognitive Level:	Memory/Fundamental	

	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b) 9	
Level of Difficulty:	2	
SRO Only Justification:	N/A	




Above
+2 in.
SP/L-3
Maintain PC water level below 16 ft:
• reject water from torus using RHR
• minimize use of systems taking suction from sources external to PC
SP/L-4
WHEN
PC water level cannot be maintained below 16 ft
concurrently
1A, 1
SP/L-5
Stop injection into RPV from sources external to PC (only if adequate core cooling can be assured) except from systems required to shut down reactor
SP/L-6
WHEN
PC water level cannot be restored and maintained below 16 ft
EMERGENCY RPV DEPRESSURIZATION IS REQUIRED
SP/L-7
Maintain PC water level below 16.5 ft

PSTG/SATG Step (Fourth PC override)

If while executing the following steps Drywell Spray is required and drywell sprays are not operating and suppression pool water level is below 16.68 ft (elevation of bottom of internal suppression chamber to drywell vacuum breakers less vacuum breaker opening pressure in feet of water) and drywell temperature is below the Drywell Spray Initiation Limit, shut down recirculation pumps and drywell cooling fans and initiate drywell sprays using only those pumps not required to assure adequate core cooling by continuous injection, defeating drywell spray interlocks if necessary. Sources external to the primary containment may be used only if primary containment water level and drywell pressure can be restored and maintained below Primary Containment Pressure Limit A.	❸ ❺

Discussion

Drywell spray is may be required in Step DW/T-2 or PC/P-2 of the Primary Containment Control guideline to reduce drywell temperature and pressure.

Drywell sprays are initiated to effect a reduction in drywell temperature and pressure, to control hydrogen and oxygen concentrations in the drywell, and to mitigate the effects of a deflagration should one occur.

Spray operation effects a drywell pressure and temperature reduction through the combined effects of evaporative cooling and convective cooling. In evaporative cooling the water spray undergoes a change of state, liquid to vapor, whereas convective cooling involves no change of state.

Evaporative cooling occurs when water is sprayed into a dry or superheated atmosphere and continues until the atmosphere is saturated. The water at the surface of each droplet flashes to steam, absorbing the heat of vaporization from the surrounding atmosphere. This process can result in a relatively large drop in drywell pressure and may occur at a rate faster than can be compensated for by operator action.

Convective cooling occurs when water is sprayed into a saturated atmosphere. The sprayed water droplets absorb sensible heat from the surrounding atmosphere through convective heat transfer, reducing drywell ambient temperature and pressure until equilibrium conditions are established. This process occurs at a rate much slower than evaporative cooling and can be controlled by terminating sprays.

The Drywell Spray Initiation Limit (DWSIL) is the highest drywell temperature at which initiation of drywell sprays will not result in an evaporative cooling pressure drop to below the high drywell pressure scram setpoint. The final pressure following evaporative cooling is limited to the scram setpoint to ensure that the operator has time to terminate sprays before convective cooling reduces pressure below 0 psig. The scram setpoint is used as a predefined, relatively low action level with sufficient margin to atmospheric pressure.

The DWSIL is a function of drywell pressure. It is utilized to preclude containment failure or de-inertion following initiation of drywell sprays. Refer to Section 16 of this appendix for a detailed discussion of the DWSIL.

CNS has internal suppression chamber-to-drywell vacuum breakers attached to the ring header at penetrations that are significantly below the top of the suppression chamber (i.e., a significant volume of noncondensibles could be trapped if the suppression chamber is flooded). If the penetrations are submerged, the vacuum breakers cannot function as designed to relieve noncondensibles into the drywell and equalize drywell and suppression chamber pressures. Spray initiation is therefore permitted only when suppression pool water level is below the bottom of the vacuum breaker openings.

The specified suppression pool water level is adjusted by subtracting a water column equivalent to the vacuum breaker opening pressure. The drywell side of the vacuum breakers will be submerged first, since water will be drawn up inside the downcomers as drywell pressure decreases. The adjustment ensures that no portion of the drywell side of the valves is submerged for any drywell below wetwell differential pressure less than or equal to the vacuum breaker opening setpoint.

Instructions to shut down recirculation pumps and drywell cooling fans is specified since continuing to operate this equipment in a spray environment has been evaluated as being more damaging than shutting it down.

Drywell spray is permitted only if the pumps to be used do not have to be operated continuously in the injection mode to assure adequate core cooling. Maintaining adequate core cooling takes precedence over initiating drywell spray since catastrophic failure of the primary containment is not expected under the conditions for which spray

Design negative pressure limits are in the 2 to 3 psig range and actual negative pressure capabilities are in excess of that. The expected pressure reduction rate is slow enough that, even considering instrument uncertainties, use of the described methodology to determine valid parameter values is sufficient to address negative pressure concerns.

Deinertion is only a concern for the Mark I&II containments. For LOCA events energy released into the drywell will pressurize the drywell and the suppression chamber. If drywell sprays are initiated, the suppression chamber to drywell vacuum breakers will operate when drywell pressure becomes less than suppression chamber

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Tech. Basis – App. B

pressure by the suppression chamber to drywell vacuum breaker opening differential pressure (typically 0.5 psid). This will not create a deinertion condition by itself because the suppression chamber atmosphere is still inerted. Drywell spray operation would need to continue until suppression chamber pressure became negative with respect to the secondary containment pressure by the reactor building to suppression chamber vacuum breaker opening differential pressure (typically 0.5 psid) before a deinertion condition could exist for the suppression chamber. Because of operator sensitivity to de-inerting the containment and the relative slow response of containment pressure in the convective cooling regime for drywell sprays, it is unlikely that this would occur.

Examination Outline Cross-Reference	Level	RO
Changed GE to Alert level (entry condition to EOP 5A) and added “ONLY” to “after” to eliminate inadvertent inclusion.	Tier#	1
	Group#	2
	K/A #	295033 EK3.02
	Rating	3.5

295033 High Secondary Containment Area Radiation Levels / 9

EK3. Knowledge of the reasons for the following responses as they apply to HIGH SECONDARY CONTAINMENT AREA RADIATION LEVELS : (CFR: 41.5 / 45.6)

EK3.02 Reactor SCRAM

Question 25

Secondary Containment radiation levels are rising due to a primary system leak while operating at power.

Which one of the following completes the statement below regarding when and why a reactor Scram is required?

A Reactor Scram is required ____(1)____ Secondary Containment radiation levels exceed Maximum Safe Operating Radiation value to ____(2)____.

A. (1) before
(2) reduce the driving head of the primary system discharging into secondary containment

B. (1) before
(2) ensure the reactor is shut down before radiation release rates exceed the values for an ALERT

C. (1) ONLY after
(2) reduce the driving head of the primary system discharging into secondary containment

D. (1) ONLY after
(2) ensure the reactor is shut down before radiation release rates exceed the values for an ALERT

Answer:
A. (1) before

(2) reduce the driving head of the primary system discharging into secondary containment

Explanation:
With a primary system discharging into secondary containment, the EPGs give the basis for when to Scram the plant and conduct an emergency depressurization. A reactor Scram is required prior to exceeding Max Safe Rad to reduce the driving head of a primary system that is discharging into the secondary containment and in anticipation of performing an emergency depressurization if radiation levels continue to rise. The rad release exceeding the ALERT level is the entry condition to EOP 5A which is easily confused with why a Scram is inserted.

Distracters:

B. This answer is incorrect due to the reason for inserting a Scram is to reduce the driving head. This choice is plausible due to radiation release rates above the ALERT level being the entry condition to EOP. The candidate that correctly identifies a Scram is required before exceeding Max Safe Rad and confuses EOP 5A entry vs. Scram reason would select this answer.

C. This answer is incorrect due to inserting a Scram being required before exceeding MSOR. This choice is plausible due to ED being required after 2 areas exceed MSOP. The candidate that confuses when a Scram is required and correctly identifies the reason would select this answer.

D. This answer is incorrect due to inserting a Scram being required before exceeding MSOR and the reason for inserting a Scram is to reduce the driving head. This choice is plausible due to ED being required after 2 areas exceed MSOP and radiation release rates above the ALERT level being the entry condition to EOP. The candidate that confuses when a Scram is required and confuses EOP 5A entry vs. Scram reason would select this answer.

Technical References:
Procedure EOP-5A Secondary Containment Control, Rev. 15
CNS PSTG/SATG Appendix B Technical Bases, Rev. 8

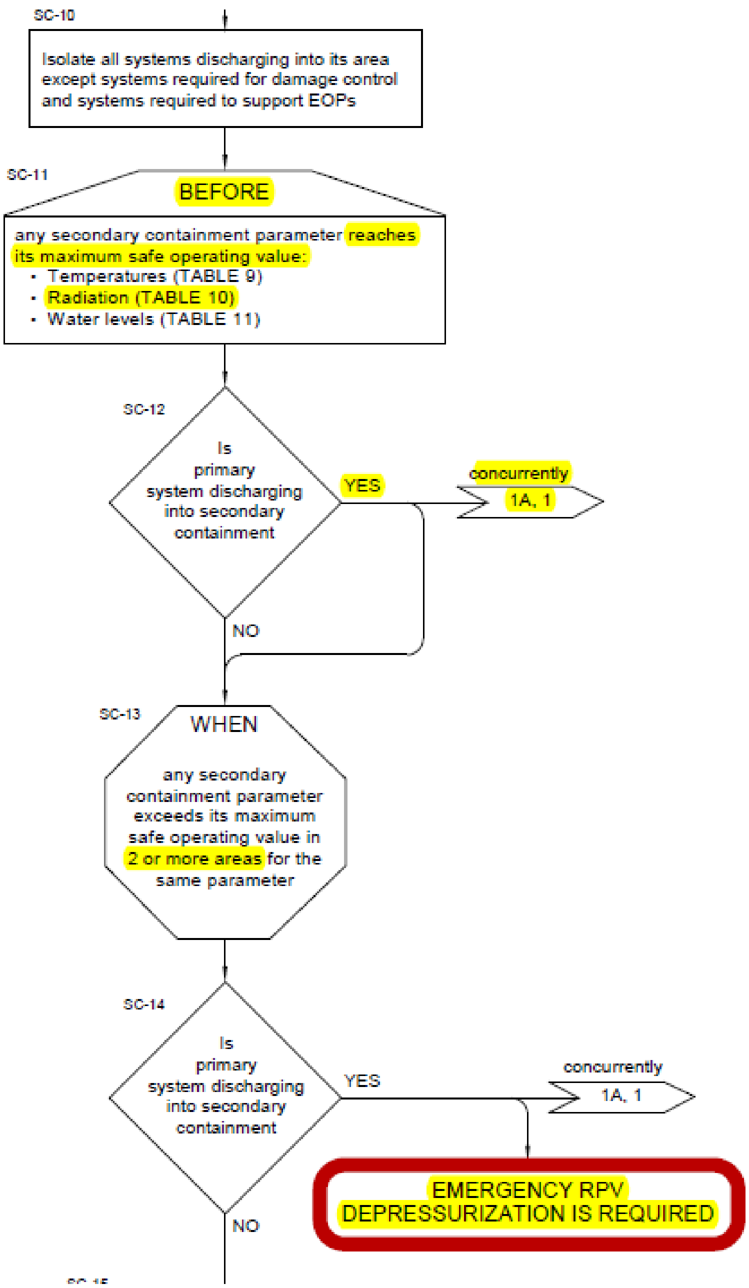
References to be provided to applicants during exam: NONE

Learning Objective:
Learning Objective: INT008-06-17 EOP Flowchart 5A Secondary Containment and Radioactivity Release Control

7. Given plant conditions and EOP flowchart 5A, SECONDARY CONTAINMENT CONTROL and RADIOACTIVITY RELEASE CONTROL, state the reasons for the actions contained in the steps.

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	X

	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b) 5	
Level of Difficulty:	3	
SRO Only Justification:	N/A	



5A	COOPER NUCLEAR STATION OPERATIONS MANUAL Emergency Operating Procedure 5.8 Attachment 1 Revision: 15

RADIOACTIVITY RELEASE CONTROL

ENTRY CONDITION:

Offsite radioactivity release rate above offsite gaseous release rate which requires Alert

16.10 Maximum Safe Operating Radiation Level

The Maximum Safe Operating Radiation Level is the highest radiation level at which neither (1) equipment necessary for the safe shutdown of the plant will fail nor (2) personnel access necessary for the safe shutdown of the plant will be precluded. The second concern defines the CNS Maximum Safe Operation Radiation Level. This level is utilized in establishing conditions under which RPV depressurization is required. Separate radiation levels are provided for each secondary containment area.

The Maximum Safe Operating Radiation Level is determined by (1) partitioning the secondary containment into areas within each of which the radiation levels during an emergency are expected to be approximately uniform, and (2) examining the equipment in and personnel access required to each area. These areas are not necessarily the same areas defined for monitoring by installed radiation monitoring systems. Neither do equipment qualification radiation levels necessarily define the radiation levels at which equipment will actually fail.

CNS input data required to calculate the Maximum Safe Operating Radiation Level are:

1. Secondary containment floor plan and equipment layout.
2. Radiation levels at which equipment necessary for the safe shutdown of the plant will fail.
3. Personnel access necessary for the safe shutdown of the plant.

The Maximum Safe Operating Radiation Level is referenced in PSTG Steps SC/R-2.1, SC/R-2.2, and SC/R-3.

PSTG/SATG Step

SC/R-2 If a primary system is discharging into secondary containment:

SC/R-2.1 Before any area radiation level reaches its maximum safe operating radiation level, enter the RPV Control Guideline at Step RC-1 and execute it concurrently with this procedure.

SC/R-2.2 When an area radiation level exceeds its maximum safe operating radiation level in more than one area, EMERGENCY RPV DEPRESSURIZATION IS REQUIRED.

Discussion

Primary systems comprise the pipes, valves, and other equipment physically connected to the RPV such that a reduction in RPV pressure will effect a decrease in the flow of steam or water being discharged through an unisolated break in the system.

A reactor scram is initiated through entry of the RPV Control guideline to reduce the primary system discharge into secondary containment and in anticipation of possible RPV depressurization in Step SC/R-2.2. If a discharge from a primary system is the source of radioactivity, the action that is directed in Step SC/R-2.1 should be adequate to terminate any further increase in secondary containment radiation levels.

A reactor scram is effected indirectly, through entry of the RPV Control guideline, rather than through an explicit direction in the Secondary Containment Control guideline to ensure that RPV water level, RPV pressure, and reactor power are properly coordinated following the scram and to avoid potential conflicts with alternate rod insertion strategies in Step RC/Q if the RPV Control guideline is already in use. (Note that Step RC-1 of the RPV Control guideline requires initiation of a reactor scram only if a scram has not yet been initiated.)

If secondary containment radiation levels continue to increase and exceed their maximum safe operating values in more than one area, the RPV must be depressurized. RPV depressurization places the primary system in its lowest possible energy state, rejects heat to the suppression pool in preference to outside the containment, and reduces the driving

PSTG/SATG Step

RR-2 Before gaseous offsite radioactivity release rate reaches that which requires a General Emergency but only if a primary system is discharging into an area outside the primary and secondary containments, EMERGENCY RPV DEPRESSURIZATION IS REQUIRED; enter the RPV Control Guideline at Step RC-1 and execute it concurrently with this procedure.

Discussion

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.

If a primary system is discharging into an area outside the primary and secondary containments, then either:

- A primary system break cannot be isolated because system operation is required to assure adequate core cooling or shut down the reactor, 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.

A reactor scram is initiated through entry of the RPV Control guideline to reduce the primary system discharge outside primary and secondary containments and to ensure that, if possible, the reactor is shut down before emergency RPV depressurization is initiated. The scram is effected indirectly, through entry of the RPV Control guideline, rather than through an explicit direction in the Radioactivity Release Control guideline, to ensure that RPV water level, RPV pressure, and reactor power are properly coordinated following the scram and to avoid potential conflicts with alternate rod insertion strategies in Step RC/Q if the RPV Control guideline is already in use. (Note that Step RC-1 of the RPV Control guideline requires initiation of a reactor scram only if a scram has not yet been initiated.)

Examination Outline Cross-Reference	Level	RO
	Tier#	1
	Group#	2
	K/A #	295035 EA2.02
	Rating	2.8

295035 Secondary Containment High Differential Pressure / 5

EA2. Ability to determine and/or interpret the following as they apply to SECONDARY CONTAINMENT HIGH DIFFERENTIAL PRESSURE: (CFR: 41.8 to 41.10)

EA2.02 Off site release rate: Plant Specific

Question 26

The plant is conducting refueling operations involving the movement of irradiated fuel assemblies in the secondary containment.

- A fuel bundle is dropped on the top of the core.
- A Group 6 isolation occurs due to bundle damage.
- Calm winds are indicated on the SPDS Weather Display.
- HV-MO-262 MG SET-1A INLET INBOARD and HV-AO-263 MG SET-1A INLET OUTBOARD fail to fully isolate.
- Reactor Building Average DP indicates -0.04” wg and stable.

Which one of the following identifies the Off-Site radiological release rate response?

A. ERP release rate rises.

B. ERP release rate lowers then rises.

C. Reactor Building monitored release rate rises.

D. Reactor Building unmonitored release rate rises.

Answer:
A. ERP release rate rises.
Explanation:
A fuel handling accident involving handling of irradiated fuel inside of the secondary

containment is one of the two principal accident scenarios for which credit is taken for secondary containment operability. Typically the Secondary Containment requires ≥0.25 inches of vacuum water gauge to maintain OPERABILITY. In this case however, the Reactor Building Average dp is below 0 inches water gauge (negative) and stable so both Standby Gas Trains are able to maintain negative building pressure under calm wind conditions and no RB unmonitored release is indicated. Since the Group 6 isolation occurred, the RB exhaust fans have terminated the release from the reactor building exhaust plenum. The ERP release rate rises because both SGTs are now operating and providing increased flow of airborne radiation due to the dropped fuel bundle damage and failure of the HV-MO-262 MG SET-1A INLET and HV-AO-263 MG SET-1A INLET to fully isolate. This flow that was being processed through the RB exhaust plenum is now routed to the ERP via the SGT system.

Distracters:

B. This answer is incorrect because the ERP release rate will not lower but only rise. The candidate may incorrectly assume that the start of the SGT causes the ERP release rate to lower because of added dilution to the ERP KAMAN calculation is based on the increased ERP flow. However, the plant is in a refueling outage so there is no radiation exiting the ERP. Once the SGT stream containing the radioactive gases from the damaged fuel bundle reach the ERP, the release rate will rise but it will not go down (for quite some time).

C. This answer is incorrect because the RB ventilation system has isolated. This choice is plausible if the candidate thinks that the failure of the HV-MO-262 MG SET-1A INLET and HV-AO-263 MG SET-1A INLET to fully isolate contribute to a rise in Reactor Building ventilation release rate. In this case, the building is being maintained negative by the SGT system and not the RB ventilation system.

D. This answer is incorrect in this instance, because the SGT system in conjunction with the partial integrity of the Secondary Containment is adequate to maintain a negative RB pressure. This answer is plausible if the candidate does not recognize that the RB pressure, though greater than the normal -0.25” wg, is still being maintained at a value at which all leakage will be inward.

Technical References:
Procedure 2.2.47, (HVAC Reactor Building) Rev 51
Procedure 2.2.73 (Standby Gas Treatment System) Rev 52.

References to be provided to applicants during exam: NONE

Learning Objective:
COR001-08-01 13. Briefly describe the following concepts as they apply to HVAC:
a. Airborne contamination control

Question Source:	Bank # 2014 NRC Retake # 26	
(note changes; attach parent)	Modified Bank #	
	New	
Question Cognitive Level:	Memory/Fundamental	

	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b) 9	
Level of Difficulty:	4	
SRO Only Justification:	N/A	

Examination Outline Cross-Reference:	Level	RO	
	Tier #	1	
	Group #	2	
	K/A #	295035 EA2.02	
	Importance Rating	2.8	

295035 Secondary Containment High Differential Pressure /5 – Ability to determine and/or interpret the following as they apply to secondary containment high differential pressure: (CFR: 41.8)
EA2.02 Off-site release rate: Plant-Specific

Question: 26

The plant is conducting refueling operations involving the movement of recently irradiated fuel assemblies in the secondary containment.

- A fuel bundle is dropped on the top of the core.
- A Group 6 isolation occurs due to bundle damage.
- Calm winds are indicated on the SPDS Weather Display.
- HV-MO-262 MG SET-1A INLET and HV-AO-263 MG SET-1A INLET fail to fully isolate.
- Reactor Building Average DP indicates -0.04" wg and stable.

Which of the following identifies the actual Off-Site radiological release rate response?

A. ERP release rate rises.

B. ERP release rate lowers then rises.

C. Reactor Building monitored release rate rises.

D. Reactor Building unmonitored release rate rises.

Answer:
A. ERP release rate rises.

2. INTERLOCKS AND SETPOINTS

2.1 Two 72" butterfly valves, HV-AO-257 and HV-MO-272, are installed in supply duct for the Reactor Building. These valves close on a Group 6 Isolation.

PROCEDURE 2.2.47	REVISION 51	PAGE 28 OF 32

ATTACHMENT 1	INFORMATION SHEET

2.2 Four 48" isolation valves, HV-MO-258, HV-AO-259, HV-MO-260, HV-AO-261, EXH FANS DISCH VLV, automatically close on a Group 6 Isolation.

1.2.8 Reactor Building 976' southeast heating is controlled by TC-1052 which modulates AS-TCV-1052 in the steam supply line to Heating Coil RHC-R-1B. This temperature controller is set at 70°F.

1.2.9 Most of Reactor Building air is normally exhausted by one of two main exhaust fans, EF-R-1A or EF-R-1B; however, during building high temperatures, two exhaust fans may be run if needed. The flow rate from each of these fans is controlled by a vortex damper. A small amount of air is drawn out of the building by the RRMG fans under normal operating conditions.

1.2.10 HV-DPIC-835A and HV-DPIC-835B consist of following:

1.2.10.1 A/M button changes system operation to AUTO or MANUAL. Illuminated light next to A indicates AUTOMATIC. Illuminated light next to M indicates MANUAL.

1.2.10.2 Display S - Automatic Setpoint Signal in dP. Displayed on S Bargraph or on upper display line by pressing D pushbutton until Parameter S is displayed on second display line.

1.2.10.3 Parameter P - Average building dP and should be near setpoint. Displayed on P Bargraph or on upper display line by pressing D pushbutton until Parameter P is displayed on second display line. Parameter P may also be read on Recorder HV-DPR-835.

1.2.10.4 Parameter V - Manual output signal in % valve position. Displayed on Open/Close Bargraph or on upper display line by pressing D pushbutton until Parameter V is displayed on second display line. 100% is open and 0% is closed.

1.2.10.5 If alarm is detected, L and S lights flash along with condition until acknowledged by depressing ACK pushbutton or alarm condition clears.

1.2.10.6 If one controller fails, other takes over both vortex dampers.

a. If in AUTO, then controller controls both vortex dampers to maintain dP setpoint.

b. If in MANUAL, then controller controls both vortex dampers at damper position in %.

1.2.11 Reactor Building to atmosphere differential pressure is normally controlled by exhaust fan vortex damper(s). The vortex dampers are controlled by a circuit which averages outside air pressure as sensed on all four sides of Reactor Building. The vortex dampers are operated to maintain Reactor Building pressure at least 0.25" wg below outside air pressure. The system DP controller is normally set at -0.30" to -0.33" wg, instead of -0.25" wg, to provide service margin between LCO entry conditions and the normal operating point of the system.

ATTACHMENT 2 INFORMATION SHEET

1. DISCUSSION

1.1 FUNCTION

1.1.1 The SGT System provides the ability to process effluent from the primary and secondary containment when required to limit the discharge of radioactive material to the environs. The system prevents out-leakage from the secondary containment during periods of primary and/or secondary containment isolation by holding it at a subatmospheric pressure of ≤ -0.25" wg. The system is also used to perform leak tests on the secondary containment.

1.2 OPERATING CHARACTERISTICS

1.2.1 The SGT System consists of two 100% capacity, identical parallel air filtration subsystems, completely enclosed within a Class 1 structure.

1.2.2 The system is designed to take suction from the Reactor Building main exhaust (normal), the Primary Containment, the HPCI System Gland Seal Exhauster, and the SGT Room.

1.2.2.1 When the HPCI System Gland Seal Exhauster is running, HPCI-AO-275 opens to admit the exhauster discharge to the SGT System. The valve fails closed on a loss of air or control power.

1.2.2.2 The suction damper from the Primary Containment to the SGT System, PC-AD-R-1B, opens automatically when a Group 6 isolation signal is received. The suction damper from the Primary Containment to the Reactor Building Exhaust Plenum, PC-AD-R-1A, automatically closes when a Group 6 isolation signal is received. PC-AD-R-1A fails closed on a loss of air or control power. PC-AD-R-1B fails open on a loss of air or control power.

1.2.2.3 Air Operated Room Air Inlet Valve SGT-AO-270 (SGT-AO-271), SGT A (B) DILUTION AIR, opens when a Group 6 isolation signal is received. The valve fails open on a loss of air or control power.

1.2.3 Each assembly consists of an air operated inlet valve, moisture separator, prefilter, air heater, high efficiency filter, activated carbon iodine adsorber, a second high efficiency filter, fan, an air operated outlet valve, and an air operated differential pressure control valve.

1.2.3.1 Air Operated Inlet Valve SGT-AO-249 (SGT-AO-250) opens when the SGT A (B) fan starts. The valve fails open on a loss of air or control power.

loss of air or control power.

1.2.4 The SGT System discharges to the ERP through two 10" underground lines. These SGT discharge lines can potentially be blocked by excessively high water level in Z sump located at the base of the ERP. Z sump pump discharge lines are heat traced to prevent blockage due to freezing during cold weather conditions. Temperature of these lines is monitored to ensure proper operation of heat trace. Upon discovery of temperature below 70°F, if power is lost to heat trace, Attachment 1 is used to assist in determining continued SGT operability. Z sump pumps and support equipment are essential in support of the SGT System.

Examination Outline Cross-Reference	Level	RO
	Tier#	1
	Group#	2
	K/A #	500000 G2.1.28
	Rating	4.1
500000 High CTMT Hydrogen Conc. / 5 G2.1.28 Knowledge of the purpose and function of major system components and controls. (CFR: 41.7)		

Question 27

Which one of the following completes the statement below regarding H_2/O_2 Monitor operation and function during normal rated power operation?

Normal H_2/O_2 Monitor alignment at power has ____(1)____ in service.
An alarm actuates if H2 concentration reaches a MINIMUM of ____(2)____.

A. (1) ONE Division
(2) 1.0%

B. (1) ONE Division
(2) 3.4%

C. (1) BOTH Divisions
(2) 1.0%

D. (1) BOTH Divisions
(2) 3.4%

Answer:
A. (1) ONE Division in service (2) 1.0%
Explanation: Normal at power alignment has ONE H_2/O_2 Monitor (Div II). If H_2 concentration reaches 1% annunciator HIGH H2 DIV II (P-2/B-1) will alarm in the control room.
Distracters: B. This answer is incorrect due to the alarm annunciating when H2 concentration

reaches 1%. This choice is plausible due to 3.4% being the High O2 Concentration level. The candidate that identifies only one division in service at power and confuses O2 with H2 concentration would select this answer.

C. This answer is incorrect due to only one division in service at power. This choice is plausible due to both divisions being put in service weekly for channel check. The candidate that confuses having both in service at power and correctly identifies the alarm setpoint would select this answer.

D. This answer is incorrect due to only one division in service at power and the alarm annunciating when H2 concentration reaches 1%. This choice is plausible due to both divisions being put in service weekly for channel check and 3.4% being the High O2 Concentration level. The candidate that confuses having both in service at power and O2 with H2 concentration would select this answer.

Technical References:
Procedure 2.2.60.1 (Containment H2/O2 Monitoring System), Rev. 22
Procedure 2.3_P-2 (Panel P-2 - Annunciator P-2), Rev. 2

References to be provided to applicants during exam: NONE

Learning Objective:
COR0020302001140D Briefly describe the following concepts as they apply to the Primary containment: Hydrogen/oxygen concentration measurement

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b) 7	
Level of Difficulty:	2	
SRO Only Justification:	N/A	

1.2.2 Each H_2/O_2 analyzer has H_2 and O_2 sensors. A sample pump draws the sample from one of four streams (sequentially three streams from drywell and one from torus), heats, analyzes, and returns sample to the torus. During normal system operation, Div 1 is in STANDBY mode. Div 2 is normally operating, processing both a H_2 and O_2 concentration signal. Sample sequencing and alarms are controlled by microprocessors in the remote process panels, within the respective divisions. Remote process panel (PC-CS-H_2/O_2II) controls sample stream indicating lights on VBD-H.

1.2.3 When CONTMT O_2 TRANSFER switch (PC-SW-CS3) is OFF, Div 2 H_2 signal is isolated from its recorder but is available on PMIS. The O_2 signal is isolated from its recorder but is available on PMIS and the digital indicator (PC-I-1) on VBD-H. With CONTMT O_2 TRANSFER switch (PC-SW-CS3) in RECORD, the Div 2 H_2 signal is processed to its recorder and PMIS. Div 2 O_2 is isolated from digital indicator (PC-I-1) on VBD-H and is processed to its recorder and PMIS.

ATTACHMENT 1	INFORMATION SHEET

1.2.4 H_2/O_2 Monitoring Systems initialize on stream 1 on system startup. On Div 1, this is observed at VBD-P1 on H_2 and O_2 recorder digital displays. On Div 2, this is observed on H_2 and O_2 recorders and on VBD-H sample stream indication lights. The O_2 indicator (PC-I-1) will display 0.00 and stream 1 indicating light will be on. Duration for each sample stream analysis is 10 minutes. When stream 1 sample analysis is completed, digital indicator (PC-I-1) will display O_2 concentration for stream 1 and stream 2 indicating light will be on. Throughout Div 2 sequencing, O_2 indicator (PC-I-1) will display the previous process sample stream analysis and the indicating light will be on for the sample stream currently being analyzed. When Switch PC-SW-CS3 is in RECORD, the recorder will print the point and value for the stream just analyzed.

1.2.5 Div 1 H_2 recorder range is 0% to 30%. Div 1 O_2 recorder range is 0% to 10%. Div 2 H_2 recorder, O_2 recorder, and O_2 digital indicator (PC-I-1) range is 0% to 30%. Wider O_2 range on Div 2 allows observing and recording containment de-inerted conditions when entry is desired. High H_2 alarm is 1.0% and high-high at 3.4% on both divisions. High O_2 alarm is 3.40% and high-high at 3.90% on both divisions. Div 1 alarms are annunciated at VBD-P1, Div 2 H_2 alarms are annunciated at VBD-P2. Div 2 O_2 alarms are annunciated at VBD-H.

1.2.6 Digital indication for H_2 and O_2 exists at remote process panels (PC-CS-H_2/O_2I and PC-CS-H_2/O_2II) in the Cable Spreading Room. Analog indication for H_2 and O_2 exists on local analyzer cabinets (PC-AN-H_2/O_2I and PC-AN-H_2/O_2II) at R-976-EAST.

2. REFERENCES

2.1 TECHNICAL SPECIFICATIONS

2.1.1 LCO 3.3.3.1, Post-Accident Monitoring (PAM) Instrumentation, Table 3.3.3.1-1, Function 6.

2.1.2 LCO 3.6.3.1, Primary Containment O_2 Concentration.

2.2 UPDATED SAFETY ANALYSIS REPORT

2.2.1 Section V-2, Primary Containment Atmospheric Control System and Containment Monitoring.

2.3 DRAWINGS

2.3.1 B&R Drawing 2022, Primary Containment Coolant and Nitrogen Inerting System.

2.3.2 Whittaker Drawing 115902, Elementary Diagram H_2O_2 Analyzer.

HIGH H2 DIV II	PANEL/WINDOW: P-2/B-1

1. OPERATOR OBSERVATION AND ACTION

1.1 Verify hydrogen content on PC-CS-H_2/O_2 II panel in Cable Spreading Room.

1.2 Start Division I H_2/O_2 Monitor System for parallel operation and monitoring per Procedure 2.2.60.1.

SETPOINT	CIC	P-2/B-1
(4971) 1%	PC-CS-H_2/O_2 II	

PROBABLE CAUSES

- Loss of coolant accident.

REFERENCES

- System Operating Procedure 2.2.60.1, Containment H_2/O_2 Monitoring System.

Examination Outline Cross-Reference	Level	RO
Clarified in explanation that injection valves automatically open as reactor pressure lowers below 436 psig.	Tier#	2
	Group#	1
	K/A #	203000 K3.01
	Rating	4.3

203000 RHR/LPCI: Injection Mode

K3. Knowledge of the effect that a loss or malfunction of the RHR/LPCI: INJECTION MODE (PLANT SPECIFIC) will have on following: (CFR: 41.7 / 45.4)

K3.01 Reactor water level

Question 28

Plant startup is in progress with Reactor Pressure at 450 psig when the following annunciator is received due to a spurious initiation signal:

RHR PMP A/B LOGIC ACTUATED	PANEL/WINDOW: 9-3-1/A-4

Which one of the following identifies the impact of this alarm under the current plant conditions if no operator action is taken?

A. Both RHR Loops will inject causing RPV water level to rise.

B. RHR Loop A ONLY will inject causing RPV water level to rise.

C. RHR Loop B ONLY will inject causing RPV water level to rise.

D. Neither RHR Loop will inject causing RPV water level to remain the same.

Answer:
D. Neither RHR Loop will inject causing RPV water level to remain the same.

Explanation:
This annunciator indicates auto start signals to both RHR pumps A & B (1 pump in each RHR loop). RHR inboard injection valves open when reactor pressure lowers below 436 psig. Flow however does not occur until reactor pressure falls below the

shutoff head of the RHR pumps at 230 psig.

Distracters:

A. This answer is incorrect due to neither RHR loop injecting. This choice is plausible due to confusing RHR pump start logic with LPCI loop logic (Loop vs. System logic). The candidate that confuses this annunciator as being Div 1 logic and does not recognize RPV pressure being above RHR shutoff head would select this answer.

B. This answer is incorrect due to neither RHR loop injecting. This choice is plausible due to confusing RHR pump start logic with LPCI loop logic (Loop vs. System logic). The candidate that confuses this annunciator as being Div 2 logic and does not recognize RPV pressure being above RHR shutoff head would select this answer.

C. This answer is incorrect due to neither RHR loop injecting. This choice is plausible due to not recognizing RPV pressure being above RHR system shutoff head. The candidate that does not recognize RPV pressure being above RHR shutoff head would select this answer.

Technical References:

Procedure 2.2.69.1(RHR LPCI Mode), Rev. 30

Procedure 2.3_9-3-1 (Panel 9-3 - Annunciator 9-3-1), Rev. 34

References to be provided to applicants during exam: NONE

Learning Objective:

COR002-23-02, OPS Residual Heat Removal System

4. Describe the interrelationship between the RHR system and the following:
   n. Reactor pressure

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b) 7	
Level of Difficulty:	2	
SRO Only Justification:	N/A	

NOTE – RHR pump operation at minimum flow should be limited to < 15 minutes or pump damage may result.

5.2.2.7 If reactor pressure is > 436 psig, injection valves will not open. Ensure minimum flow valves RHR-MO-16A and RHR-MO-16B remain open as long as respective subsystem flow is < 2107 gpm.

5.2.2.8 As reactor pressure drops, observe Annunciator 9-3-2/C-5, RX LOW PRESS 291-436 psig, alarms and injection valves open at 436 psig.

NOTE 1 – Valves RHR-MO-27A and RHR-MO-27B cannot be throttled for 5 minutes after LPCI has initiated and RPV pressure has dropped below 436 psig.

NOTE 2 – Rated flow for each RHR Subsystem is 8400 gpm if one pump is operating and 15,000 gpm if both pumps are operating.

5.2.2.9 WHEN injection valves open, THEN ensure flow through injection lines by following:

a. Flow rise on RHR-FI-133A.

b. Flow rise on RHR-FI-133B.

5.2.2.10 Ensure recorder RHR-FR-143 is turned on.

5.2.2.11 Monitor reactor vessel level. Take action as directed by EOPs. When level has recovered to operating level of +3" to +54", pumps may be shut down or outboard injection valves, RHR-MO-27A and RHR-MO-27B, may be throttled as Operator desires (RHR-MO-27A and RHR-MO-27B cannot be throttled for 5 minutes after RHR is initiated).

PROCEDURE 2.2.69.1	REVISION 30	PAGE 5 OF 9

RHR PMP A/B LOGIC ACTUATED	PANEL/WINDOW: 9-3-1/A-4

1. AUTOMATIC ACTIONS

1.1 RHR System A auto initiation.

2. OPERATOR OBSERVATION AND ACTION

2.1 IF initiation signal is not valid, THEN perform following:

2.1.1 Stop RHR Pump A and verify PUMP STOP SIG SEALED-IN amber light on, if initiation signal present.

2.1.2 Stop RHR Pump B and verify PUMP STOP SIG SEALED-IN amber light on, if initiation signal present.

2.1.3 Enter Procedure 2.4CSCS.

SETPOINT
1. (1130) RHR PUMP A LOGIC ACTUATED
2. (1131) RHR PUMP B LOGIC ACTUATED

CIC
1. RHR-REL-K18A
2. RHR-REL-K21A

9-3-1/A-4

PROBABLE CAUSES

- RHR System A auto initiation due to high drywell pressure of ≤ 1.84 psig or low reactor water level of ≥ -113".

REFERENCES

- Abnormal Procedure 2.4CSCS, Inadvertent CSCS Initiation.

RPV PRESSURE				
Pressure (psig)	Description	Notes		
1080	Lowest SRV lift pressure	SRV mechanical actuation		
1050	Hi pressure scram	Upper pressure control band		
1029 / 999	LLS SRVs open	Open signal		
940	BPVs full open	Pressure above which SRVs are used to stabilize pressure with main condenser available		
854	LLS SRVs close	Close signal		
~550	Condensate booster pump shutoff head	Beginning of condensate injection		
~350	CS pump shutoff head	Beginning of CS injection		
~300	RHR pump shutoff head	Beginning of RHR injection		
70	SDC pressure interlock	Interlock clears		
50	Decay Heat Removal Pressure	10 min. decay heat pressure Depressurized RPV state		
MINIMUM STEAM COOLING PRESSURES	Open SRVs	RPV Press. (psig)	Open SRVs	RPV Press. (psig)
	6 or more	145	3	300
	5	175	2	460
	4	225	1	930

USAR

TABLE VI-5-4

PLANT ECCS PARAMETERS

1. Low Pressure Coolant Injection (LPCI) System

	Variable	Units	Analysis * Value
a.	Maximum vessel pressure at which pumps can inject flow	psid (vessel to drywell)	263 (230)
b.	Minimum rated flow		
	• Vessel pressure for following flow rates	psid (vessel to drywell)	20
	• 2 LPCI pumps injecting into one recirculation loop	gpm	13500 (10700)
	• 2 LPCI pumps injecting into two recirculation loops	gpm	13900 (13000)
	• 4 LPCI pumps injecting into two recirculation loops	gpm	27000 (21400)
c.	Initiating Signals		
	• Low water level 1	in (above vessel zero)	358.56
	• High drywell pressure	psig	2
d.	Maximum allowable time from initiating signal to pump at rated speed and capable of rated flow (including Diesel Generator start time)	sec	59
e.	Maximum injection valve stroke time - opening	sec	45
f.	Pressure permissive at which LPCI injection valve may open	psig	275
g.	Pressure permissive at which recirculation discharge valve may close	psig	185
h.	Recirculation discharge valve stroke time - closing	sec	40
i.	Recirculation discharge valve design differential pressure	psid	200

* Values given in parentheses (#) are with minimum flow valve open.

Examination Outline Cross-Reference	Level	RO
Added Normal RPV level band while in Mode 4 is 80 to 100 inches to explanation. SDC valve names not required as these are required to be known as a normal response to lowering RPV water level.	Tier#	2
	Group#	1
	K/A #	205000 K4.03
	Rating	3.8

205000 Shutdown Cooling

K4. Knowledge of SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE) design feature(s) and/or interlocks which provide for the following: (CFR: 41.7)

K4.03 Low reactor water level: Plant-Specific

Question 29

The plant is in Mode 4.
RHR Loop A is in SDC with RHR Pump A in service.

Which one of the following completes the statement below regarding the impact on RHR Loop A as RPV Water Level lowers to -20 inches?

When RPV level reaches ____(1)____ inches, SDC valves isolate and RHR Pump A ____(2)____.

A. (1) +35
(2) trips

B. (1) +35
(2) remains running

C. (1) +3
(2) trips

D. (1) +3
(2) remains running

Answer:
C. (1) +3 (2) trips

Explanation:
Group 2 SDC isolation occurs at RPV Water level of +3 inches. This isolation closes RHR-MO-17 (SDC RHR SUPPLY OUTBD VLV), RHR-MO-18 (SDC RHR SUPPLY INBD VLV), and RHR-MO-25A (INBD INJECTION VLV). Closure of either SDC supply valve trips RHR pumps operating in SDC. Normal RPV level band while in Mode 4 is 80 to 100 inches.

Distracters:

A. This answer is incorrect due to +3 inches RPV level causes the isolation. This choice is plausible due to +35 inches being the level at which the SDC isolation resets with reactor pressure at 59 psig. The candidate that confuses the RPV water level which causes a SDC isolation and correctly identifies RHR pump trip on loss of suction path would select this answer.

B. This answer is incorrect due to +3 inches RPV level causes the isolation and the RHR pump trips. This choice is plausible due to +35 inches being the level at which the SDC isolation resets with reactor pressure at 59 psig and confusing SDC isolation interlocks with RHR pump trips. Additionally, part 2 is plausible, since core spray pumps do not trip directly from suction valve closure. The candidate that confuses the RPV water level which causes a SDC isolation and confuses RHR system alignment upon isolation or RHR pump trip logic with Core Spray pump trip logic would select this answer.

D. This answer is incorrect due to the RHR pump trips. This choice is plausible due to confusing SDC isolation interlocks with RHR pump trips. Additionally, part 2 is plausible, since core spray pumps do not trip directly from suction valve closure. The candidate that correctly identifies the RPV water level which causes a SDC isolation and confuses RHR system alignment upon isolation or RHR pump trip logic with Core Spray pump trip logic would select this answer.

Technical References:
Procedure 2.1.22 (Recovering From A Group Isolation), Rev. 60
Procedure 2.2.69.2 (RHR System Shutdown Operations), Rev. 90

References to be provided to applicants during exam: NONE

Learning Objective:
COR0022302001150D Given plant conditions, determine if the following should occur: RHR valve reposition

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b) 7	

Level of Difficulty:	3
SRO Only Justification:	N/A

5.3 Determine isolation cause:

ISOLATION	ALLOWABLE VALUE	COMMENTS
Low Reactor Water Level	≥ 3"	
Drywell Pressure	≤ 1.84 psig	Ensure Group 6 Isolation.
RPS Power Supply Failure	Loss of power	

PROCEDURE 2.1.22 REVISION 60 PAGE 7 OF 30

CAUTION – Lowering reactor vessel pressure during SDC heatup before SDC is in service could cause a spurious high pressure isolation from flashing in sensing lines.©[2]

NOTE 1 – Steps will be performed at Panel 9-3 unless specified otherwise.

NOTE 2 – SDC isolation resets at ~ 59 psig (dome pressure) with RPV level at ~ 35".

4.4 WHEN RPV pressure < 50 psig, THEN ensure following valves and breaker are aligned as specified:

4.4.1 RHR-MO-16A, LOOP A MIN FLOW BYP VLV, is tagged closed.

4.4.2 Ensure Breaker 4D on MCC-Q for RHR-MO-16A is tagged to OFF.Ⓟ[10]

4.4.3 Ensure RHR-MO-16A handwheel is tagged DO NOT OPERATE.

4.4.4 RHR-MO-39A, SUPPR POOL COOLING/TORUS SPRAY OUTBD VLV, is closed.Ⓟ[10]

4.4.5 RHR-MO-13A, PUMP A TORUS SUCT VLV, is closed.

4.4.6 RHR-MO-13C, PUMP C TORUS SUCT VLV, is closed.

4.4.7 RHR-MO-12A, HX-A OUTLET VLV, is closed.Ⓟ[12]

4.4.8 RHR-MO-15A, PUMP A SDC SUCT VLV, is open.

4.4.9 RHR-MO-15C, PUMP C SDC SUCT VLV, is open.

4.4.10 RHR-MO-65A, HX-A INLET VLV, is open.

4.4.11 RHR-MO-27A, OUTBD INJECTION VLV, is open.Ⓟ[12]

4.5 Ensure following valves are closed:

4.5.1 CM-296, LOOP A INJECTION LINE PRESSURE MAINTENANCE SHUTOFF (R-881-NW Quad).

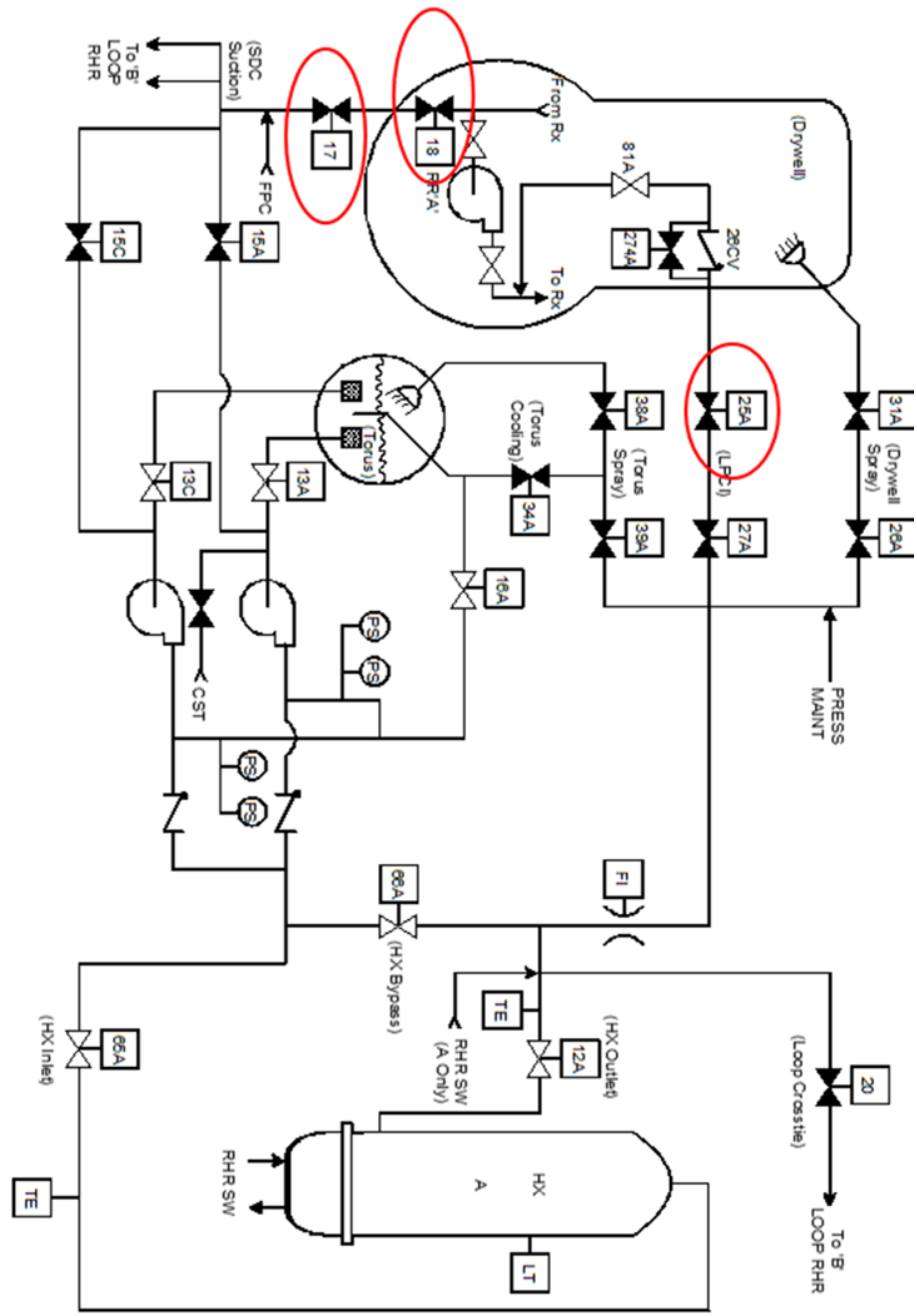
4.5.2 CM-297, LOOP A INJECTION LINE PRESSURE MAINTENANCE ROOT (R-881-NW Quad).

4.6 Ensure RHR-MO-25A, INBD INJECTION VLV, open.

4.7 Check RHR-MO-18, SHUTDOWN COOLING RHR SUPPLY INBD VLV, and RHR-MO-17, SHUTDOWN COOLING RHR SUPPLY OUTBD VLV, are open, or open by performing following:

4.7.1 Throttle open RHR-208, RHR SHUTDOWN COOLING SUPPLY LINE VENT (R-890-Torus Area south of Bent 1), to ensure piping is depressurized.

4.7.2 Ensure CM-P-RX, RX BLDG AUX COND PMP, is running.



RHR LOOP 'A'
Figure 1, Rev. 9
COR002-23-02

Examination Outline Cross-Reference	Level	RO
Candidates are required to know system alignment based upon conditions provided – no changes.	Tier#	2
	Group#	1
	K/A #	206000 A1.03
	Rating	3.5

206000 HPCI

A1. Ability to predict and/or monitor changes in parameters associated with operating the HIGH PRESSURE COOLANT INJECTION SYSTEM controls including: (CFR: 41.5 / 45.5)

A1.03 Condensate storage tank level: BWR-2,3,4

Question 30

HPCI is operating in pressure control mode following a Scram when the following annunciator is received:

HPCI SUCTION TRANSFER	PANEL/WINDOW: 9-3-2/A-4

- CM-LI-681A (EMERG COND STOR TK 1A LEVEL) indicates < 2 feet.

What is the status/alignment of HPCI five minutes later?

	MO-25 (Minimum Flow Valve)	MO-21 & MO-24 (Test Return Isolation Valves)
A.	Open	Open
B.	Open	Closed
C.	Closed	Open
D.	Closed	Closed

Answer:		
B.	Open	Closed

Explanation:
The low ECST level has initiated a transfer of the of the HPCI suction valves. With the transfer of suction to the suppression pool the pump test return isolation valves automatically close. Now the HPCI system is without a discharge path causing the minimum flow valve to open due to low flow.

Distracters:
A. This answer is incorrect due to the test return valves being closed. This choice is plausible due to not recognizing the loss of discharge flowpath. The candidate that correctly identifies min flow valve open and does not recognize the total loss of discharge path would select this answer.
C. This answer is incorrect due to the min flow valve being open and the test return valves being closed. This choice is plausible due to the min flow valve not immediately opening during the suction transfer and not recognizing the loss of discharge flowpath. The candidate that does not recognize the total loss of discharge path would select this answer.
D. This answer is incorrect due to the min flow valve being open. This choice is plausible due to the min flow valve not immediately opening during the suction transfer. The candidate that does not recognize the final system alignment but does recognize the test returns closing would select this answer.

Technical References:
Procedure 2.2.33 (High Pressure Coolant Injection System), Rev. 77
Procedure 2.3_9-3-2 (Panel 9-3 - Annunciator 9-3-2), Rev. 30

References to be provided to applicants during exam: NONE

Learning Objective:

COR0021102001120D Given plant conditions, determine if the following HPCI actions should occur: Suction transfer

COR0021102001100H Predict the consequences of the following on the HPCI system: Low ECST level

Question Source:	Bank # 19680	
(note changes; attach parent)	Modified Bank #	
	New	
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b) 5	
Level of Difficulty:	3	
SRO Only Justification:	N/A	

QUESTION: 6, 19680 (1 point(s))

The plant was operating at power with the HPCI system in full flow test per 6.HPCI.103 when the HPCI SUCTION TRANSFER annunciator was received. ECST level indicator, CM-LI-681A, was observed to be reading below the minimum indication of 2 feet.

What is the status/alignment of HPCI several minutes later?
(Assume the operator takes no action.)

	Minimum Flow Valve (MO-25)	Test Return Line Isol Valves (MO-21 & 24)
a.	open	closed
b.	closed	closed
c.	open	open
d.	closed	open

ANSWER: 6, 19680

a. open closed

The low ECST level has initiated a transfer of the of the HPCI suction valves. With the swap over to suction from the suppression pool the pump test return isolation valves automatically close. Now the HPCI system is without a discharge path the minimum flow valve opens due to low flow.

Answer source: 2.2.33 and 9-3-2 / C-4

Distractors:

b. The minimum flow valve is open.
c. The Pump Test Return Line Isolation Valves, (MO-21 & 24) are closed.
d. The minimum flow valve is open and the Pump Test Return Line Isolation Valves, (MO-21 & 24) are closed.

HPCI SUCTION TRANSFER	PANEL/WINDOW: 9-3-2/A-4

1. AUTOMATIC ACTIONS

NOTE – If a Group 4 isolation signal is present, HPCI-MO-58, TORUS PUMP SUCT VLV, will not open if closed.

1.1 For alarm message (1634) HPCI SUCT TRANSFER SUPR POOL HIGH LEVEL:

1.1.1 Open HPCI-MO-58, TORUS PUMP SUCT VLV.

1.1.2 When HPCI-MO-58 is fully open, HPCI-MO-17, ECST PUMP SUCT VLV, HPCI-MO-21, TEST BYPASS TO ECST VLV, and HPCI-MO-24, ECST TEST LINE SHUTOFF VLV, receive a close signal.

1.2 For alarm message (1650) HPCI/RCIC SUCT TRANSFER ECST A LOW LEVEL or (1651) HPCI/RCIC SUCT TRANSFER ECST B LOW LEVEL:

1.2.1 Opens HPCI-MO-58 and RCIC-MO-41, TORUS PUMP SUCT VLV.

1.2.2 When HPCI-MO-58 and RCIC-MO-41 are fully open, HPCI-MO-17, HPCI-MO-21, HPCI-MO-24, RCIC-MO-18, ECST PUMP SUCT VLV, RCIC-MO-30, TEST BYP TO ECST VLV, and RCIC-MO-33, ECST TEST LINE SHUTOFF, receive a close signal.

SETPOINT	CIC 9-3-2/A-4
1. (1634) HPCI SUCT TRANSFER SUPPR POOL HIGH LEVEL at 3.0" H_2O above normal (Tech Spec ≤ 4" above normal)	1. HPCI-REL-K19 operation caused by HPCI-LS-91A or HPCI-LS-91B
2. (1650) HPCI/RCIC SUCT TRANSFER ECST A LOW LEVEL at 24" from bottom of ECST A or 2 1/2" indicated (Tech Spec ≥ 23")	2. RCIC-REL-K39X operation caused by HPCI-LS-74A or HPCI-LS-75A
3. (1651) HPCI/RCIC SUCT TRANSFER ECST B LOW LEVEL at 24" from bottom of ECST B or 2 1/2" indicated (Tech Spec ≥ 23")	3. HPCI-REL-K18 operation caused by HPCI-LS-74B or HPCI-LS-75B

PROBABLE CAUSES

- High level in torus caused by SRV operation or systems valve leakage.
- Extended operation of HPCI and/or RCIC in the RPV injection mode.

REFERENCES

- Technical Specifications Table 3.3.5.1-1, Functions 3d and 3e.
- System Operating Procedure 2.2.7, Condensate Storage and Transfer System.

2.1.1.4 HPCI-MO-17, ECST PUMP SUCT VLV

a. Opens on either high drywell pressure or low reactor water level, if HPCI-MO-58, TORUS PUMP SUCT VLV, is not fully open.

b. Closes when HPCI-MO-58 is fully open (overrides the open signal from either high drywell pressure or low reactor water level).

2.1.1.5 HPCI-MO-58, TORUS PUMP SUCT VLV

a. Opens on emergency condensate storage tank low water level at ≥ 23" (from tank bottom) remaining or Suppression Pool high level at ≤ 4", provided there is no low steam supply pressure (≥ 107 psig) or HPCI System isolation signal present.

b. Closes on low steam supply pressure (≥ 107 psig) or HPCI System isolation signal if HPCI-MO-17 is full open and cannot be reopened until the isolation signal or low steam supply pressure signal is cleared.

2.1.1.6 HPCI-MO-20, PUMP DISCHARGE VLV

a. Opens on either high drywell pressure or low reactor water level.

b. No auto close interlocks.

2.1.1.7 HPCI-MO-19, INJECTION VLV

a. Opens on either high drywell pressure or low reactor water level.

b. No auto close interlocks.

NOTE – A pressure surge on the pressure maintenance system may cause HPCI-MO-25 to open.

2.1.1.8 HPCI-MO-25, MIN FLOW BYP VLV

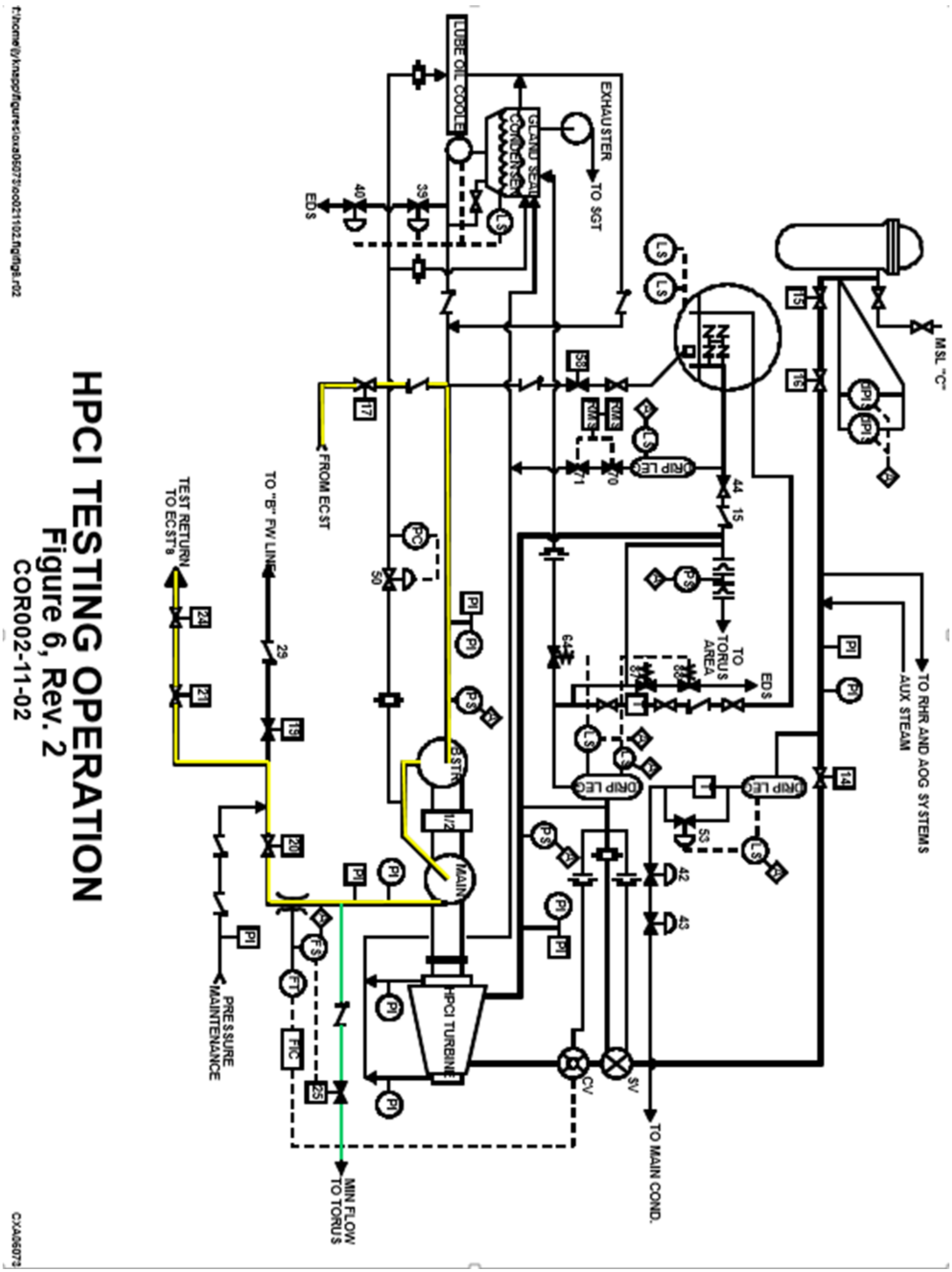
a. Opens on HPCI pump low flow, ≥ 490 gpm, and either HPCI pump discharge pressure is > 125 psig or an initiation signal is present, provided there is no HPCI turbine trip signal present.

b. Closes on HPCI pump high flow, > 800 gpm, or HPCI turbine trip.

2.1.1.9 HPCI-MO-21, TEST BYPASS TO ECST VLV

a. No auto open interlocks.

b. Closes on either high drywell pressure, low reactor water level, or when HPCI-MO-58 is fully open.


HPCI TESTING OPERATION
Figure 6, Rev. 2
COR002-11-02
LUBE OIL COOLER
GLAND SEAL CONDENSER
EXHAUSTER
TO SGT
MSL "C"
TO RHR AND AOG SYSTEMS
AUX STEAM
TO TORUS AREA
EDS
DRIP LEG
FROM ECST
TO "B" FW LINE
TEST RETURN TO ECST's
BSTR
MAIN
HPCI TURBINE
SV
CV
TO MAIN COND.
MIN FLOW TO TORUS
PRESSURE MAINTENANCE

Examination Outline Cross-Reference	Level	RO
Distractor A not changed due to being the correct answer if the HPCI turbine were at 0 rpm speed. Distractor B changed to 4, 2, and 1 ONLY.	Tier#	2
	Group#	1
	K/A #	206000G2.4.49
	Rating	4.6

206000 HPCI

G2.4.49 Ability to perform without reference to procedures those actions that require immediate operation of system components and controls. (CFR: 41.10 / 43.2 / 45.6)

Question 31

HPCI inadvertently initiates while operating at power.

- Turbine speed at 1000 rpm and rising.

Which one of the following identifies the Immediate Operator Action(s) (in the correct order if applicable) IAW 2.4CSCS (Inadvertent CSCS Initiation)?

(1) Release TURBINE TRIP button.
(2) Press and hold TURBINE TRIP button.
(3) Place AUXILIARY OIL PUMP in PULL-TO-LOCK.
(4) Ensure AUXILIARY OIL PUMP control switch in START.

A. 3 ONLY.

B. 4, 2 and then 1 ONLY.

C. 2, 3 and then 1 ONLY.

D. 4, 2, 3 and then 1.

Answer:
D. 4, 2, 3 and then 1.

Explanation:
The Immediate Operator Actions for an inadvertent HPCI initiation while operating at power IAW 2.4CSCS is:
1. Ensure AUXILIARY OIL PUMP control switch in START.
2. Press and hold TURBINE TRIP button.
3. AFTER turbine stops, THEN place AUXILIARY OIL PUMP in PULL-TO-LOCK.

4. Release TURBINE TRIP button.

“AFTER turbine stops” is a knowledge item intentionally removed from answer choices to support ACTIONs only.

Distracters:

A. This answer is incorrect due to all 4 actions being required. This choice is plausible and would be correct if the HPCI system was NOT running. The candidate that confuses actions required with HPCI shutdown vs. running would select this answer.
B. This answer is incorrect due to all 4 actions being required in the correct order. This choice is plausible if depressing and releasing the trip push button for HPCI is confused with RCIC stop & prevent actions. The candidate that confuses RCIC stop and prevent vs. HPCI immediate operator actions would select this answer.
C. This answer is incorrect due to all 4 actions being required. This choice is plausible and would be correct if the question asked for actions to Stop & Prevent HPCI IAW procedure 5.3. The candidate that confuses AOP vs. EOP actions or forgets to start the AOP would select this answer.

Technical References:
Procedure 2.4CSCS (Inadvertent CSCS Initiation), Rev. 9
Procedure 5.8 {Emergency Operating Procedures (EOPs)}, Rev. 37

References to be provided to applicants during exam: NONE

Learning Objective:
INT0320123G0G0100 2. Given plant condition(s), determine from memory all immediate operator actions required to mitigate the event(s).

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b) 10	
Level of Difficulty:	2	
SRO Only Justification:	N/A	

CNS OPERATIONS MANUAL ABNORMAL PROCEDURE 2.4CSCS INADVERTENT CSCS INITIATION	USE: CONTINUOUS QUALITY: QAPD RELATED EFFECTIVE: 6/25/13 APPROVAL: ITR-RDM OWNER: OSG SUPV DEPARTMENT: OPS

1. ENTRY CONDITIONS

1.1 Inadvertent initiation of one or more CSCS Systems.

1.2 Inadvertent initiation of RCIC.

2. AUTOMATIC ACTIONS

2.1 None.

3. IMMEDIATE OPERATOR ACTIONS

3.1 IF HPCI initiated, THEN perform following:

3.1.1 Ensure AUXILIARY OIL PUMP control switch in START.

3.1.2 Press and hold TURBINE TRIP button.

3.1.3 AFTER turbine stops, THEN place AUXILIARY OIL PUMP in PULL-TO-LOCK.

3.1.4 Release TURBINE TRIP button.

4. SUBSEQUENT OPERATOR ACTIONS

4.1 Record current time and date. Time/Date: ________/__________

4.2 IF reactor pressure ≤ 450 psig, THEN secure affected system(s) in following order:

4.2.1 CS per Attachment 1 (Page 3).

4.2.2 ADS per Attachment 2 (Page 4).

4.2.3 RHR per Attachment 3 (Page 5).

4.2.4 RCIC per Attachment 4 (Page 6).

4.3 IF reactor pressure > 450 psig, THEN secure affected system(s) in following order:

4.3.1 RCIC per Attachment 4 (Page 6).

4.3.2 ADS per Attachment 2 (Page 4).

4.3.3 RHR per Attachment 3 (Page 5).

ATTACHMENT 4 STOP AND PREVENT HARD CARD

1. STOP INJECTION

1.1 Stop HPCI by performing one of following:

1.1.1 TRIP HPCI turbine:

1.1.1.1 IF running, THEN press and hold **TURBINE TRIP** button.

1.1.1.2 WHEN the Turbine is at zero rpm, THEN place **AUXILIARY OIL PUMP** switch to **PULL-TO-LOCK**.

1.1.1.3 If applicable, release **TURBINE TRIP** button.

1.1.2 Close HPCI-MO-16, STM SUPP OUTBD ISOL VLV.

1.1.3 Depress MANUAL ISOLATION PUSHBUTTON, if initiation signal present.

1.2 Stop Feedwater by performing following:

1.2.1 Ensure RFP A is tripped.

1.2.2 Ensure RFP B is tripped.

1.2.3 IF Reactor pressure ≤ 600 psig, THEN ensure all condensate booster pumps are tripped.

1.2.3.1 CBP A.

1.2.3.2 CBP B.

1.2.3.3 CBP C.

CAUTION – If Core Spray and RHR pumps are placed in PULL-TO-LOCK before system flow is reduced to minimum, draining of system may occur.

1.3 Stop Core Spray by ensuring following:

1.3.1 CS System A secured with pump in PULL-TO-LOCK.

1.3.2 CS System B secured with pump in PULL-TO-LOCK.

1.4 Stop RHR by ensuring one of following:

1.4.1 Both RHR Systems secured with pumps in PULL-TO-LOCK.

1.4.2 RHR outboard injection valves automatic open signal bypassed per Procedure 5.8.20 (PTMs 97 through 100) with injection valves closed.

1.4.3 IF RPV pressure is maintained ≥ 500 psig, THEN operate RHR aligned to suppression pool cooling and/or containment spray per Procedure 2.2.69.3.

Examination Outline Cross-Reference	Level	RO
Randomly re-selected KA (loss of AC power impact on Core Spray). New question developed to determine impact of loss of Startup & ESST transformers following a LOCA and status of Core spray with RPV pressure greater than pump shutoff head. This question requires analyzing the impact of power loss on the Core Spray system under LOCA conditions and predicting system response with RPV pressure above the pump shutoff head. LOK is high cognitive. Added "following Bus energization" to eliminate potential challenges associated with DG start time (14 sec) not being addressed in the time delay.	Tier# Group# K/A # Rating	2 1 209001 K6.01 3.4

209001 LPCS

K6. Knowledge of the effect that a loss or malfunction of the following will have on the LOW PRESSURE CORE SPRAY SYSTEM : (CFR: 41.7 / 45.7)

K6.01 A.C. power

Question 32

The plant is operating at power with DG2 tagged out for maintenance.

A transient occurs with the following conditions present:

- The Reactor is shutdown.
- RPV Water level is -90 inches (wide range).
- Reactor pressure is 400 psig.
- DW pressure 4.5 psig.

The Startup Transformer then de-energizes and the Critical Buses fail to transfer to the ESST.

Which one of the following completes the statement below regarding the status of the Core Spray system following the loss of Startup Transformer?

Core Spray system ____(1)____ auto start(s) following a 10 second time delay following Bus energization and is/are currently ____(2)____.

A. (1) A ONLY
(2) running on minimum flow

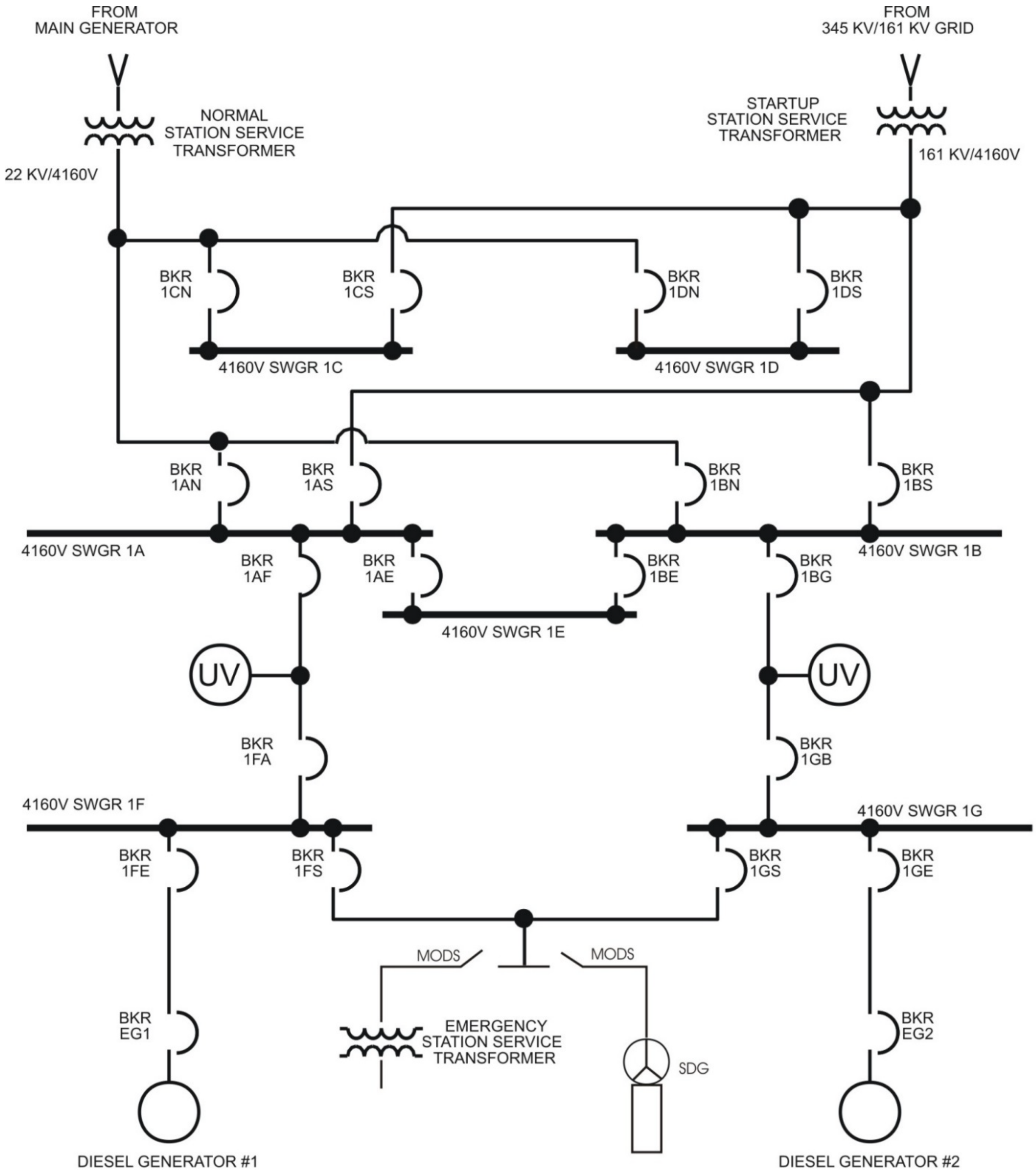
B. (1) A ONLY
(2) injecting into the RPV

C. (1) A and B
(2) running on minimum flow

D. (1) A and B
(2) injecting into the RPV

Answer:
A. (1) A ONLY (2) running on minimum flow
Explanation: Requires knowledge of the impact that a loss of startup transformer with a failure of the emergency transformer has on the Core Spray system with an auto start signal present and RPV pressure above system shutoff head pressure. The loss of startup transformer would normally auto fast transfer to the emergency transformer without critical bus loss of power. Since a loss of power to both critical buses is provided, the DGs will energize the critical buses following load shed. Since DG2 is removed from service, only DG1 ties to energize critical bus F. Once Bus F is energized, Core Spray pump A auto starts following a 10 second time delay. Due to reactor pressure being above the Core Spray pump shutoff head (350 psig), CS pump A will be running on minimum flow. Removing DG2 unavailability from the stem would make the first part of C & D correct. Providing RPV pressure below 350 psig would make second part of B & D correct.
Distracters: B. The answer is incorrect due to the Core Spray pump running on min flow. This choice is plausible due to confusing Core Spray pump shutoff head pressure or failing to recognized reactor pressure being above CS injection pressure (pressure is below injection valve open permissive of <436 psig). The candidate that correctly identifies only CS pump A operating and confuses whether the pump is injecting vs. on min flow would choose this answer. C. The answer is incorrect due to only Core Spray pump A running. This choice is plausible due to not recognizing the failure of the emergency transformer to automatically fast transfer with DG2 unavailable The candidate that confuses core spray pump status and correctly identifies the pump operating on min flow would choose this answer. D. The answer is incorrect due to only Core Spray pump A running AND the Core Spray pump running on min flow. This choice is plausible due to not recognizing the failure of the emergency transformer to automatically fast transfer with DG2 unavailable AND confusing Core Spray pump shutoff head or failing to recognized reactor pressure being above CS injection pressure. The candidate that confuses core spray pump status and confuses whether the pump is injecting vs. on min flow would choose this answer.
Technical References: Procedure 2.2.9 (Core Spray System), Rev. 78 Procedure 2.2.18 (4160V Auxiliary Power Distribution System), Rev. 181

References to be provided to applicants during exam: NONE		
Learning Objective:		
COR0020602001070A Predict the consequences a malfunction of the following would have on the Core Spray System: A.C. electrical power		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b) 7	
Level of Difficulty:	2	
SRO Only Justification:	N/A	



4160V DISTRIBUTION SYSTEM

Figure 4, Rev. 10

COR001-01

f. MCC-T (MCC-W).

2.4.5 When Relay 27X5/1F (27X5/1G) drops out, following actions occur:

2.4.5.1 A close permissive signal is applied to the remote and local manual closing logic of Breaker EG-1 (EG-2).

2.4.5.2 A close permissive signal is applied to the remote manual closing logic of Breaker 1FA (1GB).

2.4.5.3 A close permissive signal is applied to the remote manual closing logic and a close signal is applied to the automatic closing logic of Breaker 1FS (1GS).

2.4.6 At this time, Breaker 1FS (1GS) will close to energize Bus 1F (1G).

2.4.6.1 If Breaker 1FS (1GS) does <u>not</u> close, Breaker EG-1 (EG-2) will close when Relay 27X3/1F (27X3/1G) has timed out and DG1 (DG2) is at rated speed and voltage.

2.4.7 If Bus 1F (1G) has been energized by the DG, as indicated by Breakers EG-1 (EG-2) and 1FE (1GE) being closed, or the Emergency Transformer, as indicated by Breaker 1FS (1GS) being closed, following actions occur:

2.4.7.1 Relay 27/1F1 (27/1G1) picks up and causes Relays 27X/1F (27X/1G) <u>and</u> 27XX/1F (27XX/1G) to drop out.

2.4.7.2 Relay 27/1F2 (27/1G2) picks up and causes Relay 27X/1F2 (27X/1G2) to drop out.

ATTACHMENT 5 INFORMATION SHEET

1. DISCUSSION

1.1 FUNCTION

1.1.1 The CS System automatically provides low pressure injection to the top of the fuel assemblies in time and at a sufficient flow rate to cool the core, and limit fuel clad temperature when reactor vessel pressure is ≤ 436 psig and an initiation signal of reactor low water level or high drywell pressure is present.

1.2 OPERATING CHARACTERISTICS

1.2.1 Two independent subsystems are provided as a part of the CS System. Each subsystem consists of following:

1.2.1.1 One 100% capacity centrifugal pump driven by an electric motor.

1.2.1.2 A spray sparger in the reactor vessel above the core.

1.2.1.3 Piping and valves to convey water from the suppression pool to the sparger.

1.2.1.4 The associated controls and instrumentation.

1.2.2 The CS System supplies 4720 gpm (per pump) of water at a back pressure of 113 psig to the reactor vessel as part of the Core Standby Cooling (CSC) Systems.

2. INTERLOCKS AND SETPOINTS

2.1 CS Pump A(B) starts automatically on an initiation signal of high drywell pressure (≤ 1.84 psig) or low reactor vessel water level (≥ -113" indicated below instrument zero).

2.1.1 With normal power available, the pumps start after a 10 second time delay. This time delay for starting prevents voltage dips on the 4160V emergency buses due to ECCS initiation.

2.1.2 With a loss of off-site power, the pumps start 10 seconds after the restoration of power. This time delay is to prevent overloading the emergency power supply during the ECCS initiation.

2.2 If a CS initiation signal is present and a CS pump is stopped with its control switch on Panel 9-3, the amber PUMP STOP SIG SEALED-IN light will come on. The pump can be restarted manually but will not auto start. This light remains on as long as the CS initiation signal is present. It resets automatically when the initiation signal clears.

9. INJECTING TO RPV WITH CS SYSTEM

9.1 Start CS Subsystem A as follows:

9.1.1 Ensure reactor pressure < 350 psig.

9.1.2 While observing motor start criteria in Step 2.5, start CS Pump A.

9.1.3 Throttle open CS-MO-12A, INBD INJ THROTTLE VLV, as necessary, per EOPs to restore and maintain adequate core cooling.

NOTE – CS-MO-5A closes when subsystem flow has been > ~ 2120 gpm for ~ 8 seconds.

9.1.4 Ensure CS-MO-5A, MIN FLOW BYP VLV, closes.

9.1.5 IF PCIS Group 6 lights lit on Panel 9-5, at VBD-M, THEN ensure REC-MO-711 or REC-MO-714, CRITICAL LOOP SUPPLY (associated with an in service REC HX), is open.

NOTE – If a CS System initiation signal is present, Step 9.1.6 is N/A.

9.1.6 Prior to reaching desired reactor vessel level, level can be controlled at desired level. IF an initiation is not present, THEN perform following:

9.1.6.1 Throttle closed CS-MO-12A, INBD INJ THROTTLE VLV, to maintain desired vessel level.

9.1.6.2 Throttle open CS-MO-26A, TEST LINE RECIRC VLV, to maintain desired flow indicated on CS-FI-50A, PUMP FLOW.

NOTE – If Step 9.1.6 was performed, Step 9.1.7 is N/A.

9.1.7 WHEN desired reactor level is reached, THEN perform following:

9.1.7.1 Throttle CS-MO-12A, INBD INJ THROTTLE VLV, as necessary, per EOPs to maintain desired reactor vessel level.

CAUTION – CS-MO-12A may not fully close against pump flow and pressure unless pump is secured first.

9.1.7.2 IF reactor vessel level cannot be maintained below desired level, THEN close CS-MO-12A.

a. IF CS-MO-12A leaks by more than is acceptable and flow is desired to be totally secured, THEN perform Step 9.1.7.3.

9.1.7.3 IF it is anticipated that injection with subsystem will not be required for 60 minutes, THEN perform Step 9.1.8 to ensure the subsystem remains ready for restart.

Examination Outline Cross-Reference	Level	RO
Eliminated BITT reference due to not being required to correctly answer question – other parameters provided in stem to support plausibility associated with not requiring SLC initiation. Removed graphs from references. Added RCIC & CRD to stem for systems maintaining RPV water level to eliminate questions/assumptions of RCIC availability.	Tier#	2
	Group#	1
	K/A #	211000 A2.08
	Rating	4.1

211000 SLC

A2. Ability to (a) predict the impacts of the following on the STANDBY LIQUID CONTROL SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5 / 45.6)

A2.08 Failure to SCRAM

Question 33

The following plant conditions are present after a manual Scram:

- Reactor power is 25%.
- Both Reactor Recirculation pumps are tripped.
- Periodic neutron flux oscillations are 35% peak-to-peak.
- RPV Pressure is 900 psig being controlled by BPVs.
- RPV Water Level is -70 inches being maintained with HPCI, RCIC, and CRD.
- Torus Water Level is 12 feet and stable.
- Average Torus Water Temperature is 105°F and slowly rising

Which one of the following completes the statements below regarding the requirement to initiate SLC under the current conditions and **IF** SLC is initiated, the action required if both squib valves fail to fire?

SLC ____(1)____ required to be initiated at this time.

If SLC injection were required and both squib valves failed to fire, alternate boron injection via ____(2)____ is required to be performed.

A. (1) is
(2) RWCU ONLY

B. (1) is
(2) RWCU or RCIC

C. (1) is NOT
(2) RWCU ONLY

D. (1) is NOT
(2) RWCU or RCIC

Answer:

B. (1) is
(2) RWCU or RCIC

Explanation:

Requires knowledge of when SLC is required to be initiated during an ATWS. If neutron flux oscillations are in excess of 25% peak-to-peak, SLC is required to be initiated. SLC is also required to be injected prior to reaching BITT (at 25% power, BITT is 110°F SP temperature which makes NOT required at this time plausible).
If SLC has been initiated with both squib valves failing to fire requires securing running SLC pumps and aligning for alternate boron injection to RWCU or RCIC. Procedure 5.8.8 provides guidance to bypass all Group 3 isolations making RWCU available for alt boron injection.

Distracters:

A. The answer is incorrect due to Alternate boron injection being aligned to either RWCU or RCIC. This choice is plausible due to confusing system availability for alternate boron injection. The candidate that recognizes SLC being required to be injected and confuses systems available for alt boron injection would choose this answer.
C. The answer is incorrect due to SLC injection being required and Alternate boron injection being aligned to either RWCU or RCIC. This choice is plausible due to confusing when SLC injection requirements (25% peak to peak flux oscillations or prior to reaching BITT) and system availability for alternate boron injection. The candidate that confuses SLC injection conditions and systems available for alt boron injection would choose this answer.
D. The answer is incorrect due to SLC injection being required. This choice is plausible due to confusing when SLC injection requirements (25% peak to peak flux oscillations or prior to reaching BITT). The candidate that confuses SLC injection conditions and correctly identifies systems available for alt boron injection would choose this answer.

Technical References:

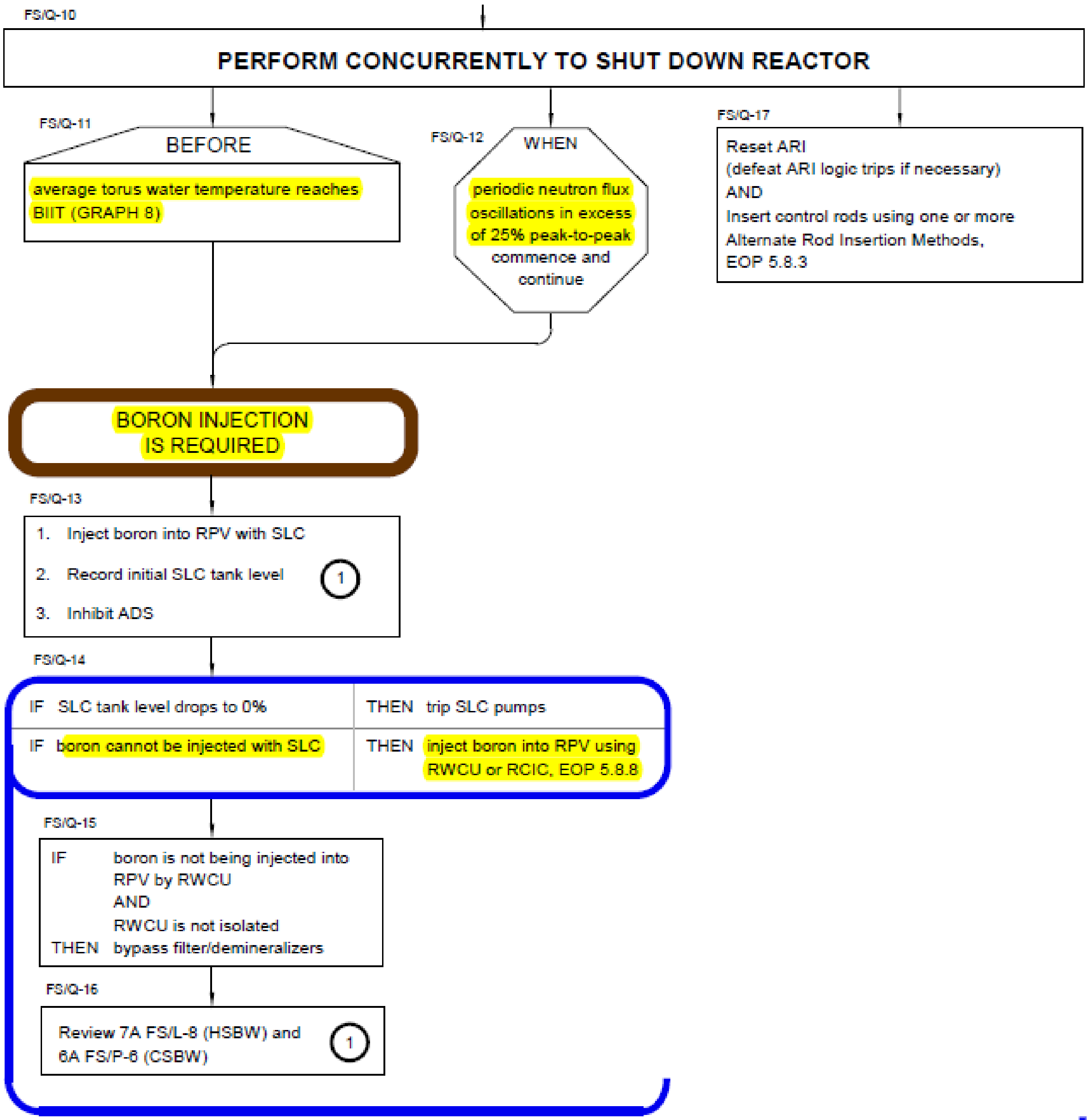
EOP 6A (ATWS Pressure & Power Control), Rev. 16
EOPSAG (EOP and SAG Graphs), Rev. 15

References to be provided to applicants during exam: NONE

Learning Objective:

INT00806060010800 Explain the basis for injecting boron before the Boron Injection Initiation Temperature is exceeded and when large periodic neutron flux oscillations in excess of 25% occur.

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b) 5	
Level of Difficulty:	3	
SRO Only Justification:	N/A	


FS/Q-10
PERFORM CONCURRENTLY TO SHUT DOWN REACTOR
FS/Q-11
BEFORE
average torus water temperature reaches BIIT (GRAPH 8)
FS/Q-12
WHEN
periodic neutron flux oscillations in excess of 25% peak-to-peak commence and continue
FS/Q-17
Reset ARI
(defeat ARI logic trips if necessary)
AND
Insert control rods using one or more Alternate Rod Insertion Methods, EOP 5.8.3
BORON INJECTION IS REQUIRED
FS/Q-13
1. Inject boron into RPV with SLC
2. Record initial SLC tank level
1
3. Inhibit ADS
FS/Q-14
IF SLC tank level drops to 0%
THEN trip SLC pumps
IF boron cannot be injected with SLC
THEN inject boron into RPV using RWCU or RCIC, EOP 5.8.8
FS/Q-15
IF boron is not being injected into RPV by RWCU
AND
RWCU is not isolated
THEN bypass filter/demineralizers
FS/Q-16
Review 7A FS/L-8 (HSBW) and 6A FS/P-6 (CSBW)
1

Examination Outline Cross-Reference	Level	RO
	Tier#	2
	Group#	1
	K/A #	212000 A3.06
	Rating	4.2

212000 RPS

A3. Ability to monitor automatic operations of the REACTOR PROTECTION SYSTEM including: (CFR: 41.7 / 45.7)

A3.06 Main turbine trip: Plant-Specific

Question 34

The plant is operating at 35% power with the following annunciator in alarm due to MS-PS-14C (TSV & TCV Closure Trip Bypassed Chan A2) failure:

TSV & TCV CLOSURE TRIP BYP CHAN A/B	PANEL/WINDOW: 9-5-2/C-4

CRT alarm message indicates:
- (2705) TSV & TCV CLOSURE TRIP BYPASSED CHAN A2

What is the status of RPS if the Main Turbine trips?

A. Only RPS A is de-energized.

B. Only RPS B is de-energized.

C. Both RPS A and B are de-energized.

D. Both RPS A and B remain energized.

Answer:
C. Both RPS A and B are de-energized.

Explanation:
With the plant operating at 35% power and MS-PS-14C (TSV & TCV Closure Trip Bypassed Chan A2) failing downscale, if the main turbine trips, 3 of the 4 auto Scram

channels will de-energize therefore de-energizing both RPS A & B.

Distracters:

A. This answer is incorrect due to RPS B also de-energizes. This choice is plausible if the stem were changes to reflect B TSV/TCV bypassed or RPS trip logic being easily confused. The candidate that confuses which RPS trip channels are currently bypassed would select this answer.

B. This answer is incorrect due to RPS B also de-energizes. This choice is plausible if the stem were changes to reflect B TSV/TCV bypassed or RPS trip logic being easily confused. The candidate that confuses which RPS trip channels are currently bypassed would select this answer.

D. This answer is incorrect due to RPS B also de-energizes. This choice is plausible if the stem were changes to reflect B TSV/TCV bypassed or RPS trip logic being easily confused. The candidate that confuses which RPS trip channels are currently bypassed would select this answer.

Technical References:

Procedure 2.3_9-5-2 (PANEL 9-5 - ANNUNCIATOR 9-5-2), Rev. 44

References to be provided to applicants during exam: NONE

Learning Objective:

COR0022102001090C Predict the consequences a malfunction of the following would have on the RPS system: Nuclear boiler instrumentation

COR0022102001040J Describe the RPS design features and/or interlocks that provide for the following: Bypassing of selected Scram signal (manually and automatically)

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b) 7	
Level of Difficulty:	2	
SRO Only Justification:	N/A	

TSV & TCV CLOSURE TRIP BYP CHAN A/B	PANEL/WINDOW: 9-5-2/C-4

1. AUTOMATIC ACTIONS

1.1 When below setpoint on all four logic channels, TCV and TSV trips are bypassed. Actuation of a TCV or TSV trip under these conditions will not result in a scram.

2. OPERATOR OBSERVATION AND ACTION

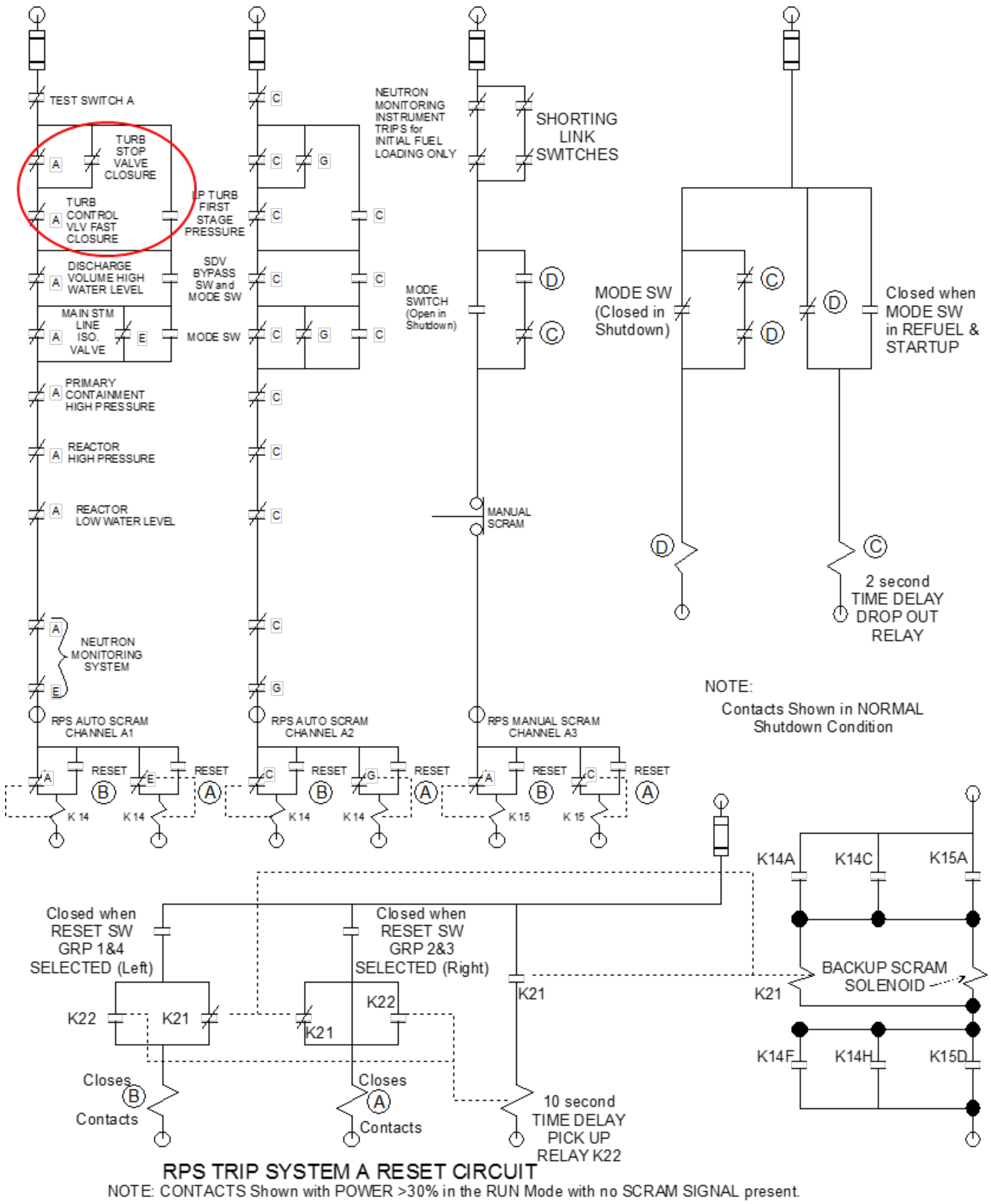
2.1 Ensure conditions of turbine and reactor are consistent with alarm.

2.2 Whenever power is raised above setpoint, ensure alarm clears.

SETPOINT	CIC	9-5-2/C-4
164.5 psig first stage turbine pressure < 29.5% RTP as measured by turbine first stage pressure (Tech Spec < 29.5% RTP/187.8 psig):		
1. (2704) TSV & TCV CLOSURE TRIP BYPASSED CHAN A1	1. MS-PS-14A	
2. (2705) TSV & TCV CLOSURE TRIP BYPASSED CHAN A2	2. MS-PS-14C	
3. (2706) TSV & TCV CLOSURE TRIP BYPASSED CHAN B1	3. MS-PS-14B	
4. (2707) TSV & TCV CLOSURE TRIP BYPASSED CHAN B2	4. MS-PS-14D	

PROBABLE CAUSES
- Lowering reactor power (i.e., reactor shutdown).

REFERENCES
- Technical Specification LCO 3.3.1.1, Reactor Protection System (RPS) Instrumentation.



RPS SCRAM RESET LOGIC

Figure 10, Rev. 13

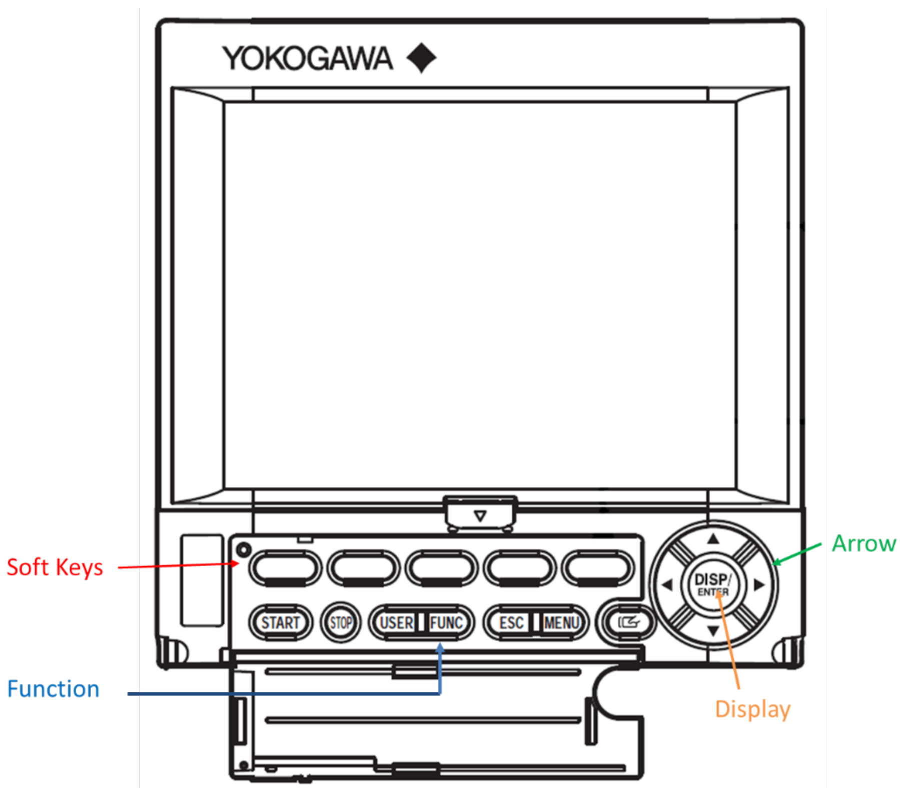
COR002-21-02

Examination Outline Cross-Reference	Level	RO
Procedure 4.19 (Honeywell and Yokogawa Digital Recorders) sections have been classified as skill of the craft which supports operation from memory.	Tier#	2
	Group#	1
	K/A #	215003 A4.01
	Rating	3.3

215003 IRM
A4. Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8)
A4.01 IRM recorder indication

Question 35

Which one of the following identifies how to place an IRM recorder in Second Speed IAW Procedure 4.19 (Honeywell and Yokogawa Digital Recorders)?



Depress the ____(1)____ key.

Utilize the applicable____(2)____ key to select SECOND SPEED.

A. (1) DISP
   (2) Arrow

B. (1) DISP
   (2) Soft

C. (1) FUNC
   (2) Arrow

D. (1) FUNC
   (2) Soft

Answer:
D. (1) FUNC
   (2) Soft

Explanation:
To place an IRM recorder in second speed requires opening the front key cover, pressing the function key, and selecting SECOND SPEED.

Distracters:
A. This answer is incorrect due to the FUNC & Soft keys being required to be depressed to select SECOND SPEED. This choice is plausible due to the DISP key being utilized to support different screen displays and the Arrow keys move around different display options. The candidate that confuses DISP and Arrow keys would select this answer.
B. This answer is incorrect due to the FUNC & Soft keys being required to be depressed to select SECOND SPEED. This choice is plausible due to the DISP key being utilized to support different screen displays. The candidate that does confuses DISP and correctly identifies soft keys would select this answer.
C. This answer is incorrect due to the FUNC & Soft keys being required to be depressed to select SECOND SPEED. This choice is plausible due to the Arrow keys move around different display options. The candidate that does correctly identifies DISP and confuses Arrow keys would select this answer.

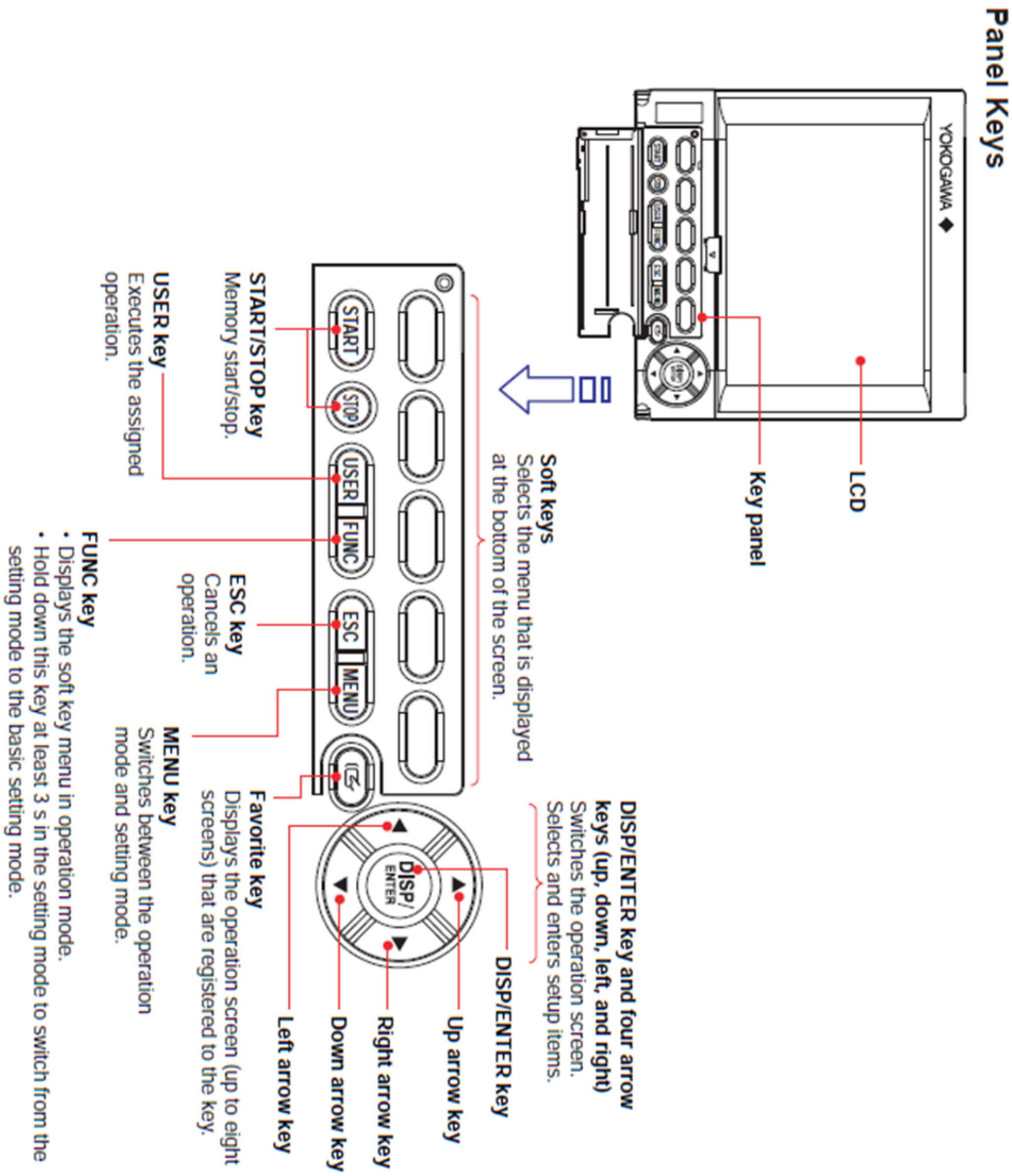
Technical References:
Procedure 4.19 (Honeywell And Yokogawa Digital Recorders), Rev. 54
Procedure 2.1.1 (Startup Procedure), Rev. 182

References to be provided to applicants during exam: NONE

Learning Objective: Task 299001P0301 - Change recorder chart paper and ink {not selected for training due to being skill of the craft (administrative)}.

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X

Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b) 7	
Level of Difficulty:	4	
SRO Only Justification:	N/A	



17. VIEWING HISTORICAL DATA ON A HONEYWELL RECORDER

NOTE – Honeywell historical data available for viewing on recorder is limited to data within recorders internal memory which is typically < 24 hours' worth of data.

17.1 Select chart of data to be viewed by rotating thumbwheel until an arrow appears on appropriate chart.

17.2 Select chart by pressing thumbwheel.

17.2.1 IF an option menu pops up asking to reset all min's and max's, THEN select EXIT using thumbwheel and repeat Step 17.1 with arrow in a different location on appropriate chart.

17.3 Select REPLAY using thumbwheel.

17.4 Chart is now in replay mode. Bottom right-hand corner will display NORMAL. This is speed in which thumbwheel will scroll through data. Pressing thumbwheel will cycle this from NORMAL to FAST to SLOW and back to NORMAL.

17.5 Rotating thumbwheel will scroll chart backwards or forwards on timeline to allow viewing of historical data.

17.6 Once historical data viewing is completed, select EXIT in lower left-hand corner by using appropriate soft key.

18. VIEWING HISTORICAL DATA ON A YOKOGAWA RECORDER

NOTE – Yokogawa historical data available for viewing on recorder is limited to data within recorders internal memory which is typically < 24 hours' worth of data.

18.1 Press DISP/ENTER button.

18.2 Using arrow keys, highlight TREND HISTORY.

18.3 Using arrow keys, highlight desired trend.

18.4 Select ENT.

14.6.23 Using appropriate soft key, select KEYLOCK.

NOTE – Key symbol should appear in top right corner of screen.

14.6.24 Perform recorder status check per Section 9.

14.7 IF PC Card was changed out and will be sent to CNS Records, THEN label PC Card with following information:

14.7.1 Recorder CIC.

14.7.2 Chart Number.

14.7.3 Date.

14.7.4 Time.

15. CHANGING HONEYWELL RECORDER CHART SPEEDS

NOTE – Chart speeds were originally set to 30 mm/hr on Honeywell recorders based upon chart speed of recorders they replaced. Honeywell recorders chart speed can be changed (from 1 mm/hr to 12000 mm/hr - typically 30 mm/hr) by Operations at any time by using appropriate steps below.

15.1 Select chart to change speed of by rotating thumbwheel until an arrow appears on appropriate chart.

15.2 Select chart by pressing thumbwheel.

15.2.1 IF an option menu pops up asking to reset all min's and max's, THEN select EXIT using thumbwheel and repeat Step 15.1 with arrow in a different location on appropriate chart.

15.3 Select appropriate chart speed using thumbwheel.

16. CHANGING YOKOGAWA RECORDER CHART SPEEDS

NOTE – Chart speeds were originally set on Yokogawa recorders based upon chart speed of recorders they replaced. Yokogawa recorders chart speed can be selected as NORMAL SPEED or SECOND SPEED using appropriate steps below; however, not all recorders have this feature enabled.

16.1 Open front key cover.

16.2 Press FUNC button.

16.3 Using appropriate soft key, select NORMAL SPEED or SECOND SPEED depending on which is applicable.

NOTE 1 – Step 4.12 satisfies SR 3.3.1.1.5 for Table 3.3.1.1-1, Function 1a.

NOTE 2 – During performance of SRM to IRM overlap, SRM high and high-high alarms may be received dependent on core conditions as count level rises while waiting for IRM response.

4.12 Verify SRM to IRM overlap as follows:

4.12.1 Verify all OPERABLE IRM channels indicate on-scale, prior to operable SRMs reaching 1×10^6 cps, with SRM detectors fully inserted.

4.12.2 AFTER proper response is received on all OPERABLE IRM channels, THEN position SRM detectors per Procedure 4.1.1 to maintain a SRM count rate of 10^3 to 10^5 cps. Closely monitor all SRM indications.

Initials/Time/Date: _______/_______/_______

4.12.2.1 IF SRM(s) cannot be withdrawn, THEN refer to Procedure 4.1.1.

Initials/Time/Date: _______/_______/_______

No procedure reference provided

4.13 WHEN positive response is observed on IRMs, THEN perform following:

4.13.1 Place one IRM recorder in each logic channel on second speed and annotate chart with time and date.

4.13.2 Place SRM recorder on normal speed and annotate chart with time and date.

Initials/Time/Date: _______/_______/_______

CAUTION – Reactor Protection System channel trip occurs at ≤ 121/125 of any IRM recorder scale and on any range.

NOTE – Rod withdrawal blocks will occur if IRM channel indication falls below downscale trip setpoint in any range except Position 1 or if upscale trip setpoint in any range is exceeded.

4.13.3 Range IRMs, as required, to maintain an indication on IRM recorders between ~ 25 to ~ 75 on 0 to 125 scale.

Initials/Time/Date: _______/_______/_______

3.2 YOKOGAWA RECORDER

3.2.1 To annotate chart with message "Recorder SAT", perform following:

3.2.1.1 Open front key cover.

3.2.1.2 Press USER button.

3.2.1.3 Close front key cover.

3.2.2 To annotate chart with any other preprogrammed message, perform following:

3.2.2.1 Open front key cover.

3.2.2.2 Press FUNC button.

3.2.2.3 Using appropriate soft key, select MESSAGE.

3.2.2.4 Using appropriate soft key, select appropriate Group and/or message from list.

Examination Outline Cross-Reference	Level	RO
Comments incorporated – New Answer is "C".	Tier#	2
	Group#	1
	K/A #	215004 G2.1.20
	Rating	4.6

215004 Source Range Monitor

G2.1.20 Ability to interpret and execute procedure steps. (CFR: 41.10 / 43.5 / 45.12)

Question 36

The reactor is critical during plant startup.

When are SRMs first allowed to be FULLY withdrawn IAW Procedure 2.1.1 (Startup Procedure)?

As soon as…

A. SRM/IRM overlap has been verified for all operable IRM channels.

B. all SRM channels have exceeded 1x10^5 cps.

C. all operable IRM channels are on range 3.

D. all operable IRM channels are on range 8.

Answer:

C. all operable IRM channels are on range 3.

Explanation:

Step 4.14 of procedure 2.1.1, Startup Procedure, states to fully withdraw SRMs when all operable IRM channels are on range 3 or above. SRM downscale, not full in functions are only required operable with IRMs on range 1 or 2. Therefore, answer A is correct due to being the FIRST condition which allows for full withdrawal..

Distracters:

A. This answer is incorrect due to SRMs are not allowed to be fully withdrawn until all operable IRMs are on range 3. This choice is plausible because SRMs are allowed to be partially withdrawn after IRM overlap has been verified by observing IRMs onscale before SRMs reach 1x10^6 cps, which is done with IRMs on range 1.

B. This answer is incorrect due to SRMs are not allowed to be fully withdrawn until all operable IRMs are on range 3. It is plausible because SRMs are partially withdrawn to maintain SRM readings below $1x10^5$ cps, after IRM overlap has been verified.

D. This answer is incorrect due to range 3 being encountered before range 8, and SRMs are FIRST allowed to be fully withdrawn on range 3. This answer is plausible because IRM range 8 bypasses ALL SRM functions, namely the SRM high and inop rod blocks.

Technical References:
Procedure 2.1.1 (Startup Procedure), Rev. 182
TS 3.3.1.2, SRM Instrumentation
TRM 3.3.1, Control Rod Block Instrumentation.

References to be provided to applicants during exam: NONE

Learning Objective:
INT032010400A0300 Describe the general sequence of events performed during a Reactor startup and RPV heatup.

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b) 10	
Level of Difficulty:	3	
SRO Only Justification:	N/A	

CAUTION – If moderator temperature is dropping when approaching criticality, caution should be exercised. Due to greater moderation, period may become shorter, requiring rod insertion until reactor power reaches heating range and moderator temperature starts rising.

NOTE – Reactor is critical when neutron flux rises with a constant (stable) period without requiring additional control rod withdrawal.

4.14 WHEN reactor is critical, as indicated by SRMs or IRMs, THEN perform following:

4.14.1 Calculate reactor period by multiplying doubling time by 1.443.

4.14.2 Log control rod number, control rod position, moderator temperature, reactor pressure, sequence, reactor period, and time on Procedure 10.13, Attachment 1, and in Control Room Operator's Log.

Initials/Time/Date: ________/________/________

4.14.3 Annotate following SRM and IRM recorders as to time and date of reactor criticality:

4.14.3.1 NM-NR-45, SOURCE RANGE LEVEL.

4.14.3.2 NM-NR-46A, IRM-APRM.

4.14.3.3 NM-NR-46B, IRM-APRM/RBM.

4.14.3.4 NM-NR-46C, IRM-APRM/RBM.

4.14.3.5 NM-NR-46D, IRM-APRM.

Initials/Time/Date: ________/________/________

NOTE – Steps 4.14.4 and 6.25 are N/A if not starting up from a refuel outage.

4.14.4 Inform Reactor Engineering to perform applicable portions of Procedure 6.REACT.603 within 4 hours of criticality.

Initials/Time/Date: ________/________/________

4.15 WHEN all OPERABLE IRMs are on Range 3 or above, THEN fully withdraw all SRM detectors per Procedure 4.1.1.

Initials/Time/Date: ________/________/________

4.15.1 IF SRM(s) cannot be withdrawn, THEN refer to Procedure 4.1.1.

Initials/Time/Date: ________/________/________

NOTE 1 – Step 4.12 satisfies SR 3.3.1.1.5 for Table 3.3.1.1-1, Function 1a.

NOTE 2 – During performance of SRM to IRM overlap, SRM high and high-high alarms may be received dependent on core conditions as count level rises while waiting for IRM response.

4.12 Verify SRM to IRM overlap as follows:

4.12.1 Verify all OPERABLE IRM channels indicate on-scale, prior to operable SRMs reaching 1×10^6 cps, with SRM detectors fully inserted.

4.12.2 AFTER proper response is received on all OPERABLE IRM channels, THEN position SRM detectors per Procedure 4.1.1 to maintain a SRM count rate of 10^3 to 10^5 cps. Closely monitor all SRM indications.

Initials/Time/Date: ________ / ________ / ________

4.12.2.1 IF SRM(s) cannot be withdrawn, THEN refer to Procedure 4.1.1.

Initials/Time/Date: ________ / ________ / ________

4.13 WHEN positive response is observed on IRMs, THEN perform following:

4.13.1 Place one IRM recorder in each logic channel on second speed and annotate chart with time and date.

4.13.2 Place SRM recorder on normal speed and annotate chart with time and date.

Initials/Time/Date: ________ / ________ / ________

CAUTION – Reactor Protection System channel trip occurs at ≤ 121/125 of any IRM recorder scale and on any range.

NOTE – Rod withdrawal blocks will occur if IRM channel indication falls below downscale trip setpoint in any range except Position 1 or if upscale trip setpoint in any range is exceeded.

4.13.3 Range IRMs, as required, to maintain an indication on IRM recorders between ~ 25 to ~ 75 on 0 to 125 scale.

Initials/Time/Date: ________ / ________ / ________

Table T3.3.1-1 (Page 1 of 3)
Control Rod Block Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	ACCEPTANCE LIMITS
1. SRM				
a. Detector Not Full In	2[a], [b]	1 per circuit loop	TSR 3.3.1.3 TSR 3.3.1.8	NA
	5	[c]	TSR 3.3.1.3 TSR 3.3.1.8	NA
b. Upscale	2[a]	1 per circuit loop	TSR 3.3.1.1 TSR 3.3.1.3 TSR 3.3.1.4 TSR 3.3.1.8	≤ 1 x 10^5 cps
	5	[c]	TSR 3.3.1.3 TSR 3.3.1.4 TSR 3.3.1.8	≤ 1 x 10^5 cps
c. Inoperative	2[a]	1 per circuit loop	TSR 3.3.1.3 TSR 3.3.1.4	NA
	5	[c]	TSR 3.3.1.3 TSR 3.3.1.4	NA
d. Downscale	2[a]	1 per circuit loop	TSR 3.3.1.1 TSR 3.3.1.3 TSR 3.3.1.4 TSR 3.3.1.8	≥ 3 cps
	5	[c]	TSR 3.3.1.3 TSR 3.3.1.4 TSR 3.3.1.8	≥ 3 cps

(continued)

(a) With IRMs on Range 2 or below.

(b) With SRM count rate < 100 cps.

(c) As required by Technical Specifications LCO 3.3.1.2.

Table 3.3.1.2-1 (page 1 of 1)
Source Range Monitor Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS
1. Source Range Monitor	2[(a)]	3	SR 3.3.1.2.1 SR 3.3.1.2.4 SR 3.3.1.2.6 SR 3.3.1.2.7
	3,4	2	SR 3.3.1.2.3 SR 3.3.1.2.4 SR 3.3.1.2.6 SR 3.3.1.2.7
	5	2[(b)(c)]	SR 3.3.1.2.1 SR 3.3.1.2.2 SR 3.3.1.2.4 SR 3.3.1.2.5 SR 3.3.1.2.7

(a) With IRMs on Range 2 or below.

(b) Only one SRM channel is required to be OPERABLE during spiral offload or reload when the fueled region includes only that SRM detector.

(c) Special movable detectors may be used in place of SRMs if connected to normal SRM circuits.

Examination Outline Cross-Reference	Level	RO
Revised question to test impact of bypassing a downscale LPRM on APRM indication and which APRM indication requires calibration. Requires analyzing current ARPM indication with an LPRM downscale and predicting the resultant indication due to bypassing this LPRM and determining which indication requires calibration by comparing provided indications to allowable deviation from CTP is HCL. Fixed distractors and correct answer is D.	Tier#	2
	Group#	1
	K/A #	215005 A4.06
	Rating	3.6

215005 APRM / LPRM

A4. Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8)

A4.06 Verification of proper functioning/ operability

Question 37

The plant is operating at 50% CTP with both Reactor Recirculation Loops in service.

- APRM B indicates 50.7%

An LPRM assigned to APRM B is required to be bypassed due to indicating downscale.

Which one of the following completes the statements below regarding the impact of bypassing this LPRM on APRM B indication and what APRM B indication requires calibration IAW Procedure 6.LOG.601 (Daily Surveillance Log - MODES 1, 2, and 3)?

APRM B indication will be ____(1)____ due to bypassing a downscale LPRM.
APRM calibration is required if APRM B indicates ____(2)____.

A. (1) lower
(2) 48.5%

B. (1) lower
(2) 52.3%

C. (1) higher
(2) 48.5%

D. (1) higher
(2) 52.3%

Answer:

D. (1) higher
(2) 52.3%

Explanation:

Requires analyzing the impact of bypassing a downscale LPRM on APRM indication and determining the APRM indication which exceeds +2% for two loop operation. Div 2 APRMs have 14 LPRMs assigned which are averaged to provide percent power indication. With APRM B indicating 50.7%, removing an LPRM reading downscale from the average will cause the APRM to indicate higher (if stem provides an LPRM reading high or upscale – indicating lower would be correct). (APRM - %CTP) readings shall be within $-2.0 \leq$ APRM - %CTP ≤ 2.0 making an indication of > 52% requiring calibration.

Distracters:

A. This answer is incorrect due APRM indication rises and being within $\pm$ 2% CTP. This choice is plausible if the stem were changed to provide the LPRM as reading high or upscale OR how APRM indication development is confused AND if indication is compared to the initial APRM indication of 50.7% (50.7 - 2.0 = 48.7) which 48.5% could be interpreted as needing calibration. The candidate that confuses APRM indication due to LPRM bypass and confuses the allowable deviation from CTP or from initial APRM indication would select this answer.

B. This answer is incorrect due APRM indication rises. This choice is plausible if the stem were changed to provide the LPRM as reading high or upscale OR how APRM indication development is confused. The candidate that confuses APRM indication due to bypassing a downscale LPRM and correctly identifies the allowable deviation from CTP or from initial APRM indication would select this answer.

C. This answer is incorrect due APRM indication being within $\pm$ 2% of CTP. This choice is plausible if indication is compared to the initial APRM indication of 50.7% (50.7 - 2.0 = 48.7) which 48.5% could be interpreted as needing calibration. The candidate that correctly identifies APRM indication rises and confuses the allowable deviation from CTP or from initial APRM indication would select this answer.

Technical References:

Procedure 10.1 (APRM Calibration), Rev. 49
Procedure 6.LOG.601 (Daily Surveillance Log - Modes 1, 2, and 3), Rev. 119

References to be provided to applicants during exam: NONE

Learning Objective:

COR0020102001110A Given an Average Power Range Monitor System control manipulation, predict the changes in the following parameters: Reactor power indication

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	

	New	X
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b) 7	
Level of Difficulty:	2	
SRO Only Justification:	N/A	

1. PURPOSE

1.1 This procedure provides instructions for Operations to calibrate the Average Power Range Monitor (APRM) System to the Core Thermal Power (CTP).

2. PRECAUTIONS AND LIMITATIONS

2.1 Prior to performing any adjustments to an APRM channel, it shall be bypassed.

2.2 Any APRM with its setting outside the two loop operation range of ± 2% RTP or corrected single loop operation range as required by LCO 3.4.1, as observed on a Periodic Case or Heat Balance Summary, shall be calibrated per this procedure.

2.3 Ensure APRM recorders are energized and APRM/IRM switches are in APRM.

3.4 Plant Startup - During plant startup, the APRM AGAFs are not very close to 1.00 because of the significantly different rod pattern from when the last calibration was performed.

3.5 Valid CTP - During reactor startup, a valid CTP (by heat balance) can be obtained when the active feedwater loop(s) flow is ≥ 0.9 Mlbm/hr. GARDEL will calculate an inaccurate heat balance if the active loop(s) flow is < 0.9 Mlbm/hr and will be swapped to calculated CTP using the APRMs. This is done for exposure accounting purposes only.

3.6 In the example below, the APRM Data section of the Heat Balance Summary or Periodic Case gives a table that directly calculates the APRM - %CTP. These parameters are determined at slightly different times by GARDEL and are rounded off, so round off differences may exist.

CTP = 99.9%	A	C	E	B	D	F
READING	100.4	100.2	98.8	100.3	99.5	99.7
AGAF	0.995	0.997	1.011	0.996	1.003	1.002
APRM - %CTP	0.5	0.3	-1.1	0.4	-0.3	-0.2

4. REFERENCES

4.1 COMMITMENTS AND OBLIGATIONS MATRIX

COMMITMENTS AND OBLIGATIONS	AFFECTED STEPS
QAPD	None
Ⓟ[1] SOER 90-3, Nuclear Instrument Miscalibration, September 11, 1990	3.2 and 4.3

4.2 TECHNICAL SPECIFICATIONS

4.2.1 3.3.1.1, Reactor Protection System (RPS) Instrumentation.

4.2.2 3.4.1, Recirculation Loops Operating.

4.3 TECHNICAL REQUIREMENTS MANUAL

4.3.1 TLCO 3.3.1, Control Rod Block Instrumentation.

4.3.2 TLCO 3.3.3, Non-Type A, Non-Category 1 Post-Accident Monitoring (PAM) Instrumentation.

ATTACHMENT 2 THERMAL LIMIT CHECKS

TIME	FLOW MLB/HR	SLO Y	SLO N	MFLCPR	MFLPD	MAPRAT	% PWR	APRM - %CTP A	C	E	B	D	F	NPP 10.1 Y	N
OPERABILITY LIMIT				≤ 1.00 (c)(d)	≤ 1.00 (c)(d)	≤ 1.00 (c)(d)	N/A	6.9 ≤ APRM - %CTP ≤ 10.9 (a) -2.0 ≤ APRM - %CTP ≤ 2.0 (b)						N/A	
ATTACHMENT 22 NOTES				7	71	6	N/A	119, 120, 121, 124, 125						N/A	

(a) Single loop operation limits (limit required to be established within 24 hours of entering single loop operation).
(b) Two loop operation limits.
(c) Applicable: ≥ 25% RTP.
(d) Surveillance frequency is once within 12 hours after ≥ 25% RTP and 24 hours thereafter.

IF any acceptance criteria is not satisfied, THEN perform Procedure 10.1 or 10.7 as applicable. ☐

Examination Outline Cross-Reference	Level	RO
Corrected explanations.	Tier#	2
	Group#	1
	K/A #	217000 K1.04
	Rating	2.6

217000 RCIC

K1. Knowledge of the physical connections and/or cause-effect relationships between REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) and the following: (CFR: 41.2 to 41.9 / 45.7 to 45.8)

K1.04 Main condenser

Question 38

Which one of the following identifies where the RCIC Steam Supply line drains are routed to during normal plant operation?

A. Main Condenser

B. Barometric Condenser

C. Floor Drain Sump

D. Equipment Drain Sump

Answer:
A. Main Condenser
Explanation: During normal system standby operation, the Steam Line Drain Isolation Valves AO-34 and 35 are open and the Trap Bypass Valve AO-32 is closed. This allows the steam to flow through the drain pot, condense in the steam trap, and drain to the Main Condenser. This arrangement maintains the steam line to the RCIC turbine as moisture-free as possible. The Steam Line Drain Isolation Valves AO-34 and 35 can be closed from the Control Room to prevent the release of radioactivity to the environment during accident conditions.
Distracters: B. This answer is incorrect due to RCIC steam line drains being routed to the main

condenser. This choice is plausible due to the barometric condenser receives leakoff steam from the turbine gland seals, trip throttle valve, and the governor valve. The candidate that does confuses steam leakoff drain with supply drains would select this answer.

C. This answer is incorrect due to RCIC steam line drains being routed to the main condenser. This choice is plausible due to the Barometric Condenser Condensate Pump pumps condensate to the Reactor Building Equipment Drain sump when RCIC is in Standby status which is easily confused with floor drains. The candidate that does confuses barometric condenser with main condenser and were it drains to would select this answer.

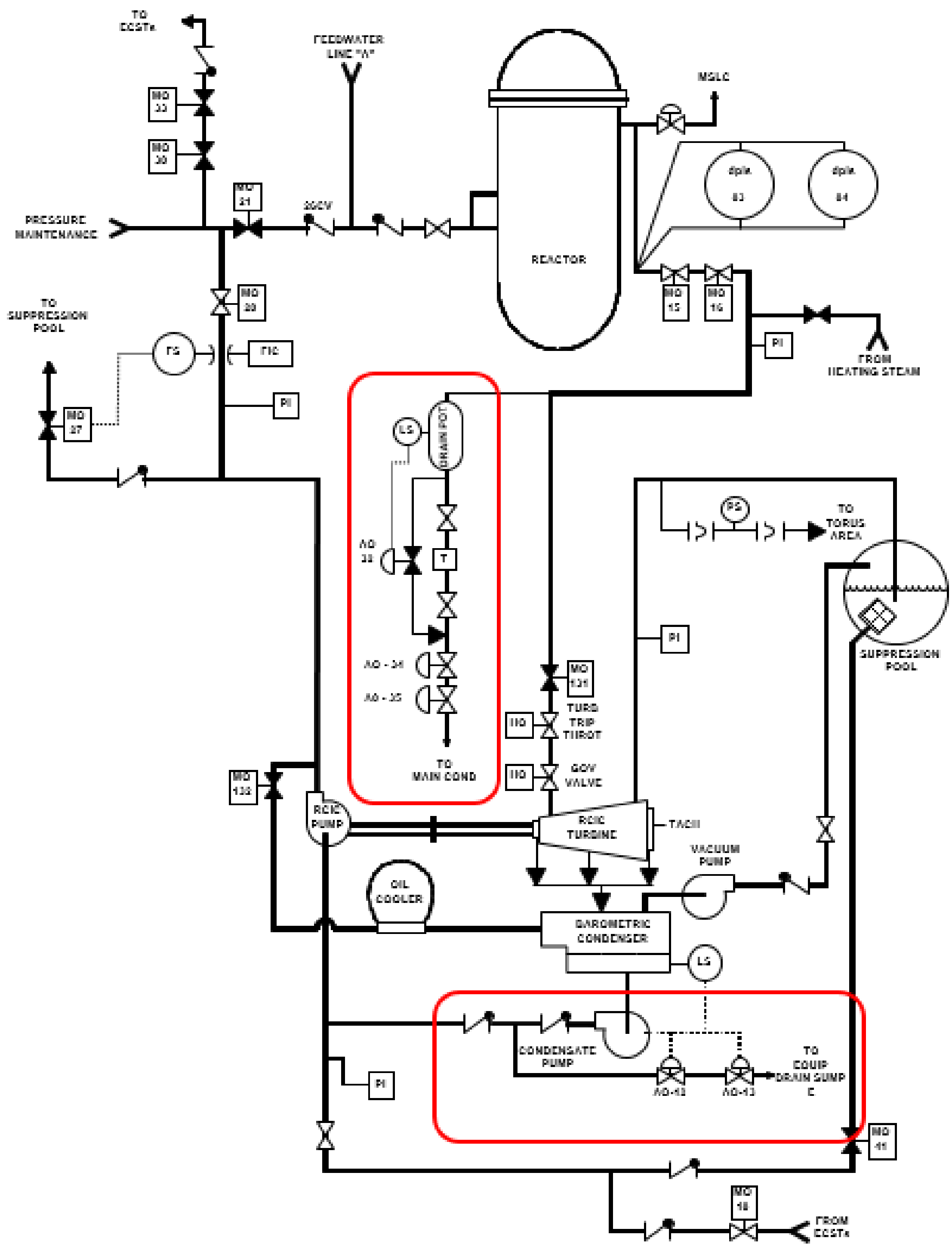
D. This answer is incorrect due to RCIC steam line drains being routed to the main condenser. This choice is plausible due to the Barometric Condenser Condensate Pump pumps condensate to the Reactor Building Equipment Drain sump when RCIC is in Standby status. The candidate that does confuses barometric condenser with main condenser would select this answer.

Technical References:
B&R Drawing 2043, Reactor Core Isolation Cooling System

References to be provided to applicants during exam: NONE

Learning Objective:
COR002-18-02 9. State how the following systems interrelate with the operation of the Reactor Core Isolation Cooling system:
j. Main Condenser

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b) 8	
Level of Difficulty:	3	
SRO Only Justification:	N/A	



SIMPLIFIED RCIC DIAGRAM

Figure 1, Rev. 1

COR002-18-02

Examination Outline Cross-Reference	**Level**	RO
	Tier#	2
	Group#	1
	K/A #	218000 K2.01
	Rating	3.1

218000 ADS

K2. Knowledge of electrical power supplies to the following: (CFR: 41.7)

K2.01 ADS logic

Examination Outline Cross-Reference	Level	RO
	Tier#	2
	Group#	1
	K/A #	218000 K2.01
	Rating	3.1

218000 ADS

K2. Knowledge of electrical power supplies to the following: (CFR: 41.7)

K2.01 ADS logic

Question 39

125V DC power has been lost to Panel AA2.

Which one of the following logics is de-energized?

A. ADS logic A

B. ADS logic B

C. LLS logic A

D. LLS logic B

Answer:

A. ADS logic A

Explanation:

ADS logic A is powered from Div 1 while B logic is powered from Div 2 with an alternate supply from Div1 DC. Both LLS logic channels are normally powered from Panel AA2, with an alternate supply from Panel BB2. Channels with alternate power supplies automatically shift to the alternate upon loss of the normal power supply.

Distracters:

B. This answer is incorrect due to ADS logic B primary power being supplied by Panel BB2. This choice is plausible due to ADS logic power supplies being easily confused. The candidate that confuses ADS logic power supplies would select this answer.

C. This answer is incorrect due to LLS logic A automatically transferring to alternate supply Panel BB2. This choice is plausible due to LLS logic power supplies being easily confused with ADS. The candidate that confuses ADS vs. LLS logic power supplies would select this answer.

D. This answer is incorrect due to LLS logic B automatically transferring to alternate supply Panel BB2. This choice is plausible due to LLS logic power supplies being easily confused with ADS. The candidate that confuses ADS vs. LLS logic power supplies would select this answer.

Technical References:
GE Drawing 791E253; Automatic Blowdown System
GE Drawing 944E689; Low-Low Set
Procedure 2.3_9-3-1 (Panel 9-3 - Annunciator 9-3-1), Rev. 34

References to be provided to applicants during exam: NONE

Learning Objective:
COR0021602001020A State the electrical power supply to the following NPR components: ADS logic

Question Source:	Bank # 7866	
(note changes; attach parent)	Modified Bank #	
	New	
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b) 7	
Level of Difficulty:	3	
SRO Only Justification:	N/A	

QUESTION: 36, 7866 (1 point(s))

125V DC power has been lost to Panel AA2.

Which of the following will be de-energized?

a. ADS logic A

b. ADS logic B

c. LLS logic A

d. LLS logic B

ANSWER: 36, 7866

a. ADS logic A

Distracters:

b. ADS logic B is normally supplied from BB2 with an alternate source of power from AA2.

c. LLS logic A is normally supplied from AA2 but will auto transfer to BB2 on lost of power.

d. LLS logic B is normally supplied from AA2 but will auto transfer to BB2 on lost of power.

REFERENCE: COR002-16 Student Text; 79/E253; 944E689

ADS CONTROL POWER FAILURE	PANEL/WINDOW: 9-3-1/E-1

1. AUTOMATIC ACTIONS

1.1 For alarm message (1008), ADS B logic control power transfers to emergency supply.

1.2 For alarm messages (1009 through 1016), associated relief valve solenoid and control power transfers to emergency supply.

SETPOINT	CIC	9-3-1/E-2
Relay operation caused by:		
1. (1038) LOW LOW SET A LOGIC POWER FAILURE	1. MS-REL-K22A	
2. (1039) LOW LOW SET B LOGIC POWER FAILURE	2. MS-REL-K22B	

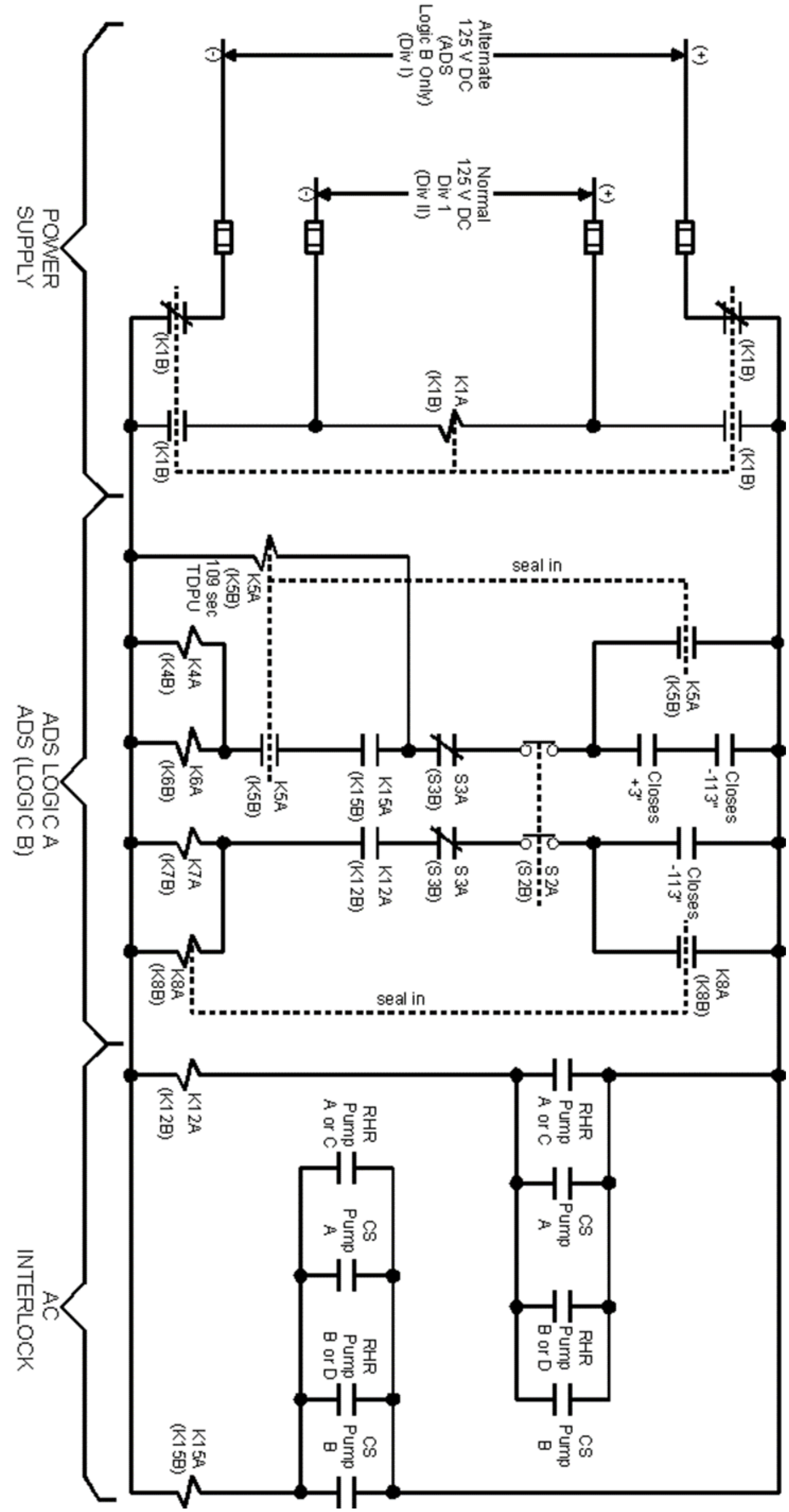
PROBABLE CAUSES

NOTE – Logic A operates SRV D and Logic B operates SRV F.

- Breaker trip; Panel AA2, Circuit 15, or Panel BB2, Circuit 8.
- Blown fuse; Panel 9-45 - 2E-F20A, 2E-F21A, 2E-F20B, or 2E-F21B.

REFERENCES

- LCO 3.3.6.3, Low-Low Set (LLS) Instrumentation.



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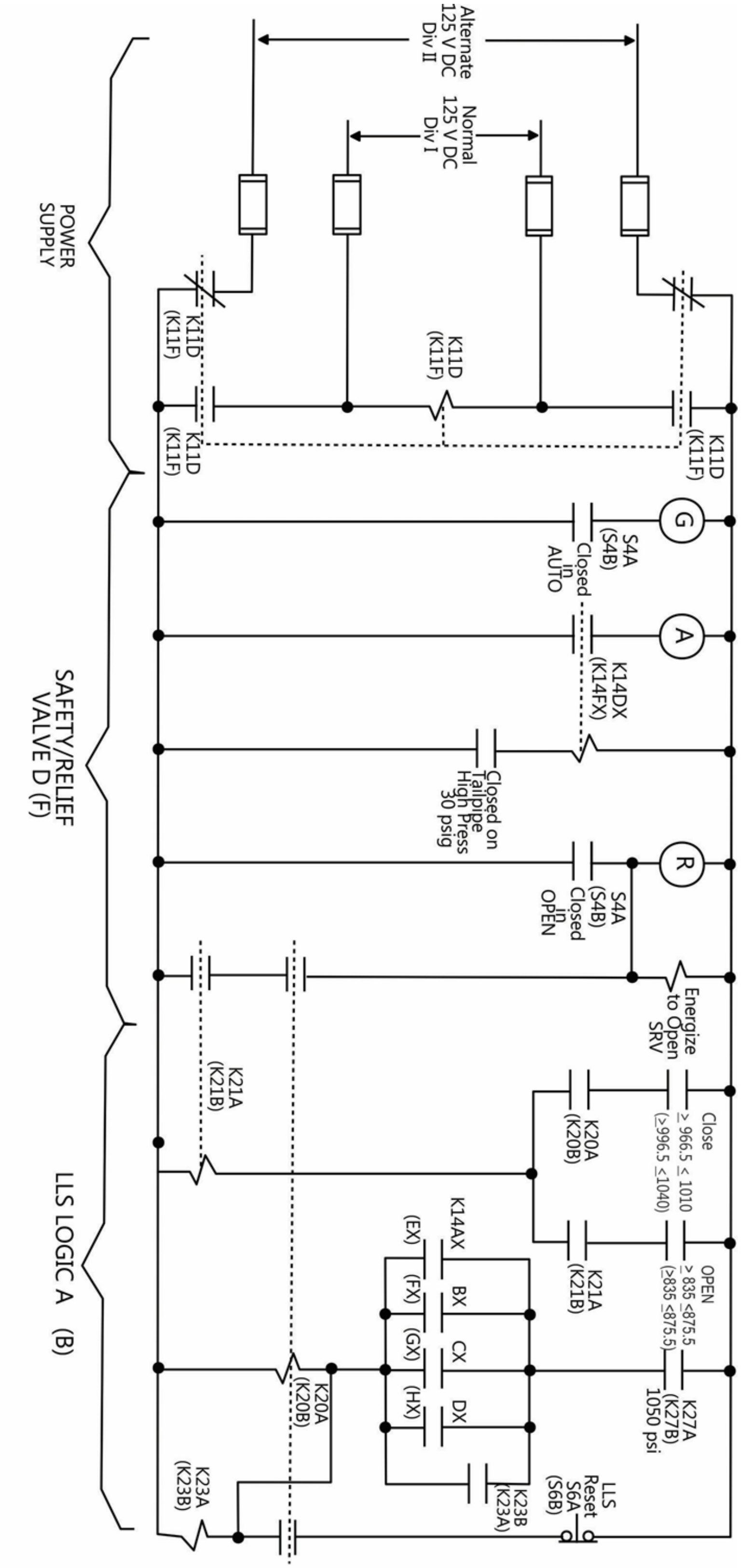
ADS
FIGURE 5, REV. 6
COR002-16

CXA04327



LLS
Figure 6, Rev. 12
COR002-16-02

Examination Outline Cross-Reference	Level	RO
	Tier#	2
	Group#	1
	K/A #	223002 K3.07
	Rating	3.7

223002 PCIS/Nuclear Steam Supply Shutoff

K3. Knowledge of the effect that a loss or malfunction of the PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF will have on following: (CFR: 41.7 / 45.4)

K3.07 Reactor pressure

Question 40

The plant is operating at rated power.

Which one of the following events will cause reactor pressure to rise and AUTOMATICALLY open a Safety Relief Valve (SRV) without any operator action?

A. One of the DEH pressure inputs fails low.

B. Both Reactor Recirculation pump speeds fail to maximum.

C. Main Steam Line Area Temperature exceeds 200°F due to ventilation failure.

D. RWCU resin intrusion causes MSL Radiation levels to exceed 4 x Normal.

Answer:

C. Main Steam Line Area Temperature exceeds 200°F due to ventilation failure.

Explanation:

MSIV isolation due to ventilation failure causing MSL temperature to rise above 195°F ($\leq$ 195°F setpoint) is considered a malfunction of PCIS due to the Main Steam Tunnel Temperature-High Function is provided to detect a break in a main steam line. With the main condenser unavailable as a heat sink, SRVs will continue to cycle to remove decay heat.

Distracters:

A. This answer is incorrect due to MSIV closure from rated power causing a pressure transient high enough to open SRVs and with decay heat continue to cycle due to the loss of the main condenser. This choice is plausible due to the DEH pressure controller compares pressure setpoint with actual throttle pressure and processes the difference through a gain stage, a LEAD/LAG compensator and a steam line compensator to develop an output pressure error signal. If all three pressure inputs are good, the selected output is equal to the median value. If two pressure inputs are good, the selected output is equal to the highest. The candidate that confuses DEH system response would select this answer.

B. This answer is incorrect due to MSIV closure from rated power causing a pressure transient high enough to open SRVs and with decay heat continue to cycle due to the loss of the main condenser. This choice is plausible due to RR pumps failing to maximum speed would raise reactor power and pressure. Reactor power or pressure exceeding the Scram setpoint would shutdown the reactor without opening SRVs due to the main condenser still being available with pressure being controlled with BPVs. The candidate that confuses run away RR pump transient with MSIV closure would select this answer.

D. This answer is incorrect due to MSIV closure from rated power causing a pressure transient high enough to open SRVs and with decay heat continue to cycle due to the loss of the main condenser. This choice is plausible due MSL Hi Hi rad requiring the operator to manually close the MSIVs & MSL drains IAW procedure 5.2FUEL only if the reactor is shutdown. The candidate that does not recognize manual operator action being required would select this answer.

Technical References:
Procedure 2.1.22 (Recovering From A Group Isolation), Rev. 60
Procedure 5.2FUEL (FUEL FAILURE), Rev.19
Procedure 2.2.77.1 {Digital Electro-Hydraulic (DEH) Control System}, Rev.37
TS 3.3.6.1

References to be provided to applicants during exam: NONE

Learning Objective:
COR0020302001210B Given plant conditions, determine if the following should have occurred: Any of the PCIS group isolations.

Question Source:	Bank # 2012 Fitz Q# 40	
(note changes; attach parent)	Modified Bank #	
	New	
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b) 7	
Level of Difficulty:	4	
SRO Only Justification:	N/A	

Examination Outline Cross-Reference:	Level	RO/SRO
	Tier #	2
	Group #	1
	K/A #	223002 K3.07
	Importance Rating	3.7/3.8

PCIS/Nuclear Steam Supply Shutoff: Knowledge of the effect that a loss or malfunction of the PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF will have on the following: Reactor pressure.

Proposed Question: 40

The plant is at 100% power. Assuming no operator actions, which one of the following would cause reactor pressure to increase and an SRV to lift?

A. The inservice ("A") EHC pressure regulator senses a lowering reactor pressure

B. An RWR controller malfunction causes speed to increase to its high limit

C. Due to an RWCU resin intrusion, Main Steam Line Radiation levels exceed 4x normal

D. A loss of main steam tunnel (MST) cooling causes MST temperatures to exceed 200°F

Proposed Answer: D, A loss of main steam tunnel (MST) cooling causes MST temperatures to exceed 200°F

4.4 Determine isolation cause:

ISOLATION	ALLOWABLE VALUE	COMMENTS
Reactor Low-Low-Low Water Level	≥ -113"	Ensure Group 7 Isolation.
Main Steam Line High Area Temperature	≤ 195°F	
Main Steam Line High Flow	≤ 142.7%	
Main Steam Line Low Pressure	≥ 835 psig while in RUN	
Main Condenser Low Vacuum	≥ 8" Hg	Loss of driving steam to SJAEs will allow OG line to be drawn back into condenser, causing loss of vacuum. May be bypassed if not in RUN and TSVs closed by placing CONDENSER LOW VACUUM LOGIC TEST bypass switches to BYPASS (Panels 9-15 and 9-17).
RPS Power Supply Failure	Loss of power	

4.5 Refer to following procedures as dictated by plant conditions:

PROCEDURE	TITLE
2.1.5	Reactor Scram
2.2.56	Main Steam System
5.8	Emergency Operating Procedures (EOPs)

4.6 IF isolation was caused by a RPS power supply failure, THEN perform following:

4.6.1 Verify affected RPS Power Panel has been transferred to its alternate power supply.

4.6.2 Reset Group 1 Isolation by turning Group ISOL RESET, CHANNEL A and CHANNEL B, switches (Panel 9-5) to left RESET position and then releasing to NOR.

4.6.3 Check Group 1, CHANNEL A Isolation lights turn on (Panel 9-5).

4.6.4 Check Group 1, CHANNEL B Isolation lights turn on (Panel 9-5).

4.6.5 Ensure MS-MO-74 open.

CNS OPERATIONS MANUAL EMERGENCY PROCEDURE 5.2FUEL FUEL FAILURE	USE: CONTINUOUS QUALITY: QAPD RELATED EFFECTIVE: 7/18/13 APPROVAL: ITR-RDM OWNER: OSG SUPV DEPARTMENT: OPS

1. ENTRY CONDITIONS

1.1 Unexplained rise in main steam line radiation.

1.2 Unexplained rise in off-gas activity.

1.3 Unexplained changes in core parameters (i.e., power, pressure, or core flow).

1.4 Unexplained significant rise in plant background or airborne radioactivity (CAM).

1.5 Higher than expected activity in reactor coolant sample.

1.6 Irradiated fuel damage with release of radioactivity to secondary containment as indicated by HIGH alarm on refueling floor ARM #2, CAM, or Reactor Building ventilation monitor.

Scram Actions

2. AUTOMATIC ACTIONS

2.1 None.

3. IMMEDIATE OPERATOR ACTIONS

3.1 None.

4. SUBSEQUENT OPERATOR ACTIONS

4.1 Record current time and date. Time/Date: ________/__________

4.2 Lower power, as required, to reduce off-gas and main steam line radiation levels.

4.3 Check OWC Injection System for proper operation.

4.4 **IF valid MAIN STM LINE HI HI RAD (9-4-1/A-4) alarm is actuated, THEN SCRAM and enter Procedure 2.1.5.**

4.5 IF OFF-GAS TIMER INITIATED (9-4-1/C-4) alarm is actuated, THEN concurrently enter Procedure 2.4OG.

4.6 IF valid MAIN STM LINE HI HI RAD (9-4-1/A-4) alarm is actuated and reactor is shut down, THEN close MSIVs and MSL drain valves.

4.7 Direct Chemistry to sample reactor coolant activity.

4.8 Notify Reactor Engineering and follow their recommendations per Procedure 10.31.

Scram Actions

Table 3.3.6.1-1 (page 1 of 3)
Primary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Main Steam Line Isolation					
a. Reactor Vessel Water Level - Low Low Low (Level 1)	1,2,3	2	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≥ -113 inches
b. Main Steam Line Pressure - Low	1	2	E	SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.6	≥ 835 psig
c. Main Steam Line Flow - High	1,2,3	2 per MSL	D	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 142.7% rated steam flow
d. Condenser Vacuum - Low	1, 2[(a)], 3[(a)]	2	D	SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.6	≥ 8 inches Hg vacuum
e. Main Steam Tunnel Temperature - High	1,2,3	2 per location	D	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 195°F
2. Primary Containment Isolation					
a. Reactor Vessel Water Level - Low (Level 3)	1,2,3	2	G	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≥ 3 inches
b. Drywell Pressure - High	1,2,3	2	G	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 1.84 psig
c. Reactor Building Ventilation Exhaust Plenum Radiation- High	1,2,3	2	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 49 mR/hr
d. Main Steam Line Radiation - High	1,2,3	2	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 3 times full power background
e. Reactor Vessel Water Level -Low Low Low (Level 1)	1,2,3	2	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≥ -113 inches

(continued)

(a) With any turbine stop valve not closed.

1.2.3 Control signals which provide proper positioning of the governor and bypass valves are initially developed by a pressure controller. The pressure controller compares pressure setpoint with actual throttle pressure and processes the difference through a gain stage, a LEAD/LAG compensator and a steam line compensator to develop an output pressure error signal.

1.2.3.1 If all three pressure inputs are good, the selected output is equal to the median value. If two pressure inputs are good, the selected output is equal to the highest. If one pressure input is good, it is selected.

1.2.3.2 If no pressure inputs are good, the pressure controller will transfer to manual control.

1.2.4 The pressure error signal, which is proportional to total steam demand, is next compared to the flow limiter setpoint logic. The flow limiter signal is an Operator adjustable signal and is normally set at 110%. Its purpose is to limit the pressure control signal in the event of a failure upstream calling for maximum steam demand. The lower of the two signals is sent to both the governor valve positioning logic and the bypass valve positioning logic.

1.2.5 The governor valve positioning logic circuit uses a Select logic to compare the pressure control signal with either the turbine speed control signal (Mode 2) or the turbine load control signal (Modes 3 and 4).

Examination Outline Cross-Reference	Level	RO
Comment incorporated – PC-PS-12A (Drywell High Pressure) is new correct answer.	Tier#	2
	Group#	1
	K/A #	223002 K6.05
	Rating	3.0

223002 PCIS/Nuclear Steam Supply Shutoff

K6. 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 : (CFR: 41.7 / 45.7)

K6.05 Containment instrumentation

Question 41

The plant is operating at rated power.

Which one of the following INOPERABLE instruments requires entry into Technical Specification 3.3.6.1 (Primary Containment Isolation Instrumentation)?

A. HPCI-LS-91A (SUPPR POOL HIGH LEVEL)

B. PC-PS-12A (DRYWELL HIGH PRESSURE)

C. PC-TR-24 (SUPP POOL WATER TEMP)

D. CS-DPIS-43A (CORE SPRAY A BREAK DETECTED)

Answer:
B. PC-PS-12A (DRYWELL HIGH PRESSURE)

Explanation:
Inoperability of PC-PS-12A (DRYWELL HIGH PRESSURE) requires entry into TS 3.3.6.1 due to providing input into Group 2 & 6 isolations

Distracters:
A. This answer is incorrect due to PC-PS-12A inoperability requiring entry into TS 3.3.6.1. This choice is plausible due to HPCI LS-91A inoperability requiring entry into TS 3.3.5.1(ECCS Instrumentation). The candidate that confuses TS 3.3.5.1 with 3.3.6.1 instruments would select this answer.

C. This answer is incorrect due to PC-PS-12A inoperability requiring entry into TS 3.3.6.1. This choice is plausible due to PC-TR-24 inoperability requiring entry into TS 3.3.3.1 (PAM Instrumentation). The candidate that confuses TS 3.3.3.1 with 3.3.6.1 instruments would select this answer.

D. This answer is incorrect due to PC-PS-12A inoperability requiring entry into TS 3.3.6.1. This choice is plausible due to CS-DPIS-43A inoperability requiring entry into TS 3.5.1 (ECCS – Operating) for CS system operability. The candidate that confuses TS 3.5.1 with 3.3.6.1 instruments would select this answer.

Technical References:
Procedure 2.3_9-4-1 (Panel 9-4 - Annunciator 9-4-1), Rev. 54
Procedure 2.3_9-5-2 (Panel 9-5 - Annunciator 9-5-2), Rev. 44
Procedure 2.3_9-3-2 (Panel 9-3 - Annunciator 9-3-2), Rev. 33
Procedure 2.3_J-1 (Panel J - Annunciator J-1), Rev. 7
Technical Specifications.

References to be provided to applicants during exam: NONE

Learning Objective:
COR0021502001060B Given a specific NBI malfunction, determine effect on any of the following: PCIS

Question Source:	Bank # 2009 Quad City NRC Q# 41	
(note changes; attach parent)	Modified Bank #	
	New	
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b) 7	
Level of Difficulty:	3	
SRO Only Justification:	N/A	

41	ID: QDC.ILT.15470	Points: 1.00

Given the following:

- Unit 1 is in Mode 1.

Which of the following instruments, if determined to be inoperable, will affect instrumentation controlled by TS LCO 3.3.6.1, Primary Containment Isolation Instrumentation?

A. LS 1-2351A, Torus Water Level - High

B. PS 1-1001-88A, Drywell Pressure - High

C. TR 1-5741-130 Point 28, Drywell Temperature

D. RE 1-1705-2A, Steam Line Radiation Monitor

Answer: B

Answer Explanation:

Answer: PS 1-1001-88A is used by the PCIS system to initiate various functions on High Drywell pressure.

Distractor 1 is incorrect: LS 1-2351A is used by HPCI and RCIC suction logic transfer to the CCSTs and is controlled by TS LCO 3.3.5.1 (ECCS Instrumentation).
Distractor 2 is incorrect: TR 1-5741-130 Point 28 inputs to annunciator 912-7 A-6 and is controlled by TS LCO 3.3.3.1 (PAM Instrumentation).
Distractor 3 is incorrect: RE 1-1705-2A, Steam Line Radiation Monitor, is used to trip the Mechanical Vacuum Pump and is controlled by TS LCO 3.3.7.2 (Mechanical Vacuum Pump Trip Instrumentation).

Reference: QCOS 1600-06 Rev 19, Table 3.3.6.1-1 Amendment No. 224/219
Reference provided during examination: N/A

Cognitive level: Memory

Level (RO/SRO): RO
Tier: 2 Group: 1

Question Source: Dresden ILT Bank (NRC-12743)
Question History: Dresden 2006 NRC exam

10 CFR Part 55 Content: 41.7

Comments: None.

Table 3.3.6.1-1 (page 1 of 3)
Primary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Main Steam Line Isolation					
a. Reactor Vessel Water Level - Low Low Low (Level 1)	1,2,3	2	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≥ -113 inches
b. Main Steam Line Pressure - Low	1	2	E	SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.6	≥ 835 psig
c. Main Steam Line Flow - High	1,2,3	2 per MSL	D	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 142.7% rated steam flow
d. Condenser Vacuum - Low	1, 2[(a)], 3[(a)]	2	D	SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.6	≥ 8 inches Hg vacuum
e. Main Steam Tunnel Temperature - High	1,2,3	2 per location	D	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 195°F
2. Primary Containment Isolation					
a. Reactor Vessel Water Level - Low (Level 3)	1,2,3	2	G	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≥ 3 inches
b. Drywell Pressure - High	1,2,3	2	G	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 1.84 psig
c. Reactor Building Ventilation Exhaust Plenum Radiation- High	1,2,3	2	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 49 mR/hr
d. Main Steam Line Radiation - High	1,2,3	2	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 3 times full power background
e. Reactor Vessel Water Level -Low Low Low (Level 1)	1,2,3	2	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≥ -113 inches

Table 3.3.5.1-1 (page 3 of 6)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. LPCI System (continued)					
g. Low Pressure Coolant Injection Pump Discharge Flow - Low (Bypass)	1,2,3, 4[(a)], 5[(a)]	1 per subsystem	E	SR 3.3.5.1.2 SR 3.3.5.1.4[(c)(d)] SR 3.3.5.1.5	≥ 2107 gpm
3. High Pressure Coolant Injection (HPCI) System					
a. Reactor Vessel Water Level - Low Low (Level 2)	1, 2[(f)], 3[(f)]	4	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.4[(c)(d)] SR 3.3.5.1.5	≥ -42 inches
b. Drywell Pressure - High	1, 2[(f)], 3[(f)]	4	B	SR 3.3.5.1.2 SR 3.3.5.1.4[(c)(d)] SR 3.3.5.1.5	≤ 1.84 psig
c. Reactor Vessel Water Level - High (Level 8)	1, 2[(f)], 3[(f)]	2	C	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.5	≤ 54 inches
d. Emergency Condensate Storage Tank (ECST) Level - Low	1, 2[(f)], 3[(f)]	2	D	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5	≥ 23 inches
e. Suppression Pool Water Level - High	1, 2[(f)], 3[(f)]	2	D	SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.5	≤ 4 inches

(continued)

(a) When the associated ECCS subsystem(s) are required to be OPERABLE per LCO 3.5.2, ECCS - Shutdown.

(c) 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.

(d) 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 Limiting Trip Setpoint and the methodologies used to determine the as-found and the as-left tolerances are specified in the Technical Requirements Manual.

(f) With reactor steam dome pressure > 150 psig.

DRYWELL HIGH PRESSURE TRIP	PANEL/WINDOW: 9-5-2/D-3

SETPOINT	CIC
1. (2698) DRYWELL HIGH PRESSURE CHAN A1 TRIP at 1.61 psig (Tech Spec ≤ 1.84 psig)	1. PC-PS-12A
2. (2699) DRYWELL HIGH PRESSURE CHAN A2 TRIP at 1.61 psig (Tech Spec ≤ 1.84 psig)	2. PC-PS-12C
3. (2696) DRYWELL HIGH PRESSURE CHAN B1 TRIP at 1.61 psig (Tech Spec ≤ 1.84 psig)	3. PC-PS-12B
4. (2697) DRYWELL HIGH PRESSURE CHAN B2 TRIP at 1.61 psig (Tech Spec ≤ 1.84 psig)	4. PC-PS-12D

PROBABLE CAUSES

- Loss of coolant accident.
- Drywell fan coil unit failure.
- Loss of REC to Drywell.
- Drywell nitrogen makeup flow controller failure.

REFERENCES

- Technical Specifications LCO 3.3.1.1, Reactor Protection System (RPS) Instrumentation.
- Technical Specifications LCO 3.3.6.1, Primary Containment Isolation Instrumentation.
- Technical Specifications LCO 3.3.6.2, Secondary Containment Isolation Instrumentation.
- Technical Specifications LCO 3.3.7.1, CREFS Instrumentation.
- General Operating Procedure 2.1.5, Reactor Scram.

SUPPR POOL DIV I WATER HIGH TEMP	PANEL/WINDOW: J-1/A-1

SETPOINT	CIC	J-1/A-1
(4850) 92°F	PC-TR-24	

PROBABLE CAUSES

- Main steam relief valve leakage or operation.
- HPCI or RCIC operation.
- High ambient temperature.
- Energizing PC-TR-24.

REFERENCES

- LCO 3.6.2.1, Suppression Pool Average Temperature.
- System Operating Procedure 2.2.69.3, RHR Suppression Pool Cooling and Containment Spray.

HPCI SUCTION TRANSFER	PANEL/WINDOW: 9-3-2/A-4

SETPOINT

1. (1634) HPCI SUCT TRANSFER SUPPR POOL HIGH LEVEL at 3.0" H_2O above normal (Tech Spec ≤ 4" above normal)
2. (1650) HPCI/RCIC SUCT TRANSFER ECST A LOW LEVEL at 24" from bottom of ECST A or 2 1/2" indicated (Tech Spec ≥ 23")
3. (1651) HPCI/RCIC SUCT TRANSFER ECST B LOW LEVEL at 24" from bottom of ECST B or 2 1/2" indicated (Tech Spec ≥ 23")

CIC 9-3-2/A-4

1. HPCI-REL-K19 operation caused by HPCI-LS-91A or HPCI-LS-91B
2. RCIC-REL-K39X operation caused by HPCI-LS-74A or HPCI-LS-75A
3. HPCI-REL-K18 operation caused by HPCI-LS-74B or HPCI-LS-75B

PROBABLE CAUSES

- High level in torus caused by SRV operation or systems valve leakage.
- Extended operation of HPCI and/or RCIC in the RPV injection mode.

REFERENCES

- Technical Specifications Table 3.3.5.1-1, Functions 3d and 3e.
- System Operating Procedure 2.2.7, Condensate Storage and Transfer System.

RCIC SUCTION TRANSFER	PANEL/WINDOW: 9-4-1/F-2

SETPOINT

1. (1650) HPCI/RCIC SUCT TRANSFER ECST A LOW LEVEL, 24" from bottom of ECST A (Tech Spec ≥ 23")
2. (1651) HPCI/RCIC SUCT TRANSFER ECST B LOW LEVEL, 24" from bottom of ECST B (Tech Spec ≥ 23")

CIC 9-4-1/F-2

1. RCIC-REL-K39X operation caused by HPCI-LS-74A or HPCI-LS-75A
2. HPCI-REL-K18 operation caused by HPCI-LS-74B or HPCI-LS-75B

PROBABLE CAUSES

- Extended operation of HPCI and/or RCIC in Vessel Injection Mode.

REFERENCES

- Technical Specification LCO 3.3.5.1, Emergency Core Cooling (ECCS) System Instrumentation.
- Technical Specification LCO 3.3.5.2, Reactor Core Isolation Cooling (RCIC) System Instrumentation.
- System Operating Procedure 2.2.7, Condensate Storage and Transfer System.
- System Operating Procedure 2.2.67.1, Reactor Core Isolation Cooling System Operations.

Examination Outline Cross-Reference	Level	RO
Changed to "complete loss of pneumatic supply to the accumulators" to eliminate misinterpretation.	Tier#	2
	Group#	1
	K/A #	239002 K4.09
	Rating	3.7

239002 SRVs

K4. Knowledge of RELIEF/SAFETY VALVES design feature(s) and/or interlocks which provide for the following: (CFR: 41.7)

K4.09 Manual opening of the SRV

Question 42

Which one of the following completes the statement below regarding the design feature associated with the SRVs which allows for opening actuations following a complete loss of pneumatic supply to the accumulators?

ADS valve accumulators are designed to allow for a minimum of ____(1)____actuations at atmospheric Drywell pressure.

LLS valve accumulators are designed to allow for a minimum of ____(2)____actuations.

A. (1) 5
   (2) 5

B. (1) 5
   (2) 14

C. (1) 14
   (2) 5

D. (1) 14
   (2) 14

Answer:

B. (1) 5
   (2) 14

Explanation:

The relief valves which are a part of the ADS are equipped with accumulators which, in the unlikely event the nitrogen pressure is lost, will provide for two valve actuations with the drywell at 70% of its design pressure. Six ADS accumulators are tested to ensure that they will provide sufficient motive force to actuate the main steam relief valves at least five times at atmospheric drywell pressure after being isolated from the nitrogen supply for one hour. The five actuations at atmospheric drywell pressure are equivalent to the two actuations at 70% drywell design pressure [103] [104]. The main steam relief valves associated with low-low set logic operation have two additional

Distracters:

A. This answer is incorrect due to the LLS accumulators designed to provide a minimum of 14 valve actuations. This choice is plausible due to being easily confused with ADS accumulator design. The candidate that confuses ADS & LLS accumulator designs would select this answer.

C. This answer is incorrect due to the ADS accumulators designed to provide a minimum of 5 and LLS 14 valve actuations. This choice is plausible due to ADS & LLS accumulator design being easily confused. The candidate that confuses ADS & LLS accumulator designs would select this answer.

D. This answer is incorrect due to the ADS accumulators designed to provide a minimum of 5 valve actuations. This choice is plausible due to ADS & LLS accumulator design being easily confused. The candidate that confuses ADS & LLS accumulator designs would select this answer.

Technical References:

USAR, Volume II, Chapter IV, Section 4.0

References to be provided to applicants during exam: NONE

Learning Objective:

COR0021602 5. Describe the Nuclear Pressure Relief system design features and/or interlocks that provide for the following:

j. Safety/Relief operating signals
k. Backup to N2 for ADS valves

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b) 7	
Level of Difficulty:	3	
SRO Only Justification:	N/A	

USAR

The adequacy of the plant design for mitigating the consequences of the most severe overpressurization event must be reevaluated with each core reload. Approved methodologies described in References 92 and 97 are used in performing the reload analysis. The analysis is performed at 102% power and 100% core flow condition to bound the fuel reload cycle operating conditions and to account for power level uncertainties specified in Regulatory Guide 1.49. Results are reported in the Supplemental Reload Licensing Report. Recent reload licensing analysis results for Cooper Nuclear Station are reported in Reference 98. Reference 98 also reported the results of an additional analysis of the MSIV closure with flux scram event to extend the applicability of Reference 96 to the current cycle. The MSIV closure with a flux scram event was analyzed for Cycle 20 with the ICF condition. The analysis is documented in Reference 106.

The impact of various SRVs out-of-service (OOS) such that they can not open is discussed in Section IV-4.9.3 for the High Neutron Flux Scram.

The automatic depressurization capability of the pressure relief system is evaluated in USAR Chapter VI, "ECCS," and in USAR Section VII-4, "ECCS Controls and Instrumentation."

Motive force for relief valve operation other than pressure relief is normally provided by nitrogen. Backup air from the instrument control air system can be supplied. The relief valves which are a part of the ADS are equipped with accumulators which, in the unlikely event the nitrogen pressure is lost, will provide for two valve actuations with the drywell at 70% of its design pressure. Six ADS accumulators are tested to ensure that they will provide sufficient motive force to actuate the main steam relief valves at least five times at atmospheric drywell pressure after being isolated from the nitrogen supply for one hour. The five actuations at atmospheric drywell pressure are equivalent to the two actuations at 70% drywell design pressure.[103] [104]. The main steam relief valves associated with low-low set logic operation have two additional and larger accumulators sized to permit fourteen valve operations.[59] A minimum of 70 psig is required to assure 14 actuations under the test conditions for the two larger accumulators. The pressure in the accumulators is continuously monitored for low pressure by means of a pressure switch located in the system downstream of the accumulators and annunciated in the control room. After a Safe Shutdown Earthquake, the ADS accumulators allow operation of the ADS valves that are credited with controlled depressurization for approximately 40 hours. For non-seismic events, the outdoor liquid nitrogen tank can be refilled, as needed, to assure ADS availability long term (100 days).

The remote pneumatic actuators are DC powered solenoid valves. These are normally closed, fail closed valves, and a power or valve malfunction will prevent the relief valve from operating for ADS. Abnormal solenoid valve operation would be detected during the operational tests of the relief valve. A complete rupture of the solenoid valve would result in a low pressure-accumulator alarm. Since ADS criteria is met with one relief valve inoperative, a double failure would be required before the ADS criteria is violated.[28]

The relief valve discharge piping was designed, installed, and tested as outlined in USAR Appendix A, and modified for increased structural safety margins during the CNS Mark I Containment Short Term Program (STP) as discussed in USAR Section V-4.1. The NRC has evaluated the impact of safety/relief valve (SRV) discharge piping on reactor safety and has concluded that the loss of SRV discharge piping may increase transient and accident consequences and probabilities. The Standard Review Plan, NUREG 75-087, also indicates that the SRV discharge piping is safety related.[57] The requirements of the piping design identified in USAR Section V-4.1.4 and USAR Appendix A, the load criteria discussed in USAR Section IV-4.5 and the NRC's evaluation are sufficient to satisfy safety design bases 3 and 4.

Examination Outline Cross-Reference	Level	RO
Comment incorporated.	Tier#	2
	Group#	1
	K/A #	259002 K5.03
	Rating	3.1

259002 Reactor Water Level Control

K5. Knowledge of the operational implications of the following concepts as they apply to REACTOR WATER LEVEL CONTROL SYSTEM : (CFR: 41.5 / 45.3)

K5.03 Water level measurement

Question 43

The plant is operating at rated power.

What is the impact on the RVLCS if ALL level signals become invalid?

A. RFPT Control shifts to MDEM ONLY.

B. RFPT Control shifts to MDVP ONLY.

C. RFPT Control shifts to MDEM and RVLCS transfers to Single Element.

D. RFPT Control shifts to MDVP and RVLCS transfers to Single Element.

Answer:

C. RFPT Control shifts to MDEM and RVLCS transfers to Single Element.

Explanation:

A loss of all valid level inputs causes the RFPT Control to shift to MDEM and RVLCS to transfer to single element.

3-Element permissives include:

1. Total Steam Flow Good (2 or more valid steam flow transmitters)
2. At least 1 RFW Flow Measurement for each RFP is valid when both RFPTs are in Auto
3. Either Feed Pump in Auto
4. The RVLCS is in Auto
5. Total Main Steam Flow is greater than 10% of rated flow (~1.0 x 10^6 lbm/hr)
6. Reactor Vessel Level Good (1 or more valid transmitters)
7. Three Element Selector Switch RFC-SW_S2 is in AUTO

Level Input Response:

1. Upon loss of validation of one level signal, the system will utilize the average of the three remaining level signals.
2. Upon loss of validation of two level signals, the system will utilize the remaining two level signals.
3. Upon loss of validation of three level signals, the system will generate an alarm and use the remaining good level as the signal for control.
4. Upon loss of all valid level signals (failed, invalid or otherwise), the system will generate an alarm and revert to master manual control.
5. In the event that only two level instruments are available and they begin to diverge such that average level deviates > 4 inches from median level, the median level will become the controlling level signal. This signal will be provided by the higher of the 2 available level instruments as the logic provides a 1000 inch high level into the calculation to ensure the higher of the 2 remaining level instruments is used for the controlling RPV level signal. This condition will also result in these 2 remaining level instruments to indicate there is > 8 inches difference between these 2 instruments. A deviation alarm will be generated to Ronan and to the HMI to alert the operator of this condition. The operator will then have to determine which instrument is good, and then bypass the failing instrument in accordance with 2.4RXLVL.

Distracters:

A. This answer is incorrect due to RVLCS transferring to single element on loss of valid level inputs. This choice is plausible due to 3-element permissives being commonly confused. The candidate that confuses 3-element control permissives would select this answer.

B. This answer is incorrect due to RFPT Control shifting to MDEM and RVLCS transferring to single element on loss of valid level inputs. This choice is plausible due to RVLCS shifting to MVDP upon loss of all valid speed inputs and 3-element permissives being commonly confused. The candidate that confuses speed vs. level inputs and 3-element control permissives would select this answer.

D. This answer is incorrect due to RFPT Control shifting to MDEM. This choice is plausible due to RVLCS shifting to MVDP upon loss of all valid speed inputs. The candidate that confuses speed vs. level inputs and correctly identifies transfer to single element would select this answer.

Technical References:

Procedure 4.4.1 (Reactor Vessel Level Control System), Rev. 7
Procedure 2.2.28 (Feedwater System Startup And Shutdown), Rev. 104

References to be provided to applicants during exam: NONE

Learning Objective:

COR002150202. Describe the interrelationship between RVLC and the following:
g. Nuclear Boiler Instrumentation

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X

Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b) 5	
Level of Difficulty:	2	
SRO Only Justification:	N/A	

2.2 The following conditions will automatically shift from 3 element to 1 element:

2.2.1 MASTER LEVEL controller taken to MAN.

2.2.2 Wide FW flow transmitter is invalid when associated RFP is in AUTO.

2.2.3 Three steam flow elements are invalid.

2.2.4 Turbine 1st stage flow invalid with less than four steam flow elements.

2.2.5 Both individual RFP controls not in AUTO.

2.2.6 Total steam flow < 1 Mlbm/hr.

2.2.7 All Reactor vessel water level elements are invalid.

PROCEDURE 4.4.1 REVISION 7 PAGE 65 OF 68

ATTACHMENT 3 INFORMATION SHEET

2.2.8 Post-scram when 3 element controller output drops to ≤ 2%.

2.3 On loss of all Reactor pressure inputs, RVLC automatically uses a 1000 psig substitute value.

2.4 On loss of all Reactor level inputs, RVLC System fails operating RFP controllers in AUTO to MDEM.

1.2.6.2 Manual Demand (MDEM) - In MDEM RFPT speed is controlled by the Operator through a RVLC/RFPT HMI. The Operator inputs desired RFPT speed. The RFPTC System adjusts governor valve position to obtain the desired RFPT speed. The RFPTC uses three (3) speed probes to determine RFPT speed. If three (3) speed probes are good, the median value is selected for controlling RFPT speed. If two speed pickups are good, the selected speed output is equal to the highest. If one speed pickup is good, it is used. If all speed probes are invalid, the RFPTC transfers to MDVP. If the speed between the 3 probes is > 10 rpm, an alarm will inform the Operator of the deviation. Transfer from MDEM to AUTO is only possible when RFPT status is on-line, RFP speed is ≥ 2800 rpm, and within 600 rpm of the AUTO signal. A green AUTO PERMISSIVE light on the RFPT controller informs the Operator that conditions are met. When the RFPTC is transferred to AUTO, the transfer is bumpless and the difference between the master level controller signal vs. actual RFPT speed is captured as a bias signal.

1.2.6.3 Manual Direct Valve Positioning (MDVP) - In Manual Direct Valve Positioning Mode (MDVP), the Operator provides an output signal at any RVLC/RFPT HMI. This signal is processed by the Tricon and sent onto I/P assemblies. The RFPT nozzle block valve will be directly positioned by the RFPT Control System without any speed feedback feature present. MDVP Mode is not generally used, but would be used for testing/calibration purposes or if all speed probes fail. MDVP with output at 0 is the mode automatically selected when the RFPT is tripped. Once RFPT is started, the mode shifts automatically to MDEM. The Operator can select MDVP when the RFPT is in any status. Transferring from any other control mode (i.e., AUTO) to MDVP is a bumpless transfer.

1.2.6.4 Reactor Pressure Follow - Speed control may be placed into Reactor Pressure Follow Control when RFPT is on-line. When this HMI selection has been made, the speed setpoint will be derived from a Reactor Vessel Pressure vs. RFPT speed schedule. When a different control mode is selected from the HMI, the RFPT will de-select Reactor Pressure Follow and enter the selected mode. If the permissives for Reactor Pressure Follow are lost, then the controller will shift to MDEM at current speed. This control will be used to bring the first RFPT to the point where it can be switched to Auto Speed Control with the speed setpoint by the master level controller during Feedwater Sequence Mode 2. After a scram has been detected, the RFP remaining in AUTO post-scram will go to the Reactor Pressure Follow Mode once the RFP discharge valves are closed and startup valves are controlling level.

1.2.18 REACTOR WATER LEVEL INPUT

1.2.18.1 Reactor vessel level instruments will be validated by first checking if the analog input signal is within the expected range 4 to 20 mA. The 4 to 20 mA is converted to machine counts, which corresponds to 819 to 4095 counts in the TSAP. The analog inputs will first be checked for a signal failed high or low by verifying the machine counts are between 778 and 4300 counts (~ -5% and +105% of the transmitter operating range 4 to 20 mA). If the input count falls outside the expected input range, the Tricon processor will consider that instrument as failed. The failed instrument will be removed from all process calculations and a main Control Room alarm is activated. If the instrument returns to the normal range, it will be validated and then may be restored by resetting the validation logic from the HMI.

1.2.18.2 Reactor vessel level signal validation is further performed by comparing each individual instrument to the selected reactor vessel level. When any instrument differs by 8" from the selected reactor vessel level, the suspect instrument will be invalid and will not be used in the processing calculations.

1.2.18.3 After development of the average level signal by using all valid level instruments, this average will be compared to the median of the valid level signals. If the average differs by more than 4" from the median value, the control signal to the MASTER LEVEL Controller will switch to the median value as selected water level and each individual level signal will be validated against the median value instead of the average value. When there is an even number of valid transmitters a dummy variable (1000") is also compared with the valid signals. This results in the higher of the two median signals will be selected. If the average value returns to a valid state (within ± 2" of the median), the controlling signal, sent to the MASTER LEVEL Controller, will switch back to the average level signal.

1.2.18.4 When there are only two valid level signals, the system will generate an alarm for a deviation of more than ± 8" between the two valid signals. Automatic invalidation of one of these signals due to a level deviation between the remaining two signals will not be performed by the RVLCS (i.e., Operator intervention is required to determine the valid signal and place the invalid signal in BYPASS). However, automatic invalidation of an individual signal on poor quality is still provided, this would result in the poor quality level signal being made invalid and the RVLC System would then control based on the remaining one level signal.

1.2.18.5 Upon a loss of all level signals, the master controller shifts to manual.

1.2.18.6 Only LT-59D is density compensated. Narrow range instruments are not density compensated due to Level 8 trips remaining uncompensated. During plants conditions, such as stratification, use of compensated level indications would maintain actual water level high and drive the uncompensated signal into a Level 8 trip. This is due to during plant cooldown, the density of water rises. If actual water level was to remain steady, the d/p sensed by level instruments would lower during plant cooldown. To uncompensated instruments, the lower d/p would indicate a rise in water level even though actual water level is constant.

1.2.18.7 The LT59D signal is enhanced with a predictive bias that will use recirc flow to determine the amount of bias to be applied to the LT59D level measurement to reduce the hydraulic effects noted at different power levels.

1.2.19 REACTOR PRESSURE

1.2.19.1 The RVLC System utilizes the existing three wide range pressure loops (NBI-PT-53A, NBI-PT-53B, and NBI-PT-53C) and the existing narrow range pressure loop (NBI-PT-58). The RVLC System generates a selected Reactor Pressure signal for density compensation of LT59D. Narrow range pressure is used to density compensate the main steam flow signal.

1.2.19.2 Each pressure input is validated in a similar method as Level inputs to verify instrument is in the 4 to 20 mA range. Each pressure instrument input is then further validated against the selected reactor pressure signal. Any wide range vessel pressure signal > ± 75 psig out of range of the selected reactor vessel pressure signals will be considered invalid. The narrow range vessel pressure signal will be considered invalid if it is > ± 50 psig out of range of the selected reactor vessel pressure signal when reactor pressure is between 925 and 1050 psig (as indicated by the selected wide range signal). Upon loss of the vessel pressure signal, the system will automatically default to the specific gravities at 1000 psig to calculate level differential pressure measurement density compensation.

1.2.19.3 Average pressure will be compared against the median. If the average differs by more than ± 75 psig from the median value, the signal used for pressure compensation will switch to the median value as the selected.

1.2.19.4 The narrow range pressure signal will be bypassed and the associated average pressure/narrow range pressure comparator alarm will be disabled when reactor pressure is < 925 psig, as indicated by the average pressure.

Examination Outline Cross-Reference	Level	RO
The absence of an abnormal switch alignment in the stem provides system in normal 100% alignment – no change.	Tier#	2
	Group#	1
	K/A #	261000 K6.04
	Rating	2.9

261000 SGTS

K6. Knowledge of the effect that a loss or malfunction of the following will have on the STANDBY GAS TREATMENT SYSTEM : (CFR: 41.7 / 45.7)

K6.04 Process radiation monitoring

Question 44

The plant is operating at rated power.

What is the impact on the Standby Gas Treatment Trains (SGTs) if ALL Reactor Building Exhaust Plenum Radiation Monitors fail downscale?

A. ONLY SGT A starts.

B. ONLY SGT B starts.

C. BOTH SGTs start.

D. Neither SGT starts.

Answer:

C. BOTH SGTs start.

Explanation:

SGT receive start signals from Group 6 isolation logic. The RB Exh Plenum trip channels are aligned such that an upscale of Div 1 logic (A or C) combined with an upscale of Div 2 (B or D) logic or ALL channels downscale will cause both SGTs to start while in the AUTO mode. With ALL channels failing downscale, both SGTs start.

Distracters:

A. This answer is incorrect due to both SGTs starting. This choice is plausible if the initial condition of the stem provided SGT B STANDBY Mode. The candidate that confuses normal rated power SGT alignment and recognizes downscale failure being SGT auto start would select this answer.

B. This answer is incorrect due to both SGTs starting. This choice is plausible if the initial condition of the stem provided SGT A STANDBY Mode. The candidate that confuses normal rated power SGT alignment and recognizes downscale failure being SGT auto start would select this answer.

D. This answer is incorrect due to both SGTs starting. This choice is plausible if the downscale failures are not recognized as an auto start. The candidate that does not recognize the downscale failures as being SGT auto start would select this answer.

Technical References:
Procedure 2.2.73 (Standby Gas Treatment System), Rev. 52

References to be provided to applicants during exam: NONE

Learning Objective:
COR0011802001120D Given plant conditions related to the Radiation Monitoring system, determine if any of the following should occur: Standby Gas Treatment Startup
COR0022802001080A Describe the Standby Gas Treatment design features and/or interlocks that provide for the following: Automatic system initiation
COR0022802001130A Given plant conditions, determine if any of the following should occur: SGT automatic initiation

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b) 7	
Level of Difficulty:	2	
SRO Only Justification:	N/A	

1.3 INTERLOCKS AND SETPOINTS

1.3.1 A Group 6 isolation is caused by following:

1.3.1.1 High drywell pressure of ≤ 1.84 psig.

1.3.1.2 Low reactor water level of ≥ -42".

1.3.1.3 Reactor Building Exhaust Plenum Trip Channel A or C <u>and</u> Channel B or D high-high at ≤ 49 mrem/hr (actual setpoint 10 mr/hr).

1.3.1.4 Reactor Building Exhaust Plenum Trip Channel A or C <u>and</u> Channel B or D MODE switch not in OPERATE.

PROCEDURE 2.2.73	REVISION 52	PAGE 21 OF 25

ATTACHMENT 2	INFORMATION SHEET

1.3.1.5 All four Reactor Building exhaust plenum trip channels downscale at 0.1 mr/hr.

1.3.2 The following conditions cause SGT-AO-249 (SGT-AO-250), SGT A (B) INLET, to open:

1.3.2.1 EF-R-1E (EF-R-1F), SGT A (B) EXHAUST FAN, starts.

1.3.2.2 Loss of air or power.

Interlocks and Trips

- Reactor Building vent exhaust plenum radiation monitor:
  - [(A↑) or (C↑) or (A↓ and C↓)] and [(B↑) or (D↑) or (B↓ and D↓)];
  - Inop Trip produces both upscale and downscale trip in affected monitor.

Examination Outline Cross-Reference	Level	RO
Comments incorporated.	Tier#	2
	Group#	1
	K/A #	262001 K6.02
	Rating	3.6

262001 AC Electrical Distribution

K6. Knowledge of the effect that a loss or malfunction of the following will have on the A.C. ELECTRICAL DISTRIBUTION: (CFR: 41.7 / 45.7)

K6.02 Off site power

Question 45

The plant is operating at rated power.

ALL off-site power voltages simultaneously lower causing ALL 4160 VAC bus voltages to stabilize at 3800 VAC.

Which one of the following completes the statements below regarding the status of Diesel Generators (DGs) and 480 VAC Load Shedding if Grid voltage remains at this value for 2 minutes?

____(1)____ DGs will be running.

480 VAC Load-Shedding ____(2)____ occur.

A. (1) Both
(2) will

B. (1) Both
(2) will NOT

C. (1) Neither
(2) will

D. (1) Neither
(2) will NOT

Answer:
A. (1) Both

(2) will

Explanation: With the NSST supplying 1A/1F & 1B/1G and bus voltage lowering to 3800 volts, breakers 1FA and 1GB will trip 12.5 seconds after voltage has lowered below 3880 VAC. The "loss of voltage" signal will start the Diesel generators and apply a close permissive to 1FS & 1GS. As the Emergency Transformer voltage is also degraded, 1FS/1GS will not automatically close and the EDGs will close onto the bus when they have reached rated voltage and speed (at least 10 seconds after the bus was de-energized). As the "loss of voltage" signal exists for > 5.5 seconds, 480 volt load-shedding will be initiated and 12 breakers will receive a trip signal.

Distracters:

B. This answer is incorrect due to 480 VAC load shed occurs. This choice is plausible if the time from reaching the first level undervoltage were reduced to less than 5.5 seconds. The candidate that correctly identifies the degraded voltage starting both DGs and confuses first & second level undervoltage logic would select this answer.

C. This answer is incorrect due to both DGs automatically starting. This choice is plausible if the undervoltage setpoints and time delays are not known or confused. The candidate that confuses degraded voltage DG starts and remember 480 VAC load shed logic would select this answer.

D. This answer is incorrect due to both DGs automatically starting and 480 VAC load shed occurring. This choice is plausible if the undervoltage setpoints and time delays are not known or confused and the time from reaching the first level undervoltage were reduced to less than 5.5 seconds. The candidate that confuses degraded voltage DG starts and first & second level undervoltage logic would select this answer.

Technical References:
Procedure 5.3GRID (Degraded Grid Voltage), Rev. 42

References to be provided to applicants during exam: NONE

Learning Objective:
COR0010102001080B Predict the consequences of the following on plant operation: 4160V Critical bus undervoltage

Question Source:	Bank # 2260	
(note changes; attach parent)	Modified Bank #	
	New	
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b) 7	
Level of Difficulty:	3	

SRO Only Justification:	N/A

QUESTION: 33, 2260 (1 point(s))

The plant is operating at rated power when 345 KV, 161 KV and 69 KV voltages simultaneously lower such that the Normal Transformer secondary voltage drops to 3700 VAC. All three (3) voltages lower at the same rate.

How will the Diesel Generators and 480 VAC load-shedding be affected by this voltage reduction over the next 1 (one) minute?

a. Both Diesel Generators will start. 480 VAC load-shedding **WILL** occur.

b. Both Diesel Generators will start. 480 VAC load-shedding will **NOT** occur.

c. Neither Diesel Generator will start. 480 VAC load-shedding **WILL** occur.

d. Neither Diesel Generator will start. 480 VAC load-shedding will **NOT** occur.

ANSWER: 33, 2260

a. Both Diesel Generators will start. 480 VAC load-shedding **WILL** occur.

With the NSST supplying 1A/1F & 1B/1G and bus voltage lowering to 3700 volts, breakers 1FA and 1GB will trip 12.5 seconds after voltage has lowered below 3880 VAC. The "loss of voltage" signal will start the Diesel generators and apply a close permissive to 1FS & 1GS. As the Emergency Transformer voltage is also degraded, 1FS/1GS will not automatically close and the EDGs will close onto the bus when they have reached rated voltage and speed (at least 10 seconds after the bus was de-energized). As the "loss of voltage" signal exists for > 5.5 seconds, 480 volt load-shedding will be initiated and 12 breakers will receive a trip signal.

1.18 Isolating non-critical 4160V buses at 3600V is based on 4 kV motors that are rated at +/- 10%. The non-critical buses have overcurrent bus breaker protection, but do not have automatic undervoltage protection. Below 3600V equipment damage is probable; therefore, it's prudent to manually isolate the buses if the voltage degrades to this value. Risk Assessment has reviewed this configuration and concurs with the action.

1.19 Actions in this procedure to place the voltage regulator to OFF are mandated under conditions indicative of a CNS voltage regulator oscillation causing 345 kV voltage variations. If the grid voltage is oscillating and voltage regulator is working as designed, as grid voltage lowers regulator will raise field amps in an attempt to maintain terminal voltage. If regulator is causing the oscillations, field amps rising will cause 345 kV volts to rise. So, if 345 kV voltage rises as field amps lower, regulator is working properly and there is no need to transfer to the base adjuster. If 345 kV voltage rises as our field amps rise, then our regulator is driving the voltage oscillations and transferring to the base adjuster should stabilize conditions.

1.20 There are two levels of undervoltage protection at CNS. The first level is a loss of voltage protection which is designed to actuate at conditions indicative of voltage rapidly collapsing to zero volts. The relays which actuate are a time undervoltage relay with inverse time characteristics (i.e., the lower the voltage, the faster the actuation). The timer starts timing at bus voltage < 2870V. The second level of undervoltage protection is for sustained low voltage conditions. This system is designed to respond to a static low voltage condition and will actuate whenever the bus voltage drops below 3880V ± 52V for a time delay of 7.5 ± 0.8 seconds.

1.21 Expected plant response to loss of all off-site power (simultaneous loss of 345 kV, 161 kV, and 69 kV lines):

If all off-site power sources are lost simultaneously, the main generator will trip, causing loss of NSST. If there is no bus lockout, 1AS and 1BS will close on to the SSST when 1AN and 1BN trip open. 1AS and 1BS will close even if there is no voltage on the SSST secondary. 1AF, 1FA, 1BG, and 1GB will trip in < 1 second (with an inverse time constant, "0" volts will result in a very short delay before the breakers trip on first level protection). The DGs will receive an auto start signal at the same time as these breakers receive a trip signal. The voltage permissive for 1FS and 1GS will not be met, so these breakers will not close. When DGs reach rated voltage and speed, and at least 10 seconds after the first level voltage protection signal was sensed, DG output breakers will close and energize 4160 VAC Bus 1F/1G. The only 4160 VAC loads that trip on undervoltage are the RRMG Sets and circ water pumps (setpoints similar to the first level undervoltage protection). The condensate booster pumps will trip on low suction pressure or low oil pressure. All 4160 VAC loads powered from 1F/1G will trip, except the 480V transformer feed. In addition, once the first level protection signal has been sensed for at least 5.5 seconds, load shedding of selected 480 VAC loads will occur. The breakers require manual operator action to restore to service.

CRD Pumps A and B, Air Compressors A and B, MCCs OG1, P, N, and M (480 Switchgear 1F), and MCCs MR, U, W, and V (480 Switchgear 1G) are the breakers that receive a trip from the load shedding relays. If reactor power is > 30%, the reactor will scram immediately when the loss of off-site power occurs.

1.22 Expected plant response to all off-site power degraded with DGs available:

If all off-site power sources degrade simultaneously at a similar rate, the first alarm will be the 1FS/1GS auto closure prohibited annunciator, and these breakers will not automatically close. As voltage lowers below 3880, a 7.5 second timer starts (no alarm). When this times out, C-1/A-7(C-4/A-2), 4160V BUS 1F (1G) LOW VOLTAGE, alarms. If a LPCI initiation signal is present, Breaker 1FA/1GB will trip immediately. If no LPCI initiation, these breakers will trip 5 seconds later. This results in Bus 1F/1G de-energizing and starting the first level undervoltage protection scheme as described in the preceding paragraph. When RPS frequency or voltage drops too low, EPAs trip and the RPS buses will de-energize. When RPS is lost, reactor scrams and group isolations occur. The RRMG Sets will not reach their trip voltage, but will trip due to loss of lube oil after a 6 second pressure switch delay. 4160 VAC BOP loads will not trip, but continue to run on degraded voltage. 1AS and 1BS will close on to the SSST when Breakers 1AN and 1BN trip open. These breakers will close even if there is low voltage on the SSST secondary. Load-shedding of the 480V breakers occur as described in preceding paragraph.

Examination Outline Cross-Reference	Level	RO
	Tier#	2
	Group#	1
	K/A #	262001 A1.05
	Rating	3.2

262001 AC Electrical Distribution

A1. Ability to predict and/or monitor changes in parameters associated with operating the A.C. ELECTRICAL DISTRIBUTION controls including: (CFR: 41.5 / 45.5)

A1.05 Breaker lineups

Question 46

The following conditions exist:

- 4160V bus 1A is energized from the Normal Transformer.
- SSST and ESST are energized.
- SYNCH SWITCH 1AN OR 1AS is in the 1AS position.

The control switch for breaker 1AS is placed in the CLOSE position.

How will the AC distribution system respond to these conditions?

A. Breaker 1AS will NOT CLOSE.

B. Breaker 1AS will CLOSE and breaker 1AN will trip OPEN immediately.

C. Breaker 1AS will CLOSE and breaker 1AN will trip OPEN after a 3 second time delay.

D. Breaker 1AS will CLOSE and Bus 1A will be energized from both the Normal and Startup Transformers.

Answer:
B. Breaker 1AS will CLOSE and breaker 1AN will trip OPEN immediately.
Explanation: The following conditions must be met to CLOSE Breaker 1AS:

- The switch for Breaker 1AS must be placed to CLOSE.
- Bus 1A must be de-energized as indicated by Relay 27X/1A (27X/1B) being dropped out **OR** SYNCH SWITCH 1AN OR 1AS must be in 1AS.
- The 86/1AN and 86/1AS lockout relays are reset.
- A Startup Transformer fault does not exist as indicated by Relay 86X 2/SB being dropped out.
- The anti-pumping circuit is dropped out.

Since all these conditions are met, 1AS breaker will close which causes the 1AN breaker to trip immediately.

Distracters:

A. This answer is incorrect due to Breaker 1AS will CLOSE and breaker 1AN will trip OPEN immediately. This choice is plausible due to having the SYNCH switch in 1AN would not allow the breaker to close. The candidate that confuses the SYNCH switch position would select this answer.

C. This answer is incorrect due to Breaker 1AS will CLOSE and breaker 1AN will trip OPEN immediately. This choice is plausible due to confusing 1AN breaker with the DG Output breaker (EG1) which receives a trip signal for 3 seconds following a diesel auto start signal with the diesel generator paralleled with the critical bus. The candidate that confuses EG1 with 1AS trip logic would select this answer.

D. This answer is incorrect due to Breaker 1AS will CLOSE and breaker 1AN will trip OPEN immediately. This choice is plausible due to critical bus breaker trip logic being easily confused with non-critical bus breaker logic. The candidate that confuses critical bus breaker logic with non-critical bus logic would select this answer.

Technical References:
Procedure 2.2.18 (4160V Auxiliary Power Distribution System), Rev. 181
Procedure 2.2.20 { Standby AC Power System (Diesel Generator)}, Rev. 92

References to be provided to applicants during exam: NONE

Learning Objective:

COR0010102001090E Describe the AC Electrical Distribution System design feature(s) and/or interlock(s) that provide for the following: Paralleling of AC sources(synchroscope)
COR0010102001090B Describe the AC Electrical Distribution System design feature(s) and/or interlock(s) that provide for the following: Circuit breaker automatic trips

Question Source:	Bank # 24313	
(note changes; attach parent)	Modified Bank #	
	New	

Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	

10CFR Part 55 Content:	55.41(b) 5
Level of Difficulty:	3
SRO Only Justification:	N/A

QUESTION: 124 24313 1 point

The following conditions exist:

- ☐ 4160V bus 1A is energized from the Normal Transformer.
- ☐ SSST and ESST are energized.
- ☐ SYNCH SWITCH 1AN OR 1AS is in the **1AS** position.
- ☐ The control switch for breaker **1AS** is placed in the CLOSE position.

How will the AC distribution system respond to these conditions?

a. Breaker 1AS will CLOSE and breaker 1AN will trip OPEN immediately.

b. Breaker 1AS will CLOSE and breaker 1AN will trip OPEN after a 3 second time delay.

c. Breaker 1AS will CLOSE and Bus 1A will be energized from both the Normal and Startup Transformers.

d. Breaker 1AS will NOT CLOSE.

ANSWER:

a. Breaker 1AS will CLOSE and breaker 1AN will trip OPEN immediately.

2.7 BREAKER 1AN (1BN), NORMAL TRANSFORMER FEED TO 4160V BUS 1A (1B)

2.7.1 All of following conditions must be met to manually close Breaker 1AN (1BN) from the Control Room:

2.7.1.1 The switch for Breaker 1AN (1BN) must be placed to CLOSE.

2.7.1.2 Bus 1A (1B) must be de-energized as indicated by Relay 27X/1A (27X/1B) being dropped out or SYNCH SWITCH 1AN OR 1AS (1BN OR 1BS) must be in 1AN (1BN).

2.7.1.3 The 86/1AN (86/1BN) and 86/1AS (86/1BS) lockout relays are reset.

2.7.1.4 A Normal Transformer fault does not exist as indicated by Relay 86X-2/NB being dropped out.

PROCEDURE 2.2.18	REVISION 181	PAGE 220 OF 231

2.7.2 Breaker 1AN (1BN) will trip on any of following conditions:

2.7.2.1 The switch for Breaker 1AN (1BN) is placed to TRIP.

2.7.2.2 The switch for Breaker 1AS (1BS) is in CLOSE and Breaker 1AS (1BS) is closed.

2.7.2.3 The 86/1AN (86/1BN) lockout relay is tripped.

2.7.2.4 86X-1/NB (aux trip ckt) lockout relay trips:

a. Main Transformer lockout.

b. Normal Transformer lockout.

c. Main Generator lockout.

d. 86ASO lockout.

e. Breaker 3310 or 3312 lockout.

2.8 BREAKER 1AS (1BS), STARTUP TRANSFORMER FEED TO 4160V BUS 1A (1B)

2.8.1 All of following conditions must be met for Breaker 1AS (1BS) to automatically close:

2.8.1.1 The switch for Breaker 1AS (1BS) must be in NORMAL AFTER TRIP (green flagged).

2.8.1.2 Breaker 1AN (1BN) is open.

2.8.1.3 The 86/1AN (86/1BN) and 86/1AS (86/1BS) lockout relays are reset.

2.8.1.4 A Startup Transformer fault does not exist as indicated by Relay 86X-2/SB being dropped out.

2.8.1.5 The anti-pump circuit is dropped out.

a. The anti-pumping circuit can be reset if it is picked up by any of the following:

1. Placing the switch for Breaker 1AS (1BS) to TRIP.

2. Unlocking the local TEST CONTROL SWITCH, taking the switch to TEST, then placing it back to CONNECT and locking it in position.

3. Opening the local breaker shutter slide (all the way to the left), then closing the breaker shutter slide (all the way to the right).

2.8.2 All of following conditions must be met to manually close Breaker 1AS (1BS) from the Control Room:

2.8.2.1 The switch for Breaker 1AS (1BS) must be placed to CLOSE.

2.8.2.2 Bus 1A (1B) must be de-energized as indicated by Relay 27X/1A (27X/1B) being dropped out or SYNCH SWITCH 1AN OR 1AS (1BN OR 1BS) must be in 1AS (1BS).

2.8.2.3 The 86/1AN (86/1BN) and 86/1AS (86/1BS) lockout relays are reset.

2.8.2.4 A Startup Transformer fault does not exist as indicated by Relay 86X-2/SB being dropped out.

2.8.2.5 The anti-pumping circuit is dropped out.

2.8.3 Breaker 1AS (1BS) will trip on any of following conditions:

2.8.3.1 The switch for Breaker 1AS (1BS) is placed to TRIP.

2.8.3.2 The switch for Breaker 1AN (1BN) is in CLOSE and Breaker 1AN (1BN) is closed.

2.8.3.3 The 86/1AS (86/1BS) lockout relay trips.

2.8.3.4 86X-1/SB (aux trip ckt) lockout relay trips:

a. Startup Transformer lockout.

b. Breaker 1604 or 1606 lockout.

c. Breakers 1604 and 1606 opened simultaneously.

2.9 BREAKER 1FA (1GB), 4160V BUS 1A (1B) FEED TO 4160V BUS 1F (1G)

2.9.1 All of following conditions must be met to manually close Breaker 1FA (1GB) from the Control Room:

2.9.1.1 The switch for Breaker 1FA (1GB) must be placed to CLOSE.

2.9.1.2 Bus 1F (1G) is de-energized as indicated by Relay 27X5/1F (27X5/1G) being dropped out, Breaker 1AF (1BG) is open, or SYNCH SWITCH 1FA OR 1AF (1BG OR 1GB) is in 1FA (1GB).

2.9.1.3 The 86/1AN (86/1BN), 86/1AS (86/1BS), 86/1FA (86/1GB), 86/1FE (86/1GE), and 86/1FS (86/1GS) lockout relays are reset.

2.7.4 Breaker EG1(EG2) will trip if Isolation Switch IS/EG-1(EG-2) is in NORM and <u>any</u> of following conditions are met:

<u>NOTE</u> – If Isolation Switch IS/EG-1(EG-2) is in ISOL, neither the control switch for Breaker EG1(EG2) on PANEL C nor 86/1FE(1GE) lockout relay will trip the breaker.

2.7.4.1 The control switch for Breaker EG1(EG2) on PANEL C is placed to TRIP.

2.7.4.2 The control switch for Breaker EG1(EG2) on the DIESEL GENERATOR #1(2) METERING Panel is placed to TRIP and the DIESEL GEN 1(2) BKR MODE switch on PANEL C is in LOCAL.

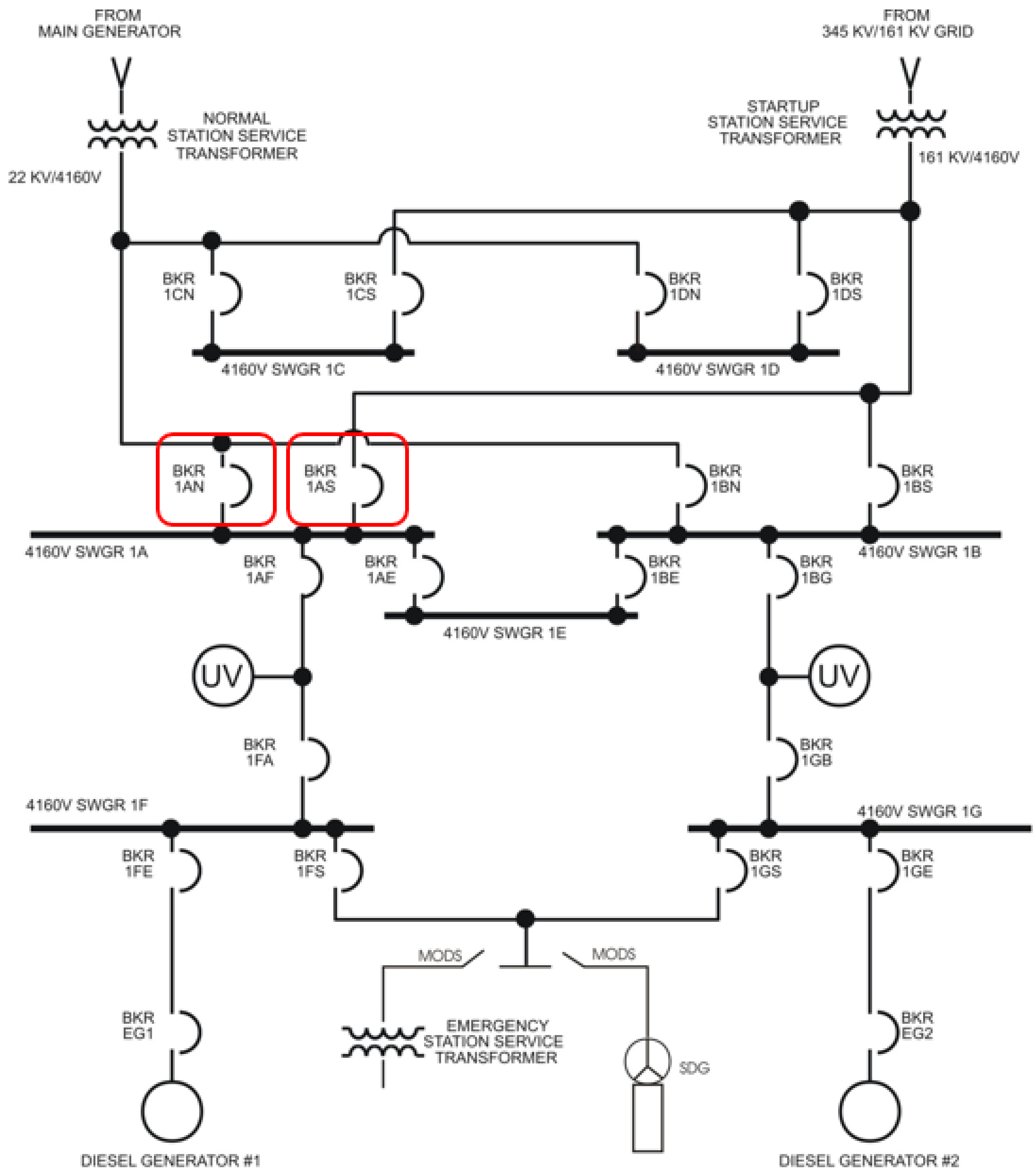
<u>NOTE</u> – If EG1(EG2) are open, they still receive a trip signal from a diesel auto start signal preventing manual closure for 3 seconds.

2.7.4.3 For 3 seconds following a diesel auto start signal with the diesel generator paralleled with the critical bus.

2.7.4.4 The VOLTAGE SHUTDOWN button on the DIESEL GENERATOR #1(2) METERING Panel is pressed.

2.7.4.5 The 86/DG1(DG2) <u>or</u> 86/1FE(1GE) lockout relay trips.

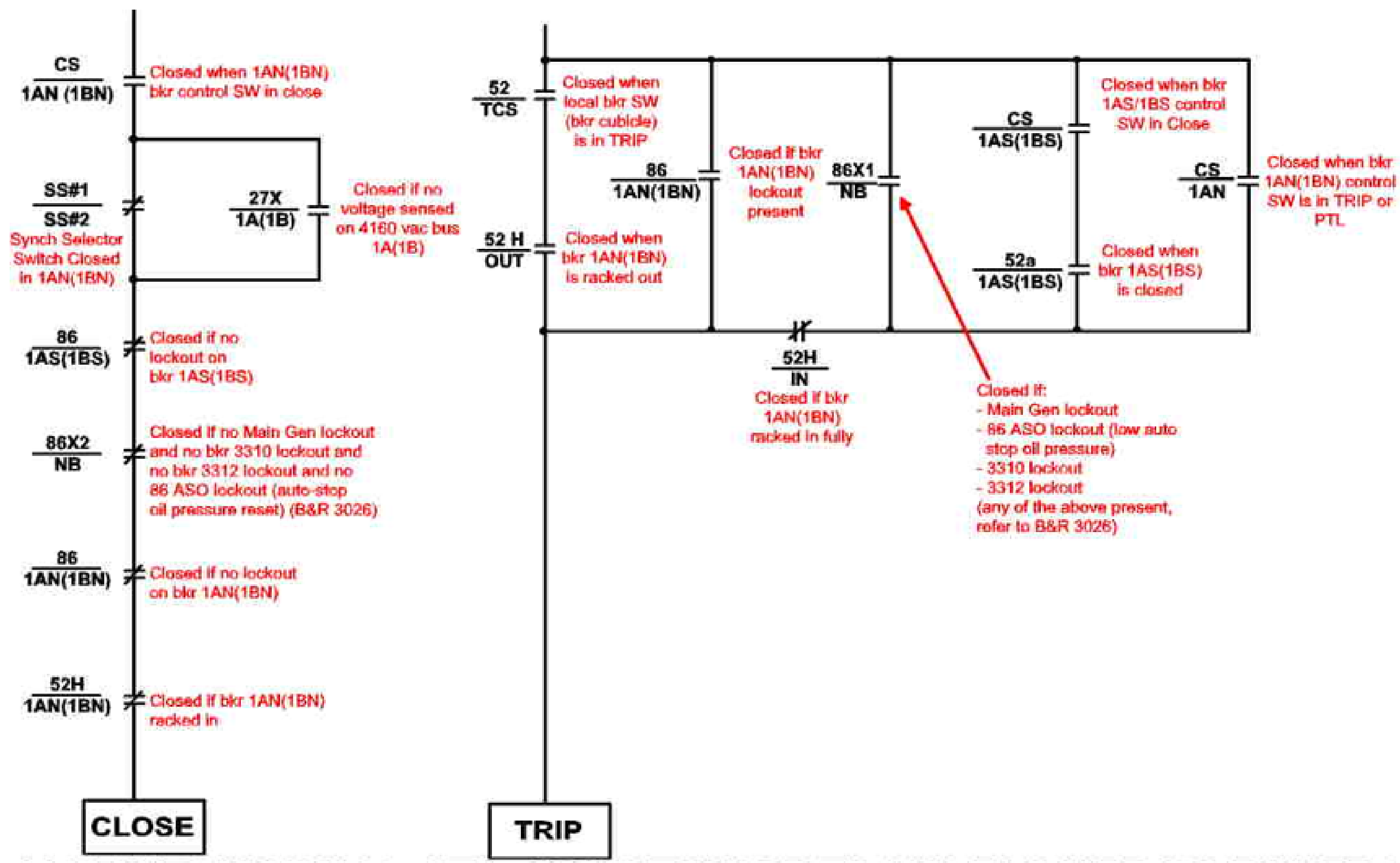
2.7.4.6 Trip caused by Diesel Generator Protection Circuit.



4160V DISTRIBUTION SYSTEM

Figure 4, Rev. 10

COR001-01



4160 VAC NORMAL STATION SERVICE TRANSFORMER BREAKERS 1AN AND (1BN) TRIP/CLOSE INTERLOCKS (SIMPLIFIED)

B&R 3017 SH 1

Figure 23, Rev. 1

COR001-01-01

CXA08410

Examination Outline Cross-Reference	Level	RO
	Tier#	2
	Group#	1
	K/A #	262002 K3.07
	Rating	2.6

262002 UPS (AC/DC)

Knowledge of the effect that a loss or malfunction of the UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.) will have on following:(CFR: 41.7 / 45.4)

K3.07 Movement of control rods: Plant-Specific

Question 47

The plant is at 8% power.

Which one of the following will be **UNAVAILABLE** if NBPP power is lost?

A. All APRM flux indication.

B. All control rod insertion capability.

C. All control rod withdrawal capability.

D. All fire protection sprinkler system automatic actuation capability.

Answer:

C. All control rod withdrawal capability.

Explanation:

Answer C is correct because NBPP supplies power to RPIS and the RMCS. Rod position information is lost. A control rod withdrawal block is generated below 10% power from the RPIS when no rod position is available. The only control rod withdrawal mechanism is via the RMCS, and with no power, control rods cannot be withdrawn.

Distracters:

A. This answer is incorrect yet plausible because APRM recorders, normally used to monitor APRMs, lose power, but APRM flux indications remain available at the APRM meters on backpanels 9-13 and 9-14.

B. This answer is plausible yet incorrect because RPIS and RMCS power are lost, which would prevent normal control rod movement from panel 9-5. However, capability to insert control rods via Scram is not affected. D. This answer is incorrect yet plausible because fire protection manual pull stations are defeated by loss of NBPP. However, for sprinkler systems utilizing heat actuated devices as for wet-pipe systems, mechanical/automatic actuation capability is unaffected. This answer was chosen during validiation.		
Technical References: Procedure 5.3NBPP (No Break Power Failure), Rev. 19 Procedure 2.2.22 (Vital Instrument Power System), Rev. 71		
References to be provided to applicants during exam: NONE		
Learning Objective:		
COR0022002001100D State the electrical power supplies to the following: RPIS		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b) 7	
Level of Difficulty:	4	
SRO Only Justification:	N/A	

- 4.4.9 Make Gaitronics announcement that fire protection manual pull stations and alarms are inoperative.
- 4.4.10 Notify FP Engineer of manual pull station impairment per Procedures 0.23 and 0.39.1.
- 4.4.11 Notify Security Shift Supervisor alternate power to SECURITY SYSTEM INVERTER 2 (SAS) ALTERNATE FEEDER is lost.
- 4.5 IF NBPP is still de-engergized and fault does not exist on NBPP:
- 4.5.1 Ensure EE-DSC-NBPP(AC), MCC-R FEED TO NBPP TRANSFMR (Cable Spreading near NBPP), is in ON.
- 4.5.2 Ensure MCC-R (R-903'-W), Breaker 2B, NO BREAK AC POWER SUPPLY, is closed and reset.
- 4.5.3 At Inverter A (125/250 A Switchgear Room), perform the following:
- 4.5.3.1 Open INVERTER OUTPUT breaker.
- 4.5.3.2 Ensure SUPPLY TO NBPP breaker is closed.
- 4.5.3.3 Ensure ALTERNATE AC INPUT TO STATIC SWITCH breaker is closed.
- 4.5.3.4 Depress ALTERNATE SOURCE SUPPLYING LOAD button.
- 4.5.4 IF NBPP is still de-energized:
- 4.5.4.1 Place MANUAL BYPASS SWITCH to ALTERNATE SOURCE TO LOAD.
- 4.5.4.2 Depress ALTERNATE SOURCE SUPPLYING LOAD button.
- 4.6 IF NBPP is still de-energized:
- 4.6.1 Open all NBPP breakers.
- 4.6.2 Ensure DLCO 3.3.1 requirements are met.
- 4.6.3 Ensure DLCO 3.3.2 requirements are met.
- 4.7 IF NBPP is energized:
- 4.7.1 Ensure NBPP Breaker 19 closed.
- 4.7.2 At IC-SW-IC218(SS) 120V AC SUPPLIES TO GAITRONICS Panel, simultaneously place both switches to NORM (Cable Spreading Room near NBPP). N/A if already in NORM.

ATTACHMENT 1 NBPP LOADS

ATTACHMENT 1 NBPP LOADS

LOAD	NOTES
Condensate booster pump and RFPT minimum flow valve control power and alarms.	CBP will recirc flow back to hotwell.
MC-FCV-17 control power.	MC-FCV-17 fails open which results in hotwell being pumped to CST.
All PANEL 9-5 Neutron Monitoring Recorders.	PNLs 9-12 and 9-14 indication are still available.
PANEL 9-5 Reactor Vessel Level Instrument, RFC-LI-94A.	
RFC-CS-83, RVLC HMI.	RVLC/RFPT controls still available on other HMIs on PANEL 9-5 and PANEL A.
PANEL 9-5 Reactor vessel level, pressure, and steam flow recorders.	
PNL 9-4 RCIC pump suction and discharge pressure indicators.	RCIC will still operate without these indications.
RCIC turbine steam inlet and exhaust pressure indicators.	RCIC will still operate without these indications.
Off-Gas High Radiation Timer, RMP-TMR-117, and isolation valve control power.	AOG-AOV-901, AOG-AOV-902, and OG-AOV-254 will not close.
RPIS and Rod Select Matrix.	
Annunciator Video Indicating Displays and Video Computers.	Master Alarm Log will be lost, but Annunciator CPUs will maintain historical record of alarms during loss of NBPP.
Reactor Building Vent Radiation Recorder RMP-RR-455.	
Elevated Release Point Flow Monitor OG-FIT-4001.	ERP Kaman cannot determine process or sample flow.
SW Effluent Radiation Monitor LPUs RMP-RM-332A and RMP-RM-332B.	Loss of process radiation high alarm inputs to Control Room RDUs RMP-RM-351A and RMP-RM-351B.
Radwaste Control Room alarms and indications.	
Drywell Radiation Monitor Optical Isolators, RMA-IO-40A and RMA-IO-40B.	
Drywell Rad Monitor Recorder RMA-RR-40.	
Main generator voltage regulator alarms.	

- 16.10 Close MC-101, FCV-12 INLET (T-882-near CBP C).
- 16.11 Issue Caution Order and tag condensate and condensate booster pump control switches, identifying that minimum flowlines are isolated.
- 16.12 Ensure Off-Site Dose Assessment Manual requirements for:
  - 16.12.1 ERP Kaman (D 3.3.2) are met due to loss of power to OG-FIT-4001 (Kaman cannot determine processor sample flow).
  - 16.12.2 SW Radiation Monitoring System (D 3.3.1) are met due to loss of power to LPUs RMP-RM-332A and RMP-RM-332B.
  - 16.12.3 ERP Alternate Sampler A (D 3.3.2) are met due to loss of power to OG-P-ERPSA.
- 16.13 Inform Shift Manager power to the following components will be lost:
  - 16.13.1 Rod Select Matrix.
  - 16.13.2 Rod Positions Information System.
  - 16.13.3 DEH backup power.
  - 16.13.4 NAWAS System.
  - 16.13.5 Annunciator VIDs and printers.
  - 16.13.6 Backup power supply to RFC-CC-3.
  - 16.13.7 Cooling fans for RFC-CC-1A and RFC-CC-1B.
  - 16.13.8 Neutron monitoring recorders.
  - 16.13.9 The following reactor vessel level indications and controls:
    - 16.13.9.1 RFC-CS-83, RVLC HMI.
    - 16.13.9.2 Reactor vessel level, pressure, and steam flow recorders.
    - 16.13.9.3 NBI-LI-94A, A reactor vessel level instrument.
  - 16.13.10 RMP-RR-455, RX BLDG VENT RADIATION RECORDER.
  - 16.13.11 RMA-RR-40, DRYWELL RAD MONITOR RECORDER.
  - 16.13.12 RCIC-PI-96, pump suction pressure indicator.
  - 16.13.13 RCIC-PI-93, RCIC pump discharge pressure indicator.
  - 16.13.14 RCIC-PI-94, RCIC turbine steam inlet pressure indicator.
  - 16.13.15 RCIC-PI-95, RCIC turbine steam exhaust pressure indicator.

1.2.6 The static switch can also be operated with the NBPP PWR TRANSFER switch on Panel C or by pressing the ALTERNATE SOURCE SUPPLYING LOAD button on the inverter. The NBPP power can also be transferred by placing the MANUAL BYPASS switch on the inverter to ALTERNATE SOURCE TO LOAD. The static switch and manual bypass switch transfer the NBPP power supply in a make before break logic.

1.2.7 NBPP feeds the following major loads: reactor vessel level recorders, NBI-LI-94A, high off-gas activity isolation logic timers and valve control power, ERP flow indicating transmitter which sends process flow signal to ERP Kaman, SW radiation monitors, Gaitronics, rod select power, rod position information system, NAWAS System, Annunciator VIDs and printers, neutron monitoring recorders, condensate pump, and condensate booster pump minimum flow valve control power and alarms, Reactor Building exhaust plenum and drywell high range radiation recorders, main generator voltage regulator alarms, fire protection manual pull stations and alarms, and the SGT System low flow to stack alarm.

1.2.8 NBPP also supplies backup power to DEH and second power source to RFC-CC-3. NBPP can also supply power to drywell fan coil unit temperature recorders, drywell nitrogen purge controls, drywell temperature indicators and alarm units, main condenser hotwell level indicators and controls, cooling fans for RFC-CC-1A and RFC-CC-1B, RVLC HMI Kaman RICs and recorders, and PC-TR-24, SUPPR POOL TEMP RECORDER, when the NORMAL/ALTERNATE POWER SUPPLY - MANUAL TRANSFER SWITCH is placed to ALT.

2. INTERLOCKS AND SETPOINTS

2.1 RPS transfer switch prevents simultaneous connection of MG Set and alternate power supply to RPSPP1A(1B).

2.2 EPAs trip on overvoltage ≤ 131 V with time delay set to ≤ 3.8 seconds or undervoltage ≥ 109V with time delay set to ≤ 3.8 seconds or under-frequency ≥ 57.2 Hz, with time delay set to ≤ 3.8 seconds. If one of these conditions exists, the LED on the front of the panel will be lit.

2.3 Static switch in Inverter A will automatically transfer NBPP to its alternate power supply if the inverter output voltage or frequency is abnormal.

3. REFERENCES

3.1 TECHNICAL SPECIFICATIONS

3.1.1 LCO 3.3.8.2, Reactor Protection System (RPS) Electric Power Monitoring.

3.1.2 LCO 3.4.8, Residual Heat Removal (RHR) Shutdown Cooling System - Cold Shutdown.

Examination Outline Cross-Reference	**Level**	RO
	Tier#	2
	Group#	1
	K/A #	262002 A2.01
	Rating	2.6

262002 UPS (AC/DC)

A2. Ability to (a) predict the impacts of the following on the UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.) ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5 / 45.6)

A2.01 Under voltage

Question 48

The plant is operating at rated power:

NBPP inverter output voltage lowers below the LOW INVERTER AC VOLTAGE setpoint.

Which one of the following completes the statements below regarding the impact low voltage has on NBPP and the procedure utilized for removing the inverter from service **for corrective maintenance**?

NBPP will____(1)____.

NBPP is required to be locally Transferred to the Alternate AC source IAW Procedure ____(2)____.

A. (1) de-energize
(2) 2.2.22 (Vital Instrument Power System)

B. (1) de-energize
(2) 5.3NBPP (No Break Power Failure)

C. (1) remain energized
(2) 2.2.22 (Vital Instrument Power System)

D. (1) remain energized
(2) 5.3NBPP (No Break Power Failure)

Answer:

C. (1) remain energized
(2) 2.2.22 (Vital Instrument Power System)

Explanation:

The No Break Power Panel (NBPP) operates from power supplied by an inverter and an AC supply (MCC-R) as alternate source. A static switch inside the inverter transfers to the alternate power source when inverter output voltage or frequency is not within limits specified for safe system operations (setpoint 212 to 220 VAC) automatically without power interruption. Since NBPP remains energized, normal transfer to the alternate source is performed IAW SOP 2.2.22 (Vital Instrument Power System). No entry conditions exist for 5.3NBPP.

Distracters:

A. This answer is incorrect due to the NBPP remaining energized. This choice is plausible due to not recognizing the inverter having auto transfer capability due to low output voltage. The candidate that does confuses low voltage transfer with output breaker trip and correctly identifies the procedure utilized to transfer would select this answer.

B. This answer is incorrect due to the NBPP remaining energized and procedure 2.2.22 utilized to transfer power. This choice is plausible due to not recognizing the inverter having auto transfer capability due to low output voltage and NBPP de-energized being an entry condition into 5.3NBPP. The candidate that confuses low voltage transfer with output breaker trip and incorrectly identifies the procedure utilized to transfer would select this answer.

D. This answer is incorrect due to procedure 2.2.22 utilized to transfer power. This choice is plausible due to NBPP de-energized being an entry condition into 5.3NBPP. The candidate that correctly identifies low voltage transfer and incorrectly identifies the procedure utilized to transfer would select this answer.

Technical References:

Procedure 5.3NBPP (No Break Power Failure), Rev. 19
Procedure 2.2.22 (Vital Instrument Power System), Rev. 71
Procedure 7.3.29.1 (No Break Power Panel Inverter Maintenance), Rev.13
Procedure 2.3_C-4 (Panel C - Annunciator C-4), Rev. 34

References to be provided to applicants during exam: NONE

Learning Objective:

COR0010102001090G Describe the AC Electrical Distribution System design feature(s) and/or interlock(s) that provide for the following: Transfer from preferred power to alternate power supplies

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X

Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b) 5	
Level of Difficulty:	2	
SRO Only Justification:	N/A	

1.2.5 The No Break Power Panel (NBPP) operates from power supplied by an inverter and an AC supply (MCC-R) as alternate source. The alternate AC supply can automatically feed NBPP without interruption. A static switch inside the inverter transfers to the alternate power source when inverter output voltage or frequency is not within limits specified for safe system operations.

PROCEDURE 2.2.22	REVISION 71	PAGE 61 OF 64

ATTACHMENT 1	INFORMATION SHEET

1.2.6 The static switch can also be operated with the NBPP PWR TRANSFER switch on Panel C or by pressing the ALTERNATE SOURCE SUPPLYING LOAD button on the inverter. The NBPP power can also be transferred by placing the MANUAL BYPASS switch on the inverter to ALTERNATE SOURCE TO LOAD. The static switch and manual bypass switch transfer the NBPP power supply in a make before break logic.

1.2.7 NBPP feeds the following major loads: reactor vessel level recorders, NBI-LI-94A, high off-gas activity isolation logic timers and valve control power, ERP flow indicating transmitter which sends process flow signal to ERP Kaman, SW radiation monitors, Gaitronics, rod select power, rod position information system, NAWAS System, Annunciator VIDs and printers, neutron monitoring recorders, condensate pump, and condensate booster pump minimum flow valve control power and alarms, Reactor Building exhaust plenum and drywell high range radiation recorders, main generator voltage regulator alarms, fire protection manual pull stations and alarms, and the SGT System low flow to stack alarm.

1.2.8 NBPP also supplies backup power to DEH and second power source to RFC-CC-3. NBPP can also supply power to drywell fan coil unit temperature recorders, drywell nitrogen purge controls, drywell temperature indicators and alarm units, main condenser hotwell level indicators and controls, cooling fans for RFC-CC-1A and RFC-CC-1B, RVLC HMI Kaman RICs and recorders, and PC-TR-24, SUPPR POOL TEMP RECORDER, when the NORMAL/ALTERNATE POWER SUPPLY - MANUAL TRANSFER SWITCH is placed to ALT.

2. INTERLOCKS AND SETPOINTS

2.1 RPS transfer switch prevents simultaneous connection of MG Set and alternate power supply to RPSPP1A(1B).

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2.3 Static switch in Inverter A will automatically transfer NBPP to its alternate power supply if the inverter output voltage or frequency is abnormal.

NO BREAK PWR PNL LOSS OF VOLTAGE	PANEL/WINDOW: C-4/D-7

1. AUTOMATIC ACTIONS

1.1 NBPP transfers to emergency AC supply from MCC-R upon loss of voltage from Inverter A.

2. OPERATOR OBSERVATION AND ACTION

2.1 Verify EE-VI-NBPP voltage in green band.

2.2 IF NBPP is de-energized, THEN perform following:

2.2.1 Enter Procedure 5.3NBPP.

15.2.3.3 Transfer power locally at NBPP Static Inverter by performing following:

a. At EE-IVTR-1A NBPP Static Inverter (125/250 VDC A Switchgear Room), perform following:

1. Check IN SYNC light is on. IF light is off, THEN stop here and inform Electricians of problem.

2. Place MANUAL BYPASS switch to ALTERNATE SOURCE TO LOAD.

3. Press ALTERNATE SOURCE SUPPLYING LOAD button.

4. Check ALTERNATE SOURCE SUPPLYING LOAD light is on.

b. At Panel C, place NO-BREAK POWER Panel VOLTS switch to 1-N and then to 2-N to check voltages are matched.

c. Place NO-BREAK POWER Panel VOLTS switch to 1-2.

16. REMOVING NBPP FROM SERVICE

16.1 Ensure reactor is in MODE 4 or 5 condition.

16.2 Inform Security Shift Supervisor alternate power to SECURITY SYSTEM INVERTER 2 (SAS) ALTERNATE FEEDER will be lost.

16.3 At 120V AC SUPPLIES FOR GAITRONICS (Cable Spreading Room north of NBPP), simultaneously place both switches to ALT.

16.4 Make Gaitronics announcement that fire protection manual pull stations will be inoperative.

16.5 Initiate Fire Protection Impairment for Loss of Power Supply to manual pull stations per Procedures 0.23 and 0.39.1.

16.6 Shut down all radwaste operations, except condensate and fuel pool filter demineralizers, due to loss of instrumentation and alarms on Main Radwaste Control Panel.

16.7 Close MC-37, FCV-17 INLET (T-882-east of TEC HXs), to prevent pumping hotwell to CST.

16.8 Close MC-99, FCV-8 INLET (T-882-near CBP A).

16.9 Close MC-100, FCV-10 INLET (T-882-near CBP B).

CNS OPERATIONS MANUAL EMERGENCY PROCEDURE 5.3NBPP NO BREAK POWER FAILURE	USE: CONTINUOUS QUALITY: QAPD RELATED EFFECTIVE: 10/17/13 APPROVAL: ITR-RDM OWNER: OSG SUPV DEPARTMENT: OPS

1. ENTRY CONDITIONS

   1.1 NBPP de-energized.

2. AUTOMATIC ACTIONS

   2.1 None.

3. IMMEDIATE OPERATOR ACTIONS

   3.1 None.

4. SUBSEQUENT OPERATOR ACTIONS

   4.1 Record current time and date. Time/Date: ________/__________

   4.2 Verify RVLC System is controlling level.

   4.3 IF MCC-R available, THEN place NBPP PWR TRANSFER switch to MCC-R.

   4.4 IF NBPP is still de-energized:

   4.4.1 Use PNLs 9-12 and 9-14 indication to monitor reactor power.

   4.4.2 Notify Electricians and Engineering to assist with determining and isolating fault.

   4.4.3 At IC-SW-IC218(SS) 120V AC SUPPLIES TO GAITRONICS (Cable Spreading Room near NBPP), simultaneously place both switches to ALT.

   4.4.4 Close MC-37, FCV-17 INLET (T-882-N east of TEC HXs).

   4.4.5 IF A CBP is not running, THEN close MC-99, FCV-8 INLET (T-882-near A CBP).

   4.4.6 IF B CBP is not running, THEN close MC-100, FCV-10 INLET (T-882-near B CBP).

   4.4.7 IF C CBP is not running, THEN close MC-101, FCV-12 INLET (T-882-near C CBP).

   4.4.8 Shut down all radwaste operations except condensate and fuel pool filter demineralizers.

ATTACHMENT 2 INFORMATION SHEET

1. DISCUSSION

1.1 This procedure outlines the actions to be taken for a loss of power from the NBPP. Significant loads affected by loss of NBPP are identified in Attachment 1 (Page 5).

1.2 PROBABLE CAUSE

1.2.1 Inverter 1A Static switch failure.

1.2.2 Blown fuses.

1.2.3 Electrical fault on NBPP.

1.3 PROBABLE ANNUNCIATORS

1.3.1 C-4/F-7, NO BREAK SYS EMERGENCY AC FAILURE.

1.3.2 C-4/D-7, NO BREAK PWR PNL LOSS OF VOLTAGE.

1.3.3 C-4/E-7, NO BREAK INVERTER 1A VOLT FAILURE.

1.4 PROBABLE INDICATIONS

1.4.1 No voltage indicated on EE-VI-NBPP.

1.4.2 No amperage indicated on EE-AI-NBPP.

1.4.3 NBPP power available lights for Inverter A or MCC-R on Bd-C off.

1.4.4 Loss of various Control Room indications, identified by Attachment 1 (Page 5).

2. REFERENCES

2.1 Administrative Procedure 0.23, CNS Fire Protection Plan.

2.2 Administrative Procedure 0.39.1, Fire Watches and Fire Impairments.

2.3 Conduct of Operations Procedure 2.0.1.2, Operations Procedure Policy.

2.4 General Operating Procedure 2.1.5, Reactor Scram.

2.5 System Operating Procedure 2.2.22, Vital Instrument Power System.

2.6 System Operating Procedure 2.2.33.1, High Pressure Coolant Injection System Operations.

2.7 System Operating Procedure 2.2.67.1, Reactor Core Isolation Cooling System Operations.

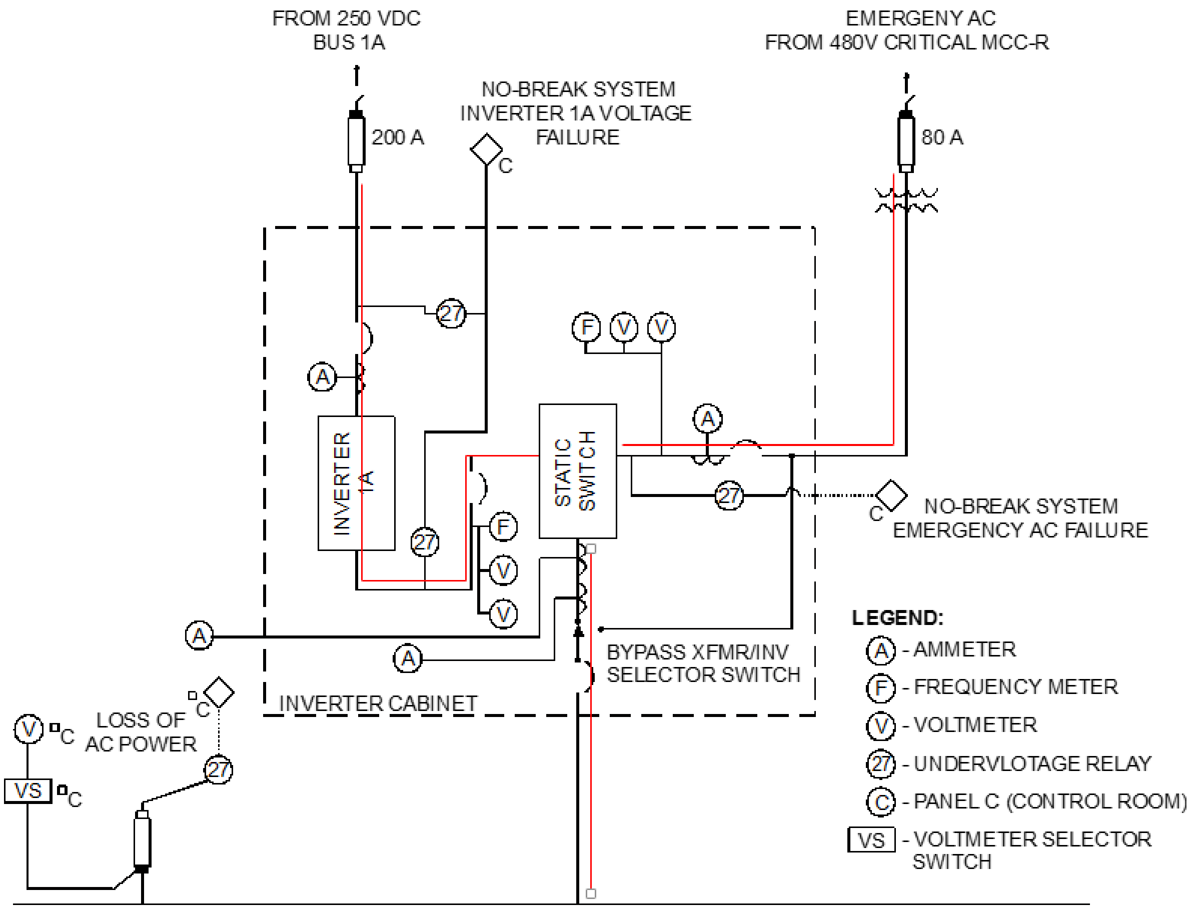
TABLE 6 - AC SENSING BOARDS

LOW INVERTER AC VOLTAGE BOARD X7			
DESIRED VAC	AS FOUND VAC	AS LEFT VAC	RANGE VAC
216			212 to 220

HIGH INVERTER AC VOLTAGE BOARD X8			
DESIRED VAC	AS FOUND VAC	AS LEFT VAC	RANGE VAC
264			259 to 269

LOW ALTERNATE SOURCE AC VOLTAGE BOARD X9			
DESIRED VAC	AS FOUND VAC	AS LEFT VAC	RANGE VAC
216			212 to 220

HIGH ALTERNATE SOURCE AC VOLTAGE BOARD X10			
DESIRED VAC	AS FOUND VAC	AS LEFT VAC	RANGE VAC
264			259 to 269



NO-BREAK POWER SYSTEM

Figure 7, Rev. 9

COR001-01

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CXA06278

Examination Outline Cross-Reference	Level	RO
	Tier#	2
	Group#	1
	K/A #	263000 K2.01
	Rating	3.1

263000 DC Electrical Distribution

K2. Knowledge of electrical power supplies to the following: (CFR: 41.7)

K2.01 Major D.C. loads

Question 49

The plant is operating at rated power.

Which one of the following will lose power if 125V DC Panel 1B is de-energized?

A. HPCI Initiation Logic

B. RCIC Initiation Logic

C. 4160V BUS 1A Breaker Control Power

D. 4160V BUS 1F Breaker Control Power

Answer:
A. HPCI Initiation Logic

Explanation:
125V DC Panel 1B provides power to HPCI Initiation Logic.

Distracters:
B. This answer is incorrect due to 125V DC Panel 1B providing power to HPCI Initiation Logic. RCIC initiation logic is powered from 125V DC bus 1A. This choice is plausible due to being easily confused as logic power source. The candidate that confuses RCIC Initiation logic power source would select this answer.
C. This answer is incorrect due to 125V DC Panel 1B providing power to HPCI Initiation Logic. 4160V BUS 1A Breaker Control Power is powered from 125V DC bus 1A. This choice is plausible due to being easily confused as Control power

source. The candidate that confuses 4160V BUS 1C Breaker Control Power source would select this answer. D. This answer is incorrect due to 125V DC Panel 1B providing power to HPCI Initiation Logic. 4160V BUS 1F Breaker Control Power is powered from 125V DC bus 1A. This choice is plausible due to being easily confused as Control power source. The candidate that confuses 4160V BUS 1F Breaker Control Power source would select this answer.		
Technical References: Procedure 5.3DC125 (LOSS OF 125 VDC), Rev. 35		
References to be provided to applicants during exam: NONE		
Learning Objective:		
COR0020702001080B Given a specific DC Electrical Distribution system malfunction, determine the effect on any of the following: Components using DC control power (i.e., breakers)		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank # 4025	X
	New	
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b) 7	
Level of Difficulty:	2	
SRO Only Justification:	N/A	

QUESTION: 109 4025 (1 point(s))

A loss of 125 VDC Distribution Panel A will result in loss of power to which of the following?

a. HPCI control logic.

b. Main Generator Trip logic.

c. 4160V breaker control power for Bus 1A and 1E.

d. RFC-CS-83, RFC OPERATOR CONTROL STATION HMI.

ANSWER: 109 4025

d. 4160V breaker control power for Bus 1A and 1E.

ATTACHMENT 13 INFORMATION SHEET

1. DISCUSSION

1.1 125 VDC Distribution Panels supply control and instrument power for annunciators, control logic power, and protective relaying. Because of its importance, a loss of a 125 VDC Distribution Panel can only be tolerated for a very short period. This procedure provides guidance to protect plant electrical equipment and maintain reactor safety should the loss of an entire 125 VDC Distribution or individual Distribution Panel occur. The procedure is written assuming the plant is operating at 100% power in normal operating electrical configurations. Some steps may not be required if the plant is operating at < 100% power.ⓟ[1]

1.2 If 125 VDC Distribution Panel A is lost, the following major equipment is affected: RRMG A speed and breaker control, 4160V Bus 1A, 1E, and 1F breaker control and undervoltage logics, 480V Bus 1A and 1F breaker control, the right light in all Control Room annunciators, annunciator panels for Water Treatment, RHR A Gland Water, Auxiliary Steam Boiler C, DG-1 starting and breaker control logics, CS A, RCIC, and RHR A control logics, TIP valve control monitors, main generator voltage regulation, RFPT A trip logic, and ARI solenoid valve power.

1.3 If 125 VDC Distribution Panel B is lost, the following major equipment is affected: RRMG B speed and breaker control, 4160V Bus 1B and 1G breaker control and undervoltage logics, 480V Bus 1B and 1G breaker control, the left light in all Control Room annunciators, annunciator panels for ALRW, RHR B Gland Water, Auxiliary Steam Boiler D, DG-2 starting and breaker control logics, CS B, HPCI, and RHR B control logics, main generator trip logic, main generator and transformer protective relaying, bypass valves fail to control pressure after turbine trip and RFPT B trip logic.

1.4 If Panel AA2 is lost, RFPT Trip Logic Relay 30TTS will de-energize (RFPT trip monitor). RCIC will not operate. High Level trip on HPCI will not function. Core Spray Pump A, RHR Pump A, and RHR Pump B will not start automatically, but can be started from Control Room. DG-1 will not auto start on High Drywell Pressure or Low Vessel Level. The low pressure permissive opening logic for CS-MO-11A, CS-MO-12A, and RHR-MO-27A will not function; these valves may be driven closed but may not be opened from the Control Room. RFP A has no trip protection other than mechanical overspeed, cannot be tripped from Control Room, and must be tripped locally.

Examination Outline Cross-Reference	Level	RO
	Tier#	2
	Group#	1
	K/A #	263000 A3.01
	Rating	3.2

263000 DC Electrical Distribution

A3. Ability to monitor automatic operations of the D.C. ELECTRICAL DISTRIBUTION including: (CFR: 41.7 / 45.7)

A3.01 Meters, dials, recorders, alarms, and indicating lights

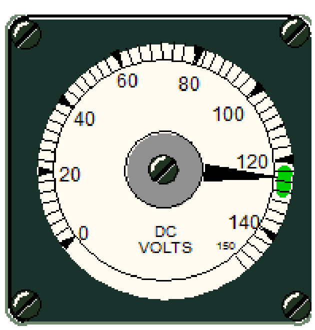
Question 50

While operating at rated power the following annunciator is received:

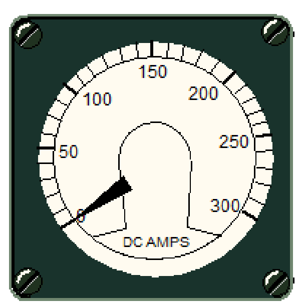
125V DC BATT CHARGER 1A TROUBLE	PANEL/WINDOW: C-1/C-2

CRT alarm message indicates:
(3765) 125V DC BATTERY CHARGER 1A DC VOLTAGE HIGH (in and reset)
(3762) 125V DC BATTERY CHARGER 1A AC VOLTAGE FAILURE.
(3764) 125V DC BATTERY CHARGER 1A DC VOLTAGE LOW

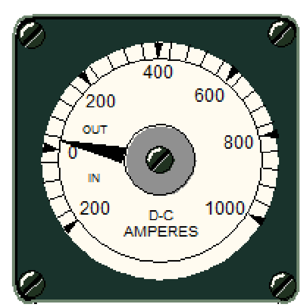
The following indications are observed for 125V DC Bus 1A:



VOLTS



CHARGER 1A AMPS



BATTERY AMPS

What is the position of the 125V Charger 1A AC input and DC output breakers and why?

A. ONLY the AC input breaker is OPEN due to a loss of AC power.

B. ONLY the AC input breaker is OPEN due to Battery Charger 1A DC voltage high.

C. Both the AC input and DC output Breakers are OPEN due to a loss of AC power.

D. Both the AC input and DC output Breakers are OPEN due to Battery Charger 1A DC voltage high.

Answer:
B. ONLY the AC input breaker is OPEN due to Battery Charger 1A DC voltage high.

Explanation:
The AC input breaker has tripped open due to battery charger 1A DC voltage high. DC output over voltage causes the AC input breaker on a 125V CHARGER to trip. The DC output breaker does NOT automatically trip open. Neither breaker automatically trips open due to loss of AC power to the chargers.

Distracters:
A. This answer is incorrect due to the AC Input breaker being open due to high voltage. This choice is plausible due to low voltage being a cause for the 125V DC BATT CHARGER 1A TROUBLE annunciator. The candidate that recognizes the charger breaker status and confuses the cause would select this answer.
C. This answer is incorrect due to the DC output breaker remaining closed and the AC Input breaker being open due to high voltage. This choice is plausible due to undervoltage trips being common breaker trips and low voltage being a cause for the 125V DC BATT CHARGER 1A TROUBLE annunciator. The candidate that does not know battery charger breaker trips and confuses the cause would select this answer.
D. This answer is incorrect due to the DC output breaker remaining closed during high voltage. This choice is plausible due to high voltage trip causing the AC input breaker to trip. The candidate that does not know specific battery charger breaker trips and recognizes the high voltage trip would select this answer.

Technical References:
Procedure 2.2.25.1 125 VDC Electrical System (Div 1)}, Rev. 19
3058 DC One Line Diagram
Procedure 2.3_C-1 (Panel C - Annunciator C-1), Rev. 30

References to be provided to applicants during exam: NONE

Learning Objective:
COR0020702001080D Given a specific DC Electrical Distribution system malfunction, determine the effect on any of the following: Battery chargers

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank # 12518	X
	New	

Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b) 7	
Level of Difficulty:	4	
SRO Only Justification:	N/A	

QUESTION: 15 12518 (1 point(s))

The plant is at 100% power with the 125 VDC electrical distribution system aligned for normal lineup. The following occur:

- Annunciator C-4/C-7, 125 VDC BATT CHARGER 1B TROUBLE alarms
- CRT alarm message indicates:

(3765) 125V DC BATTERY CHARGER 1B DC VOLTAGE HIGH (in and reset)
(3762) 125V DC BATTERY CHARGER 1B AC VOLTAGE FAILURE.
(3764) 125V DC BATTERY CHARGER 1B DC VOLTAGE LOW

The following 125 VDC indications are observed:

- 125 VDC Bus 1B indicates approximately 120 volts and stable
- 125 VDC Battery 1B indicates approximately 75 amps OUT and stable
- 125 VDC Charger 1B indicates 0 amps

What is the position (open or closed) of the 125V charger 1B AC input and DC output breakers and why?

a. AC input breaker tripped open due to a loss of AC power to the chargers.
DC output breaker remains closed.

b. AC input breaker tripped open due to battery charger 1B DC voltage high.
DC output breaker remains closed.

c. AC input breaker **AND** DC output breaker tripped open due to a loss of AC power to the chargers.

d. AC input breaker **AND** DC output breaker tripped open due to battery charger 1B DC voltage high.

ANSWER: 15 12518

b. AC input breaker tripped open due to battery charger 1B DC voltage high.
DC output breaker remains closed.

SETPOINT	CIC	C-1/C-2
Relay operation caused by:		
1. (3725) 125 VDC BATTERY CHARGER 1A AC VOLTAGE FAILURE	1. EE-CHG-125A(27K2)	
2. (3726) 125 VDC BATTERY CHARGER 1A OUTPUT BKR TRIP	2. EE-CHG-125A(DCB)	
3. (3727) 125 VDC BATTERY CHARGER 1A DC VOLTAGE LOW	3. EE-CHG-125A(K5L)	
4. (3728) 125 VDC BATTERY CHARGER 1A DC VOLTAGE HIGH	4. EE-CHG-125A(K5H)	

PROBABLE CAUSES

- Charger internal problem (failure to regulate).
- AC voltage failure.
- Output breaker tripped.

REFERENCES

- Technical Specification LCO 3.8.4, DC Sources - Operating.
- Technical Specification LCO 3.8.5, DC Sources - Shutdown.
- Technical Specification LCO 3.8.6, Battery Cell Parameters.
- Ⓟ[1] SCR 2003-0881, Increased Trend in Missed Technical Specification Surveillances. Affects Step 1.3.

2. INTERLOCKS AND SETPOINTS

2.1 DC output over voltage causes the AC input breaker on 125V Charger to trip.

3. REFERENCES

3.1 TECHNICAL SPECIFICATIONS

3.1.1 LCO 3.6.1.3, Primary Containment Isolation Valves (PCIVs).

3.1.2 LCO 3.8.1, AC Sources - Operating.

3.1.3 LCO 3.8.4, DC Sources - Operating.

3.1.4 LCO 3.8.5, DC Sources - Shutdown.

3.1.5 LCO 3.8.7, Distribution System - Operating.

3.1.6 LCO 3.8.8, Distribution System - Shutdown.

3.2 UPDATED SAFETY ANALYSIS REPORT

3.2.1 Section VIII-6.0, 125/250 Volt D-C Power Systems.

3.3 DRAWINGS

3.3.1 B&R Drawing 3002, Auxiliary One Line Diagram.

3.3.2 B&R Drawing 3006, Auxiliary One Line Diagram.

3.3.3 B&R Drawing 3058, DC One Line Diagram.

3.3.4 B&R Drawing 3059, DC Panel Schedules.

3.3.5 B&R Drawing 3401, Auxiliary One Line Diagram.

3.4 VENDOR MANUALS

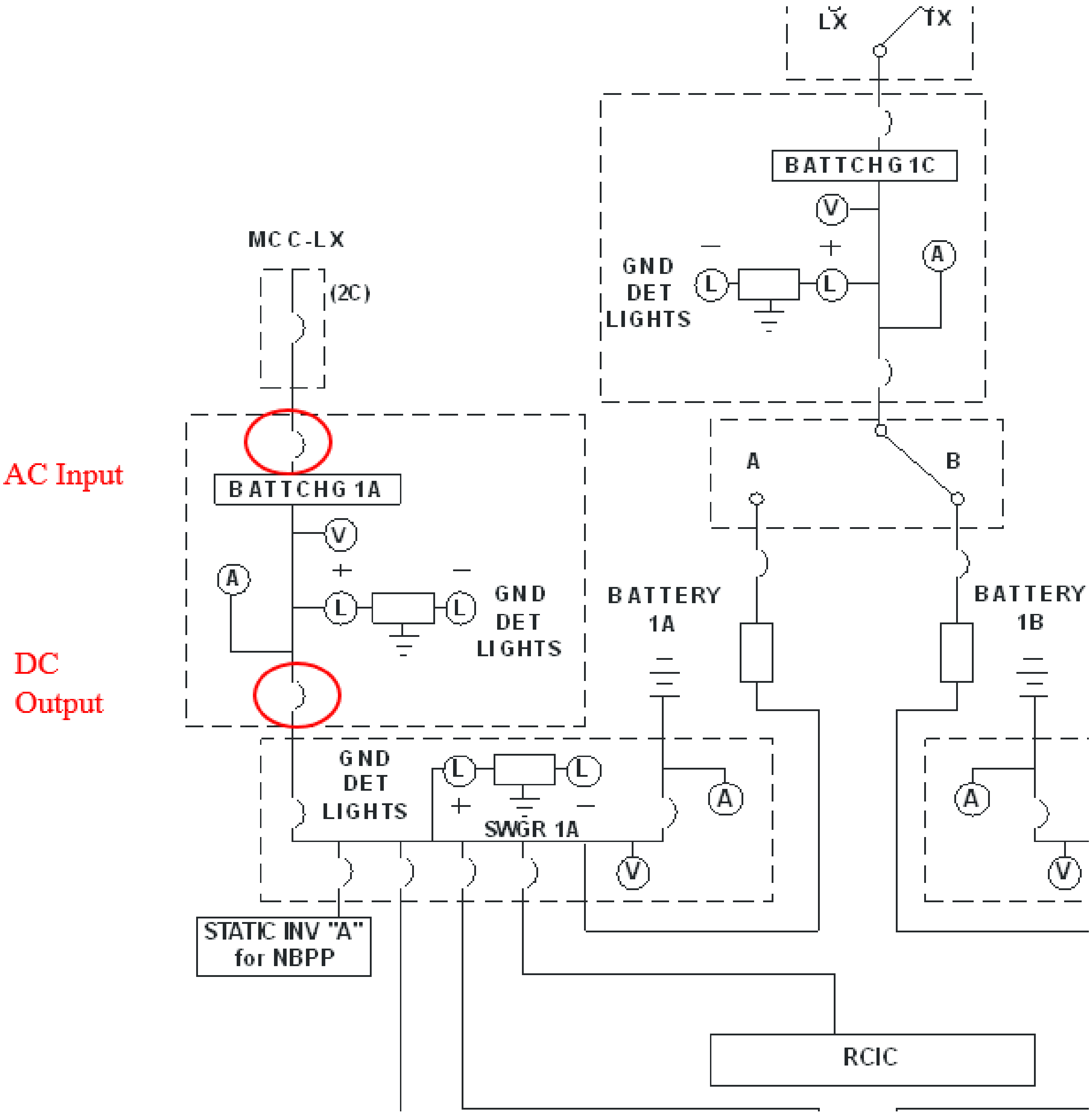
3.4.1 CNS Number 1188, 125 Volt Batteries and Battery Chargers.

3.5 PROCEDURES

3.5.1 Administrative Procedure 0.39.1, Fire Watches and Fire Impairments.

3.5.2 System Operating Procedure 2.2A_125DC.DIV1, 125 VDC Power Checklist (Div 1).

3.5.3 System Operating Procedure 2.2.2, Carbon Dioxide Systems.


LX
TX
BATTCHG 1C
V
A
GND
DET
LIGHTS
L
MCC-LX
(2C)
AC Input
BATTCHG 1A
DC
Output
A
B
BATTERY
1A
BATTERY
1B
SWGR 1A
STATIC INV "A"
for NBPP
RCIC

Examination Outline Cross-Reference	**Level**	RO
Comments incorporated – revised to remote ONLY and OR local.	**Tier#**	2
	Group#	1
	K/A #	264000 A4.04
	Rating	3.7

264000 EDGs

A4. Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8)

A4.04 Manual start, loading, and stopping of emergency generator: Plant-Specific

Question 51

The plant is operating at rated power.

DG1 is operating in parallel with off-site power from the Control Room.

Which one of the following completes the statements below regarding the impact of taking the governor control switch to raise and the action required to shutdown the DG IAW Procedure 2.2.20 {Standby AC Power System (Diesel Generator)}?

Taking the Governor Control Switch to RAISE will cause DG ____(1)____ to rise.

The DG is shutdown by placing the DIESEL GEN STOP/START switch to STOP on Panel C ____(2)____.

A. (1) WATT loading
(2) ONLY

B. (1) WATT loading
(2) OR pressing the DG-SW-DG1(5L), LOCAL ENGINE STOP button

C. (1) output frequency
(2) ONLY

D. (1) output frequency
(2) OR pressing the DG-SW-DG1(5L), LOCAL ENGINE STOP button

Answer:
A. (1) WATT loading

(2) ONLY

Explanation:
With the DG in parallel with off site power, taking the governor control switch to raise will cause the DG to pick up additional load therefore raising watt loading. IAW procedure 2.2.20, while operating the DG remotely from the control room requires placing the DIESEL GEN STOP/START switch to STOP on Panel C to shutdown the DG.

Distracters:

B. This answer is incorrect due to stopping the DG when operating from the control room requires placing the DIESEL GEN STOP/START switch to STOP on Panel C ONLY. This choice is plausible due to pressing the local STOP button being a method to shutdown the DG but not IAW procedure 2.2.20 or if the stem were changed to having the DG being operated locally (making this the correct answer). The candidate that correctly identifies the WATT loading rise and does not recall the procedure requirement when operating from the control room would select this answer.

C. This answer is incorrect due to WATT loading rising. This choice is plausible if being parallel with off site power response is confused with isolated bus. The candidate that confuses isolated bus with parallel operations and correctly identifies the procedure requirement to shutdown the DG when operating from the control room would select this answer.

D. This answer is incorrect due to WATT loading rising and stopping the DG when operating from the control room requires placing the DIESEL GEN STOP/START switch to STOP on Panel C. This choice is plausible if being parallel with off site power response is confused with isolated bus and due to pressing the local STOP button being a method to shutdown the DG but not IAW procedure 2.2.20 or if the stem were changed to having the DG being operated locally (making this the correct answer). The candidate that confuses isolated bus with parallel operations and does not recall the procedure requirement when operating from the control room would select this answer.

Technical References:
Procedure 2.2.20 (Standby AC Power System (Diesel Generator), Rev. 92

References to be provided to applicants during exam: NONE

Learning Objective: SKL012420800A030K Given plant conditions, predict changes in the following DG system components/parameters: Manual start, load, and stop of DG

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X

Question Cognitive Level:	Memory/Fundamental	

	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b) 7	
Level of Difficulty:	3	
SRO Only Justification:	N/A	

4. DG SHUTDOWN METHODS ®[2]

<u>NOTE</u> – The following is a list of methods that may be used to shut down DG during emergency conditions. Methods are <u>not</u> listed in any particular order. See DISCUSSION for shutdown interlock criteria.

4.1 DG1 SHUTDOWN

4.1.1 At ENGINE #1 CONTROL PANEL:

4.1.1.1 IF DG-SW-DG1(43CM), DG1 CONTROL MODE SELECTOR, switch is in LOCAL, THEN perform following:

a. Press DG-SW-DG1(5L), LOCAL ENGINE STOP button until STOP light turns on.

4.1.1.2 IF DG-SW-DG1(43CM), DG1 CONTROL MODE SELECTOR, switch is in REMOTE, THEN perform following:

a. At PANEL C, place DIESEL GEN STOP/START switch to STOP for 1 to 2 seconds, then release.

4.1.2 At ENGINE #1 CONTROL PANEL, press EMERGENCY STOP button.

4.1.3 At Left Bank Air Start Panel, press DG EMERGENCY SHUTDOWN long enough for control air to bleed off safety trip valve.

4.1.4 Manually shut off fuel racks.

<u>NOTE</u> – See Attachment 2 for location of Span Adjustment Lever.

4.1.4.1 Place open end wrench (located SE corner of DG Room by EG 1 breaker) over top of Span Adjustment Lever (catwalk; south end of engine on fuel rack linkage).

4.1.4.2 Rotate wrench and lever clockwise to right and hold until engine has stopped.

Examination Outline Cross-Reference	Level	RO
	Tier#	2
	Group#	1
	K/A #	300000 K1.05
	Rating	3.1

300000 Instrument Air

K1 Knowledge of the connections and / or cause effect relationships between INSTRUMENT AIR SYSTEM and the following: (CFR: 41.2 to 41.9 / 45.7 to 45.8)

K1.05 Main Steam Isolation Valve air

Question 52

The plant is operating at 50% power.

Which one of the following identifies the impact that a complete loss of Instrument Air will have on the Main Steam Isolation Valves (MSIVs)?

A. The Inboard MSIVs drift closed.

B. The Inboard MSIVs fast close IMMEDIATELY.

C. The Outboard MSIVs drift closed.

D. The Outboard MSIVs fast close IMMEDIATELY.

Answer:

C. The Outboard MSIVs drift close.

Explanation:

With the plant at power the inboard MSIVs are supplied pneumatics by nitrogen and are unaffected by the loss of instrument air. The outboard MSIVs will lose pneumatics and will drift close when the accumulators bleed off.

Distracters:

A. This answer is incorrect due to the outboard MSIVs drifting closed while at 50% power. This choice is plausible if the stem were changed to having the plant shutdown (DW open for access) making it the correct answer. The candidate does not recall that the INBD MSIVs are supplied with N2 at power or does not recognize the plant at power would select this answer.

B. This answer is incorrect due to the outboard MSIVs drifting closed while at 50% power. This choice is plausible if the stem were changed to having the plant shutdown (DW open for access) and not recognizing MSIVs having accumulators keeping them open. The candidate that does not recall which MSIVs have accumulators and are supplied with IA at power or does not recognize the plant at power would select this answer.

D. This answer is incorrect due to the outboard MSIVs drifting closed while at 50% power. This choice is plausible if it is not recognized that the ALL MSIVs have accumulators keeping them open. The candidate that does not recall MSIVs having accumulators would select this answer.

Technical References:
Procedure 5.2AIR (Loss Of Instrument Air), Rev. 21

References to be provided to applicants during exam: NONE

Learning Objective:
COR0021402001030F Describe the interrelationships between the Main Steam System and the following: Plant air systems/N2 systems

Question Source:	Bank # 19073	
(note changes; attach parent)	Modified Bank #	
	New	
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b) 7	
Level of Difficulty:	2	
SRO Only Justification:	N/A	

QUESTION: 2, 19073 (1 point(s))

Given the following:

- The plant is operating at 50% power.
- A complete loss of instrument air occurs.

Which one of the following describes the impact on the Main Steam Isolation Valves (MSIVs) due to the loss of instrument air?

a. Inboard MSIVs remain open; Outboard MSIVs drift close.

b. Inboard MSIVs remain open; Outboard MSIVs close immediately.

c. Outboard MSIVs remain open; Inboard MSIVs drift close.

d. Outboard MSIVs remain open; Inboard MSIVs close immediately.

ANSWER: 2, 19073

a. is correct. With the plant at power the inboard MSIVs are supplied pneumatics by nitrogen and are unaffected by loss of instrument air; the outboard MSIVs will lose pneumatics and drift close when the accumulators bleed off.

1.7 Relief valve accumulators provide nitrogen for five actuations of MS-RV-71A, MS-RV-71B, MS-RV-71C, MS-RV-71E, MS-RV-71G, and MS-RV-71H. Accumulators for MS-RV-71D and MS-RV-71F are larger and provide for fourteen actuations since these valves cycle at lower setpoints as part of the ADS/LLS logic.

1.8 The time that SA-MO-81 is open should be minimized to avoid the introduction of moisture and dirt to IA System.

1.9 PROBABLE CAUSE

1.9.1 Multiple compressor failures.

1.9.2 System component failures.

1.9.3 Air dryer cyclic controls failure.

1.10 PROBABLE ANNUNCIATORS

1.10.1 9-3-1/C-2, DRYWELL PNEUMATIC HDR LOW PRESSURE (if supplied from instrument air).

1.10.2 9-5-1/C-4, ROD DRIFT (rods inserting due to scram valves opening).

1.10.3 9-5-2/B-2, MSIV NOT FULL OPEN TRIP.

1.10.4 9-5-2/F-5, SCRAM VALVE PILOT AIR LOW PRESSURE.

1.10.5 A-4/A-4, AIR RECEIVER A OR B LOW PRESSURE.

1.10.6 A-4/A-5, CONTROL AIR LOW PRESSURE.

1.10.7 A-4/B-4, SERVICE AIR ISOLATION PCV-609.

1.10.8 A-4/B-5, SERVICE AIR LOW PRESSURE.

1.10.9 A-4/F-5, AIR DRYER TROUBLE.

1.10.10 A-4/G-4, INTAKE BLDG CONTROL AIR LOW PRESSURE.

1.11 PROBABLE INDICATIONS

1.11.1 MSIVs supplied by Instrument Air, drifting closed.

1.11.2 Scram valves opening.

2. REFERENCES

2.1 PROCEDURES

2.1.1 Administrative Procedure 0.23, CNS Fire Protection Plan.

Examination Outline Cross-Reference	Level	RO
	Tier#	2
	Group#	1
	K/A #	400000 K2.01
	Rating	2.9
400000 Component Cooling Water K2. Knowledge of electrical power supplies to the following: (CFR: 41.7) K2.01 CCW pumps		

Question 53

Which MCC powers REC Pump B?

A. MCC-K

B. MCC-R

C. MCC-S

D. MCC-Y

Answer:
A. MCC-K
Explanation: REC Pump B is powered from MCC-K.
Distracters: B. This answer is incorrect due to REC Pump B is powered from MCC-K. This choice is plausible due to other REC components being powered from MCC-R (REC MO 700, NON-CRITICAL HEADER SUPPLY VALVE). The candidate that does not know where REC Pump B is powered from would select this answer. C. This answer is incorrect due to REC Pump B is powered from MCC-K. This choice is plausible due to other REC components being powered from MCC-S (REC Pump C & D). The candidate that confuses where REC Pump B is powered from would select this answer. D. This answer is incorrect due to REC Pump B is powered from MCC-K. This choice is plausible due to other REC components being powered from MCC-Y

(REC MO 712, HX A OUTLET VLV). The candidate that does not know where REC Pump B is powered from would select this answer.

Technical References:
Procedure 2.2.65 (Reactor Equipment Cooling Water System) Rev. 64
Procedure 2.2A.REC.DIV1 {Reactor Equipment Cooling Water System Component Checklist (DIV 1)}, Rev. 00
Procedure 2.2A.REC.DIV2 {Reactor Equipment Cooling Water System Component Checklist (DIV 2)}, Rev. 01

References to be provided to applicants during exam: NONE

Learning Objective:
COR0021902001020B State the electrical power supplies to the following REC components: Pump Motors

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b) 7	
Level of Difficulty:	3	
SRO Only Justification:	N/A	

4.5 Open Breaker 7C on MCC-R for REC-MO-700, NON-CRITICAL HEADER SUPPLY VALVE.

4.6 Open Breaker 4B on MCC-RB for REC-MO-713, HX B OUTLET VLV (R-903-W), or Breaker 4C on MCC-Y for REC-MO-712, HX A OUTLET VLV (R-903-SW).

4.7 Manually throttle open REC-MOV-700MV (R-931-N REC HX area) far enough to fill downstream piping (~ 25% open).

4.8 Manually throttle open REC-MO-1329MV (R-931-N REC HX area) far enough to fill downstream piping (~ 25% open).

ATTACHMENT 2 POWER SUPPLY CHECKLIST

DESCRIPTION	POWER SUPPLY	NORMAL POSITION	PERFORMED BY	VERIFIED BY	COMMENTS
AUXILIARY RELAY CABINET CCP-1A (PCV-609)	EE-PNL-CCP1A(13)	ON			
4160V CRITICAL SWITCHGEAR 1F	EE-PNL-CCP1A(17)	ON			
VBD-M (REC VALVE CONTROLS) REC-701AV	EE-PNL-CCP1A(8)	ON			
REC PUMP A	EE-MCC-K(3C)	ON			
REC PUMP B	EE-MCC-K(4D)	ON			
REC PUMP A MOTOR HEATER	EE-PNL-LPR1F(33)	ON			
REC PUMP B MOTOR HEATER	EE-PNL-LPR1F(34)	ON			
REC-MO-711	EE-MCC-Q(7B)	ON			
REC-MO-697	EE-MCC-R(9C)	ON			

PROCEDURE 2.2A.REC.DIV1 REVISION 0 PAGE 5 OF 15

ATTACHMENT 2 POWER SUPPLY CHECKLIST

DESCRIPTION	POWER SUPPLY	NORMAL POSITION	PERFORMED BY	VERIFIED BY	COMMENTS
AUXILIARY RELAY CABINET CCP-1B (H&V CONTROL UNIT B)	EE-PNL-CCP1B(10)	ON			
AUXILIARY RELAY CABINET CCP-1B (REC SYSTEM) (CIRCULATING WATER PUMP)	EE-PNL-CCP1B(14)	ON			
REC-MO-698	EE-MCC-R(9D)	ON			
4160 V SWITCHGEAR G (RELAY 1GR POWER SUPPLY)	EE-PNL-CPP2(11)	ON			
REC-MO-714	EE-MCC-Y(7C)	ON			
REC PUMP C	EE-MCC-S(4C)	ON			
REC PUMP D	EE-MCC-S(5C)	ON			
REC PUMP C MOTOR HEATER	EE-PNL-LPR1G(23)	ON			
REC PUMP D MOTOR HEATER	EE-PNL-LPR1G(25)	ON			

PROCEDURE 2.2A.REC.DIV2 REVISION 1 PAGE 6 OF 17

Examination Outline Cross-Reference	Level	RO
Revised to determine cause of HCU accumulator alarm and procedure used to correct.	Tier#	2
	Group#	2
	K/A #	201003 A2.08
	Rating	3.8

201003 Control Rod and Drive Mechanism

A2. Ability to (a) predict the impacts of the following on the CONTROL ROD AND DRIVE MECHANISM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5 / 45.6)

A2.08 Low HCU accumulator pressure/high level

Question 54

The plant is operating at power when the following alarm is received:

CRD ACCUM LOW PRESS OR HIGH LEVEL	PANEL/WINDOW: 9-5-2/G-6

The building operator reports the lighted pushbutton for HCU 22-19 remained lit when depressed on the local HCU Trouble Panel.

Which of the following completes the statements below regarding the cause of the HCU Accumulator alarm and the procedure used to correct this condition?

The HCU accumulator alarm is due to ____(1)____.

The specific steps to correct this condition are contained in Procedure ____(2)____.

A. (1) High Level
   (2) 2.4CRD (CRD Trouble)

B. (1) High Level
   (2) 2.2.8 (Control Rod Drive Hydraulic System)

C. (1) Low Pressure
   (2) 2.4CRD (CRD Trouble)

D. (1) Low Pressure
(2) 2.2.8 (Control Rod Drive Hydraulic System)

Answer:		
D. (1) Low Pressure (2) 2.2.8 (Control Rod Drive Hydraulic System)		
Explanation: Requires knowledge of HCU alarm panel indications and abnormal procedure entry conditions (or lack of entry condition). HCU accumulator Hi level and/or Low Pressure cause Control Room annunciation and lights a pushbutton indicating light located on the end of each HCU bank (panels 25-22 & 25-4). When the annunciator is received, a Station Operator is dispatched to determine the cause. If the light goes out when depressed – the alarm is due to High Accumulator water level (requires draining to correct). If the light stays lit when depressed the alarm is due to Low Accumulator pressure (requires charging to correct). Requires knowledge of Abnormal Procedure entry conditions which do not exist due to this CRD alarm. Draining/Recharging scram accumulators is governed by procedure 2.2.8, Control Rod Drive Hydraulic System. Therefore, Answer D is correct.		
Distracters: Answers that state High Level are plausible due to commonly confusing the light indications when the pushbutton is depress. This includes answers A and B. These answers are wrong, because the alarm is due to low pressure. Answers that state procedure 2.4CRD contains steps to recharge the scram accumulator are plausible, because 2.4CRD contains mitigating actions for a variety of anomalies/malfunctions (CRD high temperature) affecting the CRD system. This includes answers A and C. These are wrong, because scram accumulator low pressure is not an entry condition for 2.4CRD, and 2.4CRD does not contain these steps, procedure 2.2.8 does.		
Technical References: Procedure 2.3_9-5-2 (Panel 9-5 - Annunciator 9-5-2), Rev. 33 2.2.8, Control Rod Drive Hydraulic System Procedure 2.4CRD (CRD Trouble), Rev. 15		
References to be provided to applicants during exam: NONE		
Learning Objective:		
COR002-04-02 9. Given a CRDH system component manipulation, predict and explain the changes in the following parameters: e. HCU pressure/level		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X

Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b) 5	
Level of Difficulty:	3	
SRO Only Justification:	N/A	

re-coupled.

11. HCU ACCUMULATOR ALARM CORRECTIVE ACTION

11.1 Determine cause of alarm by depressing lighted pushbutton for HCU XX-XX on Panel 25-22 (R-903-N) or Panel 25-4 (R-903-S).

11.2 IF light stays on while button is depressed, THEN trouble is low gas pressure.

11.2.1 Report pressure indicated on PI-131 of associated HCU to Control Room.

11.2.2 Charge accumulator per Section 6, 7, 8, or 9.

11.3 IF light goes out while button is depressed, THEN trouble is water leakage. Proceed as follows to remove water from HCU accumulator instrumentation block:

11.3.1 Close CRD-111(XX-XX), INSTRUMENT BLOCK SHUTOFF HCU XX-XX.

CRD ACCUM LOW PRESS OR HIGH LEVEL	PANEL/WINDOW: 9-5-2/G-6

1. OPERATOR OBSERVATION AND ACTION

    1.1 Enter applicable LCO 3.1.5, Conditions and Required Actions.

    1.2 Correct cause per Procedure 2.2.8.

SETPOINT	CIC	9-5-2/G-6
1. (2756) CRD ACCUM LOW PRESSURE at 960 psig	1. CRD-PS-130(XX-YY)	
2. (2756) CRD ACCUM HIGH LEVEL at 37 cc	2. CRD-LS-129(XX-YY)	

PROBABLE CAUSES

- Instrument block seal leakage.

REFERENCES

- Technical Specifications LCO 3.1.5, Control Rod Scram Accumulators.
- System Operating Procedure 2.2.8, Control Rod Drive Hydraulic System.

CNS OPERATIONS MANUAL SYSTEM OPERATING PROCEDURE 2.2.8 CONTROL ROD DRIVE HYDRAULIC SYSTEM	USE: CONTINUOUS QUALITY: QAPD RELATED EFFECTIVE: 3/27/15 APPROVAL: ITR-RDM OWNER: OSG SUPV DEPARTMENT: OPS

CNS OPERATIONS MANUAL ABNORMAL PROCEDURE 2.4CRD CRD TROUBLE	USE: CONTINUOUS QUALITY: QAPD RELATED EFFECTIVE: 12/2/10 APPROVAL: ITR-RDM OWNER: OSG SUPV DEPARTMENT: OPS

P&I: 2

Scram Actions

1. ENTRY CONDITIONS

1.1 Changing RPIS indications when drive is not moved intentionally.

1.2 ROD DRIFT light (red) on full core display.

1.3 Reactor power or flux indication does not change when a control rod is moved.

1.4 Control rod position indication does not change when drive movement is attempted.

1.5 Control rod fails to insert when given a SCRAM signal.

1.6 CRD high temperature alarm on PMIS.

1.7 Abnormal insert/withdrawal drive flows/ΔPs.

1.8 Abnormal cooling water flow/ΔPs which are unexplained or cannot be corrected.

1.9 Continuous blank 4-rod display position indication on Panel 9-5 and control rod cannot be determined to be in required position.

2. AUTOMATIC ACTIONS

2.1 None.

3. IMMEDIATE OPERATOR ACTIONS

3.1 None.

Scram Actions

Examination Outline Cross-Reference	Level	RO
	Tier#	2
	Group#	2
	K/A #	201002 A3.02
	Rating	2.8

201002 RMCS

A3. Ability to monitor automatic operations of the REACTOR MANUAL CONTROL SYSTEM including:
(CFR: 41.7 / 45.7)

A3.02 Rod movement sequence lights

Question 55

The plant is at 95% power.

- Control Rod 34-35 is being inserted using the Emergency Notch Override switch on Panel 9-5.
- As soon as Control Rod 34-35 reaches position 11, the Emergency Notch Override switch is released.

Which one of the following completes the statements below regarding how the Rod Settle light on Panel 9-5 and Control Rod 34-35 respond when the Emergency Notch Override switch is released?

The Rod Settle light ____(1)____ illuminate.

Control Rod 34-35 final position will indicate____(2)____.

A. (1) will
(2) 11

B. (1) will
(2) 12

C. (1) will **NOT**
(2) 11

D. (1) will **NOT**
(2) 12

Answer:
D. (1) will **not**
(2) 12

Explanation:
Emergency Rod Insert bypasses the rod sequence timer of RMCS, directly applying an insert signal, and only an insert signal, to the selected control rod. When this switch position is used, the CRD has no settle function, so the settle light will not illuminate. The settle function energizes the 120 valve for the control rod HCU and directs water from below piston to the exhaust header, allowing the control rod to settle downward (outward) to the next notch (even) position. However, even with120 valve not energized, the weight of the control rod will result in water being forced past seals in CRDM and settle into notched position, although settling will take longer. In the case given, the control rod is between notch positions 12 and 10 when the insert signal is removed; therefore it will settle at position 12. Position 11 was selected by an operator during validation.

Distracters:
Distracters that in Part 1 reflect the settle light will illuminate are plausible, because for normal notch insert with the sequence timer active the settle light will illuminate. This is also true for continuous rod withdrawal using the Emergency Notch Override hand switch. This includes answers A and B. This is wrong, because Emergency Rod Insert bypasses the rod sequence timer of RMCS, and the settle function does not occur.

Answers that in Part 2 reflect the control rod will settle at position 11 are plausible, because the stem states the control rod is at position 11 when the insert signal is removed. This includes answers A and C. This is wrong, because position 11 is not a notch position. Only even positions are notch positions, where the CRDM collet fingers latch.

Technical References:
Procedure 4.3 (Reactor Manual Control System And Rod Position Information System), Rev. 28

References to be provided to applicants during exam: NONE

Learning Objective:
COR0022002001060B Given a RMCS control manipulation, predict and explain the response of the following: Rod movement sequence lights

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X

Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b) 6	
Level of Difficulty:	4	
SRO Only Justification:	N/A	

4.3 At ROD SELECT MATRIX, select rod to be withdrawn by pressing applicable button and ensuring following:Ⓟ[1]

4.3.1 Only button on ROD SELECT MATRIX that backlights brightly is selected rod.

4.3.2 Only select light on FULL CORE DISPLAY that backlights is selected rod.

4.3.3 LPRM signals on the 16 LPRM readout windows are normal or as expected for power level and rod selected.

NOTE – Rod movement timer will complete cycle but will not reset until ROD MOVEMENT CONTROL switch is released.

4.4 While monitoring reactor power, momentarily place ROD MOVEMENT CONTROL switch to NOTCH OUT and ensure rod stops at next even notch position before ROD SETTLE light turns off.

4.4.1 IF rod fails to settle in ~ 10 seconds, THEN issue Condition Report for System Engineer to investigate cause and Fuels Engineer to track as potential channel bow which could impact thermal limit.Ⓢ[1]

4.5 IF control rod is withdrawn to notch Position 48, THEN perform coupling test per Section 9 or 10.

4.6 If required, repeat Steps 4.3 through 4.5 to select and withdraw next control rod.

4.7 Place ROD SELECT POWER switch to OFF to deselect control rods.

5. CONTINUOUS CONTROL ROD WITHDRAWAL

5.1 Check ROD SELECT POWER switch to ON. If REACTOR MODE switch in REFUEL, ROD SELECT POWER switch must be placed to OFF and then to ON to select a different rod.

5.2 Check white ROD OUT PERMIT light on.

5.3 At ROD SELECT MATRIX, select rod to be withdrawn by pressing applicable button and ensuring following:Ⓟ[1]

5.3.1 Only button on ROD SELECT MATRIX that backlights brightly is selected rod.

5.3.2 Only select light on FULL CORE DISPLAY that backlights is selected rod.

5.3.3 LPRM signals on the 16 LPRM readout windows are normal or as expected for power level and rod selected.

5.4 While monitoring reactor power, simultaneously place and hold ROD MOVEMENT CONTROL switch to OUT NOTCH and EMERGENCY NOTCH OVERRIDE switch to OVERRIDE.

5.5 IF control rod is withdrawn to notch Position 48, THEN perform coupling test per Section 9 or 10.

1.2.6.2 Activates the 4-rod display to provide the Operator with position data for the group considered.

1.2.6.3 Energizes Rod Select Relay K32 (Panel 9-28) which allows an enabling logic for the directional control valves.

1.2.7 A provision in the Reactor Manual Control System permits emergency control rod insertion. This is accomplished by positioning the EMERGENCY NOTCH OVERRIDE switch to EMER ROD IN. This bypasses the timer switch and its associated relays and applies power directly to the drive-in circuits until the EMERGENCY NOTCH OVERRIDE switch is returned to OFF. The switch cannot be used if an RWM insert block exists. Use of the switch bypasses the settle function of the directional control valves.

1.2.8 The direction of movement for the selected control rod is made by momentarily positioning the ROD MOVEMENT CONTROL switch (Panel 9-5) to either IN or OUT NOTCH. When this switch is momentarily moved to IN, the PLC (Panel 9-28) will automatically program single-notch drive-in and settle movement. When the ROD MOVEMENT CONTROL switch is moved to OUT NOTCH, the PLC will automatically program unlatch, single-notch drive out, and settle CRD movement. Following either of these sequences the PLC will automatically reset in preparation for another command. The PLC and associated rod manual control relays also provide rod-selected and rod-driving-signal data to the Rod Position Information System (RPIS) (Panel 9-27).

1.2.9 Continuous drive-in movement is obtained by holding the ROD MOVEMENT CONTROL switch in the IN position. The control rod will drive in until the switch is released and returns to the OFF position, after which the PLC will automatically program the CRD settle function, then reset. To move a control rod in the continuous-out mode, two distinct operations must take place; the EMERGENCY NOTCH OVERRIDE switch must be positioned to OVER RIDE at the same time (simultaneously) the ROD MOVEMENT CONTROL switch is positioned and held on OUT NOTCH. The selected control rod will drive out until either switch is released and spring returned to the OFF position, following which the PLC will automatically program the CRD settle function, then reset.

1.2.10 The solid-state RPIS (Panel 9-27) receives rod information from the 137 CRDs. The system processes this information and performs required logic functions and delivers it through output buffers to various Control Room displays and annunciators and to the RWM and PMIS.

<table>
<tr><td>Examination Outline Cross-Reference</td><td>Level</td><td>RO</td></tr>
<tr><td rowspan="5">Revised question to determine if RWM light is lit or extinguished and modified second part to “restore the RWM to service.
Added RWM aide disclaimer.</td><td>Tier#</td><td>2</td></tr>
<tr><td>Group#</td><td>2</td></tr>
<tr><td>K/A #</td><td>201006 A4.02</td></tr>
<tr><td>Rating</td><td>2.9</td></tr>
<tr><td></td><td></td></tr>
<tr><td colspan="3">201006 RWM

A4. Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8)

A4.02 Pushbutton indicating switches: P-Spec(Not-BWR6)</td></tr>
</table>

Examination Outline Cross-Reference	**Level**	RO
Revised question to determine if RWM light is lit or extinguished and modified second part to “restore the RWM to service. Added RWM aide disclaimer.	**Tier#**	2
	Group#	2
	K/A #	201006 A4.02
	Rating	2.9

201006 RWM

A4. Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8)

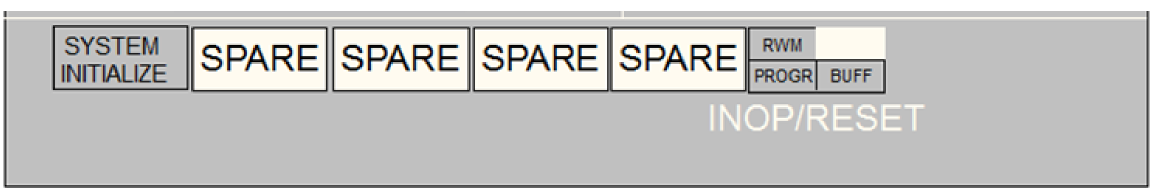
A4.02 Pushbutton indicating switches: P-Spec(Not-BWR6)

Question 56

The plant is operating at rated power.

The RWM is manually bypassed with the keylock switch for maintenance and then returned to NORMAL.

Which one of the following completes the statement below regarding the current RWM indications and restoration steps performed at Panel 9-5 IAW Procedure 4.2 (Rod Worth Minimizer)?



(Panel 9-5 shown as a physical layout aid ONLY – not representative of current plant conditions)

The RWM light is currently ____(1)____ due to current plant conditions.

Depressing the INOP/RESET push button will restore the RWM to service only if the SYSTEM INITIALIZE push button is depressed ____(2)____.

A. (1) lit
   (2) first

B. (1) lit
   (2) at the same time

C. (1) extinguished
   (2) first

D. (1) extinguished
(2) at the same time

Answer:
A. (1) lit (2) first
Explanation: The INOP/RESET pushbutton will indicate RWM alarm due to the RWM output buffer is not in an operating status and that RWM is/was OFF-LINE or manual bypassed with the keylock switch. Since the RWM was manually bypassed and returned to NORMAL – the RWM light is lit. To return the RWM to service requires performing procedure 4.2, Section 7.3 (Remove RWM from manual bypass) which requires pressing the SYSTEM INITIALIZE button followed by pressing the INOP/RESET button. The RWM requires initializing prior to resetting. Pushing both INOP/RESET & SYSTEM INITIALIZE button at the same time does not extinguish the RWM light and is not authorized IAW procedure 4.2.
Distracters: B. This answer is incorrect due to depressing the INOP/RESET at the same time as the SYSTEM INITIALIZE button does not extinguish the RWM light. This choice is plausible due the common misconception that the RWM would not need to be initialized prior to resetting the RWM light. The candidate that confuses the order in which the RWM is restored from being bypassed would select this answer. C. This answer is incorrect due to RWM light being lit under the given conditions. This choice is plausible due the confusing the cause of the other indicating lights (BUFF & PROGR) to illuminate. The candidate that confuses the RWM indicating lights and correctly identifies the order in which the RWM is restored from being bypassed would select this answer. D. This answer is incorrect due to RWM light being lit under the given conditions and depressing the INOP/RESET at the same time as the SYSTEM INITIALIZE button does not extinguish the RWM light. This choice is plausible due the confusing the cause of the other indicating lights (BUFF & PROGR) to illuminate and the common misconception that the RWM would not need to be initialized prior to resetting the RWM light. The candidate that confuses RWM indicating lights and the order in which the RWM is restored from being bypassed would select this answer.
Technical References: Procedure 4.2 (Rod Worth Minimizer), Rev. 29
References to be provided to applicants during exam: NONE
Learning Objective:
COR002-26-02

7. Given a RWM component manipulation, predict and explain the changes in the following parameters: b. Status of rod blocks		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b) 7	
Level of Difficulty:	3	
SRO Only Justification:	N/A	

7.3 Remove RWM from manual bypass by performing following:

7.3.1 (Independent Verification) On Panel 9-5, place RWM BYPASS keylock switch to NORMAL.

Performed By: ________

Verified By: ________

7.3.2 On Panel 9-5, check MANUAL BYPASS light turns off.

7.3.3 IF operating below the LPAP, THEN check Annunciator 9-5-1/A-5 alarms.

7.3.4 On Panel 9-5, press SYSTEM INITIALIZE button.

7.3.5 Check Annunciator 9-5-1/A-5 is clear.

7.3.6 On Panel 9-5, press INOP/RESET button.

7.3.7 On Panel 9-5, check RWM quadrant of INOP/RESET light turns off.

7.3.8 At an IDT, demand RWM and check RWM MODE display indicates mode based on plant conditions.

8. CORRECTION OF WITHDRAW ERRORS

8.1 IF Rod Test function is active, deselect selected control rod THEN select and insert withdrawn control rod.

8.2 IF RWM is in normal operation, THEN select rod causing withdraw error and insert it to correct position.

ATTACHMENT 1 INFORMATION SHEET

1.3.6 INOP/RESET pushbutton will indicate following alarm conditions when associated light turns on:

1.3.6.1 RWM - This light indicates that RWM output buffer is not in an operating status and that RWM is/was OFF-LINE or manual bypassed with keylock switch.

1.3.6.2 PROGR - This light indicates that either RPIS or RWM Program is not operating.

1.3.6.3 BUFF - This light indicates that one of three contacts associated with each of block/permissive functions does not agree with the other two contacts.

1.3.7 After condition causing alarm has been corrected, INOP/RESET pushbutton can be depressed to turn off alarm light(s). While pushbutton is depressed, the three active lights associated with pushbutton will turn on.

1.3.8 RWM RESPONSE TO FAILURES

1.3.8.1 The RPIS Data-Acquisition System does not always "fail safe". Depending on the specific component or power supply failure, RWM may fail to enforce the control rod sequence or may cause control rod blocks even if the sequence is met.

1.3.8.2 IF the RWM computer program aborts, THEN an exit handler will toggle the state of the outputs so insert and withdrawal blocks will exist. This will occur regardless of reactor power level. These blocks can be bypassed by use of the Manual Bypass keylock switch if necessary.

1.3.8.3 IF power is lost to the PMIS computer, THEN RWM insert and withdrawal block status will remain in the last state demanded by the RWM Program, irrespective of the actual control rod pattern or reactor power level.

1.3.8.4 IF power is lost to the RVLC platform in Panel 9-18, THEN the total steam flow and feed flow inputs to RWM will fail to LPSP condition and rod blocks will be enforced.

2. INTERLOCKS AND SETPOINTS

2.1 RWM enforces adherence to operating sequence by applying control rod withdraw and insert blocks when reactor power is below LPSP (20% steam or feed flow) (Tech Spec Allowable Value ≤ 9.85% RTP).

2.2 RWM auto bypasses when reactor power is above LPAP (35% steam flow).

Examination Outline Cross-Reference	Level	RO
	Tier#	2
	Group#	2
	K/A #	202001 G2.4.30
	Rating	2.7

202001 Recirculation

G2.4.30 Knowledge of events related to system operation/status that must be reported to internal organizations or external agencies, such as the State, the NRC, or the transmission system operator. (CFR: 41.10 / 43.5 / 45.11)

Question 57

Jet Pump #7 ΔP is NOT within the established pattern curve limits while operating at rated power.

Who is required to be IMMEDIATELY notified IAW 6.LOG.601 (Daily Surveillance Log - MODES 1, 2, and 3)?

A. GMPO

B. NRC Resident

C. Reactor Engineering

D. Operations Manager

Answer:
C. Reactor Engineering

Explanation:
IAW 6.LOG.601 (Daily Surveillance Log - MODES 1, 2, And 3) - IF any Jet Pump ΔP vs. established pattern is not within curve limits, THEN immediately notify Reactor Engineering.

Distracters:
A. This answer is incorrect due to 6.LOG.601 requiring immediate notification of Reactor Engineering. This choice is plausible due Operations Instruction #12 (Notification Guideline) providing guidance to notify the GMPO as soon as

reasonably possible. The candidate that does not know the 6.LOG.601 Jet Pump Operability notification requirement would select this answer.

B. This answer is incorrect due to 6.LOG.601 requiring immediate notification of Reactor Engineering. This choice is plausible due Operations Instruction #12 (Notification Guideline) providing guidance to notify the NRC Resident as soon as reasonably possible. The candidate that does not know the 6.LOG.601 Jet Pump Operability notification requirement would select this answer.

D. This answer is incorrect due to 6.LOG.601 requiring immediate notification of Reactor Engineering. This choice is plausible due Operations Instruction #12 (Notification Guideline) providing guidance to notify the Operations Manager as soon as reasonably possible. The candidate that does not know the 6.LOG.601 Jet Pump Operability notification requirement would select this answer.

Technical References:
6.LOG.601 (Daily Surveillance Log - MODES 1, 2, And 3), Rev. 117
Operations Instruction #12 (Notification Guideline), Rev. 17

References to be provided to applicants during exam: NONE

Learning Objective:
INT00705050010300 Given a set of plant conditions that constitutes non-compliance with a Chapter 3.4 LCO, determine the ACTIONS that are required.

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b) 10	
Level of Difficulty:	3	
SRO Only Justification:	N/A	

ATTACHMENT 13 JET PUMP OPERABILITY

ATTACHMENT 13 JET PUMP OPERABILITY

NOTE – If in single loop operation, mark idle loop N/A.

	ITEMS	LOOP A	LOOP B	JET PUMP ΔP (%) JP #	LOOP B	JP #	LOOP A
A	Core Flow (10^6 lb/hr) NBI-FRDPR-95			1		11	
B	RR Pump Flow (10^3 gpm) RR-FR-163			2		12	
C	RRMG Set Speed (%) RRFC-SI-1A/B			3		13	
D	JP Flow (10^6 lb/hr) NBI-FI-92A/B			4		14	
				5		15	
				6		16	
				7		17	
				8		18	
				9		19	
				10		20	
				LOOP B Avg		LOOP A Avg	

	CHECKS		SAT	UNSAT	OPERABILITY LIMIT	APPLICABLE MODE	ATT. 22 NOTE
1	Item B and C values within curve limits [a]	Loop A: ✓			SAT	1 [b], 2 [b]	50
		Loop B: ✓					
2	Item C and D values within curve limits [a]	Loop A: ✓					
		Loop B: ✓					
3	Jet Pump ΔP differs by ≤ 20% from established patterns [a, c]	Loop A: ✓					
		Loop B: ✓					
	Checks 1 and 2 SAT or Check 3 SAT	Loop A: ✓			SAT		
		Loop B: ✓					

[a] Refer to Jet Pump Operability Curves maintained in Control Room.

[b] Required to be performed within 24 hours after > 25% RTP and required to be performed within 4 hours after associated recirculation loop is in service.

[c] IF any Jet Pump ΔP vs. established pattern is not within curve limits, THEN immediately notify Reactor Engineering.

OPERATIONS INSTRUCTION #12 NOTIFICATION GUIDELINES	CLASS: INFORMATION USE EFFECTIVE: 05/05/14

NOTIFICATION MATRIX

Condition Requiring Notification (Refer to OI Step for follow-up details)	Site VP/ GMPO	Operations Manager	AOM-SHIFT	Shift Outage Manager Step 7.3	Licensing Manager Step 7.5	NRC Resident /QA Mgr	DNSA DOE (Note 4)
Abnormal, Emergency, HAZMAT Response, or EOPs (6.4.1)® [1]	Note 1	X	X			X	
Significant plant schedule deviations. (6.4.2)	Note 1	X	X				
CRs that fit or are in the CAT A category.(6.4.3)	Note 1	X	X				
Operational activities which warrant heightened awareness or unplanned entry into Orange/Red Risk window. (6.4.4 or 6.4.5)	Note 1	X	X			X	
Unplanned LCO entry of ≤7 days, any violation of CNS T.S. or Safety Limit. (7.9.1) **See Note 3**	X	X	X	X	X	X	X
Condition which results in an unplanned power reduction, or limits plant power capability (7.9.2)	X	X	X	X		X	
Any personnel injury and/or entry into 5.7.24 (7.9.3)	X	X	X	X		X	
Significant Radiological Event (7.9.4) -Spill - Contamination -Unmonitored release - Exposure -Uncontrolled Rad material	X	X	X	X		X	X
Unplanned loss of shutdown cooling or forced recirculation in RPV during plant shutdown. (7.9.5)	X	X	X	X	X	X	X
Mispositioned fuel bundles, unauthorized or improper core alterations, or potential for damaged fuel. (7.9.6)	X	X	X	X		X	
NRC notification per Procedure 2.0.5, except for ENS failure or Notification of any offsite agency. (7.9.7)	X	X	X	X	X	X	X
Significant road conditions affecting Emergency Plan emergency routes, (7.9.8) **See Note 2**	X	X	X	X		X	
Reactivity mis-management events (7.9.9)	X	X	X	X	X	X	X
Problem Identification Reports prompted by NRC. (7.9.10)	X	X	X	X	X		X
Damage to plant equipment caused by personnel error. (7.9.11)	X	X	X	X		X	
Events that may prompt media attention (7.9.12)	X	X	X	X	X	X	X
Personnel Events - badges pulled, fitness for duty, test for cause, etc. (7.9.13)	X	X	X	X		X	
Any events that exceed Table 3 Limits in 9.ENV.3 procedure, including Groundwater Tritium (7.9.14)	X	X	X	X	X	X	X
Significant Security Threats or Incidents (7.9.15)	X	X	X	X	X	X	X
Entry and exit from Transmission Loading Relief Level TLR-5b or higher. (7.9.16)	X	X	X	X			

Examination Outline Cross-Reference	Level	RO
	Tier#	2
	Group#	2
	K/A #	219000 K1.06
	Rating	3.2

219000 RHR/LPCI: Torus/Pool Cooling Mode

K1. Knowledge of the physical connections and/or cause-effect relationships between RHR/LPCI: TORUS/SUPPRESSION POOL COOLING MODE and the following: (CFR: 41.2 to 41.9 / 45.7 to 45.8)

K1.06 Keep fill system

Question 58

Which one of the following completes the statements below regarding the location pressure maintenance connects to the RHR system and the action required to prevent filling the Suppression Pool while operating the RHR Loop in Suppression Pool Cooling (SPC)?

Pressure Maintenance connects to the RHR Loop on the ____(1)____ side of the RHR pumps.

RHR system pressure is required to be maintained ____(2)____ than Condensate Transfer system pressure while operating in SPC.

A. (1) suction
(2) less

B. (1) suction
(2) greater

C. (1) discharge
(2) less

D. (1) discharge
(2) greater

Answer:
D. (1) discharge (2) greater

Explanation:
Pressure maintenance (Keepfill) is provided to the RHR system via the Condensate transfer system to the RHR pump discharge. During SPC operations, guidance is provided to prevent filling the SP by either

1. Isolating pressure maintenance during operation OR
2. Maintaining RHR system pressure greater than Condensate Transfer System pressure.

Distracters:

A. This answer is incorrect due to keepfill being provided on the pump discharge and RHR pressure is required to be maintained greater than Condensate Transfer pressure. This choice is plausible due to the common misconception that keepfill is provided on the pump suction for cavitation concerns and RHR system pressure being required to be maintained below the system relief valve lift setpoint (450 psig). The candidate that incorrectly identifies the keepfill tie to the suction side of the RHR pump and confuses system pressure control requirements would select this answer.

B. This answer is incorrect due to keepfill being provided on the pump discharge. This choice is plausible due to the common misconception that keepfill is provided on the pump suction for cavitation concerns. The candidate that incorrectly identifies the keepfill tie to the suction side of the RHR pump and correctly identifies system pressure control requirements would select this answer.

C. This answer is incorrect due to RHR pressure being required to be maintained greater than Condensate Transfer pressure. This choice is plausible due confusing RHR system pressure being required to be maintained below the system relief valve lift setpoint (450 psig). The candidate that correctly identifies the keepfill tie to the discharge side of the RHR pump and confuses system pressure control requirements would select this answer.

Technical References:
Procedure 2.2.69.3 (RHR Suppression Pool Cooling And Containment Spray) Rev. 46

References to be provided to applicants during exam: NONE

Learning Objective:
COR0022302001060B Given an RHR control manipulation, predict and explain changes in the following: RHR pump/system flow and pressure

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X

Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X

10CFR Part 55 Content:	55.41(b) 7
Level of Difficulty:	3
SRO Only Justification:	N/A

4.24 Perform one of the following:

4.24.1 Close CM-296, LOOP A INJECTION LINE PRESSURE MAINTENANCE SHUTOFF (R-881-NW Quad).

4.24.2 Maintain RHR Subsystem A pressure greater than Condensate Transfer System pressure to prevent filling Torus.

4.25 Throttle closed RHR-MO-66A, HX BYPASS VLV, to obtain desired cooling rate.

4.26 IF PCIS Group 6 lights lit on Panel 9-5, at VBD-M, THEN ensure REC-MO-711 or REC-MO-714, CRITICAL LOOP SUPPLY (associated with an in service REC HX), open.

4.27 IF additional cooling required, THEN ensure RHR Pump B or D running in Suppression Cooling Mode, if available.

4.28 IF additional cooling still required, THEN perform following:

4.28.1 Start non-running RHR Pump A or C.

NOTE – If Suppression Cooling Mode is being operated for accident condition, maintaining flow rate ≤ 11,550 gpm is not required.

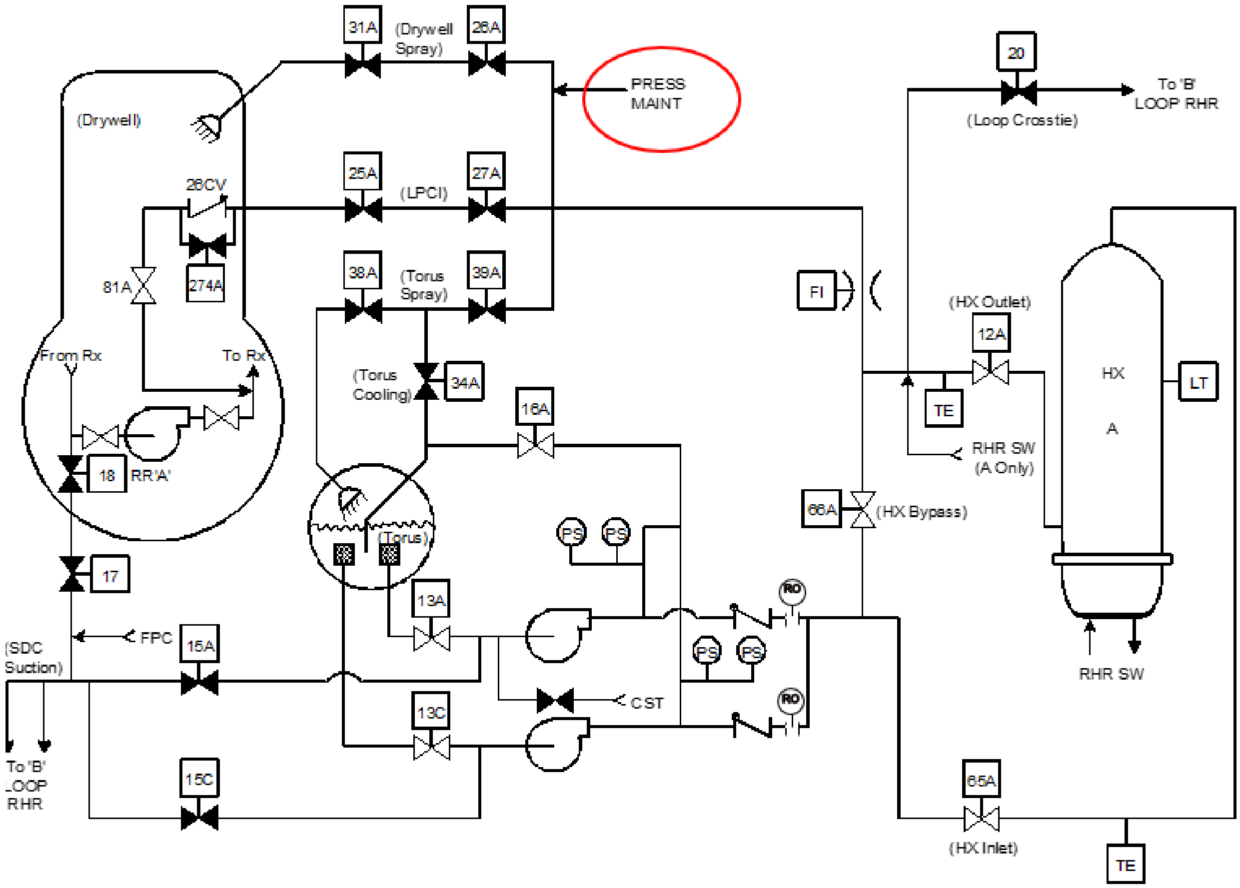
4.28.2 Throttle RHR-MO-34A to obtain a flow rate of ≤ 11,550 gpm. Ⓟ[3]

5. PLACING RHR SUBSYSTEM A SUPPRESSION POOL COOLING TO RHR SUBSYSTEM B IN SERVICE USING RHR-MO-20

NOTE 1 – No RHR loop will be available for LPCI Mode of operation.

NOTE 2 – Use only those RHR pumps not required to assure adequate core cooling by continuous operation.

5.1 EOP/SAGs flowcharts have directed to operate all available suppression pool cooling.



RHR LOOP 'A'

Figure 1, Rev. 9

COR002-23-02

\\cxa05157\co022302.fig\fig1.r09

Examination Outline Cross-Reference	Level	RO
Question is not evaluating NPSH or Vortex limits (No reference required) – no change	Tier#	2
	Group#	2
	K/A #	226001 K1.03
	Rating	3.5

226001 RHR/LPCI: CTMT Spray Mode

K1. Knowledge of the physical connections and/or cause-effect relationships between RHR/LPCI: CONTAINMENT SPRAY SYSTEM MODE and the following: (CFR: 41.2 to 41.9 / 45.7 to 45.8)

K1.03 LPCI/RHR pumps

Question 59

The following conditions exist during a LOCA:

- Suppression Pool Level is 13 feet.
- Suppression Pool Temperature is 145°F
- DW and Torus Sprays are in service with RHR Loop A.
- RHR Loop A flow is 16,000 gpm (within NPSH & Vortex Limits).

RHR Pump A trips.

Which one of the following completes the statement below regarding the impact on RHR Pump C under these conditions?

RHR Pump C ____(1)____ is prevented/minimized by an orifice installed on the ____(2)____.

A. (1) runout
(2) pump discharge

B. (1) runout
(2) containment spray line

C. (1) vortexing
(2) pump discharge

D. (1) vortexing
(2) containment spray line

Answer:

A. (1) runout
   (2) pump discharge

Explanation:
RHR pumps are initially being operated within the NPSH & Vortex limitations for 2 pumps. Upon loss of 1 pump, the remaining RHR pump flow rises but is limited by an orifice specifically installed on the pump discharge to prevent pump runout.

Distracters:
B. This answer is incorrect due to the orifice being installed on the pump discharge. This choice is plausible due to the loop flow indicator being installed on the combined HX outlet & bypass line upstream of the containment spray line. The candidate that correctly identifies the orifice installation being provided to protect the pump from runout and confuses the location would select this answer.
C. This answer is incorrect due to the orifice being installed to protect the pump from runout. This choice is plausible because vortexing is a function of pump flow rate, but based upon suppression pool level vortexing would not be the primary concern. The candidate that incorrectly identifies the orifice installation being provided to protect the pump from vortexing and correctly identifies the location would select this answer.
D. This answer is incorrect due to the orifice being installed to protect the pump from runout and located on the pump discharge. This choice is plausible because vortexing is a function of pump flow rate, though based upon suppression pool level vortexing would not be experienced, and because of the loop flow indicator being installed on the combined HX outlet & bypass line. The candidate that incorrectly identifies the orifice installation being provided to protect the pump from vortexing and confuses the location would select this answer.

Technical References:
USAR Chapter IV, Section 8
Procedure 2.2.69.3 (RHR Suppression Pool Cooling And Containment Spray) Rev. 46
EOP and SAG Graphs, Rev. 14

References to be provided to applicants during exam: None

Learning Objective:
COR0022302001030I Describe RHR System design feature(s) and/or interlocks which provide for the following: Pump runout protection

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b) 7	

Level of Difficulty:	3
SRO Only Justification:	N/A

2.4 RHR pump operation at minimum flow should be limited to < 15 minutes or pump damage may result.

2.5 Avoid RHR pump runout. Rated RHR pump flow rate is 7700 to 8400 gpm.

2.6 When reactor is in MODE 4 or 5, RPV level may rise due to leakage past RHR-MO-27A(B) which causes RHR pump discharge pressure to flex outboard disc of RHR-MO-25A(B) open. When outboard disc of RHR-MO-25A(B) flexes, which is an expected design condition for valve, a flow path to RPV is created through RHR-66(74).

2.7 RHR-MO-20 shall be closed when LPCI Mode required to be OPERABLE.

2.8 Following indicated closure of any limitorque motor operated throttle valve, switch should be held in close for an additional 5 seconds. This will ensure open-to-close stroke terminated by torque switch and not close limit switch.

2.9 Switch for any seal-in limitorque motor operated valve should not be held in CLOSE any longer than momentarily after green indicating light turns on. Placing or holding switch in close after valve is closed can cause hammering of valve.

2.10 When operating any limitorque motor operated valve, a minimum of 3 seconds should elapse after releasing switch before reversing its direction of travel.

2.11 Following manual engagement and operation of any Essential limitorque motor operated valve, it shall be operated electrically to ensure electrical geartrain is engaged. This operation may be a partial stroke.

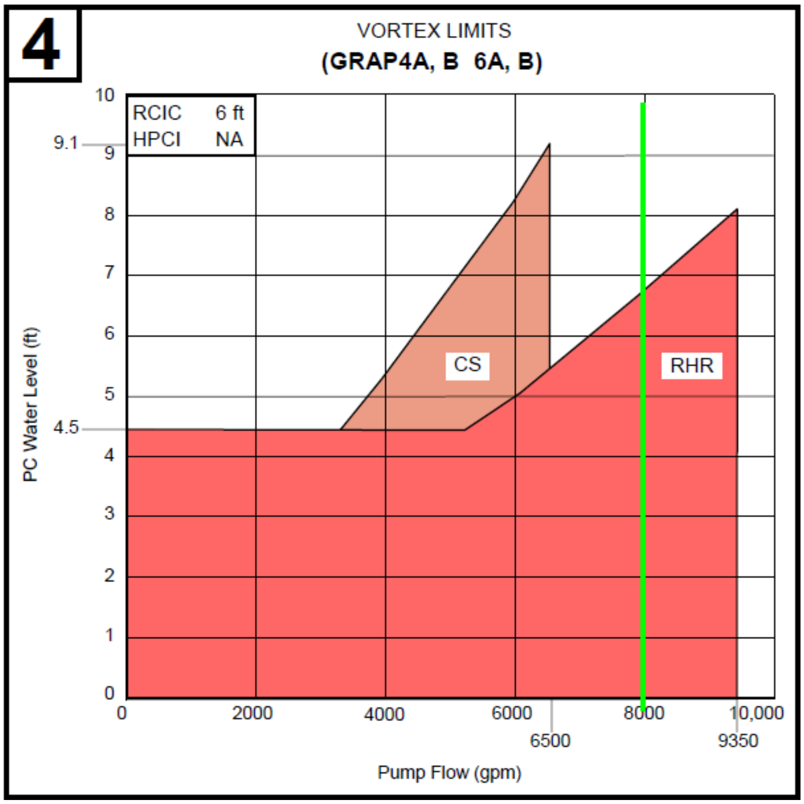
The major equipment of the RHR system consists of two heat exchangers and four main system pumps. The heat exchangers (tube side) are cooled by the Service Water System (USAR Section X-7). The system is equipped with piping, valves, controls and instrumentation, which are provided for proper system operation. A schematic diagram of the RHR system is shown in Burns and Roe Drawing 2040, Sheets 1 and 2.

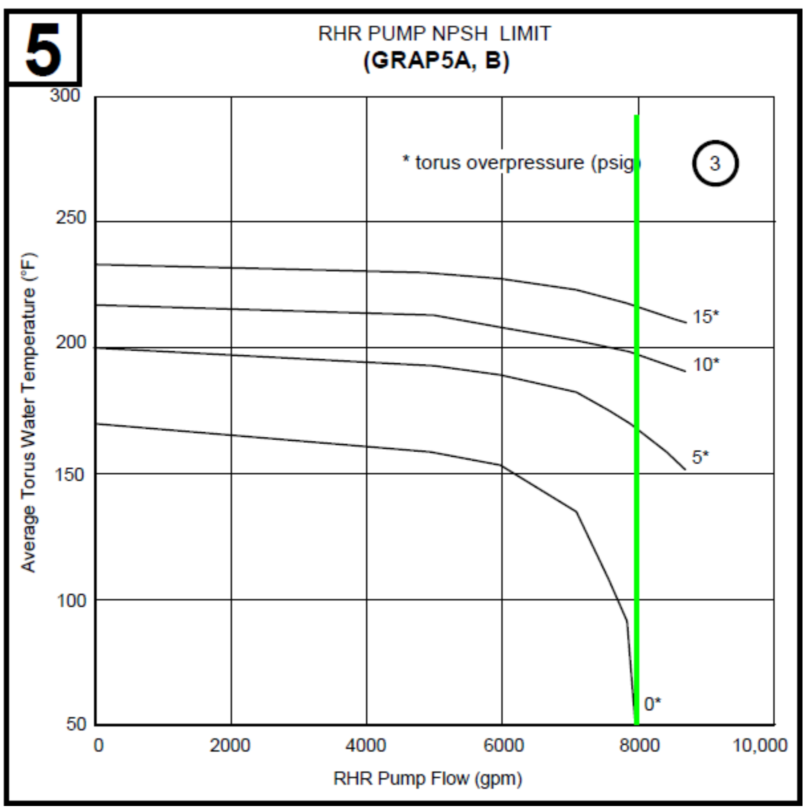
The main system pumps were sized on the basis of the flow required during the low pressure coolant injection (LPCI) mode of operation (as specified by the LOCA accident analysis in effect at the time of issuance of the original Operating License), which is the mode requiring the maximum system flow rate. (The original LOCA accident analysis has been superseded as a result of the current ECCS performance criteria in 10CFR50.46. Integrated ECCS availability resulting from the limiting single failure, together with assumed pump performance, is used to demonstrate that Peak Cladding Temperature (PCT) remains within the prescribed limit.) An orifice is installed in the discharge piping of each RHR pump, to prevent pump runout to cavitation and the resultant possible loss of long-term containment cooling.[35] The heat exchangers are sized on the basis of their required duty for the containment cooling function which is the mode requiring the maximum heat exchanger capacity. A summary of the design requirements of the main system pumps and the heat exchangers is presented in Table IV-8-1.

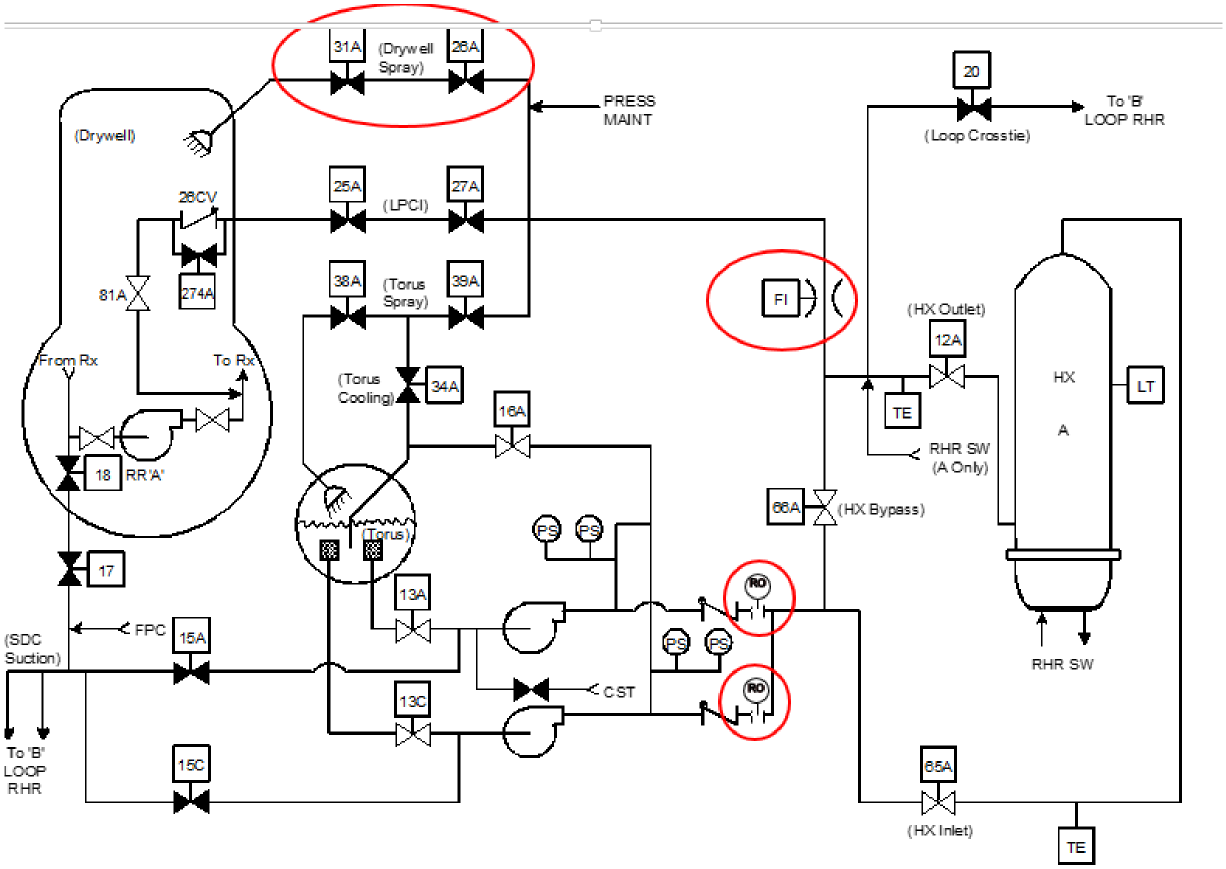
One loop, consisting of one heat exchanger, two main system pumps in parallel, and associated piping, is located on one side of the reactor building. The two pumps are located in a compartment which is provided with a duplex sump pump. The other heat exchanger, pumps, and piping, forming a second loop, are located on the other side of the reactor building to minimize the possibility of a single physical event causing the loss of the entire system. These two pumps are also located in a compartment with a duplex sump pump.

RHR system equipment is designed in accordance with Class I seismic design criteria (see USAR Chapter XII and Appendix C). The system is assumed to be filled with water for the seismic analysis.

The system piping and main system pumps are designed, constructed, and tested in accordance with the requirements of USAR Appendix A.


4
VORTEX LIMITS
(GRAP4A, B 6A, B)
RCIC 6 ft
HPCI NA
CS
RHR
PC Water Level (ft)
Pump Flow (gpm)
10
9.1
9
8
7
6
5
4.5
4
3
2
1
0
0
2000
4000
6000
6500
8000
10,000
9350


5
RHR PUMP NPSH LIMIT
(GRAP5A, B)
* torus overpressure (psig)
3
15*
10*
5*
0*
Average Torus Water Temperature (°F)
RHR Pump Flow (gpm)
300
250
200
150
100
50
0
2000
4000
6000
8000
10,000



RHR LOOP 'A'

Figure 1, Rev. 9

COR002-23-02

Examination Outline Cross-Reference	Level	RO
	Tier#	2
	Group#	2
	K/A #	239001 K2.01
	Rating	3.2

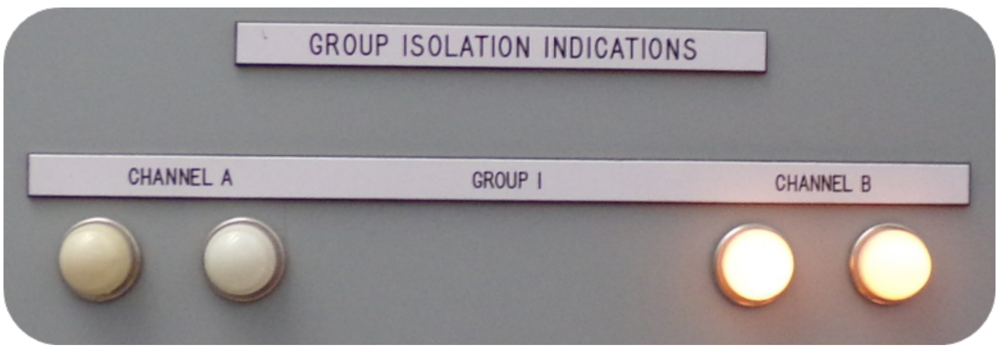
239001 Main and Reheat Steam

K2. Knowledge of electrical power supplies to the following: (CFR: 41.7)

K2.01 Main steam isolation valve solenoids

Question 60

The below indication for MSIV solenoid logic is present in the Control Room.



Which one of the following identifies the power supply that has been LOST to the MSIV solenoids?

A. Division 1 AC

B. Division 1 DC

C. Division 2 AC

D. Division 2 DC

Answer:
A. Division 1 AC

Explanation:
Panel 9-5 Group 1 isolation indicating lights provide indication of solenoid power availability from Div 1 & 2 AC {2 lights per AC supply – G1W1A & G1W2A (Div 1 AC Inboard) and G1W1B & G1W2B (Div 2 AC Outboard)}. A loss of Div 1 AC will cause both Channel A (Inboard) lights to extinguish.
Panel 9-41 has 4 indicating lights which provide Div 1/2 AC and DC solenoid power availability (1 light per power supply – Lights DS-175 – 178).

Each MSIV operator contains two AC solenoids and one DC solenoid. One of the AC solenoids is used for valve stroke testing at power and is called the Slow Closure Test Solenoid. The other two solenoids (one AC and one DC) determine the position of the MSIV by porting or venting the pneumatic source to or from the operator. Both of these solenoids deenergize to close the MSIV. The AC solenoids are powered from the Reactor Protection System and the DC solenoids are powered from the Station Battery System

Distracters:

B. The answer is incorrect due to both lights being powered from Div 1 AC. This choice is plausible due to the common misconception of each indicating light providing power indication of each solenoid (Div 1 & 2 AC and DC). The candidate that confuses Panel 9-41 with Panel 9-5 light indications would select this answer.
C. The answer is incorrect due to both lights being powered from Div 1 AC. This choice is plausible due to the common misconception of each indicating light providing power indication of each solenoid (Div 1 & 2 AC and DC). The candidate that confuses Panel 9-41 with Panel 9-5 light indications would select this answer.
D. The answer is incorrect due to both lights being powered from Div 1 AC. This choice is plausible due to the common misconception of each indicating light providing power indication of each solenoid (Div 1 & 2 AC and DC). The candidate that confuses Panel 9-41 with Panel 9-5 light indications would select this answer.

Technical References:
GE Drawing 791E266, Sheets 4, 7, 9, & 10

References to be provided to applicants during exam: NONE

Learning Objective:
COR0021402001020A State the electrical power supply to the following Main Steam components: Main steam isolation valve solenoids

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b) 7	

Examination Outline Cross-Reference	Level	RO
	Tier#	2
	Group#	2
	K/A #	241000.K3.01
	Rating	4.1

241000 Reactor/Turbine Pressure Regulating System

K3. Knowledge of the effect that a loss or malfunction of the REACTOR/TURBINE PRESSURE REGULATING SYSTEM will have on following: (CFR: 41.7 / 45.4)

K3.01 Reactor power

Question 61

The plant is operating at 80% with the following equipment tagged out for maintenance:

- DEH pump A
- TEC pump C

The following annunciator is received:

4160V BUS 1B BKR SS1B TRIP	PANEL/WINDOW: C-3/F-7

Which one of the following identifies the impact of this annunciator under the current plant conditions?

A. The reactor automatically stabilizes at a lower power level due to the reduced Feedwater flow.

B. Reactor power is required to be reduced due to degraded Main Condenser vacuum.

C. A manual reactor Scram is required due to the loss of Turbine Equipment Cooling.

D. The reactor will automatically Scram due to a Main Turbine trip.

Answer:
D. The reactor will automatically Scram due to a Main Turbine trip.

Explanation:
The loss of 480V B results in the loss of MCC-F and the loss of the only remaining DEH pump. A turbine trip at this power level results in an automatic reactor Scram.

Changed answer order (Long to short) due to too many “A” answers in a row.

Distracters:

A. The answer is incorrect due to the reactor automatically Scramming due to loss of DEH causing a MT trip. This choice is plausible due to a Condensate & Condensate Booster pump being powered from Bus 1B. The candidate that confuses which power supply is lost would select this answer.
B. The answer is incorrect due to the reactor automatically Scramming due to loss of DEH causing a MT trip. This choice is plausible due to 2 Circulating pumps being powered from Bus 1B. The candidate that confuses which power supply is lost would select this answer.
C. The answer is incorrect due to the reactor automatically Scramming due to loss of DEH causing a MT trip. This choice is plausible due to TEC pump 1C losing power but already tagged out for maintenance. The candidate that confuses TEC pump power supplies would select this answer.

Technical References:
Procedure 2.3_C-3 (Panel C - Annunciator C-3), Rev. 45
Procedure 5.3AC480 (480 VAC BUS FAILURE), Rev.43
Procedure 2.4TEC (TEC Abnormal), Rev.25
Procedure 2.4VAC (Loss Of Condenser Vacuum), Rev.25

References to be provided to applicants during exam: NONE

Learning Objective:
INT0320131S0S0100 Given plant condition(s), determine from memory the appropriate Abnormal/Emergency Procedure(s) to be utilized to mitigate the event(s).
COR0012502001020A State the electrical power supplies to the following Turbine HP Fluid components: DEH Pumps

Question Source:	Bank # 10579	
(note changes; attach parent)	Modified Bank #	
	New	
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b) 7	
Level of Difficulty:	3	

SRO Only Justification:	N/A

QUESTION: 1 10579 (point(s))

The plant is operating at 80% power in January.

- "A" DEH pump is deenergized and tagged out for repairs.
- "C" TEC pump is deenergized and tagged out for repairs.
- Annunciator C-3/F-7, 4160V BUS 1B BKR SS1B TRIP then alarms.

Which, from the following list, applying only these given conditions, will cause a reactor scram?

a. Feedwater

b. Service Water

c. Turbine High Pressure Fluid

d. Turbine Equipment Cooling

ANSWER: 1 10579

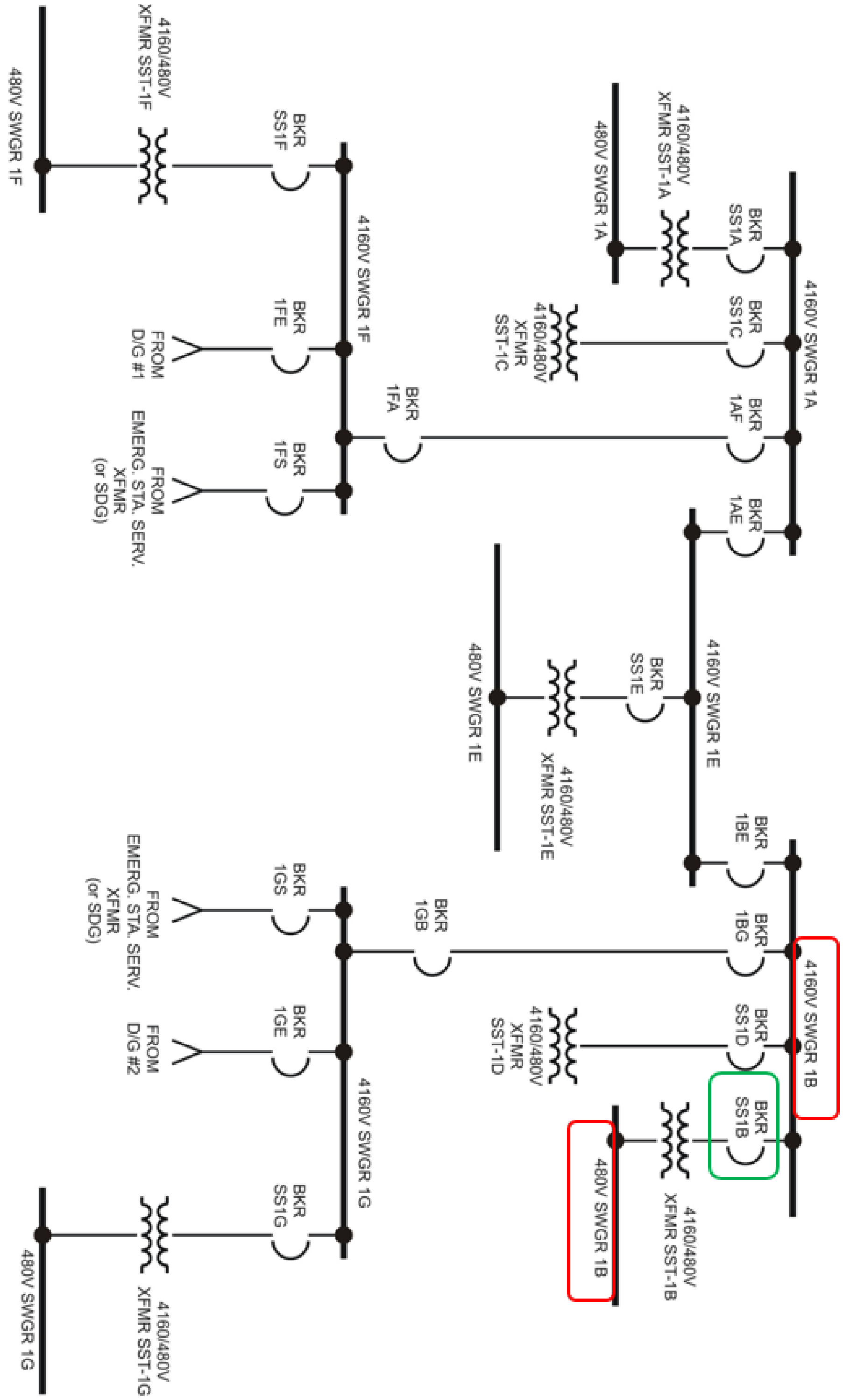
c. Turbine High Pressure Fluid

PROCEDURE 2.3_C-3	REVISION 45	PAGE 112 OF 137

4160V BUS 1B BKR SS1B TRIP	PANEL/WINDOW: C-3/F-7

1. OPERATOR OBSERVATION AND ACTION

1.1 Enter Procedure 5.3AC480 for 480 Bus 1B.



480V DISTRIBUTION SYSTEM
Figure 5, Rev. 10
COR001-01

4160V BUS 1B UNDERVOLTAGE	PANEL/WINDOW: C-3/A-7

1. AUTOMATIC ACTIONS

1.1 Breaker 1BS closes if open.

2. OPERATOR OBSERVATION AND ACTION

2.1 IF 4160V Bus 1B de-energizes <u>and</u> is <u>not</u> restored by fast transfer to SSST, THEN perform following:

2.1.1 SCRAM and enter Procedure 2.1.5.

2.1.2 Verify 480V Bus 1C has picked up Switchyard loads by observing ammeter on Panel C.

2.1.3 Momentarily place following switches to TRIP:

2.1.3.1 CIRC WATER PUMP C.

2.1.3.2 CIRC WATER PUMP D.

2.1.3.3 CONDENSATE PUMP B.

2.1.3.4 CONDENSATE BOOSTER PUMP B.

2.1.4 Enter Procedure 5.3AC480.

Scram Actions

Scram Actions

ATTACHMENT 4 480V BUS 1B - MAJOR LOADS

LOADS	NOTES
480V BUS 1B	
Warehouse	
SAC 1C	
SWBD MSA Machine Shop	
Mechanical Vacuum Pump 1B	
MCC-F	Causes loss of MCC-J and MCC-H.
TEC Pump 1C	
MCC-G	
MCC-F	
MCC-J	
MCC-H	
Lighting Panel LPCB2	Causes power loss to: SMA-3 Recorder, TS-3A Seismic Trigger, and Control Panel. DC Switchgear Room 1B Supply Fan. Air Compressor Cooling Water Selector Panel. Annunciator Cabinets 1, 5, 6, and 7 fans.
Turbine Building Starter Rack PP-TG-SB	
Heating Boiler Feed Pump A	
Lube Oil Conditioner	
AR-MO-164 Gland Exhaust B Discharge	
Turbine Building Supply Fans SF-T-1B-B and SF-T-1A-B	
EH Pump B	TGF System will be unable to support BPV operation shortly after scram.
RF-T-1B TG Area Recirc Fan B	
CP-BR-1A2 Auxiliary Cond Pump A2	
Gland Steam Exhauster B	
CP-BR-1A1 Auxiliary Cond Pump A1	
RF-MO-33 RFP B Startup Valve Outlet	
RF-MO-34 RFP B Startup Valve Inlet	

4. SUBSEQUENT OPERATOR ACTIONS

4.1 Record current time and date. Time/Date: ________/__________

NOTE – Highest TEC heat loads: Main Generator Hydrogen Coolers, Main Lube Oil Coolers, Exciter Air Coolers, Bus Duct Heat Exchangers.

4.2 **IF at any time TEC PRESSURE cannot be restored and maintained above 55 psig and component temperatures rise and do not stabilize, THEN perform following:**

4.2.1 **SCRAM and enter Procedure 2.1.5.**

4.2.2 Trip Main Turbine.

4.2.3 Rapidly reduce reactor pressure to 500 to 600 psig using main turbine BPVs per Procedure 2.2.77.1.

4.3 **IF SW cooling to TEC lost and cannot be restored, THEN perform following:**

4.3.1 **SCRAM and enter Procedure 2.1.5.**

4.3.2 Trip Main Turbine.

4.3.3 Rapidly reduce reactor pressure to 500 to 600 psig using main turbine BPVs per Procedure 2.2.77.1.

Scram Actions

4.4 IF abnormal generator H_2 gas or stator temperatures occur due to TEC conditions, THEN concurrently enter Procedure 2.4GEN-H2.

ATTACHMENT 2	LOSS OF TWO TEC PUMPS

ATTACHMENT 2 LOSS OF TWO TEC PUMPS

1. LOSS OF TWO TEC PUMPS

1.1 Close TEC-MO-130, TEC HX BYPASS VALVE (VBD-M).

1.2 While maintaining SW System header pressure ≥ 38 psig on SW-PI-2715A, A HEADER PRESS (PANEL A), and/or SW-PI-2715B, B HEADER PRESS (PANEL B), slowly throttle SW outlet valve on in service HX.

1.2.1 SW-55, TEC HEAT EXCHANGER A OUTLET (T-882'-N).

1.2.2 SW-64, TEC HEAT EXCHANGER B OUTLET (T-882'-N).

1.3 Reduce power, as necessary, per Procedure 2.1.10 to maintain TEC temperature < 100°F.

1.4 IF 2nd TEC pump cannot be restored and power reduction does not maintain TEC temperature < 100°F, THEN exit this Attachment and enter Attachment 1, COMPLETE LOSS OF TEC OR SW COOLING TO TEC.

PROCEDURE 2.4TEC	REVISION 25	PAGE 6 OF 14

2.2.2 5 MIN DELAY LINE for 5 consecutive minutes.

Scram Act

3. IMMEDIATE OPERATOR ACTIONS

NOTE – Vacuum can be monitored on any DEH HMI, Group 2, CONDENSER PRESSURE TRIP GRAPH.

3.1 For lowering condenser vacuum:

3.1.1 Reduce power per Procedure 2.1.10 to maintain vacuum ≥ 23" Hg.

PROCEDURE 2.4VAC	REVISION 25	PAGE 1 OF 4

Examination Outline Cross-Reference	Level	RO
Comments incorporated.	Tier#	2
	Group#	2
	K/A #	256000 K4.03
	Rating	2.8

256000 Reactor Condensate

K4. Knowledge of REACTOR CONDENSATE SYSTEM design feature(s) and/or interlocks which provide for the following: (CFR: 41.7)

K4.03 Condensate and/or booster pump protection

Question 62

A plant startup is in progress with the following conditions:

- Condensate Pump A is running.
- Condensate Booster Pump C is running.
- Condensate Booster Pump Suction Pressure is 55 psig.

The Condensate Booster Pump B control switch is placed in START.

Which one of the following completes the statement below?

If Condensate Booster Pump B control switch is placed in START, the pump will.....

A. NOT start because less than two condensate pumps are running.

B. NOT start because the low suction pressure interlock has not been met.

C. start, but then, if suction pressure falls below 25 psig for AT LEAST 9 seconds, will trip.

D. start, but then, if suction pressure falls below 25 psig for AT LEAST 12 seconds, will trip

Answer:
A. NOT start because less than two condensate pumps are running.

Explanation:
The Condensate pumps are interlocked with the Condensate Booster Pumps to allow for booster pump startup. At least one Condensate pump must be operating in order to start the first Condensate booster pump and two Condensate pumps must be running in order to start the second booster pump.

Distracters:

B. The answer is incorrect due to the pump not starting due to less than two CPs running. This choice is plausible due to the CBPs having a suction interlock of > 50 psig and confusing first or second CBP start. The candidate that recognizes the CBP will not start but confuses the reason would select this answer.

C. The answer is incorrect due to the pump not starting due to less than two CPs running. This choice is plausible due to not recognizing pump start permissives NOT being met and confusing which CBP trips on low suction pressure for 9 seconds (CBP A). The candidate that confuses CBP start permissives and trips would select this answer.

D. The answer is incorrect due to the pump not starting due to less than two CPs running. This choice is plausible due to not recognizing pump start permissives NOT being met and recognizing CBP B trips on low suction pressure for 12 seconds. The candidate that confuses CBP start permissives and trips would select this answer.

Technical References:
Procedure 2.2.6 (Condensate System), Rev. 89

References to be provided to applicants during exam: NONE

Learning Objective:
COR0020202001060A Describe the Condensate and Feedwater design features and/or interlocks that provide for the following: Condensate and Booster Pump interlocks

Question Source:	Bank # 18313	
(note changes; attach parent)	Modified Bank #	
	New	
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b) 7	
Level of Difficulty:	3	
SRO Only Justification:	N/A	

QUESTION: 44 18313 (1 point(s))

Given the following:

A plant startup is in progress.
Condensate Booster Pump 1C is running.
Condensate Pump 1A is running, supplying approximately 55 psig suction pressure to the booster pumps.

Then, the Condensate Booster Pump 1B control switch is placed in START.

Which one of the following describes the Condensate Booster Pump 1B response?

Condensate Booster Pump 1B will.....

a. NOT start because less than two condensate pumps are running.

b. NOT start because the low suction pressure interlock has not been met.

c. start, but then, if suction pressure falls below 25 psig for 9 seconds, it will trip.

d. start, but then, if suction pressure falls below 25 psig for AT LEAST 12 seconds, it will trip.

ANSWER: 44 18313

a. is correct. The Condensate pumps are interlocked with the Condensate Booster Pumps to allow for booster pump startup. At least one Condensate pump must be operating in order to start the first Condensate booster pump and two Condensate pumps must be running in order to start the second booster pump.

2. INTERLOCKS AND SETPOINTS

2.1 The first condensate booster pump will start when its switch is placed to START if the following conditions are satisfied:

2.1.1 At least one condensate pump is running.

2.1.2 Pump suction pressure > 50 psig.

2.1.3 Pump oil pressure > 5 psig.

2.2 The second and third condensate booster pump starts when its switch on is placed to START if the following conditions are satisfied:

2.2.1 At least two condensate pumps are running.

2.2.2 Pump suction pressure > 50 psig.

2.2.3 Pump oil pressure > 5 psig.

2.3 A condensate booster pump will trip if any of the following conditions occur:

2.3.1 All condensate pumps are not running.

2.3.2 Pump oil pressure < 3 psig.

2.3.3 Condensate booster pump suction pressure < 25 psig after a time delay of 9 seconds for Pump A, 12 seconds for Pump B, and 15 seconds for Pump C.

Examination Outline Cross-Reference	Level	RO
	Tier#	2
	Group#	2
	K/A #	268000 K5.01
	Rating	2.7

268000 Radwaste

K5. Knowledge of the operational implications of the following concepts as they apply to RADWASTE : (CFR: 41.5 / 45.3)

K5.01 Units of radiation, dose and dose rate

Question 63

Which one of the following completes the statement below regarding the operational implications of backwashing a Condensate Filter Demineralizer?

Normal RWP access to the RW ____(1)____ will not be available because the posting is required to be changed to a HRA due to potential dose rates above ____(2)____ mRem/hr, at a minimum, during the backwash.

A. (1) 903' F/D Valve Room
(2) 100

B. (1) 903' F/D Valve Room
(2) 1000

C. (1) 934' F/D Septum Area
(2) 100

D. (1) 934' F/D Septum Area
(2) 1000

Answer:
A. (1) 903' F/D Valve Room
(2) 100

Explanation:
Requires knowledge of resin transfer flowpath and plant areas impacted by this transfer due to rising dose rates. Water/resin backwashed from condensate filters travels from the Filter Demineralizers located on RW 934' in the associated filter pit

through the Filter Demineralizer Valve Room "U-uniform" piping to the RW 877' backwash tank room and into the backwash tank (this room remains posted HRA CA SWP SRP). The water/resin in the Backwash Tank is then transferred through the Lab Drain Tank Room via the Condensate Backwash Transfer Pump and discharge line in the overhead to the Condensate Phase Separator Tank Room and into the Condensate Phase Separator Tanks. Any water containing condensate resins will have the potential to cause increased dose rates in their surrounding areas. A Special Work Permit (SWP) is required for HRA access. The 934' F/D Septum Area is normally posted as a Radiation Area due to shielding provided by lead on the F/Ds. RW 903' Filter Demineralizer Valve Room & RW 877' Lab Drain Tank Room Door are required to be posted HRA.

Distracters:

B. The answer is incorrect due to dose rates rising above a minimum of 100 mRem/hr require posting of HRA. This choice is plausible if HRA & LHRA dose rates are confused. The candidate that correctly identifies the location and confuses the dose rate would select this answer.

C. The answer is incorrect due to 903' F/D Valve Room required to be posted HRA. This choice is plausible if F/D Septum Area dose rates are confused. The candidate that confuses the area required to be posted HRA and correctly identifies the minimum dose rate for a HRA would select this answer.

D. The answer is incorrect due to dose rates rising above a minimum of 100 mRem/hr require posting of HRA and 903' F/D Valve Room required to be posted HRA. This choice is plausible if F/D Septum Area dose rates are confused and if HRA & LHRA dose rates are confused. The candidate that confuses the location to be posted HRA and confuses the dose rate would select this answer.

Technical References:

Procedure 2.2.5 (Condensate Filter Demineralizer System), Rev. 61

RP-74c (Radiological Controls for Condensate Filter Backwash System Operation)

References to be provided to applicants during exam: NONE

Learning Objective:

COR0012002001040I Describe the interrelationship between the RWCU system and the following: Radwaste

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	

10CFR Part 55 Content:	55.41(b) 5
Level of Difficulty:	2

SRO Only Justification:	N/A

12. F/D MANUAL BACKWASH

WARNING – During backwash, resin transfer from F/D to Condensate Backwash Tank could raise radiation levels in the Condensate F/D Hold Pump Room.

12.1 IF not previously performed in Section 8 or 9, THEN notify Radiation Protection of intent to backwash Condensate F/D (RP signature required on Attachment 1).Ⓟ[1]

12.1.1 Contact Radiation Protection to establish radiological controls in F/D Hold Pump and Lab Drain Tank Rooms per RP-74C Data Form prior to backwashing condensate filters.

12.2 Ensure selected Waste Sample Tank level is > 60% and has been sampled and meets Chemistry requirements per Procedure 2.5.1.6 for transfer to a Condensate Storage Tank.

NOTE – Common and individual F/D Panels are located on Condensate F/D Panel in Radwaste Control Room.

12.3 Ensure F/D is removed from service per Section 6 or has been automatically backwashed per Section 8 or 9.

12.4 Check COND BACKWASH TANK level (Condensate Solid Waste Panel) and transfer to a Condensate Phase Separator Tank per Procedure 2.5.1.1 if level is > 5% for first backwash or > 30% if performing a second backwash.

12.5 Record before cleaning COND BACKWASH TANK LEVEL (Condensate Solid Waste Panel) in CLEANING DATA section of Attachment 1.

12.6 Close OUTLET VALVE "S" (individual F/D Panel).

12.7 AFTER OUTLET VALVE "S" has been closed for ~ 1 minute, THEN close INLET VALVE "A" (individual F/D Panel).

12.8 AFTER INLET VALVE "A" has been closed for ~ 1 minute, THEN place MODE SELECTOR switch (individual F/D Panel) to HAND or MANUAL.

NOTE – If currently performing a double backwash and first backwash was performed immediately prior to double backwash, Step 12.9 is N/A and proceed to Step 12.10.

12.9 Verify Backwash Air Blower discharge piping does not contain water by performing following:

12.9.1 Open CF-V-62, AIR BLOWER DISCHARGE VENT, until no water flows from hose into drain (RW-877 at Backwash Air Blower).

Radiological Controls for Condensate Filter Backwash System Operation

Note: Posting requirements are minimum requirements only. Radiological surveys may indicate other postings are necessary.

System flow path and affected areas:

Water/resin backwashed from condensate filters travels from the Filter Demineralizers located on RW 934' in the associated filter pit through the Filter Demineralizer Valve Room "U-uniform" piping to the RW 877' backwash tank room and into the backwash tank (this room remains posted HRA CA SWP SRP). The water/resin in the Backwash Tank is then transferred through the Lab Drain Tank Room via the Condensate Backwash Transfer Pump and discharge line in the overhead to the Condensate Phase Separator Tank Room and into the Condensate Phase Separator Tanks. Any water containing condensate resins will have the potential to cause increased dose rates in their surrounding areas.

Pre-Operation/Flushing/Draining Survey:

Note: RP Supervision approval required if pre-operation/flushing/draining survey is waived.

Performed By (initial)	N/A

Waived by: RP Supervision

Justification for Waiver:

Routine surveys are sufficient for use as pre-evolution surveys for the affected areas.

Postings, Surveys, and Controls:

Posting, Survey, or Control	Initial completed
Determine which filter will be backwashed and record here "_________"	
Post the RW 903' Filter Demineralizer Valve Room HRA SWP SRP prior to backwash.	

Post the RW 877' Lab Drain Tank Room Door HRA SWP SRP prior to backwash.	
Sign the applicable Procedure 2.2.5 Attachment 1 for the filter being backwashed. (usually located in the RW 903' control room on magnetic clip board)	
Document a log entry in the RP Shift Log to document posting changes and which filter is being backwashed.	
Following completion of the backwash, enter the Filter Demineralizer Valve Room and perform a dose rate survey of the "U" pipe ("U" is the letter designation for components on the pipe not the shape) for the filter system being used. Compare these dose rates to the routine survey of the area to determine whether dose rate changes are present.	
Following completion of the backwash, enter the Lab Drain Tank Room and perform a dose rate survey of the general area around the condensate backwash pump (note: discharge pipe runs through the overhead). Compare these dose rates to the routine survey of the area to determine whether dose rate changes are present.	
Once dose rates have been verified to have returned to routine survey conditions, down post the Filter Demineralizer Valve Room and Lab Drain Tank Room to pre-backwash conditions	
Document post evolution surveys with a log entry in the RP shift log.	

Comments:

RPT(s) performing actions (print/sign/initial/Date):

____________________/______________________/______/_________

____________________/______________________/______/_________

____________________/______________________/______/_________

RP Supervisor(print/sign/date):______________/_______________/_________

Examination Outline Cross-Reference	Level	RO
	Tier#	2
	Group#	2
	K/A #	272000 K6.03
	Rating	2.8

272000 Radiation Monitoring

K6 Knowledge of the effect that a loss or malfunction of the following will have on the RADIATION MONITORING SYSTEM : (CFR: 41.7 / 45.7)

K6.03 A.C. power

Question 64

What is one impact of de-energizing RPSPP1B on the Radiation Monitoring System?

A. RMP-RM-251D (MSL RAD MONITOR CH D) fails downscale.

B. RMP-RM-150A (OFFGAS RAD MONITOR CH A) fails downscale.

C. RMV-RM-40 (RX BLDG NORMAL RANGE KAMAN) fails downscale.

D. RMP-RM-20A (TURBINE BUILDING NORMAL RANGE MONITOR) fails downscale.

Answer:
A. RMP-RM-251D (MSL RAD MONITOR CH D) fails downscale.

Explanation:
Main Steam Line Rad Monitor channels B and D are powered from RPSPP1B.

Distracters:
B. The OG rad monitor is powered from RPSPP1A.
C. The RB Kaman is powered from CCP-1A/NBPP.
D. The TB Normal Range Monitor is powered from CCP-1A/NBPP.

Technical References:
Procedure 2.2.22 (Vital Instrument Power System), Rev. 71
Procedure 2.2.61A (Primary Coolant Leakage Detection System Component Checklist), Rev. 11

References to be provided to applicants during exam: NONE

Learning Objective:		
OPS Radiation Monitoring/COR001-18-01 6. State the electrical power supply to the following: a. Process Liquid Radiation Monitoring system b. Area Radiation Monitors c. Kamans d. Reactor Building Ventilation Exhaust monitors e. Control Room Ventilation monitors f. Main Steam Line Monitoring system g Off-Gas Radiation Monitoring system i. Drywell Ventilation monitors		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b) 7	
Level of Difficulty:	3	
SRO Only Justification:	N/A	

10.6.5.5 Ensure RW-AO-83, DISCH VLV, is closed.

10.6.5.6 Ensure RW-AO-82, DISCH ROOT VLV, is closed.

10.6.5.7 Ensure RW-AO-95, DISCH VLV, is closed.

10.6.5.8 Ensure RW-AO-94, DISCH ROOT VLV, is closed.

10.6.5.9 Ensure RHR-SSV-95, RHR SAMPLE VALVE, is closed.

10.6.5.10 Ensure RHR-SSV-96, RHR SAMPLE VALVE, is closed.

10.6.5.11 Ensure RHR-SSV-80, RHR SAMPLE VALVE, is closed.

10.6.5.12 Ensure RHR-SSV-81, RHR SAMPLE VALVE, is closed.

10.6.5.13 Ensure PC-MO-1311, DRYWELL N_2 SUPPLY ISOLATION VLV, is closed.

10.6.5.14 Ensure PC-MO-1302, TORUS N_2 SUPPLY ISOLATION VLV, is closed.

10.6.5.15 Ensure PC-MO-1308, TORUS VENT ISOLATION VLV, is closed.

10.6.5.16 Ensure PC-MO-1306, DRYWELL N_2 SUPPLY ISOLATION VLV, is closed.

10.6.5.17 Ensure PC-MO-1304, TORUS N_2 SUPPLY ISOLATION VLV, is closed.

10.6.6 At Panel 9-15, place RPS/PCIS BUS A POWER switch to OFF.

10.6.7 At RPSPP1A (Cable Spreading Room), open all closed breakers on panel.

10.6.8 Remove RPS MG Set A from service per Section 11.

10.6.9 Remove RPSPP1A alternate power supply from service per Section 12.

10.7 Remove RPSPP1B from service by performing following:

10.7.1 Power will be lost to the following equipment; ensure Shift Manager has determined all applicable Technical Specifications requirements for:

10.7.1.1 APRMs B, D, and F, and LPRMs associated with these APRMs.

10.7.1.2 LPRM Group B.

10.7.1.3 Recirculation Flow Unit B.

10.7.1.4 RMP-RM-150B, OFFGAS RAD MONITOR.

10.7.1.5 RMP-RM-251B, MSL RAD MONITOR CH B.

10.7.1.6 RMP-RM-251D, MSL RAD MONITOR CH D.

Lesson/BET Number:	COR001-18-01	Revision: 25

LO-05s,06c E. Power Supplies
LO-09a,b,c
SO-06c 1. KAMANS
SO-08a,b,c

KAMAN	POWER SUPPLY			
	RIC	SMIC AND MICROPROCESSOR	FLOW DETECTION	SAMPLE PUMPS
ERP	CCP-1B	LPGB-1	NBPP	PPGB-1
REACTOR BLDG	CCP-1A/NBPP	CPP-2	CPP	MCC-U
RE/ARW BLDG	CCP-1A/NBPP	LPRW-3	CPP	MCC-W
TURBINE BLDG	CCP-1A/NBPP	CCP-2B	CPP	MCC-DG2

LO-05p,6d,f,g
SO-02p;06d,f,g

2. RPSPP-1A and 1B: Main Steam Line monitors, Reactor Building Vent Exhaust Plenum Radiation monitors, and Air Ejector Off Gas Radiation monitor

LO-06a
SO-06a

3. Process Liquid Effluent monitors
   a. REC monitor - LPREMG
   b. SW monitor - NBPP
   c. Radwaste monitor - LPRW

LO-06b
SO-06b

4. Area Radiation Monitors - See procedure 4.8 for local locations; ARM's Panel 9-11 CPP-120 VAC

LO-6i

5. Drywell ventilation monitor (RMV-RM-4) - CPP2

6. MPF Ventilation Radiation Monitoring – PPMP1

SO-6d
LO-6d

7. 24 VDC, Aux Trip Units for Reactor Building Vent Exhaust Plenum Radiation monitors and trip units for REC & SW rad monitors.

XI. SUMMARY

A. Gaseous Monitor Setpoints

LO-02;03;11e

The first topic is a discussion of gaseous effluent per Offsite Dose Assessment Manual. Section D3.2.1 of the ODAM states that the dose rate beyond the SITE AND EXCLUSION AREA BOUNDARY due to radioactive noble gases shall not exceed 500 mrem/yr to the total body or 3000 mrem/yr to the skin. If the dose were exactly according to ODAM, anything over 500 mrem/yr to the skin would be contributed to beta (under normal conditions, CNS does not have any alpha contributing to dose).

10.6 Remove RPSPP1A from service by performing following:

10.6.1 Power will be lost to the following equipment. Ensure Shift Manager has determined all applicable Technical Specifications requirements for:

10.6.1.1 APRMs A, C, and E, and LPRMs associated with these APRMs.

10.6.1.2 LPRM Group A.

10.6.1.3 Recirculation flow Unit A.

10.6.1.4 RMP-RM-150A, OFFGAS RAD MONITOR.

10.6.1.5 RMP-RM-251A, MSL RAD MONITOR CH A.

10.6.1.6 RMP-RM-251C, MSL RAD MONITOR CH C.

10.6.1.7 RMP-RM-452A, RX BLDG VENT RAD MON CH A INDICATOR/TRIP UNIT.

10.6.1.8 RMP-RM-452C, RX BLDG VENT RAD MON CH C INDICATOR/TRIP UNIT.

Examination Outline Cross-Reference	Level	RO
Improved explanation to clarify interrelation of primary and secondary containment.	Tier#	2
	Group#	2
	K/A #	290001 A1.01
	Rating	3.1

290001 Secondary CTMT

A1. Ability to predict and/or monitor changes in parameters associated with operating the SECONDARY CONTAINMENT controls including: (CFR: 41.5 / 45.5)

A1.01 System lineups

Question 65

The following plant conditions exist:

- Reactor Building pressure = - 0.32 inches wg
- Torus Pressure = 0.03 psig
- Drywell Pressure = 1.4 psig

Which one of the following completes the statements below regarding MANUAL operation of Vacuum Breakers under the current plant conditions?
(Consider the effect of each action separately with air valved in to all AOVs.)

If AO-NRV-31 (TORUS TO DRYWELL VACUUM RELIEF) is opened at Panel VBD-J, Drywell to Torus DP will __(1)__.

If PC-AO-243 (TORUS VAC RELIEF VLV) is opened at Panel VBD-H, Torus to Reactor Building DP will __(2)__.

A. (1) lower
(2) lower

B. (1) lower
(2) remain constant

C. (1) remain constant
(2) lower

D. (1) remain constant
(2) remain constant

Answer:

B. (1) lower
   (2) remain constant

Explanation:

Containment vacuum breaker operation ultimately impacts Secondary Containment pressure under certain accident conditions to protect the primary containment.
If the DW is sprayed while in the unsafe region of the DWSIL curve, the rapid reduction in DW pressure along with the rapid reduction of torus pressure will cause air introduction to the primary containment and lower secondary containment (Rx Building) pressure through operation of all vacuum breakers. Understanding of the system design and arrangement is required to determine the impact on secondary containment.
The Torus-to-Drywell Vacuum Breakers relieve pressure from the Torus to the Drywell if there is a pressure differential greater than 0.5 psid. Each vacuum breaker has an air test operator located on Panel-J, which when manipulated allows the Control Room operator to fully cycle the valve. There is also a master control switch on the same panel which allows the opening of all twelve valves at the same time.
Reactor Building to Torus Vacuum Breakers
The Reactor Building to torus vacuum breaker system relieves pressure from the Reactor Building to the torus if torus pressure were to drop to 0.5 psi below Reactor Building pressure. Operation of either vacuum breaker will maintain a pressure differential of less than 2 psid, the external design pressure of containment.
The system consists of 2 separate lines which are open to the Reactor Building atmosphere. They then combine into a common line before going to the torus. Both lines contain a spring-tensioned check valve, and a 100% capacity air operated butterfly valve in series. Each butterfly valve is controlled by a three way switch which is located on Control Room Panel H.
Opening AO-NRV-31 under the current conditions will allow Torus & DW pressures to equalize therefore lowering the DP. Opening PC-AO-243 with Torus pressure greater than RB pressure will have no change on the DP due to the Vacuum Relief check valve (PC-CV-13CV) remaining closed with less than .5 psid.

Distracters:

A. This answer is incorrect due to Torus to Reactor Building DP remaining constant. This choice is plausible due to not recognizing or remembering the Vacuum Relief check valve (PC-CV-13CV) exists or will not open under the current plant conditions ($P_{Torus} > P_{RB}$). The candidate that correctly recognizes Torus to DW DP lowers and does not remember the RB to Torus vacuum breakers requires both the AOV & CV to open to reduce DP would select this answer.
C. This answer is incorrect due to Torus to DW DP lowering and Torus to Reactor Building DP remaining constant. This choice is plausible due to confusing the DP required to open the vacuum breaker and not recognizing or remembering the Vacuum Relief check valve (PC-CV-13CV) exists or will not open under the current plant conditions ($P_{Torus} > P_{RB}$). The candidate that confuses Torus to DW DP remaining the same vs. lowering and does not remember the RB to Torus vacuum

breakers requires both the AOV & CV to open to reduce DP would select this answer. D. This answer is incorrect due to Torus to DW DP lowering. This choice is plausible due to confusing the DP required to open the vacuum breaker. The candidate that confuses Torus to DW DP remaining the same vs. lowering and correctly recognizes Torus to RB DP remaining the same would select this answer.		
Technical References: Procedure 2.2.60 (Primary Containment Ventilation And Nitrogen Inerting System), Rev. 94		
References to be provided to applicants during exam: NONE		
Learning Objective:		
OPS Containment/COR002-03-02 LO-12 Describe the Containment design features and/or interlocks that provide for the following: f. Reactor Building to Torus D/P		
Question Source:	Bank # Quad 2011 NRC	
(note changes; attach parent)	Modified Bank #	
	New	
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b) 5	
Level of Difficulty:	3	
SRO Only Justification:	N/A	

EXAMINATION ANSWER KEY

Quad Cities 2011 ILT NRC Exam (RO Portion)

65	ID: QDC.ILT.16496	Points: 1.00

The following plant conditions exist on Unit 1:

Rx Bldg to Atmos DP	=	- 0.2 in. H_2O
Torus Pressure	=	0.01 psig
Drywell Pressure	=	1.4 psig

Complete the following statements:

If 1-1601-32A, DRYWELL TO TORUS VACUUM BKR is opened at panel 2251-24, Drywell to Torus d/p will __(1)__.

If AO 1-1601-20A, RX BLDG TO TORUS VACUUM BKR is opened at panel 901-3, Torus to Reactor Building d/p will __(2)__.

(Note: Consider the effect of each action separately.)

A. (1) lower
(2) remain constant

B. (1) lower
(2) lower

C. (1) remain constant
(2) remain constant

D. (1) remain constant
(2) lower

Answer: A

2.8 Following conditions cause PC-AO-243 and PC-AO-244, TORUS VAC RELIEF VLVs, to open:

2.8.1 Switch on VBD-H is held in OPEN.

2.8.2 Torus pressure falls 0.5 psig below Secondary Containment pressure and switch on VBD-H is not placed to CLOSED.

2.8.3 Loss of power or air.

2.9 Following conditions cause AO-NRV-20 through AO-NRV-31, TORUS TO DRYWELL VACUUM RELIEFs, to open:

2.9.1 Individual switch on VBD-J is placed to OPEN after instrument air is valved in. PC valves will normally be valved out.

2.9.2 All twelve valves open when MASTER CONTROL switch on VBD-J is placed to OPEN after instrument air is valved in.

2.9.3 When Drywell pressure is 0.5 psig less than Torus pressure.

2.10 Sutorbilt Blowers A and B will not start if PC-MO-1301 or PC-MO-1302 is closed.

2.11 PC-TMR-VTI, RUN TIME INTEGRATOR (VBD-H), may be used for monitoring Drywell and Torus purge and vent times and will start when any of following conditions are met:

2.11.1 Reactor coolant temperature > 200°F and Drywell supply isolation valves PC-MO-232 and PC-AO-238, and Drywell exhaust isolation valves PC-MO-231 and PC-AO-246 are open.

2.11.2 Reactor coolant temperature > 200°F and Drywell supply isolation valves PC-MO-232 and PC-AO-238, and Torus exhaust isolation valves PC-MO-230 and PC-AO-245 are open.

2.11.3 Reactor coolant temperature > 200°F and Torus supply isolation valves PC-MO-233 and PC-AO-237, and Torus exhaust isolation valves PC-MO-230 and PC-AO-245 are open.

2.11.4 Reactor coolant temperature > 200°F and Torus supply isolation valves PC-MO-233 and PC-AO-237, and Drywell exhaust isolation valves PC-MO-231 and PC-AO-246 are open.

Examination Outline Cross-Reference	**Level**	RO
Changed distractor 85 inches to 110 inches to support plausibility associated with incorrectly using the 400 psig curve. Correct answer changed to B. C distractor corrected.	**Tier#**	3
	Group#	
	K/A #	G2.1.25
	Rating	3.9

G2.1.25 Ability to interpret reference materials, such as graphs, curves, tables, etc. (CFR: 41.10 / 43.5 / 45.12)

Question 66

Reference Provided

The following LOCA Conditions exist:

- Fuel Zone Water Level (uncorrected) -140 inches
- RPV Pressure 600 psig
- DW Temperature 260°F

Which one of the following identifies the ACTUAL RPV Water Level under these conditions?

A. -90 inches

B. -100 inches

C. -110 inches

D. -172 inches

Answer:
B. -100 inches
Explanation: Requires interpreting the Fuel Zone (FZ) Range Correction Graph (Graph 14). With FZ RPV level indicating -140 inches with RPV pressure at 600 psig – actual RPV water is -100 inches.
Distracters: A. This answer is incorrect due to actual level being -100 inches. This choice is plausible due to being the correct level with RPV pressure at 800 psig. The

candidate that interpolates the wrong RPV pressure curve would select this answer.

C. This answer is incorrect due to actual level being -100 inches. This choice is plausible due to being the correct level with RPV pressure at 400 psig (which is adjacent to the 600 psig curve). The candidate that interpolates the wrong RPV pressure curve would select this answer.

D. This answer is incorrect due to actual level being -100 inches. This choice is plausible due to interpolating actual level (reversing axis) at -140 inches on the 600 psig curve. The candidate that interpolates backwards between indicated and actual RPV level the with the correct RPV pressure curve would select this answer.

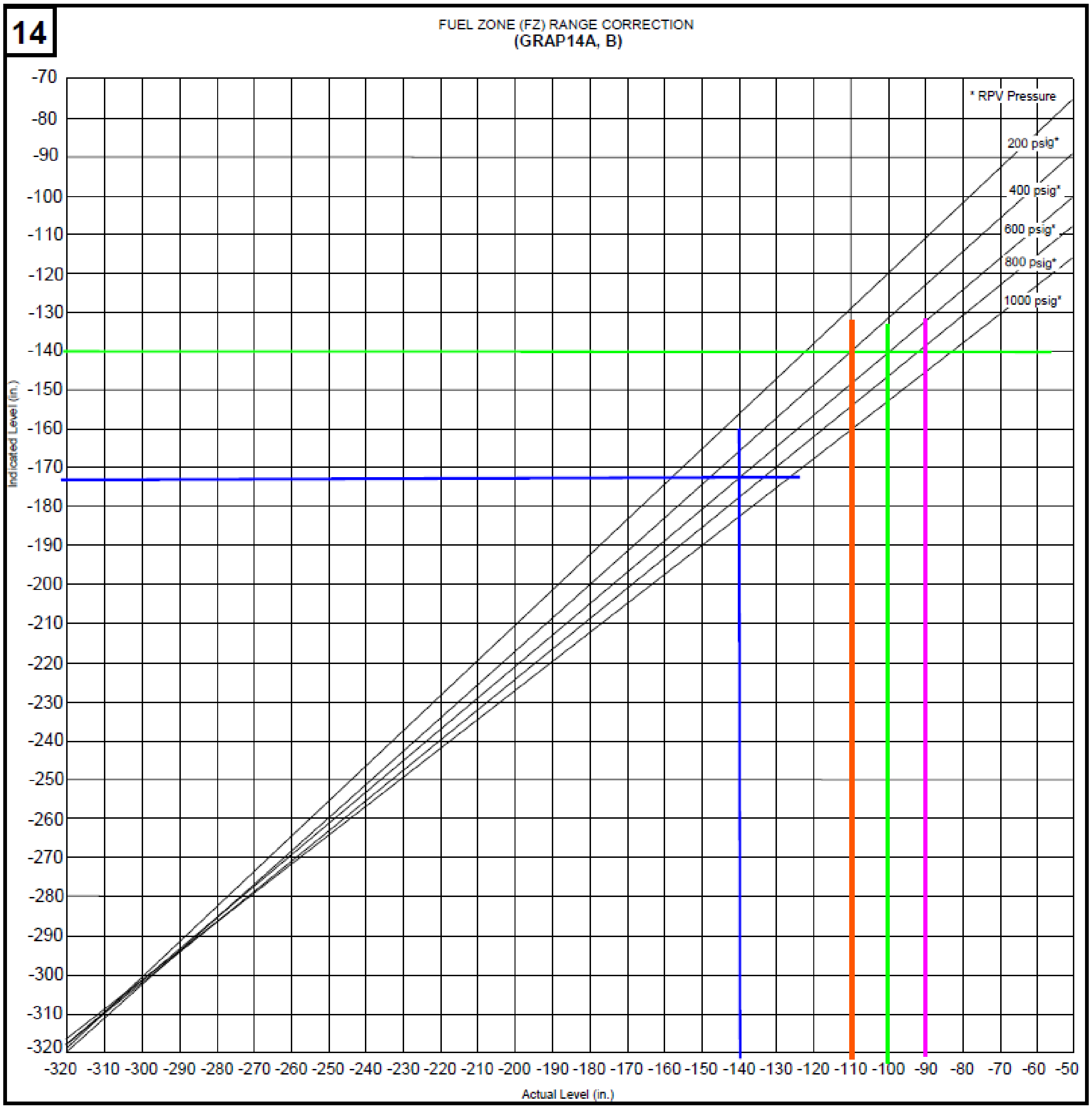
Technical References:
Emergency Operating Procedure 5.8, Attachment 2, Rev. 15

References to be provided to applicants during exam: Fuel Zone (FZ) Range Correction Graph (Graph 14).
And RPV Saturation Graph 01

Learning Objective:
INT00806090011300 Given plant conditions, assess if RPV water level can be determined or not.

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41(b) 10	
Level of Difficulty:	2	
SRO Only Justification:	N/A	

14


FUEL ZONE (FZ) RANGE CORRECTION
(GRAP14A, B)
* RPV Pressure
200 psig*
400 psig*
600 psig*
800 psig*
1000 psig*
Indicated Level (in.)
Actual Level (in.)
-70
-80
-90
-100
-110
-120
-130
-140
-150
-160
-170
-180
-190
-200
-210
-220
-230
-240
-250
-260
-270
-280
-290
-300
-310
-320
-320 -310 -300 -290 -280 -270 -260 -250 -240 -230 -220 -210 -200 -190 -180 -170 -160 -150 -140 -130 -120 -110 -100 -90 -80 -70 -60 -50

Examination Outline Cross-Reference	Level	RO
	Tier#	3
	Group#	
	K/A #	G2.1.30
	Rating	4.4
G2.1.30 Ability to locate and operate components, including local controls. (CFR: 41.7 / 45.7)		

Question 67

Which one of the following procedure types DOES NOT provide specific component location information utilized to operate Safety Related equipment in the plant?

A. Alarm Procedures

B. Abnormal Procedures

C. General Operating Procedures

D. Emergency Plan Implementing Procedures

Answer:

D. Emergency Plan Implementing Procedures

Explanation:

The primary source of specific component location information is the System Operating Procedure Component Checklists. Of the procedure choices provided, ONLY Emergency Plan Implementing Procedures do not provide specific component location information utilized to operate equipment in the plant.

The following types of procedure were evaluated for inclusion in this question which required the NOT statement in the stem (not enough NOT procedures to make a positive stem question):

- System Operating Procedure Component Checklists - do
- Emergency Plan Implementing Procedures - do not
- Abnormal Operating Procedures - do
- General Operating Procedures - do
- Administrative Procedures – do not
- Alarm Procedures - do

Distracters:

A. This answer is incorrect due to Alarm procedures providing specific component location information. This choice is plausible due to Alarm procedures not normally being utilized to locate components in the plant. The candidate that does not recall this procedure providing location information would select this answer.

B. This answer is incorrect due to Abnormal Operating procedures providing specific component location information. This choice is plausible due to Abnormal Operating procedures not normally being utilized to locate components in the plant. The candidate that does not recall this procedure providing location information would select this answer.

C. This answer is incorrect due to General Operating procedures providing specific component location information. This choice is plausible due to General Operating procedures not normally being utilized to locate components in the plant. The candidate that does not recall this procedure providing location information would select this answer.

Technical References:

Alarm Procedure 2.3_9-3-2 (Panel 9-3 - Annunciator 9-3-2), Rev. 30
EPIP Procedure 5.7.15 (OSC Team Dispatch), Rev. 20
General Operating Procedure 2.1.1 (Startup Procedure), Rev. 181
Abnormal Procedure 2.4crd (CRD Trouble), Rev. 15

References to be provided to applicants during exam: NONE

Learning Objective:

INT032010300G010B Procedure 2.0.1.2, "Operations Procedure Policy - Discuss the following as described in Procedure 2.0.1.2, Operations Procedure Policy; System and Instrument Operating Procedures

INT032010300G010E Procedure 2.0.1.2, "Operations Procedure Policy - Discuss the following as described in Procedure 2.0.1.2, Operations Procedure Policy; Abnormal and Emergency Procedures

INT032010300E010A Discuss the following as described in Alarm Procedure 2.3.1, General Alarm Procedure: Precautions and limitations

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b) 7	
Level of Difficulty:	3	
SRO Only Justification:	N/A	

HPCI SUCTION TRANSFER	PANEL/WINDOW: 9-3-2/A-4

1. AUTOMATIC ACTIONS

NOTE – If a Group 4 isolation signal is present, HPCI-MO-58, TORUS PUMP SUCT VLV, will not open if closed.

1.1 For alarm message (1634) HPCI SUCT TRANSFER SUPR POOL HIGH LEVEL:

1.1.1 Open HPCI-MO-58, TORUS PUMP SUCT VLV.

1.1.2 When HPCI-MO-58 is fully open, HPCI-MO-17, ECST PUMP SUCT VLV, HPCI-MO-21, TEST BYPASS TO ECST VLV, and HPCI-MO-24, ECST TEST LINE SHUTOFF VLV, receive a close signal.

1.2 For alarm message (1650) HPCI/RCIC SUCT TRANSFER ECST A LOW LEVEL or (1651) HPCI/RCIC SUCT TRANSFER ECST B LOW LEVEL:

1.2.1 Opens HPCI-MO-58 and RCIC-MO-41, TORUS PUMP SUCT VLV.

1.2.2 When HPCI-MO-58 and RCIC-MO-41 are fully open, HPCI-MO-17, HPCI-MO-21, HPCI-MO-24, RCIC-MO-18, ECST PUMP SUCT VLV, RCIC-MO-30, TEST BYP TO ECST VLV, and RCIC-MO-33, ECST TEST LINE SHUTOFF, receive a close signal.

2. OPERATOR OBSERVATION AND ACTION

2.1 Provide makeup to ECST per Procedure 2.2.7 as conditions permit.

2.2 As directed by CRS, WHEN suppression pool level and ECST level allow, THEN restore HPCI suction to ECST as follows:

2.2.1 IF EOPs have been entered, THEN restore HPCI suction to ECST as directed by EOPs.

2.2.2 IF not using EOPs, THEN perform following:

2.2.2.1 Notify SM that HPCI System is inoperable.

2.2.2.2 On front of Panel 9-39, place boot on Contact 9-10 of Relay 23A-K20.

2.2.2.3 Open HPCI-MO-17, ECST PUMP SUCT VLV.

2.2.2.4 Close HPCI-MO-58, TORUS PUMP SUCT VLV.

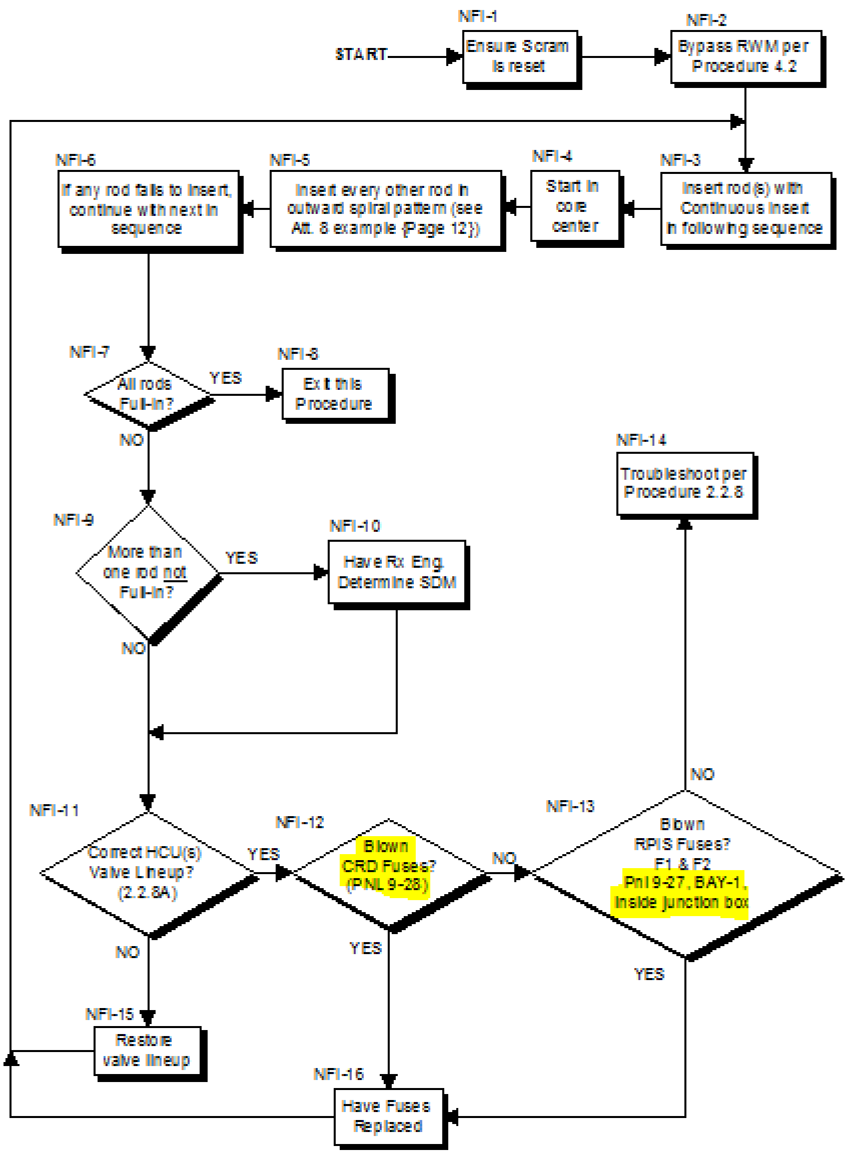
2.2.2.5 Verify Relay 23A-K20 (Panel 9-39) drops out and Contact 7-8 closes.

2.2.2.6 Remove boot from Contact 9-10 of Relay 23A-K20 (Panel 9-39).

2.2.3 Notify SM that HPCI System suction path is in normal line-up.

ATTACHMENT 2 ROD(S) NOT FULL-IN

ATTACHMENT 2 ROD(S) NOT FULL-IN



CNS OPERATIONS MANUAL EPIP PROCEDURE 5.7.15 OSC TEAM DISPATCH	USE: REFERENCE QUALITY: QAPD RELATED EFFECTIVE: 6/2/15 APPROVAL: ITR-RDM OWNER: K. A. TANNER DEPARTMENT: EP

1. PURPOSE

1.1 This procedure provides guidance to dispatch survey, repair, and rescue teams while maintaining personnel accountability and safety.©[1]

1.2 Topics covered are:

1.2.1 Dispatch and control of survey, repair, and rescue teams.

1.2.2 Precautions to be observed by survey, repair, and rescue teams.

1.2.3 Equipment to be carried during survey, repair, or rescue operations.

3. PREPARATION FOR STARTUP

NOTE 1 – Steps in Section 3 may be performed in any order.

NOTE 2 – Section 10 (Operation in MODE 4) of Procedure 2.1.4 will remain in effect until MODE 2 is entered.

NOTE 3 – Steps marked with **[MODE 2]** flag are those steps that must be completed prior to entering MODE 2.

3.1 Each Crew performing plant startup in Control Room shall receive an IPTE briefing prior to performing steps in Section 4 or 5.Ⓟ[7]

3.2 **[MODE 2]** Procedure 2.1.1.1 is complete for authorization to startup reactor.

Initials/Time/Date: ______/______/______

NOTE 1 – If AOM - Operating Shift determines Attachment 1 does not need to be performed, Step 3.3 is N/A.

NOTE 2 – AOM - Operating Shift may determine only parts of Attachment 1 will need to be performed. All parts of Attachment 1 which do not need to be performed are N/A.

3.3 Initiate Attachment 1 to ensure station systems are OPERABLE for reactor startup.

Initials/Time/Date: ______/______/______

3.4 Ensure RWCU-395, RWCU SUBCOOLING SHUTOFF (RWCU HX Room west between heat exchangers), is CLOSED.

Initials/Time/Date: ______/______/______

3.5 **[MODE 2]** Ensure RWCU-FPC Crosstie Spool Piece is removed from system prior to entry into Mode 1, 2, or 3.

Initials/Time/Date: ______/______/______

NOTE – Step 3.6 ensures Startup Level Control System is functional.

3.6 Perform Procedure 15.RF.201. Signature/Date: ____________________

3.7 Ensure Procedure 2.1.1.2 initiated to ensure Technical Specifications Surveillance Requirements are satisfied for plant startup.

Initials/Time/Date: ______/______/______

3.8 **[MODE 2]** Surveillance Coordinator or designee has verified all Surveillance Requirements listed in Procedure 2.1.1.2, Tech Spec Pre-Startup Checks, is complete or documented as a discrepancy.Ⓒ[3,4]

Signature/Date: ____________________

Examination Outline Cross-Reference	Level	RO
	Tier#	3
	Group#	
	K/A #	G2.1.31
	Rating	4.6
G2.1.31 Ability to locate control room switches, controls, and indications, and to determine that they correctly reflect the desired plant lineup. (CFR: 41.10 / 45.12)		

Question 68

The following plant conditions exist during reactor cooldown:

- Reactor MODE Switch is in SHUTDOWN.
- RWCU is in service.
- No Reactor Recirculation pumps are in service.
- RHR Loop A is in Shutdown Cooling
- 6.RCS.601 (Technical Specification Monitoring of RCS Heatup/Cooldown Rate) is in progress.

Which one of the following completes the statement below regarding the indication and indication location that confirms the reactor is in Cold Shutdown during the performance of 6.RCS.601?

____(1)____ Temperature indicates 200°F on Panel ____(2)____.

A. (1) Vessel Drain
(2) 9-4

B. (1) Vessel Drain
(2) 9-21

C. (1) RR "B" Suction
(2) 9-4

D. (1) RR "B" Suction
(2) 9-21

Answer:
B. (1) Vessel Drain (2) 9-21

Explanation:
Requires knowledge of indication location and indication which supports the plant being in Mode 4. IAW TS Definitions - Cold Shutdown (Mode 4) is determined by Mode Switch position (Shutdown) and Average Reactor Coolant Temperature ($\leq$212°F) with all reactor vessel head closure bolts fully tensioned. The RR suction temperature is an accurate measure of average reactor coolant temperature but only with a RR pump operating and one of the temperatures monitored on reactor cooldown.

Distracters:

A. This answer is incorrect due to Vessel Drain indication being located on Panel 9-21. This choice is plausible due to the common misconception that Water to RWCU is the same temperature element as the vessel bottom drain. The candidate that recognizes vessel drain temperature and confuses the location would select this answer.

C. This answer is incorrect due to Vessel Drain used to determine average reactor coolant temperature IAW 6.RCS.601 and the indication being located on Panel 9-21. This choice is plausible due to RR suction would be the preferred indication if the applicable RR pump were in operation and this indication is located on Panel 9-4. The candidate that thinks RR suction temperature is accurate without a RR pump in operation and correctly identifies the location would select this answer.

D. This answer is incorrect due to Vessel Drain used to determine average reactor coolant temperature IAW 6.RCS.601. This choice is plausible due to RR suction would be the preferred indication if the applicable RR pump were in operation. The candidate that thinks RR suction temperature is accurate without a RR pump in operation and confuses the location would select this answer.

Technical References:
Procedure 6.RCS.601 (Technical Specification Monitoring of RCS Heatup/Cooldown Rate), Rev. 21
TS Table 1.1-1

References to be provided to applicants during exam: NONE

Learning Objective:
INT00705010010400 From memory, given a set of plant conditions, determine the plant MODE

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b) 10	

Level of Difficulty:	3
SRO Only Justification:	N/A

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	> 212
4	Cold Shutdown[(a)]	Shutdown	≤ 212
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.

ATTACHMENT 1 REACTOR VESSEL TEMPERATURE HEATUP AND COOLDOWN LOG

ATTACHMENT 1 REACTOR VESSEL TEMPERATURE HEATUP AND COOLDOWN LOG

Valid with RWCU in-service

Valid with RR Pump in-service

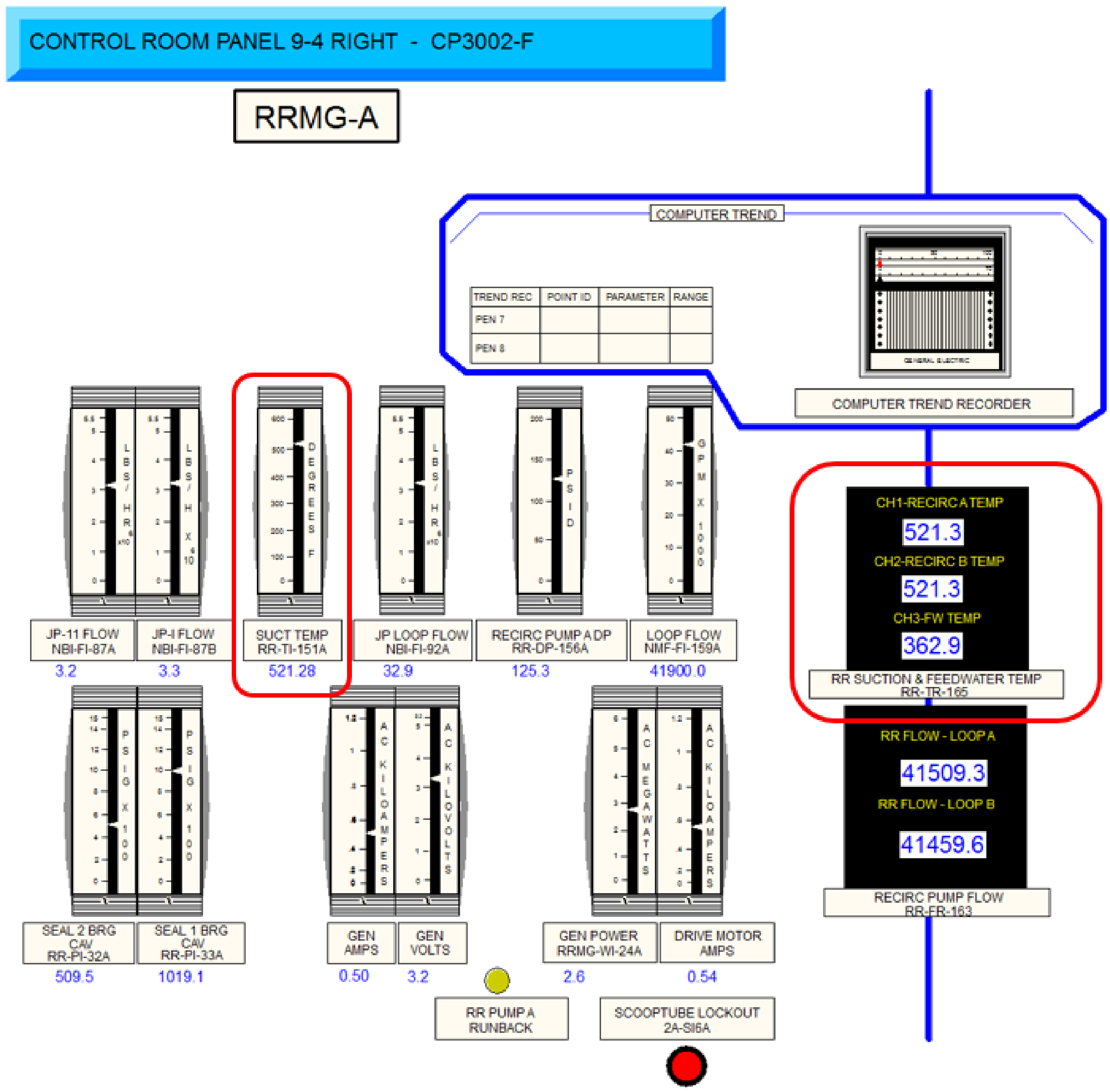
Date: ____________ Purpose/Condition: ______________________________ Page _____ of _____

TIME	VESSEL DRAIN M180 NBI-TR-89 CH 7	UPPER VESSEL M178 or M179 NBI-TR-89 CH 6	BELTLINE M182 or M181 NBI-TR-89 CH 9	BOTTOM HEAD M184 or M183 NBI-TR-89 CH 11	RR "A" SUCTION CRR222 RR-TR-165 CH 1 RR-TI-151A	RR "B" SUCTION CRR223 RR-TR-165 CH 2 RR-TI-151B	REACTOR PRESSURE B025 RFC-R-LRPR97 CHANNEL 1 RED TRACE	T-SAT (ONLY IF SATURATED)	P/T CURVE SAT ✓	HUR CDR SAT ✓	INT	CB

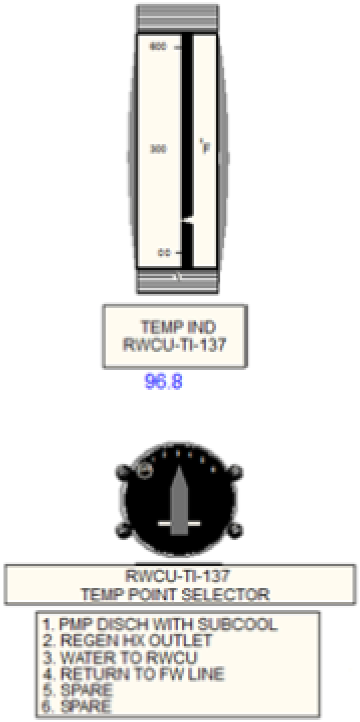
Panel 9-21

Panel 9-4

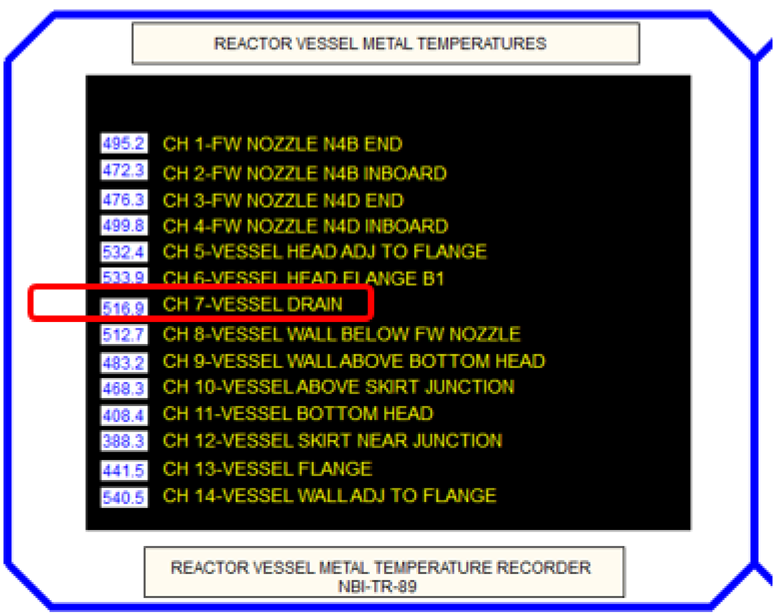
Panel 9-5


CONTROL ROOM PANEL 9-4 RIGHT - CP3002-F
RRMG-A
COMPUTER TREND
TREND REC
POINT ID
PARAMETER
RANGE
PEN 7
PEN 8
COMPUTER TREND RECORDER
CH1-RECIRC A TEMP
521.3
CH2-RECIRC B TEMP
521.3
CH3-FW TEMP
362.9
RR SUCTION & FEEDWATER TEMP
RR-TR-165
RR FLOW - LOOP A
41509.3
RR FLOW - LOOP B
41459.6
RECIRC PUMP FLOW
RR-FR-163
JP-11 FLOW
NBI-FI-87A
3.2
JP-I FLOW
NBI-FI-87B
3.3
SUCT TEMP
RR-TI-151A
521.28
JP LOOP FLOW
NBI-FI-92A
32.9
RECIRC PUMP A DP
RR-DP-156A
125.3
LOOP FLOW
NMF-FI-159A
41900.0
SEAL 2 BRG CAV
RR-PI-32A
509.5
SEAL 1 BRG CAV
RR-PI-33A
1019.1
GEN AMPS
0.50
GEN VOLTS
3.2
GEN POWER
RRMG-WI-24A
2.6
DRIVE MOTOR AMPS
0.54
RR PUMP A RUNBACK
SCOOPTUBE LOCKOUT
2A-SI6A

Panel 9-4



Panel 9-21



Examination Outline Cross-Reference	Level	RO
Significantly modified the stem and changed the correct answer to now be considered Modified. Spelled out all position titles.	Tier#	3
	Group#	
	K/A #	G2.2.12
	Rating	3.7
G2.2.12 Knowledge of surveillance procedures. (CFR: 41.10 / 45.13)		

Question 69

Which one of the following completes the statement below regarding SM delegation IAW Procedure 0.26 (Surveillance Program)?

The Shift Manager may delegate the authority to authorize performance of Surveillance Tests to the Work Control Center Administrator …

A. ONLY.

B. or the Shop Superintendent.

C. or the Work Control Operator.

D. or the Control Room Supervisor.

Answer:
D. or the Control Room Supervisor.
Explanation: IAW Procedure 0.26 the authority to authorize performance of the test may be delegated to the CRS or the WCCA as the SM deems necessary.
Distracters: A. This answer is incorrect due to the WCCA or CRS being able to be delegated by the SM. This choice is plausible due to CRSs primarily authorizing the start of surveillance tests being confused with position responsibilities already afforded the CRS. The candidate that confuses the CRS position authority vs. being delegated to authorize starting of tests would select this answer. B. This answer is incorrect due to the Shop Superintendent not authorized to be delegated by the SM IAW 0.26. This choice is plausible due to Shop Superintendent being delegated to authorize WORK start. The candidate that

confuses starting work vs. starting of tests would select this answer.

C. This answer is incorrect due to the Work Control Operator not authorized to be delegated by the SM IAW 0.26. This choice is plausible due to confusing WCCA and WCO position authorities/responsibilities (WCO does work, WCCA authorizes work). The candidate that confuses WCO as being delegated to authorize WORK start would select this answer.

Technical References:
Procedure 0.26 (Surveillance Program), Rev. 67

References to be provided to applicants during exam: NONE

Learning Objective:
G. Procedure 0.26, Surveillance Program
1. Discuss the following as described in Administrative Procedure 0.26, Surveillance Program:
   i. Surveillance Test Authorization

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank # 69 from 8/2014 NRC Exam	
	New	
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b) 10	
Level of Difficulty:	2	
SRO Only Justification:	N/A	

2.2.12 Knowledge of surveillance procedures.

Question: 69

What position, by title, can the Shift Manager delegate to authorize performance of a surveillance test?

a. Work Week Director (WWD)

b. Work Control Operator (WCO)

c. Work Control Center Supervisor (WCCS)

d. Work Control Center Administrator (WCCA)

Answer:

d. Work Control Center Administrator (WCCA)

5. SURVEILLANCE TEST AUTHORIZATION

NOTE – The authority to authorize performance of the test may be delegated to the CRS or the WCCA as the SM deems necessary.

5.1 Shift Manager shall:

5.1.1 Understand intent of test.

5.1.2 Understand direct and indirect actions of the test signals.

5.1.3 Ensure plant conditions are as required per the following:

5.1.3.1 If entire test is to be performed, ensure plant conditions are as required by the prerequisites of the test before authorizing test performance.

5.1.3.2 If only a portion of test is being performed (example, performing portions of a Surveillance Procedure for PWT), ensure applicable prerequisites are met prior to authorizing partial test performance.

5.1.4 If a scheduled divisional test, ensure it is authorized in its designated week.

5.1.5 Ensure performance of test will not compromise divisional protection of ECCS Systems.

5.1.6 Ensure no other tests are in-progress on the same system such that the combination of tests would result in erroneous signals or inadvertent initiation.

5.1.7 Be aware of any other systems affected by the test and how they are affected.

5.1.8 Ensure equivalent instrumentation, when used for Inservice Testing (IST) of pumps, meet the criteria as stated in the definition of equivalent instrument of this procedure.

NOTE – The use of delayed entry time (DET) does not circumvent the requirement to declare the SSC inoperable.

5.1.9 If a TS SR allows the use of delayed entry into an associated Condition and Required ACTION, use Attachment 1 or any similar documentation/electronic log, as defined in this procedure, to facilitate tracking the TS delayed entry time (DET).

Examination Outline Cross-Reference	Level	RO
	Tier#	3
	Group#	
	K/A #	G2.2.37
	Rating	3.6
G2.2.37 Ability to determine operability and/or availability of safety related equipment. (CFR: 41.7 / 43.5 / 45.12)		

Question 70

Which one of the following completes the statements below regarding the impact of losing 125 VDC control power to a CLOSED 4160 VAC safety-related load breaker?

The safety related load ____(1)____ .

Breaker operation can be accomplished ____(2)____.

A. (1) is unavailable
(2) locally ONLY

B. (1) is unavailable
(2) locally or from the Control Room

C. (1) remains energized
(2) locally ONLY

D. (1) remains energized
(2) locally or from the Control Room

Answer:
C. (1) remains energized (2) locally ONLY
Explanation: The loss of 125 VDC control power to a closed 4KV safety related breaker will cause loss of remote breaker indication and the inability to operate the breaker remotely. With the breaker closed, the load continues to be energized maintaining availability.
Distracters: A. This choice is incorrect due to the load remaining energized. This choice is plausible due to loss of indicating lights being commonly confused as the load

being de-energized. The candidate that confuses loss of indicating lights as being de-energized and correctly identifies remote operation is lost would select this answer.

B. This choice is incorrect due to the load remaining energized and remote operation not being available. This choice is plausible due to loss of indicating lights being commonly confused as the load being de-energized and confusing local vs. remote operation. The candidate that confuses loss of indicating lights as being de-energized and confuses remote operation with local operation would select this answer.

D. This choice is incorrect due to remote operation not being available. This choice is plausible due to confusing local vs. remote operation. The candidate that recognizes the load availability is maintained with the breaker closed and confuses remote operation with local operation would select this answer.

Technical References:
Procedure 5.3DC125 (Loss OF 125 VDC), Rev 36

References to be provided to applicants during exam: NONE

Learning Objective:

COR0010102001130N Predict the consequences of the following events on the AC Electrical Distribution System: Loss of DC Power

COR0020702001080B Given a specific DC Electrical Distribution system malfunction, determine the effect on any of the following: Components using DC control power (i.e., breakers)

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b) 7	
Level of Difficulty:	2	
SRO Only Justification:	N/A	

V. **SYSTEM INTERRELATIONSHIPS**

A. The AC Electrical Distribution System supplies operational or emergency power to almost every other plant system.

CAUTION: (LER 88-01) Do not open Potential Transformer (PT) compartments while the bus is energized. Opening of PT compartments while the bus is energized will cause the bus to lose voltage indication which might cause unwanted automatic functions to occur.

SO-14d B. 125 VDC

LO-13n Provides power for control and indication and protective relaying of 4160V and 480V breakers.

Due to loss of control power, the breakers could not be operated from the Control Room. They can be operated manually at the breaker. The breaker position indicating lights would be lost in the Control Room and locally.

The loss of a single 125V bus should not affect plant safety because of the ability to use transfer switches. Restoration of control power to 4160V and 480V breakers may cause the undervoltage devices to trip the breaker if the trip coil is energized before the undervoltage device.

The loss of the 125 VDC system also results in the loss of the automatic power transfer and load shedding features on 4160V busses 1F and 1G.

Lesson Title: OPS AC Electrical Distribution	
Lesson Number: COR001-01-01	Revision Number: 47

ATTACHMENT 1 LOSS OF 125 VDC DISTRIBUTION PANEL A

1. LOSS OF 125 VDC DISTRIBUTION PANEL A

1.1 IF 1AE, BUS 1E TIE BKR, is flagged closed, THEN place 1BE, BUS 1E TIE BKR, to PULL-TO-LOCK.

1.2 Verify RVLC System is maintaining reactor water level.

1.3 Send Operator to R-976-W to standby to open RRMG A field breaker locally.

1.4 Send Operator to Non-Critical Switchgear Room to standby to open RRMG A drive motor breaker locally.

1.5 WHEN Operators in place, THEN perform following simultaneously:

1.5.1 Open RRMG A drive motor Breaker EE-CB-4160C(1CN) or EE-CB-4160C(1CS) by depressing TRIP button.

1.5.2 Open RRMG A field breaker by depressing TRIP button on RRMG-CB-CB11A.

NOTE – Breaker 1CN has two auxiliary relays. One is located in the upper cubicle of Breaker 1CN, the other is located in the upper cubicle of Breaker 1CS.

1.6 IF Breaker EE-CB-4160C(1CN) opened, THEN perform following:

1.6.1 Inside EE-CB-4160C(1CN) upper cubicle, press TRIP button in center of Relay EE-REL-1CN(52AR) to open relay contacts.

1.6.2 Inside EE-CB-4160C(1CS) upper cubicle, press TRIP button in center of Relay EE-REL-1CN(52ARA) to open relay contacts.

1.7 IF Breaker EE-CB-4160C(1CS) opened, THEN perform following:

1.7.1 Inside EE-CB-4160C(1CS) upper cubicle, press TRIP button in center of Relay EE-REL-1CS(52AR) to open relay contacts.

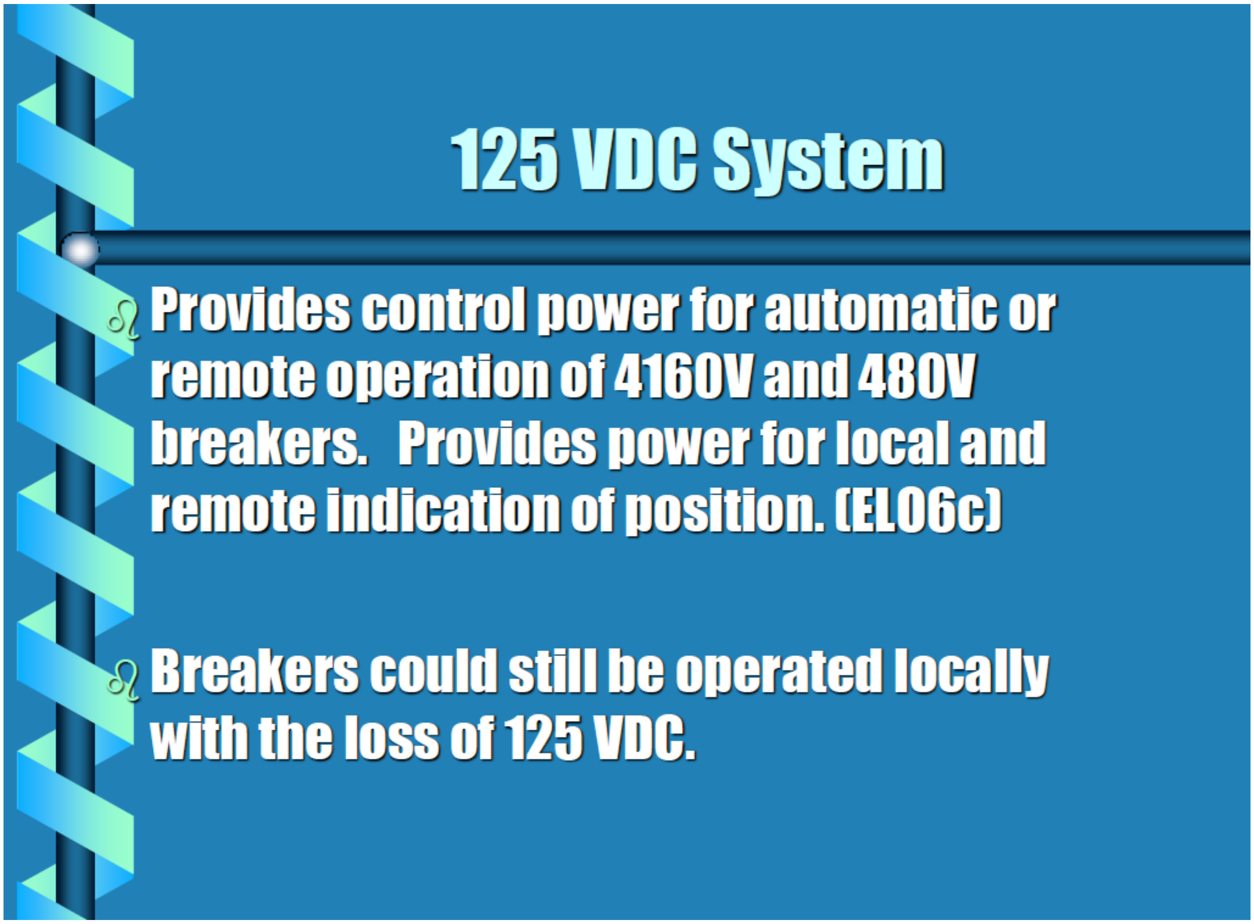
1.7.2 Inside EE-CB-4160C(1CS) upper cubicle, press TRIP button in center of Relay EE-REL-1CS(52ARA) to open relay contacts.

1.8 Enter Procedure 2.4RR for RR Pump Trip.

1.9 Secure SAC 1A.

1.9.1 At SA-LCU-SAC1A, on Elektronikon panel, press STOP.

1.9.2 At 480V Bus 1F, open Breaker EE-CB-480F(SAC-A) by depressing silver TRIP button.


125 VDC System
Provides control power for automatic or remote operation of 4160V and 480V breakers. Provides power for local and remote indication of position. (EL06c)
Breakers could still be operated locally with the loss of 125 VDC.

Examination Outline Cross-Reference	Level	RO
	Tier#	3
	Group#	
	K/A #	G2.3.7
	Rating	3.5
G2.3.7 Ability to comply with radiation work permit requirements during normal or abnormal conditions. (CFR: 41.12 / 45.10)		

Question 71

Which one of the following activities is authorized to be performed while on a “Tours and Inspections” RWP **WITHOUT** prior Radiation Protection approval/notification IAW 9.EN-RP-101 (Access Control for Radiologically Controlled Areas)?

A. Valve lineups.

B. Routing cables in the overhead.

C. Draining a system for maintenance.

D. Adding lube oil to a CRD Pump motor.

Answer:
A. Valve lineups.
Explanation: IAW 9.EN-RP-101 The following are allowed to be performed during a Tour and Inspection: • Personnel in the performance of log and/or data collection. • Personnel in the performance of valve lineups. • Radiation Protection personnel in the performance of radiological surveys. • Manager, Supervisory personnel, and Inspectors while monitoring personnel performance. Not allowed: Climbing, crawling through, under, or on components, reaching through contaminated components, venting and draining operations, or any function not specifically listed in during tour and inspection is not allowed.
Distracters: B. This choice is incorrect due to valve lineups being the only identified activity allowed to be performed un a “Tours & Inspections” RWP without prior RP

permission. This choice is plausible due to overlooking overhead areas being considered contaminated requiring RP support & permission. The candidate that does not recognize the additional RP permission would select this answer.

C. This choice is incorrect due to valve lineups being the only identified activity allowed to be performed un a “Tours & Inspections” RWP without prior RP permission. This choice is plausible due to draining systems is a normal operations activity but still requires RP notification prior to performance. The candidate that does not recognize the additional RP notification would select this answer.

D. This choice is incorrect due to valve lineups being the only identified activity allowed to be performed un a “Tours & Inspections” RWP without prior RP permission. This choice is plausible due to being a routine maintenance activity but not a standard tours & inspection function. The candidate that confuses routine maintenance being authorized on a tours & inspection RWP would select this answer.

Technical References:
Procedure 9.EN-RP-101 (Access Control For Radiologically Controlled Areas), Rev 15.

References to be provided to applicants during exam: NONE

Learning Objective:

INT032011000D0100 Procedure 9.ALARA.4, Radiation Work Permits: Discuss the precautions and limitations associated with Radiation Work Permits (RWP's).

INT032011000D0200 Procedure 9.ALARA.4, Radiation Work Permits: Discuss the compliance and use requirements associated with RWP's.

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b) 12	
Level of Difficulty:	2	
SRO Only Justification:	N/A	

16.2 The following activities are considered tour and inspection:

16.2.1 Personnel in the performance of log and/or data collection.

16.2.2 Personnel in the performance of valve lineups.

16.2.3 Radiation Protection personnel in the performance of radiological surveys.

16.2.4 Managers, Supervisory personnel, and Inspectors while monitoring personnel performance.

16.3 Climbing, crawling through, under, or on components, reaching through contaminated components, venting and draining operations, or any function not specifically listed in Step 16.2 during tour and inspection is not allowed.

16.4 Activities not listed in Step 16.2 may be allowed to be performed as a tour and inspection with prior Radiation Protection approval.

17. MANUAL ENTRY/EXIT

17.1 Use CNS Data Form RP-1B for any of the following conditions:

17.1.1 To track dose when the RCA Electronic Access System is not available; or

17.1.2 Any other situation specified by RP.

17.2 When entering the RCA using a CNS Data Form RP-1B, workers should complete the entry section of the Dose Tracking Form. Normally, the form will remain at the control point while the individual is in the RCA.

17.3 When exiting the RCA, workers shall complete the exit section of CNS Data Form RP-1B and give the Direct Reading Dosimeter (DRD) to RP personnel.

18. ACCESS CONTROL GUARD

NOTE – An Access Control Guard can only guard one posted and barricaded access point at a time.

18.1 If an entry into a LHRA/VHRA is made using the conditions of an Access Control Guard, then the Access Control shall be stationed or may be one of the entrants using the following guidelines:

18.1.1 Non-RP individuals must read, be briefed by a RPS/RPT, and sign a copy of Attachment 4, Responsibilities For The Access Control Guard.

18.1.2 RP personnel are not required to complete Attachment 4, but are expected to comply with its requirements.

18.1.3 The Access Control Guard shall have a signed Attachment 4 present with them.

Examination Outline Cross-Reference	Level	RO
Comment incorporated.	Tier#	3
	Group#	
	K/A #	G2.3.13
	Rating	3.4

G2.3.13 Knowledge of radiological safety procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc. (CFR: 41.12 / 43.4 / 45.9 / 45.10)

Question 72

An Operator has to enter a room to close a valve to stop a small water leak.

- The affected work area general radiation level is 1600 mRem/hour.

What type of entry permit is required and is continuous RP coverage required during the entry?

	Entry Permit	Continuous RP Coverage
A.	Special Work Permit (SWP)	Required
B.	Special Work Permit (SWP)	NOT Required
C.	Radiological Work Permit (RWP)	Required
D.	Radiological Work Permit (RWP)	NOT Required

Answer:
A. Special Work Permit (SWP) Required

Explanation:
Requires knowledge of LHRA entry requirements. IAW 9.EN-RP-101, entry into LHRAs require a SWP and continuous HP coverage. A Locked High Radiation Area is an area accessible to individuals in which radiation levels from sources external to the body could result in an individual receiving a deep dose equivalent > 1 rem (10 mSv) in 1 hour at 30 cm (~ 12") from the radiation source or from any surface that the radiation penetrates. A Special Work Permit Area (SWP) is an area where a SWP has been issued to control access to, and work within, which involves any **one** or

combination of the conditions: a Very High Radiation Area, **Locked High Radiation Area**, a High Radiation Area, High Contamination Area, Discrete Radioactive Particle Area, or an Airborne Radiation Area.

Distracters:

B. This answer is incorrect because continuous RP coverage is required. This answer is plausible because an SWP permit is required for entry and under some conditions Operations personnel can take action to protect the health and safety of the public without continuous RP coverage. The isolation of a small leak is not an instance where health and safety of the public is an issue. The candidate who believes continuous RP coverage is NOT required would select this answer.

C. This answer is incorrect because an SWP is required for entry. This answer is plausible because there are instances where Operations personnel can take actions to protect the health and safety of the public without signing on a permit or signing on the SWP. However, the action to isolate a small leak is not an instance where the health and safety of the public is an issue. The candidate who understands continuous RP coverage is required but doesn't realize an SWP is required would select this answer.

D. This answer is incorrect because an SWP and Continuous RP coverage are required to enter a LHRA. This answer is plausible because there are instances where Operations personnel can take actions to protect the health and safety of the public without signing on a permit or signing on the SWP and continuous HP coverage not being required. The candidate who believes the action can be taken and doesn't realize the requirements of a locked high radiation area entry would select this answer.

Technical References:
Procedure 9.EN-RP-101 (Access Control For Radiologically Controlled Areas), Rev 15

References to be provided to applicants during exam: NONE

Learning Objective:

INT032-01-100 OPS CNS Administrative Procedures Radiation Protection

H. 9-EN-RP-101, Access Control for Radiologically Controlled Areas
   1. Precautions and limitations
   2. RCA access and egress

Question Source:	Bank # Q73 2014 NRC Retake	
(note changes; attach parent)	Modified Bank #	
	New	
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b) 12	
Level of Difficulty:	2	

SRO Only Justification:	N/A

Question: 73

An Operator has to enter a room to close a valve to stop a small water leak.

- The affected work area general radiation level is 1600 mrem/hour.

What type of entry permit is required?
Is continuous RP coverage required during the entry?

	Entry Permit	Continuous RP Coverage
A.	SWP	NOT Required
B.	SWP	Required
C.	RWP	Required
D.	RWP	NOT Required

Answer:
B. SWP Required

6.8 Specific monitoring and radiological controls for access to "Locked High Radiation Areas" shall be made by RP personnel and listed on the appropriate SWP. As a minimum, each person entering a "Locked High Radiation Area" shall have:

6.8.1 DLR.

6.8.2 Alarming direct reading dosimeter (electronic dosimeter).

6.8.3 Approved SWP.

6.8.4 RP Shift Technician or Radiation Protection Supervision (RPS) approval.

6.8.5 If an individual is working in a dose rate of ≥ 1000 mR/hr, then continuous RP coverage with the use of CNS RP-56, Radiological Stay Time Verification Sheet, is required. Ⓟ[4]

6.8.6 Radiation Protection Manager or designee approval for entry into LHRAs with general area dose rates > 1.5 rem/hr in the actual work area. Ⓟ[4]

6.8.7 Documented pre-job brief for entry, using the CNS RP-800, given by RP personnel. This brief shall cover radiological conditions in the immediate work areas using the most recent survey data available and the scope of the work to be performed. RPS performs the pre-job brief for entry into the LHRAs with general dose rates > 1.5 rem/hr in the actual work area. Ⓟ[4,5,6]

6.8.7.1 All briefings shall be performed by RP Supervision, ANSI 18.1, or ANSI 3.1 RP Technicians. Ⓟ[1]

6.9 While LHRAs are open, the access to the LHRA shall be controlled per Technical Specifications.

6.10 Turnover of radiological coverage by RP personnel during Locked High Radiation Area work should be avoided. Whenever transfer of key is required, perform per Section 19.

6.11 The following verification shall follow the initial check by the Access Control Guard, RPT, or entrant and be documented on CNS RP-11D, LHRA Entry Door/Gate Check:

6.11.1 Upon re-establishing any LHRA boundary controls following work that required access into these areas, a second person shall verify the access point(s) are securely locked. This verification shall consist of ensuring the locking mechanism has been replaced on the access point. The person performing the verification shall be a RPT.

6.11.2 Where keys are required to lock doors, verify the door is closed and secured/locked with a physical challenge of the door.

6.11.3 If the door is locked using a padlock and chain or cable, then inspect the lock and chain or cable for defects and physically challenge the lock.

Examination Outline Cross-Reference	**Level**	RO
	Tier#	3
	Group#	
	K/A #	G2.4.18
	Rating	3.3
G2.4.18 Knowledge of the specific bases for EOPs. (CFR: 41.10 / 43.1 / 45.13)		

Question 73

Which one of the following completes the statement below regarding the bases for the Emergency Operating Procedures (EOPs)?

The EOPs are ____(1)____ based procedures utilized together to protect fission product barriers until ____(2)____ Flooding is required.

A. (1) event
(2) RPV

B. (1) event
(2) Primary Containment

C. (1) symptom
(2) RPV

D. (1) symptom
(2) Primary Containment

Answer:
D. (1) symptom (2) Primary Containment
Explanation: The CNS EOPS (PSTGs/SATGs) are symptomatic in that events need not be diagnosed before action is taken. Instead, entry conditions and operator actions are based upon the values and trends of key plant parameters, irrespective of the initiating event. The PSTG defines strategies for responding to emergencies and events that may degrade into emergencies up until primary containment flooding is required.
Distracters:

A. This answer is incorrect due to EOPs being symptom based and utilized until Primary Containment Flooding is required. This answer is plausible because AOPs being event specific based and RPV Flooding being easily confused with Primary Containment Flooding entry. The candidate that confuses AOP vs. EOP bases and RPV vs. Primary Containment Flooding would select this answer.
B. This answer is incorrect due to EOPs being symptom based. This answer is plausible because AOPs being event specific based. The candidate that correctly identifies AOP vs. EOP bases and confuses RPV vs. Primary Containment Flooding would select this answer.
C. This answer is incorrect due to EOPs being utilized until Primary Containment Flooding is required. This answer is plausible because RPV Flooding is easily confused with Primary Containment Flooding entry. The candidate that correctly identifies AOP vs. EOP bases and confuses RPV vs. Primary Containment Flooding would select this answer.

Technical References:
AMP-TBD00 (CNS PSTG/SATG Appendix B Technical Bases) Rev. 8

References to be provided to applicants during exam: NONE

Learning Objective:
INT00806010010400 Identify the required characteristics of Emergency Operating Procedures (EOP's) prepared from the BWR EPG/SAGs.

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b) 10	
Level of Difficulty:	3	
SRO Only Justification:	N/A	

3. PSTG/SATG Introduction

PSTG/SATG Step

The Cooper Nuclear Station (CNS) Plant-Specific Technical Guideline and Severe Accident Technical Guideline (PSTG/SATG) provide symptomatic direction for plant emergency response and severe accident mitigation. The guidelines were developed from generic technical guidance prepared by the BWR Owners' Group in response to NUREG-0737 Item I.C.1, "Guidance for the Evaluation and Development of Procedures for Transients and Accidents," and NEI 91-04, "Severe Accident Issue Closure Guidelines," Revision 1, Section 5.

Discussion

NUREG-0737, Item I.C.1 established a requirement for emergency operating procedure upgrades that led to the creation of generic Emergency Procedure Guidelines. NEI 91-04, Revision 1, Section 5, describes the industry initiative for severe accident management which led to the creation of generic Severe Accident Guidelines.

The CNS PSTGs/SATGs are symptomatic in that events need not be diagnosed before action is taken. Instead, entry conditions and operator actions are based upon the values and trends of key plant parameters, irrespective of the initiating event.

PSTG/SATG Step

The PSTG/SATG are divided into a Plant-Specific Technical Guideline (PSTG) and a Severe Accident Technical Guideline (SATG). The PSTG defines strategies for responding to emergencies and events that may degrade into emergencies up until primary containment flooding is required. It comprises four guidelines and five associated contingencies:

Guidelines:

- RPV Control
- Primary Containment Control
- Secondary Containment Control
- Radioactivity Release Control

PSTG Contingencies:

- #1 Alternate Level Control
- #2 Emergency RPV Depressurization
- #3 Steam Cooling
- #4 RPV Flooding
- #5 Level/Power Control

The SATG defines strategies applicable after primary containment flooding is required. It comprises two guidelines:

- RPV and Primary Containment Flooding
- Containment and Radioactivity Release Control

The PSTG and SATG function together as an integrated set of instructions. Each PSTG guideline protects one of the principal barriers to radioactivity release through control of key plant parameters. The PSTG contingencies form extensions to the top-level guidelines, providing more detailed instructions for controlling individual parameters under more degraded conditions. The SATG extends the PSTG still further, addressing severe accident conditions.

The Cautions section contains warnings applicable to certain steps within the guidelines. It is common to both the PSTG and SATG. Cross-references, consisting of circled numbers in reverse type, appear in the right margin of the steps to which the cautions apply.

Examination Outline Cross-Reference	Level	RO
	Tier#	3
	Group#	
	K/A #	G2.4.31
	Rating	4.2
G2.4.31 Knowledge of annunciator alarms, indications, or response procedures. (CFR: 41.10 / 45.3)		

Question 74

The plant is operating at rated power.

- A Surveillance Procedure that began during your shift is in progress.
- All alarms associated with the surveillance have been flagged with pink tape during your shift.

A flagged annunciator then alarms (the first annunciation of this alarm for the shift) due to the Surveillance Procedure at the appropriate time.

What is/are the minimum action(s) required IAW Procedure 2.3.1 (General Alarm Procedure)?

A. Acknowledge the annunciator ONLY.

B. Acknowledge and report the annunciator ONLY.

C. Acknowledge and reference the annunciator procedure ONLY.

D. Acknowledge and report the annunciator followed by referencing the annunciator procedure.

Answer:
A. Acknowledge the annunciator ONLY.
Explanation: Section 4.9 - Expected alarms, such as those associated with a Surveillance Procedure or other maintenance activities, or as determined by the CRS, may be identified with a "flag" to signify an expected alarm. This will be communicated to the CRS <u>prior</u> to receiving the alarm. Once "flagged" as an expected alarm for a SP or maintenance activity, the communication of each time the alarm comes in and resets

is **not required** as long as the alarm is received at the appropriate time or is caused by identified maintenance activity. Since the question asks for the MINIMUM actions required – acknowledging the annunciator is the only correct option.

Distracters:

B. This choice is incorrect due to reporting a flagged annunciator as Expected is not required. This choice is plausible due to procedure 2.0.3 (Conduct of Operations) provides guidance for CROs to update the Crew prior to receiving an expected actuation or annunciator that could impact operations and procedure 2.1.3 providing the option to announce the alarm as expected. The candidate that confuses when to update the crew regarding expected actuations or annunciations would choose this answer.

C. This choice is incorrect due to referencing a flagged annunciator procedure not being required. This choice is plausible due to the annunciator procedure being required to be referenced prior to flagging. The candidate that confuses when the annunciator procedure is required to be referenced when flagged would choose this answer.

D. This choice is incorrect due to announcing the flagged alarm and referencing the annunciator procedure not being required. This choice is plausible due to being actions required for an unexpected alarm. The candidate that confuses un-expected vs. expected alarms would choose this answer.

Technical References:
Procedure 2.3.1 (General Alarm Procedure), Rev. 63.
Procedure 2.0.3 (Conduct of Operations), Rev. 88.

References to be provided to applicants during exam: NONE

Learning Objective:
INT032010300E010B Discuss the following as described in Alarm Procedure 2.3.1, General Alarm Procedure: Alarm acknowledgement

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b) 10	
Level of Difficulty:	2	
SRO Only Justification:	N/A	

4.9 Expected alarms, such as those associated with a Surveillance Procedure or other maintenance activities, or as determined by the CRS, may be identified with a "flag" to signify an expected alarm. This will be communicated to the CRS prior to receiving the alarm. Once "flagged" as an expected alarm for a SP or maintenance activity, the communication of each time the alarm comes in and resets is not required as long as the alarm is received at the appropriate time or is caused by identified maintenance activity. If the "flagged" annunciator were to come in at a time not specified in the procedure or alarm was not caused by identified maintenance activity, it is to be treated as an unexpected annunciator.

4.10 During a plant transient, multiple annunciators may be coming in rapidly. The panel Operator will acknowledge the annunciators as soon as reasonably possible, commensurate with the importance of the activities that are on-going. It is not necessary to announce all annunciators that come in during a transient, but operationally significant events that are indicated by annunciators should be announced. Announcements that are made shall be prioritized according to their relative merit to other communications that are on-going.

4.11 Annunciator steps should be performed in sequence unless mitigating circumstances warrant altering the sequence. To support priorities during event mitigation, its acceptable to perform steps out-of-sequence. The procedures may not address all possible plant conditions and therefore, some steps may not apply. If steps are performed out-of-sequence or not performed, the user and CRS or SM shall ensure all applicable steps are performed and procedure intent is not altered. If steps are not performed, justification for non-performance shall be documented/logged. If necessary, an IDOCS request should be entered to evaluate the adequacy of the procedure guidance.

4.12 When an alarm is energized, audio will activate and appropriate window will flash. After determining which annunciator is alarming, press ACKNOWLEDGE button to silence alarm.

4.13 Announce alarm in a timely manner. Back panel alarms may be announced as read from the VID or the Master Alarm Log prior to responding around back or once the Operator has returned from the back panels. Unexpected alarms should be announced by reading or paraphrasing the annunciator descriptor. Expected alarms may be announced as such.

4.14 Enter applicable alarm card. If alarm is repetitive, card has been entered by the acknowledger, cause of alarm understood, and actions are simple and known, then the steps may be performed without continually referring to the card.

4.15 If alarm is repetitive, card has been entered, cause of alarm is understood, and actions are being carried out to correct (i.e., Screen Wash Manual Strainer High D/P alarming in and out), CRS may waive requirement to announce alarm each time alarm is received.

4.16 If alarm card Reference Section contains reference to TS/TRM/ODAM, communicate reference to Shift Manager or CRS.

2.9.2.7 CROs perform walkdowns of Control Room panels. The CRO-RO is responsible for ensuring panel walkdowns are completed as described in Step 7.1.10.

2.9.2.8 CROs should update the Crew prior to receiving an expected actuation or annunciator that could impact operations (examples would include a half scram, initiating CREFS, generator hydrogen low pressure due to venting, etc.).

2.9.2.9 CROs are responsible to keep comprehensive, accurate Control Room Logs that contain sufficient details such that they accurately describe all shift activities. Any major evolution, causes of abnormal conditions, and actions taken to correct abnormal conditions should be documented. Accurate logs can provide useful information in re-constructing events leading to unusual plant occurrences and facilitates good Operator turnover.

2.9.2.10 Regarding protective trip devices and alarms, the CRO should make every effort to understand the cause of any trip or alarm before it is reset.

2.9.2.11 CROs should ensure a thorough shift turnover is given. This means giving a clear and accurate chronology of significant Control Room activities, and the current status of the plant and main control boards.

2.9.2.12 The CROs shall immediately investigate any abnormal conditions found or caused during maintenance activities to determine their potential impact on plant operation. Electrical arcing incidents should receive extra attention since the consequences are not readily visible (e.g., blown fuses).Ⓟ[14]

2.9.2.13 CROs are expected to be aware of key parameters and significant values, and report them when approached or reached. Examples include:

a. All Abnormal/Emergency Procedure entries.

b. All EOP entry conditions.

c. Automatic scram and isolation setpoints.

d. 110°F suppression pool temperature.

e. All rods in.

2.9.2.14 The CRO will be assigned one (1) of three (3) positions as designated at the beginning of shift.

a. CRO - Reactor Operator (RO) (Normally "At-The-Controls" as described in Step 2.9.4).

Examination Outline Cross-Reference	Level	RO
Revised question to distinguish prioritize alarm by color and identify which color alarm may require a plant shutdown or rad release. Incorporated review comments and is now a Modified question.	Tier#	3
	Group#	
	K/A #	G2.4.45
	Rating	4.1
G2.4.45 Ability to prioritize and interpret the significance of each annunciator or alarm. (CFR: 41.10 / 43.5 / 45.3 / 45.12)		

Question 75

The plant is at power when a transient occurs.

- Multiple YELLOW and BLACK outlined annunciators are in alarm.

IAW Procedure 2.3.1 (General Alarm Procedure):

(1) Which colored alarm takes precedence?
(2) Which colored alarm indicates conditions that may require a plant shutdown or radiation release?

A. (1) BLACK;
(2) BLACK.

B. (1) BLACK;
(2) YELLOW.

C. (1) YELLOW;
(2) YELLOW.

D. (1) YELLOW;
(2) BLACK.

Answer:
C. (1) YELLOW; (2) YELLOW.
Explanation: The window box assembly is a matrix of divided lamp windows with engraved legend plates and multi-colored window bezels. Each window has been given a "PRIORITY" signifying the importance of the alarm:

Priority I - RED; alarms that alert of EOP entry conditions or conditions requiring or causing an automatic or manual plant shutdown, or significant system setpoints.

Priority II - YELLOW; alarm conditions which may require or rapidly cause a plant shutdown or radiation release.

Priority III - BLACK; alarms that indicate off normal plant conditions that affect plant or component operability but should not lead to plant shutdown or radiation release.

Distracters:

A. This option is incorrect because YELLOW outlined alarms take precedence over BLACK, and BLACK alarms are NOT expected to lead to a plant shutdown. Plausible that an applicant may confuse the hierarchy of annunciator response between YELLOW and BLACK, and BLACK annunciators do indicate off-normal conditions.
B. This option is incorrect because YELLOW outlined alarms take precedence over BLACK. Plausible that an applicant may confuse the hierarchy of annunciator response between YELLOW and BLACK; for example by overestimating the significance attached to black alarms.
D. This option is incorrect because BLACK alarms are NOT expected to lead to a plant shutdown. This option is plausible because YELLOW outlined alarms do have a higher priority than BLACK, and the applicant may overestimate the significance attached to black alarms.

Technical References:
Procedure 2.3.1 (General Alarm Procedure), Rev. 63

References to be provided to applicants during exam: NONE

Learning Objective:
COR002-35-02, Plant Annunciator System
LO-02 State the purpose of the following components related to the Plant Annunciator System.:
k. Alarm Window Boxes

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank # 74 from 2014 NRC Retake	
	New	
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b) 10	
Level of Difficulty:	3	

SRO Only Justification:	N/A

Question: 74

The plant is at power when a transient occurs.

- Multiple Black and Yellow outlined annunciators are in alarm.

Which colored alarm takes <u>precedence</u> and <u>why</u> IAW Procedure 2.3.1 (General Alarm Procedure)?

A. BLACK; an EOP entry is required ONLY.

B. YELLOW; a plant shutdown condition may be present ONLY.

C. BLACK; BOTH an EOP entry is required AND a plant shutdown condition may be present.

D. YELLOW; BOTH an EOP entry is required AND a plant shutdown condition may be present.

Answer:
B. YELLOW; a plant shutdown condition may be present ONLY.

2.5 Any alarm denoting a plant condition that might result in a hazard to personnel, the environs, or equipment, shall be acknowledged immediately, investigated, and corrective action taken as soon as possible.

2.6 Test all annunciator panels once/day as per Section 6 of this procedure.

PROCEDURE 2.3.1	REVISION 63	PAGE 1 OF 31

annunciator window will go dark and the audible will silence.

1.2.3 Control Room annunciator windows are engraved using black lettering and are color-prioritized as follows:

1.2.3.1 Priority I - Red; alarms that alert of EOP entry conditions, conditions requiring or causing an automatic or manual plant shutdown, or significant system setpoints.

1.2.3.2 Priority II - Yellow; alarm conditions that may require or rapidly cause a plant shutdown or radiation release and may be precursors to Priority 1 alarms.

1.2.3.3 Priority III - Black; alarms that indicate off normal plant conditions that affect plant or component operability but should not lead to plant shutdown or radiation release.

1.2.4 The Control Room Fire Protection Panel has alarm windows that flash either red or white. Area alarm windows that flash red identify locations where safe shutdown or safety-related equipment could be affected. For these areas, the Fire Brigade is dispatched immediately upon receipt of the alarm.

1.2.5 The AOG Control Room alarm panels have windows that flash either red or white. Windows that flash red identify alarms that, after reviewing the status of the system, require bypassing AOG Charcoal Beds or bypassing and securing AOG Train.

U.S. Nuclear Regulatory Commission

Site-Specific SRO Written Examination

Applicant Information	
Name:	
Date:	Facility/Unit: Cooper Nuclear Station
Region: I ☐ II ☐ III ☐ IV ■	Reactor Type: W ☐ CE ☐ BW ☐ GE ■
Start Time:	Finish Time:

Instructions

Use the answer sheets provided to document your answers. Staple this cover sheet on top of the answer sheets. To pass the examination you must achieve a final grade of at least 80.00 percent overall, with 70.00 percent or better on the SRO-only items if given in conjunction with the RO exam; SRO-only exams given alone require a final grade of 80.00 percent to pass. You have 8 hours to complete the combined examination, and 3 hours if you are only taking the SRO portion.

Applicant Certification

All work done on this examination is my own. I have neither given nor received aid.

____
Applicant's Signature

Results	
RO/SRO-Only/Total Examination Values	75 / 25 / 100 Points
Applicant's Score	____ / ____ / ____ Points
Applicant's Grade	____ / ____ / ____ Percent

Examination Outline Cross-Reference	Level	SRO
Revised question to determine TS allowable water level and the thermal limit protected. Annunciator 9-5-2 and OI-8 provide high level setpoint. Added OI-8 reference.	Tier#	1
	Group#	1
	K/A #	295005 AA2.07
	Rating	3.6

295005 Main Turbine Generator Trip
AA2. Ability to determine and/or interpret the following as they apply to MAIN TURBINE GENERATOR TRIP : (CFR: 41.10 / 43.5 / 45.13)
AA2.07 Reactor water level

Question 76

The plant is at rated power when Feedwater Control fails to maximum demand causing reactor water level to rise.

Which one of the following completes the statement below regarding this transient?

According to Tech Spec Bases, Main Turbine Trip is credited to occur before reactor water level exceeds the Tech Spec Allowable Value of less than or equal to ____(1)____ inches to mitigate the reduction in ____(2)____.

A. (1) 42.5
(2) Minimum Critical Power Ratio

B. (1) 42.5
(2) Average Planar Linear Heat Generation Rate

C. (1) 54.0
(2) Minimum Critical Power Ratio

D. (1) 54.0
(2) Average Planar Linear Heat Generation Rate

Answer:
C. (1) 54.0
(2) Minimum Critical Power Ratio

Explanation:
This question requires knowledge of the TS Allowable Value for the Level 8 trip, which

is a surveillance requirement, and TS bases for the Level 8 safety function. The actual setting of the level high turbine trip is 52.5 inches, whereas the TS allowable value is 54 inches. TS Bases 3.3.2.2 states the main turbine trip on reactor water level high, level 8, is credited in the accident analysis for producing and indirect scram for the feedwater controller failure to maximum demand. The indirect scram due to turbine stop/control valve closure mitigates the reduction in MCPR due to increased core inlet subcooling. The allowable value of the level 8 main turbine trip setpoint per TS SR3.3.2.2.2 is ≤54 inches. LHGR actually rises.

Distracters:

Answer A is wrong because it lists the wrong allowable value for the reactor water level high, Level 8 trip. It is plausible because 42.5 inches is the reactor water level high alarm point. The alarm response procedure for the associated alarm contains scram criteria, turbine trip criteria, and reactor feed pump trip criteria in the event level cannot be maintained below 50 inches. The unprepared student could confuse the high water level alarm and trip setpoints, since both values are familiar and each can be related to main turbine trip, reactor feed pump trip, and reactor scram.

Answer B is wrong because it lists the wrong allowable value for the reactor water level high, Level 8 trip and does not mitigate the reduction in APLHGR. It is plausible because 42.5 inches is the reactor water level high alarm point. The alarm response procedure for the associated alarm contains scram criteria, turbine trip criteria, and reactor feed pump trip criteria in the event level cannot be maintained below 50 inches AND commonly confusing APLHGR change with MCPR. The candidate that confuses the high water level alarm vs. trip setpoints and impact on APLHGR vs. MCPR would select this answer.

Answer D is incorrect because the Level 8 trip does not mitigate the reduction in APLHGR. It is plausible because APLHGR vs. MCPR change is commonly confused. The candidate that correctly identifies the high water level trip setpoint and confuses the impact on APLHGR vs. MCPR would select this answer.

Technical References:

TS 3.3.2.2 and Bases

Procedure 2.3_9-5-2 (Panel 9-5 - Annunciator 9-5-2), Rev. 44

References to be provided to applicants during exam: none

Learning Objective:

INT00705040010200 Discuss the applicable Safety Analysis in the Bases associated with each Section 3.3 Specification.

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X

Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.43(b)(2)	
Level of Difficulty:	3	
SRO Only Justification:		
This question requires knowledge of Tech Spec SR3.3.2.2.2 and Bases 3.3.2.2. The actual setting of the level high turbine trip is 52.5 inches. This question requires knowledge of the TS Allowable Value, which is a surveillance requirement, and thus, SRO only.		

Operations Instruction #8 GUIDELINE FOR SUCCESSFUL TRANSIENT MITIGATION	Class: Information Use Effective: 06/10/15

REACTOR LEVEL	AUTO ACTION
52.5" (Tech Spec ≤ 54")	RFP Trip RCIC shutdown due to RCIC-MO-131 closure signal HPCI Trip Main Turbine Trip
42.5"	High Level Alarm
27.5"	Low Level Alarm RR pump runback if either RFP feed flow is ≤ to 1 Mlbm/hr or a RFP is tripped and total steam flow > 9 Mlbm/hr
7.81" (Tech Spec ≥ 3")	Reactor Scram Group 2 Isolation ADS Logic Confirmatory
-33.43" (Tech Spec ≥ -42")	ARI Initiation Group 3 and 6 Isolations HPCI and RCIC Initiation
-33.43" after 9 second time delay (Tech Spec ≥ -42")	ATWS RPT
-104.39" (Tech Spec ≥ -113")	Group 1 and Group 7 Isolations ADS Timers Start CS, RHR, and Diesel Generator Initiation Drywell FCUs Trip
≥ -185.59" Fuel Zone (Tech Spec ≥ -193.19")	Containment Spray Permissive (2/3 core height interlock)

Feedwater and Main Turbine High Water Level Trip Instrumentation
3.3.2.2

3.3 INSTRUMENTATION

3.3.2.2 Feedwater and Main Turbine High Water Level Trip Instrumentation

LCO 3.3.2.2 Three channels of feedwater and main turbine high water level trip instrumentation shall be OPERABLE.

APPLICABILITY: THERMAL POWER ≥ 25% RTP.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One feedwater and main turbine high water level trip channel inoperable.	A.1 Place channel in trip.	7 days
B. Two or more feedwater and main turbine high water level trip channels inoperable.	B.1 Restore feedwater and main turbine high water level trip capability.	2 hours
C. Required Action and associated Completion Time not met.	C.1 Reduce THERMAL POWER to < 25% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

-------------------------------------NOTE-------------------------------------
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 feedwater and main turbine high water level trip capability is maintained.

SURVEILLANCE		FREQUENCY
SR 3.3.2.2.1	Perform CHANNEL CHECK.	24 hours
SR 3.3.2.2.2	Perform CHANNEL CALIBRATION. The Allowable Value shall be ≤ 54.0 inches.	24 months
SR 3.3.2.2.3	Perform LOGIC SYSTEM FUNCTIONAL TEST including valve actuation.	24 months

B 3.3 INSTRUMENTATION

B 3.3.2.2 Feedwater and Main Turbine High Water Level Trip Instrumentation

BASES

BACKGROUND

The feedwater and main turbine high water level trip instrumentation is designed to detect a potential failure of the Feedwater Level Control System that causes excessive feedwater flow.

With excessive feedwater flow, the water level in the reactor vessel rises toward the high water level, Level 8 reference point, causing the trip of the two feedwater pump turbines and the main turbine.

Reactor Vessel Water Level-High, Level 8 signals are provided by level sensors that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level in the reactor vessel (variable leg). Three channels of Reactor Vessel Water Level-High, Level 8 instrumentation are provided as input to a two-out-of-three initiation logic that trips the two feedwater pump turbines and the main turbine. Each channel consists of a level transmitter loop and a trip relay that compares measured input signals with pre-established setpoints. When the setpoint is exceeded, the channel outputs a main feedwater and main turbine trip signal to the trip logic.

A trip of the feedwater pump turbines limits further increase in reactor vessel water level by limiting further addition of feedwater to the reactor vessel. A trip of the main turbine and closure of the stop and control valves protects the turbine from damage due to water entering the turbine.

APPLICABLE SAFETY ANALYSES

The feedwater and main turbine high water level trip instrumentation is assumed to be capable of providing a turbine trip in the design basis transient analysis for a feedwater controller failure, maximum demand event (Ref. 1). The Level 8 trip indirectly initiates a reactor scram from the main turbine trip (above 30% RTP) and trips the feedwater pumps, thereby terminating the event. The reactor scram mitigates the reduction in MCPR.

Feedwater and main turbine high water level trip instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 2).

REACTOR HIGH WATER LEVEL	PANEL/WINDOW: 9-5-2/F-1

1. AUTOMATIC ACTIONS
   1.1 RFPT operation to control reactor level.
2. OPERATOR OBSERVATION AND ACTION
   2.1 Check reactor level by use of other level instruments.
   2.2 Check RFP(s) operating properly.
   2.3 IF RPV level cannot be maintained below +50" on narrow range, THEN perform following:
      2.3.1 SCRAM and enter Procedure 2.1.5.
      2.3.2 Ensure following not operating:
         2.3.2.1 Main Turbine.
         2.3.2.2 Both RFPs.
         2.3.2.3 HPCI.
         2.3.2.4 RCIC.
   2.4 IF RPV level cannot be maintained below +90" on NBI-LI-92, STEAM NOZZLE LVL, close, THEN perform following:
      2.4.1 Inboard MSIVs.
      2.4.2 MS-MO-74, INBD ISOL VLV.
      2.4.3 HPCI-MO-15, STM SUPP ISOL VLV.
      2.4.4 RCIC-MO-15, INBD STM SUPP ISOL VLV.
   2.5 IF annunciator due to level control malfunction, THEN enter Procedure 2.4RXLVL.

Scram Actions

Scram Actions

PROCEDURE 2.3_9-5-2	REVISION 44	PAGE 61 OF 91

SETPOINT	CIC	9-5-2/F-1
(2730) 42.5"	RFC-R-LRPR97	

PROBABLE CAUSES
- Reactor feed control malfunction.
- Condensate and/or feedwater component malfunction.

REFERENCES
- General Operating Procedure 2.1.5, Reactor Scram.
- Abnormal Procedure 2.4RXLVL, RPV Water Level Control Trouble.

Examination Outline Cross-Reference	Level	SRO
Removed TS 3.3.1.1 reference from exam. Provided TS 3.3.1.1 Condition A & C conditions as part of the stem. Added 92 days to stem.	Tier#	1
	Group#	1
	K/A #	295006, G2.2.37
	Rating	4.6
295006 SCRAM 2.2.37 Ability to determine operability and/or availability of safety related equipment. (CFR: 41.7 / 43.5 / 45.12)		

Question 77

The plant is at rated power.

It is discovered that **quarterly** surveillance (92 days) 6.2RPS.301 {Manual Scram Functional Test (DIV 2)} has exceeded its Tech Spec late date due to an administrative error.

Which of the following statements identifies the MINIMUM Tech Spec requirements regarding RPS operability pending performance of the surveillance?

TS 3.3.1.1:

- *Condition A - One or more required channels inoperable.*
- *Condition C - One or more Functions with RPS trip capability not maintained.*

A. ONLY TS 3.3.1.1 Condition A must be immediately entered.

B. BOTH TS 3.3.1.1 Condition A and Condition C must be entered immediately.

C. RPS may be considered operable for up to 92 days from discovery only if an acceptable risk evaluation is performed within 24 hours and the risk managed.

D. RPS may be considered operable for up to 115 days from discovery only if an acceptable risk evaluation is performed within 24 hours and the risk managed.

Answer: C. RPS may be considered operable for up to 92 days from discovery only if an acceptable risk evaluation is performed within 24 hours and the risk managed.
Explanation:

This question requires a determination of operability given a missed RPS surveillance. An exception to SR 3.0.1, SR 3.0.3 states when failure to comply with a surveillance frequency is discovered, declaring the LCO not met may be delayed for 24 hours or up to the surveillance frequency, whichever is greater, to allow time to perform the surveillance. If delayed greater than 24 hours, a risk evaluation must be performed and the risk managed. The surveillance in question has a 92 day frequency. Therefore, the minimum requirement is to perform a risk evaluation within 24 hours to allow delaying entry into TS 3.3.1.1 action for up to 92 days, and answer C is correct.

Distracters:

Answers A and B are plausible since the unprepared candidate may assume failure to comply with the specified surveillance frequency is the same as failing a surveillance requirement and requires immediately declaring the equipment inoperable per SR 3.0.1. Answer A reflects TS 3.3.1.1 Condition A, which is plausible because only one channel per RPS trip system exists for the manual scram function. Answer B is also plausible because only one channel per RPS trip system exists for the manual scram function, and trip capability would be lost if only one channel was inoperable and untripped, requiring both Conditions A and C. Both Answers A and B are wrong because the stem asks for the minimum requirement, which would be to delay entry into TS 3.3.1.1 actions as allowed by SR 3.0.3 and SR 3.0.1.

Answer D is plausible because the unprepared student may attempt to apply surveillance frequency extension per SR 3.0.2, 1.25 times the specified frequency, which is inappropriate in this situation. The 115 days in answer D is derived by multiplying 92 times 1.25.

Technical References:

SR 3.0.1, SR 3.0.3, TS 3.3.1.1, TS SR 3.3.1.1.9;
6.2RPS.301, Manual Scram Functional Test (DIV 2)

References to be provided to applicants during exam: none

Learning Objective:

INT00705010010200 Given plant conditions and a Specification, apply the rules of Section 3.0 to determine appropriate actions.

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X

Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X

10CFR Part 55 Content:	
	55.43(b)(2)
Level of Difficulty:	3

SRO Only Justification:	
This requires application of Tech Spec SR 3.0.3.	

3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

SR 3.0.1 SRs shall be met during the MODES or other specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO. Failure to perform a Surveillance within the specified Frequency shall be failure to meet the LCO except as provided in SR 3.0.3. Surveillances do not have to be performed on inoperable equipment or variables outside specified limits.

SR 3.0.2 The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met.

For Frequencies specified as "once," the above interval extension does not apply.

If a Completion Time requires periodic performance on a "once per . . ." basis, the above Frequency extension applies to each performance after the initial performance.

Exceptions to this Specification are stated in the individual Specifications.

SR 3.0.3 If it is discovered that a Surveillance was not performed within its specified Frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, whichever is greater. This delay period is permitted to allow performance of the Surveillance. A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed.

(continued)

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.

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. OR A.2 Place associated trip system in trip.	12 hours 12 hours
B. One or more Functions with one or more required channels inoperable in both trip systems.	B.1 Place channel in one trip system in trip. OR B.2 Place one trip system in trip.	6 hours 6 hours
C. One or more Functions with RPS trip capability not maintained.	C.1 Restore RPS trip capability.	1 hour

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
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	Adjust the channel to conform to a calibrated flow signal.	31 days
SR 3.3.1.1.8	Calibrate the local power range monitors.	1000 MWD/T average core exposure
SR 3.3.1.1.9	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.1.1.10 .	-----NOTES----- 1. Neutron detectors and recirculation loop flow transmitters are excluded. 2. For Function 2.a, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. ----- Perform CHANNEL CALIBRATION.	184 days

(continued)

Table 3.3.1.1-1 (page 3 of 3)
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
7. Scram Discharge Volume Water Level - High					
a. Level Transmitter	1,2	2	G	SR 3.3.1.1.4 SR 3.3.1.1.9 SR 3.3.1.1.12(a,b) SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 90 inches
	5(c)	2	H	SR 3.3.1.1.4 SR 3.3.1.1.9 SR 3.3.1.1.12 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 90 inches
b. Level Switch	1,2	2	G	SR 3.3.1.1.4 SR 3.3.1.1.9 SR 3.3.1.1.12 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 90 inches
	5(c)	2	H	SR 3.3.1.1.4 SR 3.3.1.1.9 SR 3.3.1.1.12 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 90 inches
8. Turbine Stop Valve — Closure	≥ 29.5% RTP	2	E	SR 3.3.1.1.4 SR 3.3.1.1.9 SR 3.3.1.1.12 SR 3.3.1.1.13 SR 3.3.1.1.14 SR 3.3.1.1.15	≤ 10% closed
9. Turbine Control Valve Fast Closure, DEH Trip Oil Pressure — Low	≥ 29.5% RTP	2	E	SR 3.3.1.1.4 SR 3.3.1.1.9 SR 3.3.1.1.12(a,b) SR 3.3.1.1.13 SR 3.3.1.1.14 SR 3.3.1.1.15	≥ 1018 psig
10. Reactor Mode Switch — Shutdown Position	1,2	1	G	SR 3.3.1.1.11 SR 3.3.1.1.13	NA
	5(c)	1	H	SR 3.3.1.1.11 SR 3.3.1.1.13	NA
11. Manual Scram	1,2	1	G	SR 3.3.1.1.9 SR 3.3.1.1.13	NA
	5(c)	1	H	SR 3.3.1.1.9 SR 3.3.1.1.13	NA

(a) 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.

(b) 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 Limiting Trip Setpoint and the methodologies used to determine the as-found and the as-left tolerances are specified in the Technical Requirements Manual.

(c) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

(from 6.2RPS.301)

.ATTACHMENT 2 INFORMATION SHEET

1. SURVEILLANCE REQUIREMENTS - TECHNICAL SPECIFICATIONS

   1.1 This procedure satisfies the requirements of SR 3.3.1.1.9 for Table 3.3.1.1-1, Function 11.

2. DISCUSSION

   2.1 This procedure provides steps to test the manual scram function by manually initiating a half scram.

   2.2 When Procedure 6.RPS.301 is performed, this procedure is not required to be performed.

3. REFERENCES

   3.1 TECHNICAL SPECIFICATIONS

      3.1.1 LCO 3.3.1.1, Reactor Protection System (RPS) Instrumentation.

   3.2 UPDATED SAFETY ANALYSIS REPORT

      3.2.1 Section VII-2.0, Reactor Protection System.

   3.3 DRAWINGS

      3.3.1 GE Drawing 791E256, RPS.

   3.4 VENDOR MANUALS

      3.4.1 CNS Number 1750, GE Type CR2940.

   3.5 PROCEDURES

      3.5.1 Surveillance Procedure 6.RPS.301, Mode Switch in Shutdown, SDV Valve Timing, and Manual Scram Functional Test.

   3.6 MISCELLANEOUS

      3.6.1 GEK-34560, RPS.

      3.6.2 Ⓟ[1] CR-CNS-2008-5767 CA-84, Add Risk Significant Flag. Affects Risk Flag.

Examination Outline Cross-Reference	Level	SRO
Question asks for the "REQUIRED" procedure to mitigate. Entry conditions ONLY exist for 5.2REC and is therefore the only procedure required to be utilized IAW conduct of operations. Added highlight reference 2.0.1.2 and Corrected highlighted reference.	Tier#	1
	Group#	1
	K/A #	295018, AA2.03
	Rating	3.5

295018 Partial or Total Loss of CCW

AA2. Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER : (CFR: 41.10 / 43.5 / 45.13)

AA2.03 Cause for partial or complete loss

Question 78

The plant is operating at 100% power with 'A' REC Heat Exchanger in service and 'B' REC Heat Exchanger in standby when the following conditions occur:

- SW Pressure on both Divisions has risen but is still in the green band.
- REC system pressure is steady and is in the green band.
- REC Surge Tank Level High alarms.
- RR MG Set Oil Temperatures are rising.
- Drywell temperature and pressure are rising.
- RWCU F/D Inlet Temp High alarms.

Alarm Procedures for REC Surge Tank Level High and RWCU F/D Inlet Temp High have been entered.

Which of the following identifies the cause of these conditions and the REQUIRED transition of procedures by the CRS to mitigate the condition?

A. SW-TCV-451A, REC HX A SW OUTLET TCV has closed. Transition from Alarm Procedures to 5.2REC (Loss of REC) to shift REC heat exchangers.

B. REC-MOV-712, REC HX A OUTLET VALVE has closed. Transition from Alarm Procedures to 5.2REC (Loss of REC) to shift REC heat exchangers.

C. SW-TCV-451A, REC HX A SW OUTLET TCV has closed. Transition from Alarm Procedures to 5.2SW (Service Water Casualties) to shift REC heat exchangers.

D. REC-MOV-712, REC HX A OUTLET VALVE has closed. Transition from Alarm Procedures to 5.2SW (Service Water Casualties) to shift REC heat exchangers.

Answer:
A. SW-TCV-451A, REC HX A SW OUTLET TCV has closed. Transition from Alarm Procedures to 5.2REC (Loss of REC) to shift REC heat exchangers.
Explanation: A loss of SW flow to the REC Heat Exchanger recovery is covered in both 5.2REC and 5.2SW (assumes loss of SW pumps or piping). If the pressure of the service water system lowers to < 38 psig the system will isolate non-critical loads. The subsequent steps will have the Operators place the other loops REC Heat Exchanger in service. For the given condition, SW Pressure rising indicates that there was some restriction in the flow path. 5.2SW shift is to use a good SW loop vs. 5.2REC due to REC cooling issues. The actions for shifting REC HXs in 5.2SW allow for bypassing Group 6 signal to SW-MO-650 (REC HX A SERVICE OUTLET), opening REC-19 & 21 (REC HX INLETs), and transferring REC-TIC-451B to MANUAL. These steps differ from the steps in 5.2REC due to shifting heat exchangers for different reasons and would not be appropriate (see highlighted differences provided).
Distracters: B. This answer is incorrect because the REC outlet valve closing will not cause SW pressure to rise and 5.2SW not being the procedure utilized to shift heat exchangers under the provided conditions. This answer is plausible because closure of this valve would provide the other indications and if the stem were changed to reflect REC pressure rising would be correct and 5.2SW provides guidance to shift (use a good SW loop vs. 5.2REC due to REC cooling issues). The candidate who confuses indications provided and which procedure provides the correct guidance would select this answer. C. This answer is incorrect because 5.2SW is not the procedure utilized to shift heat exchangers under the provided conditions. This answer is plausible because 5.2SW provides guidance to shift (use a good SW loop vs. 5.2REC due to REC cooling issues). The candidate who correctly identifies the cause and confuses which procedure provides the correct guidance would select this answer. D. This answer is incorrect because the REC outlet valve closing will not cause SW pressure to rise. This answer is plausible because closure of this valve would provide the other indications and if the stem were changed to reflect REC pressure rising would be correct along. The candidate who confuses indications provided and correctly identifies the procedure providing the correct guidance would select this answer.

Technical References: Emergency Procedure 5.2REC, Loss of REC, Rev. 16 Emergency Procedure 5.2SW, Service Water Casualties, Rev. 24. Procedure 2.0.1.2 (Operations Procedure Policy), Rev. 44		
References to be provided to applicants during exam: none		
Learning Objective: INT0320126L0L0100 Given plant condition(s), determine from memory the appropriate Abnormal/Emergency Procedure(s) to be utilized to mitigate the event(s).		
Question Source:	Bank # last NRC Exam #77, 2014 NRC retake	X
(note changes; attach parent)	Modified Bank	
	New	
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:		
	55.43(b)(5)	
Level of Difficulty:	3	
SRO Only Justification:		
10CFR55.43 b (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. Requires assessment of plant conditions and selection of procedure to mitigate the conditions.		

Question: 77

The plant is operating at 100% power when the following conditions occur:

- SW Pressure on both Divisions has risen but is still in the green band.
- REC system pressure is steady and is in the green band.
- REC Surge Tank Level High alarms.
- RR MG Set Oil Temperatures are rising.
- Drywell temperature and pressure are rising.
- RWCU F/D Inlet Temp High alarms.

Which of the following identifies the cause of these conditions and the required action to correct the problem IAW Procedure 2.0.1.2 (Operations Procedure Policy)?

The REC Heat Exchanger…

A. SW Outlet valve has closed, shift REC heat exchangers IAW 5.2REC (Loss of REC).

B. REC Outlet valve has closed, shift REC heat exchangers IAW 5.2REC (Loss of REC).

C. SW Outlet valve has closed, shift REC heat exchangers IAW 5.2SW (Service Water Casualties).

D. REC Outlet valve has closed, shift REC heat exchangers IAW 5.2SW (Service Water Casualties).

Answer:

A. SW Outlet valve has closed, shift REC heat exchangers IAW 5.2REC (Loss of REC).

be capitalized and scram actions bolded.ⓟ

6.3 ENTRY AND EXIT CONDITIONS

6.3.1 This section is used to list plant conditions or indications that are indicative of expected abnormal operational conditions or transients.

6.3.2 APs and EPs shall be entered from any of following:

6.3.2.1 When directed by another plant procedure.

6.3.2.2 When abnormal or emergency plant conditions are consistent with Procedure Entry Conditions:

a. Entry conditions are formatted as a list. Generically, if any entry condition is met, the procedure should be entered unless the entry condition specifies entry based on a logic term (e.g., "and", "or", "if", "if not", and "when").

PROCEDURE 2.0.1.2	REVISION 44	PAGE 5 OF 15

6.3.3 APs and EPs may be entered without a specific entry condition if guidance is useful in mitigating a degraded plant condition.

6.3.4 APs and EPs may be exited by any of following:

6.3.4.1 When directed by procedure.

6.3.4.2 When all applicable steps have been completed.

6.3.4.3 When conditions that required entry no longer exist and plant conditions are stable.

CNS OPERATIONS MANUAL EMERGENCY PROCEDURE 5.2SW SERVICE WATER CASUALTIES	USE: CONTINUOUS QUALITY: QAPD RELATED EFFECTIVE: 8/7/13 APPROVAL: ITR-RDM OWNER: OSG SUPV DEPARTMENT: OPS

P&L 1

1. ENTRY CONDITIONS

1.1 Service water header pressure lowering.

1.2 SW-MO-36, LOOP CROSSTIE VLV, and/or SW-MO-37, LOOP CROSSTIE VLV, automatic isolation.

1.3 Indication/report of service water piping failure.

1.4 Service water pump trip causing low system pressure isolation.

1.5 River level ≤ 873' or forecasted to lower to ≤ 873'.

Scram Actions

4.11 IF only one loop of SW available, THEN perform following:

4.11.1 Ensure REC HX in service for the operating SW loop by entering Attachment 1 (Page 6).

4.11.2 Perform following, as necessary, to restore cooling to non-critical header:

4.11.2.1 Close SW-1490, NON-CRIT HEADER ISOLATION (SW Pump Room west).

4.11.2.2 Ensure system pressure > 38 psig in active loop on SW-PI-2715A(B).

4.11.2.3 Open SW-MO-36 or SW-MO-37 in active loop.

4.11.2.4 Maintain system pressure > 38 psig as follows:

a. Start available SW pump(s) as necessary.

b. IF SW pump(s) will not start, THEN perform following:

1. Ensure control switch of affected pump in NORMAL AFTER STOP (green flagged).

2. Place affected pump MODE SELECTOR switch to MAN.

3. Start pump.

4. Ensure MODE SELECTOR switch(es) aligned per Procedure 2.2.71.

c. Slowly throttle open SW-1490 while maintaining system pressure > 38 psig.

d. Ensure MODE SELECTOR switch(es) aligned per Procedure 2.2.71.

PROCEDURE 5.2SW	REVISION 24	PAGE 3 OF 20

CNS OPERATIONS MANUAL EMERGENCY PROCEDURE 5.2REC LOSS OF REC	USE: CONTINUOUS QUALITY: QAPD RELATED EFFECTIVE: 7/18/13 APPROVAL: ITR-RDM OWNER: OSG SUPV DEPARTMENT: OPS

P&I: 1

Scram Actions

1. ENTRY CONDITIONS

1.1 REC HEADER PRESSURE ≤ 62 psig.

1.2 Rising temperatures on equipment cooled by REC.

1.3 Multiple low REC flow alarms on VBD-M.

1.4 Multiple REC pump alarms on VBD-M.

2. AUTOMATIC ACTIONS

4.9 IF SW cooling lost to single REC HX, THEN perform following:

4.9.1 IF available, THEN place standby REC HX in service per Attachment 3 (Page 7).

4.9.2 Monitor REC HX outlet temperature from REC-TI-452, REC HEADER TEMPERATURE, or PMIS Point M136 if REC-TI-452 not available.

4.9.3 IF REC HX outlet temperature approaches 98°F, THEN reduce REC heat load with one or both of following:

4.9.3.1 Reduce reactor power, as necessary, to maintain REC HX outlet temperature to ≤ 98°F per Procedure 2.1.10.

4.9.3.2 Rapidly remove RWCU from service per Procedure 2.2.66.

4.9.4 IF at any time REC HX outlet temperature cannot be maintained ≤ 98°F, THEN shut down per Procedure 2.1.4.

4.10 IF REC temperature indication is not available, REC non-critical loads are in service and a PCIS Group 2 isolation due to a primary coolant leak is present, THEN perform one of the following:

4.10.1 Maintain SW flow through REC HX A(B) ≥ 3500 gpm.

4.10.2 Isolate non-critical REC headers.

4.11 IF REC to Non-Regen HX isolated, THEN rapidly remove RWCU from service per Procedure 2.2.66.

4.12 IF REC cooling to FPC HXs lost, THEN enter Procedure 2.4FPC.

ATTACHMENT 3 PLACING STANDBY REC HX IN SERVICE

1. PLACING STANDBY REC HX IN SERVICE

1.1 Place A REC HX in service as follows:

1.1.1 Ensure SW-TCV-451A, REC HX A SW OUTLET TEMPERATURE CONTROL, switch in OPEN.

1.1.2 While maintaining SW header pressure ≥ 38 psig, slowly throttle open SW-MO-650, REC HX A SERVICE WATER OUTLET, to desired flow rate.

1.1.3 While maintaining REC HEADER PRESSURE ≥ 62 psig, slowly throttle open REC-MO-712, HX A OUTLET VLV.

1.1.4 Ensure REC-MO-713, HX B OUTLET VLV, closed.

1.1.5 Ensure SW-MO-651, REC HX B SERVICE WATER OUTLET, closed.

1.1.6 Position tags to indicate REC HX A is in service and REC HX B is in STANDBY.

1.2 Place B REC HX in service as follows:

1.2.1 Ensure SW-TCV-451B, REC HX B SW OUTLET TEMPERATURE CONTROL, switch in OPEN.

1.2.2 While maintaining SW header pressure ≥ 38 psig, slowly throttle open SW-MO-651, REC HX B SERVICE WATER OUTLET, to desired flow rate.

1.2.3 While maintaining REC HEADER PRESSURE ≥ 62 psig, slowly throttle open REC-MO-713, HX B OUTLET VLV.

1.2.4 Ensure REC-MO-712, HX A OUTLET VLV, closed.

1.2.5 Ensure SW-MO-650, REC HX A SERVICE WATER OUTLET, closed.

1.2.6 Position tags to indicate REC HX B in service and REC HX A in STANDBY.

1.1.1.7 Position tags to indicate REC HX A in service and REC HX B in standby.

1.1.2 Place REC HX B in service as follows:

1.1.2.1 Ensure SW-TCV-451B, REC HX B SW OUTLET TEMPERATURE CONTROL, switch in OPEN.

1.1.2.2 Throttle SW-MO-651, REC HX B SERVICE OUTLET, to maintain SW Subsystem B pressure ≥ 38 psig on SW-PI-2715B or until flow through valve is 400 gpm if SW pressure < 38 psig and REC non-critical loops are isolated.

a. IF 400 gpm desired and cannot be obtained with a Group 6 isolation, present, THEN perform following:

1. In Isolation Relay Cabinet B (Cable Spreading Room NE wall), lift and tape Lead TB-1, Terminal 8/SW 651-30.

2. Throttle SW-MO-651 to maintain SW Subsystem B flow of 400 gpm.

1.1.2.3 While maintaining REC header pressure ≥ 62 psig on REC-PI-452, slowly throttle open REC-MO-713, HX B OUTLET VLV; N/A if both REC HXs in service.

1.1.2.4 IF both REC HXs in service, THEN perform following:

a. Open REC-19, REC HX B INLET (R-931-N).

b. Open REC-21, REC HX A INLET (R-931-N).

PROCEDURE 5.2SW	REVISION 24	PAGE 7 OF 20

1.1.2.5 Ensure REC-MO-712, HX A OUTLET VLV, closed.

1.1.2.6 Ensure SW-MO-650, REC HX A SERVICE WATER OUTLET, closed.

CAUTION – Transferring REC-TIC-451A from AUTO to MANUAL is not a bumpless transfer. SW-TCV-451A may change position considerably during transfer.

a. IF Group 6 isolation signal present, at REC-TIC-451A, SW TO REC HX A (R-931-NE), THEN place controller to MANUAL as follows and close:

1. Depress BLUE button in lower left corner.

NOTE – Mode will automatically change about 2 seconds after mode is selected.

2. Depress UP arrow or DOWN arrow to switch from AUTO to MANUAL.

3. Depress BLUE button in lower left corner to transfer to MANUAL, if required.

NOTE – Lower display window indicating -100.0 = Full Open (0.0 = Full Closed).

4. Verify lower display window indicates manual percent output.

5. Using DOWN arrow, close SW-TCV-451A.

6. At VBD-M, mark REC-TIC-451A MODE label to indicate it is in MANUAL.

1.1.2.7 Position tags to indicate REC HX B in service and REC HX A in standby.

1.2 Use demineralized water, if necessary, to cool one REC heat exchanger.

1.2.1 Close SW-195, SW/DW CROSSTIE TELL-TALE DRAIN (R-931-N).

1.2.2 Open DW-145, REC HXS SUPPLY (R-931-N).

1.2.3 IF cooling REC HX A, THEN perform following:

1.2.3.1 Ensure SW-132, REC HX A INLET, closed (R-931-N).

1.2.3.2 Ensure SW-133, REC HX A OUTLET, closed (R-931-N).

1.2.3.3 Ensure SW-135, REC HX A BACKWASH OUTLET, closed (R-931-N).

1.2.3.4 Open SW-136, SW/DW CROSSTIE, REC HX A FLUSH (R-931-N).

6.1.7 AP and EP steps should be performed in sequence unless mitigating circumstances warrant altering the sequence. To support priorities during event mitigation, it's acceptable to perform steps out of sequence. The procedures are typically written given the plant is at 100% power. Therefore, some actions in the procedure that are performed at power may not be applicable in other modes of operation (e.g., tripping the main turbine and scramming the reactor). The procedures may not address all possible plant conditions and therefore, some steps may not apply. If steps are performed out of sequence or not performed, the user and CRS or SM shall ensure all applicable steps are performed and procedure intent is not altered. If steps are not performed, justification for non-performance shall be documented/logged. If necessary, an IDOCS request should be entered to evaluate the adequacy of the procedure guidance.Ⓟ[5]

6.3 ENTRY AND EXIT CONDITIONS

6.3.1 This section is used to list plant conditions or indications that are indicative of expected abnormal operational conditions or transients.

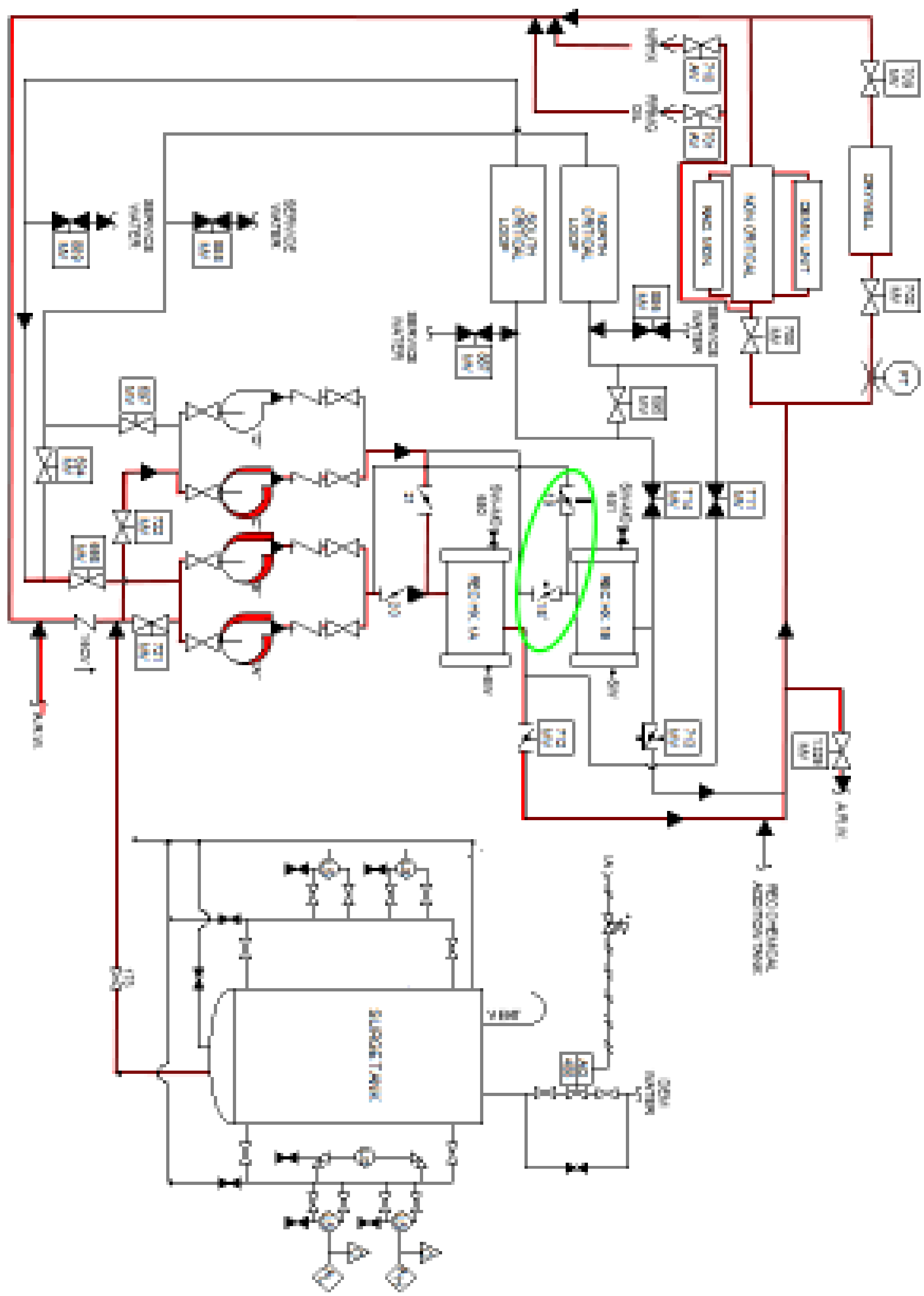
6.3.2 APs and EPs shall be entered from any of following:

6.3.2.1 When directed by another plant procedure.

6.3.2.2 When abnormal or emergency plant conditions are consistent with Procedure Entry Conditions:

a. Entry conditions are formatted as a list. Generically, if any entry condition is met, the procedure should be entered unless the entry condition specifies entry based on a logic term (e.g., "and", "or", "if", "if not", and "when").

6.3.3 APs and EPs may be entered without a specific entry condition if guidance is useful in mitigating a degraded plant condition.



REC SYSTEM Normal Power Operation
Figure 1, Rev. 11
COR002-19

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Examination Outline Cross-Reference	Level	SRO
Revised question to determine when a scram is required based upon IA pressure and when Attachment 2 (IA Pressure Loss) is implemented.	Tier#	1
	Group#	1
	K/A #	295019,G2.1.7
	Rating	4.7

295019 Partial or Total Loss of Inst. Air

G2.1.7 Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation. (CFR: 41.5 / 43.5 / 45.12 / 45.13)

Question 79

The plant is operating at rated power when a leak on the Instrument Air (IA) header occurs in the Reactor Building.

Which one of the following completes the statement below regarding the **HIGHEST** Instrument Air pressure that requires performing Procedure 2.1.5 (Reactor Scram) and when Attachment 2 (IA Pressure Loss) is required to be performed IAW 5.2AIR Procedure (Loss of Instrument Air)?

Entry into Procedure 2.1.5 is required if IA pressure lowers below ____(1)___.
Attachment 2 (IA Pressure Loss) is required to be performed ____(2)____.

A. (1) 85 psig
(2) anytime Procedure 5.2AIR is entered

B. (1) 85 psig
(2) ONLY when system pressure is considered to be too low to support continued operation

C. (1) 77 psig
(2) anytime Procedure 5.2AIR is entered

D. (1) 77 psig
(2) ONLY when system pressure is considered to be too low to support continued operation

Answer:
D. (1) 77 psig

(2) ONLY when system pressure is considered to be too low to support continued operation

Explanation:
This question requires knowledge of AOP subsequent actions and when to implement abnormal procedure attachments.

Procedure 5.2AIR requires the CRS to enter Procedure 2.1.5, Reactor Scram, which implements reactor shutdown, when air header pressure is $\leq$ 77 psig based on the operating characteristics of systems supplied by instrument air. (Service Air automatically isolates <77 psig). Attachment 2 is also required to be implemented concurrently with the 5.2AIR body when IA pressure lowers below 77 psig which is the pressure considered too low to support continued operation.

Distracters:
Answer A is incorrect due to IA pressure being $\leq$ 77 psig requires entry into procedure 2.1.5 and Attachment 2 only being required to be implemented when IA pressure is considered too low to support continued operation. This choice is plausible due to $\leq$ 85 psig being one of the lowering IA pressure milestones in the supplemental actions AND confusing Attachment 2 title (IA Pressure Loss) with when to implement the attachment. The candidate that confuses IA lowering pressure milestones and confuses when to implement Attachment 2 would select this answer.
Answer B is incorrect due to IA pressure being $\leq$ 77 psig requires entry into procedure 2.1.5. This choice is plausible due to $\leq$ 85 psig being one of the lowering IA pressure milestones in the supplemental actions. The candidate that confuses IA lowering pressure milestones and recalls when to implement Attachment 2 would select this answer.
Answer C is incorrect due to Attachment 2 only being required to be implemented when IA pressure is considered too low to support continued operation. This choice is plausible due to confusing Attachment 2 title (IA Pressure Loss) with when to implement the attachment. The candidate that correctly recalls the IA lowering pressure milestones and confuses when to implement Attachment 2 would select this answer.

Technical References: 5.2AIR, Loss of Instrument Air.

References to be provided to applicants during exam: none

Learning Objective:
COR0011702001070A Given a specific Plant Air system malfunction, determine the effect on any of the following: a. Plant operation

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X

Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:		
	55.43(b)(5)	
Level of Difficulty:	2	
SRO Only Justification:		
Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps.		

CNS OPERATIONS MANUAL EMERGENCY PROCEDURE 5.2AIR LOSS OF INSTRUMENT AIR	USE: CONTINUOUS QUALITY: QAPD RELATED EFFECTIVE: 5/4/15 APPROVAL: ITR-RDM OWNER: OSG SUPV DEPARTMENT: OPS

Scram Actions

1. ENTRY CONDITIONS

1.1 INSTRUMENT AIR PRESSURE below green band and does not recover back into green band.

1.2 SERVICE AIR PRESSURE below green band and does not recover back into green band.

2. AUTOMATIC ACTIONS

2.1 1st Standby SACs loads when system pressure 93 to 105 psi.

2.2 2nd Standby SACs loads when system pressure 90 to 100 psi.

2.3 SA-PCV-609, SERVICE AIR SYSTEM ISOLATION, closes when service air pressure < 77 psig.

3. IMMEDIATE OPERATOR ACTIONS

3.1 None.

4. SUBSEQUENT OPERATOR ACTIONS

4.1 Record current time and date. Time/Date: ________/__________

4.2 **IF more than one rod is drifting, THEN SCRAM and concurrently enter Procedure 2.1.5.**

4.3 IF air drying/filtering components at fault, THEN perform following:

4.3.1 Open SA-MO-81, SA TO IA CROSSTIE (PANEL A).

4.3.2 Place standby dryer and filters in service per Procedure 2.2.59.

4.3.3 IF necessary, THEN manually bypass any obstructed component(s).

4.3.4 WHEN dryer and filter flow returned to service, THEN close SA-MO-81.

Scram Actions

4.4 IF SACs tripped, THEN have Operator locally reset per Procedure 2.2.59.

4.5 Make following announcement twice:

"All personnel using breathing equipment supplied by plant air move to an area with a clean atmosphere."

4.6 IF Primary Containment inerting in-progress, THEN secure inerting per Procedure 2.2.60.

4.7 Stop any work involving large air system loads.

4.8 At INSTRUMENT AIR PRESSURE ≤ 85 psig:

4.8.1 Ensure Steps 4.3 through 4.7 actions are completed.

4.8.2 Place BF-C-1A, EMERG BSTR FAN, as follows:

4.8.2.1 Place switch for BF-C-1A, EMERG BSTR FAN, to RUN, then allow it to spring-return to AUTO (VBD-R).

4.8.2.2 AFTER BF-C-1A has started, THEN verify following:

a. EF-C-1B, TOILET EXH FAN, stops.

b. HV-270AV, CONTROL ROOM HVAC INLET VALVE, closes.

c. HV-271AV, CONTROL ROOM HVAC EMERGENCY BYPASS SYSTEM INLET VALVE, opens.

d. HV-272AV, CONTROL ROOM PANTRY EXHAUST FAN ISOLATION VALVE, closes.

4.8.3 IF radwaste discharge in-progress, THEN secure discharge as follows:

4.8.3.1 IF from Waste Sample Tanks, THEN perform following:

a. Close RW-AO-141A (B), WASTE SMPL PUMP DISCH.

b. Place switch for Waste Sample Pump A (B) in STOP.

c. Complete securing discharge per Procedure 2.5.1.6 as time permits.

4.8.3.2 IF from Floor Drain Sample Tank, THEN perform following:

a. Hold FLOOR DRAIN SAMPLE PUMP control switch in STOP position until pump stops running, then return to AUTO.

b. Place switch for RW-AO-227, FL DR SAMPLE DISCH VALVE, in CLOSE.

c. As time permits, complete securing discharge per Procedure 2.5.2.3.

4.8.4 Stop all radwaste processing.

4.8.5 Stop all fuel handling.

4.8.6 Bypass FPC F/Ds per Procedure 2.2.32.

loss.

4.9 At INSTRUMENT AIR PRESSURE ≤ 77 psig:

4.9.1 Concurrently enter Procedure 2.1.5.

4.9.2 Transfer level control to HPCI/RCIC per Procedure 2.2.33.1 or 2.2.67.1.

4.9.3 Close all MSIVs.

4.9.4 Close MS-MO-74, INBD ISOL VLV.

4.9.5 Close MS-MO-77, OUTBD ISOL VLV.

4.9.6 To prevent RPV overfill, perform the following:

4.9.6.1 Ensure both reactor feed pumps are tripped.

4.9.6.2 Trip condensate booster pumps, if necessary.

4.9.6.3 Isolate RFP startup valves by performing following at PANEL A:

<u>NOTE</u> – Startup FCV isolation valves open when main turbine trip/scram signal is received if SETPOINT SETDOWN switch is in ENABLE position and main steam flow is < 4 Mlbm/Hr.

a. At PANEL 9-5, place SETPOINT SETDOWN switch to DISABLE.

b. Close RF-MO-31, RFP A STARTUP VALVE OUTLET.

c. Close RF-MO-32, RFP A STARTUP VALVE INLET.

d. Close RF-MO-33, RFP B STARTUP VALVE OUTLET.

PROCEDURE 5.2AIR	REVISION 21	PAGE 3 OF 17

e. Close RF-MO-34, RFP B STARTUP VALVE INLET.

4.9.6.4 IF RWCU-FCV-55 was being used for RPV level control, THEN perform following:

a. Ensure RWCU pumps off.

b. Throttle open RWCU-MO-74, DEMIN SUCTION BYPASS VLV.

c. Establish Shutdown Cooling transfer to Main Condenser or Radwaste, as necessary, for RPV level control per Procedure 2.2.69.2.

4.9.6.5 IF NBI-LT-61, REACTOR VESSEL SHUTDOWN RANGE, reference leg is pressurized by IA System, THEN issue TCC to connect an alternate air source.

4.9.7 Concurrently perform Attachment 2 (Page 9).

ATTACHMENT 3 INFORMATION SHEET

1. DISCUSSION

1.1 This procedure provides instructions for a loss of Instrument Air (IA). The procedure is comprised of the main body and two Attachments.

1.1.1 The procedure body provides instructions that attempt to restore system pressure, reduce system load, and anticipate a subsequent pressure loss. The procedure body instructions are to be performed whether Attachment 2 is entered or not.

1.1.2 The procedure body actions are grouped according to observed Instrument Air pressure values. The procedure is structured to address a steadily lowering IA pressure. If the rate of IA pressure decay does not allow completion of actions at the specified pressure, the next grouping contains a step to trigger the performance of previous actions (e.g., if pressure drops rapidly to < 77 psig), Attachment 2 contains a step to trigger performance of actions that would have been performed at > 77 psig.

1.1.3 Attachment 2 provides instructions that are performed when system pressure is considered to be too low to support continued operation. Attachment 2 is to be performed in conjunction with the procedure body instructions.

1.1.4 Attachment 1 provides a list of loads and what effect the pressure loss will have on the load. The list is not all inclusive and only includes those loads considered to be significant or important.

1.2 A loss of IA, while in the process of inerting primary containment, would cause the containment isolation valves to fail closed, the nitrogen purge supply rupture disc to fail and/or the associated relief valve to open. This would result in the release of N_2 directly to the Reactor Building.

1.3 The loss of IA for more than a very short time will necessitate emergency plant shutdown from power. The air operated valves and components are designed to a fail safe and allow the plant to be shut down and placed in a MODE 4 condition. Some local manual valve positioning will be necessary.

1.4 Attachment 2 provides instructions that ensure a nitrogen supply to the drywell header; help to ensure availability of the relief valves. If in MODE 3 or 4, three (3) safety relief valves are required to remain available for Alternate Decay Heat Removal.

1.5 A non-critical instrument air isolation will cause a severe reduction in feedwater heating, resulting in unsafe plant operating conditions.Ⓟ[1]

1.6 Large air loads referred to in the Subsequent Actions are those which require prior Control Room authorization per Procedure 0.31.

Examination Outline Cross-Reference	Level	SRO
Provided to provide clarity during exam administration – no change.	Tier#	1
	Group#	1
	K/A #	295021, AA2.04
	Rating	3.6

295021 Loss of Shutdown Cooling

AA2. Ability to determine and/or interpret the following as they apply to LOSS OF SHUTDOWN COOLING : (CFR: 41.10 / 43.5 / 45.13)
AA2.04 Reactor water temperature

Question 80

Reference Provided

The plant is in **Mode 4** at the start of a refueling outage with the following conditions:

- The reactor was shutdown 25 hours ago.
- Reactor water level is at the reactor head flange.
- Primary containment has been relaxed for outage activities.

The operating Shutdown Cooling subsystem pump trips.

No decay heat removal systems are operating.

Reactor coolant temperature is 150°F and rising.

(1) Which one of the following is the time until Mode 3 will be entered if shutdown cooling or alternate decay heat removal systems cannot be restored?

AND

(2) What is the Emergency Classification 22 minutes after Mode 3 is entered, if reactor pressure has risen to 5 psig?

A. (1) 0.8 hour
(2) Unusual Event

B. (1) 0.8 hour
(2) Alert

C. (1) 1.3 hours
   (2) Unusual Event

D. (1) 1.3 hours
   (2) Alert

Answer:
C (1) 1.3 hours, (2) Unusual Event
Explanation: Following a plant shut down, if no decay heat removal systems are operating, reactor water temperature will rise. This question requires determining that a rise in reactor coolant temperature will result in a mode change from Mode 4 to Mode 3 when temperature reaches 212ºF. It also requires determining the estimated time until 212ºF will be reached. Based on the decay heat curve for 150 ºF with water level at the flange from procedure 2.4SDC Attachment 5, time to 212 ºF is approximately 1.3 hours. Declaration of an Unusual Event is required as soon as RPV Temperature exceeds 212°F per EAL CU3.1. CA3.1 is not applicable if the reactor in Mode 4 with the RCS intact.
Distracters: Answer A is plausible yet wrong since it reflects the time to boil if the candidate selects the 150 ºF curve from the wrong initial water level graph, the one for level at the high level trip. The graph for water level at the high level trip was chosen because it is the first graph encountered in the procedure. Stopping and focusing on the first graph encountered is a common error. It is wrong because the actual time to 212 ºF (Mode 3) is ~ 1.3 hours. Part 1 of Answer B is plausible and wrong for the same reasons stated for answer A. Additionally, part 2 is plausible and wrong because an Alert classification would be required per EAL CA3.1 if containment closure is not established within 20 minutes of reaching 212 ºF (Mode 3) and the RCS was not intact. Part 2 is wrong because the stem asks for the emergency classification 22 minutes after Mode 3 is entered, (Unusual Event), not 60 minutes later, and because the stem implies the RCS is intact by giving the initial condition as Mode 4. Answer D is plausible because 1.3 hours is the correct time to Mode 3 and for the same reasons stated why an Alert classification is plausible for Answer B. It is wrong because an Unusual Event classification is correct for the conditions given in the stem.
Technical References: 2.4SDC, Shutdown Cooling Abnormal; TS Table 1.1-1; Procedure 5.7.1, Emergency Classification: EPIPEALCOLD (EAL Wall Chart –

Modes 4,5)		
References to be provided to applicants during exam: 2.4SDC, Attachment 5; EPIPEALCOLD (EAL Wall Chart – Modes 4,5)		
Learning Objective:		
INT0320126Q0Q0100 Given plant condition(s), and the applicable Abnormal/Emergency Procedure, determine the correct subsequent actions required to mitigate the event(s). GEN0030401C0C050E Concerning event classification: Given a copy of EPIP 5.7.1 and hypothetical abnormal plant symptoms, indications, or events, determine any and all EALs which have been exceeded and specify the appropriate emergency classification.		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:		
	55.43(b)(5)	
Level of Difficulty:	3	
SRO Only Justification:		
This requires emergency classification, an Emergency Director (SRO) duty.		

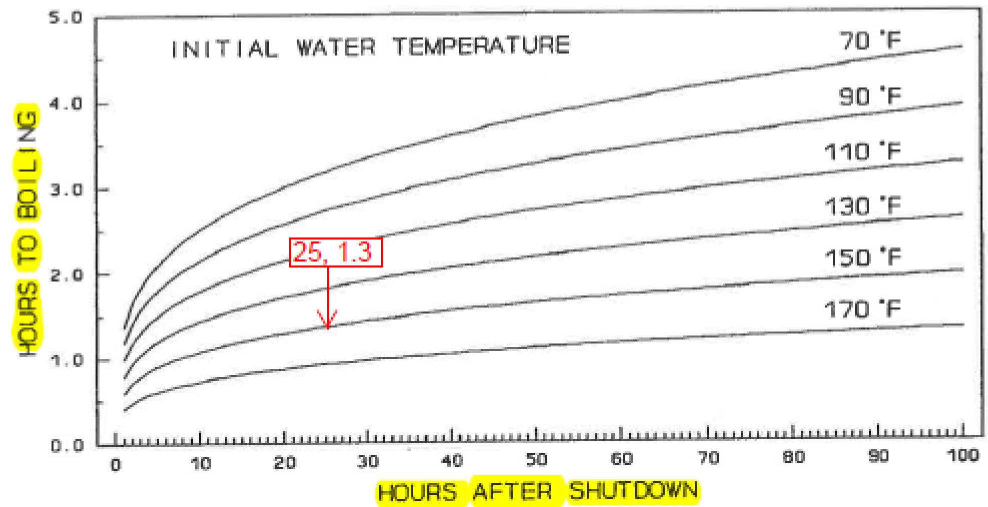
Table 1.1-1 (page 1 of 1)
MODES

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	> 212
4	Cold Shutdown[(a)]	Shutdown	≤ 212
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.

ATTACHMENT 5 TIME TO CORE BOILING/TIME TO CORE UNCOVERY



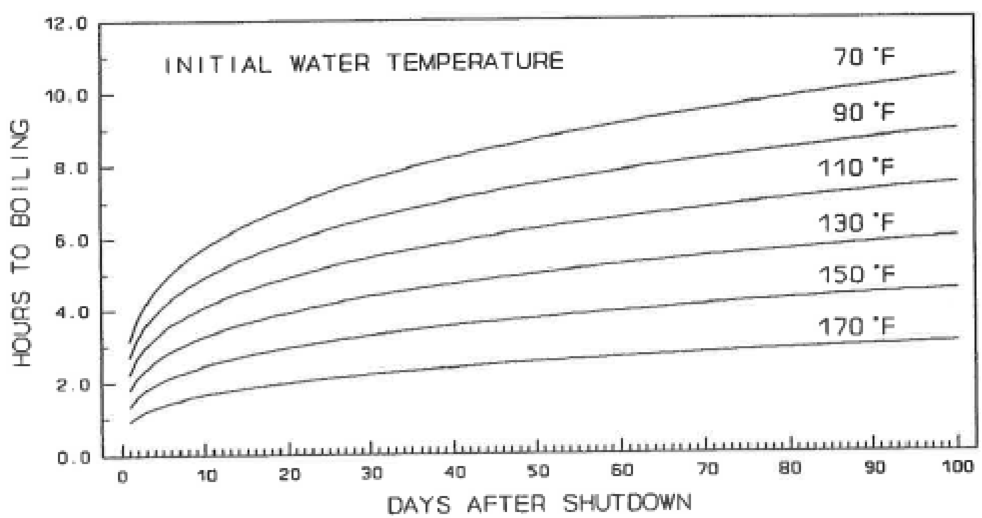
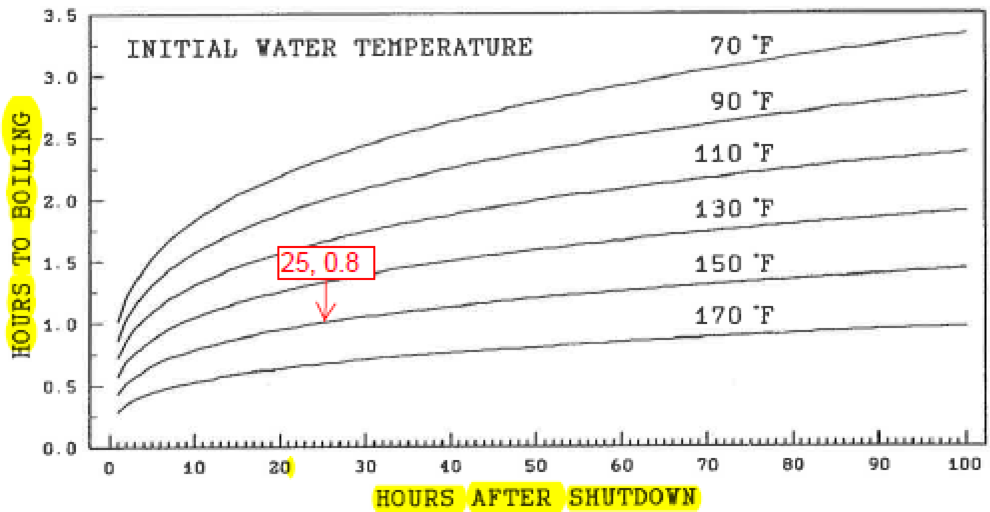


Figure 2 - TIME TO BOILING - WATER LEVEL AT FLANGE

ATTACHMENT 5 TIME TO CORE BOILING/TIME TO CORE UNCOVERY

ATTACHMENT 5 TIME TO CORE BOILING/TIME TO CORE UNCOVERY



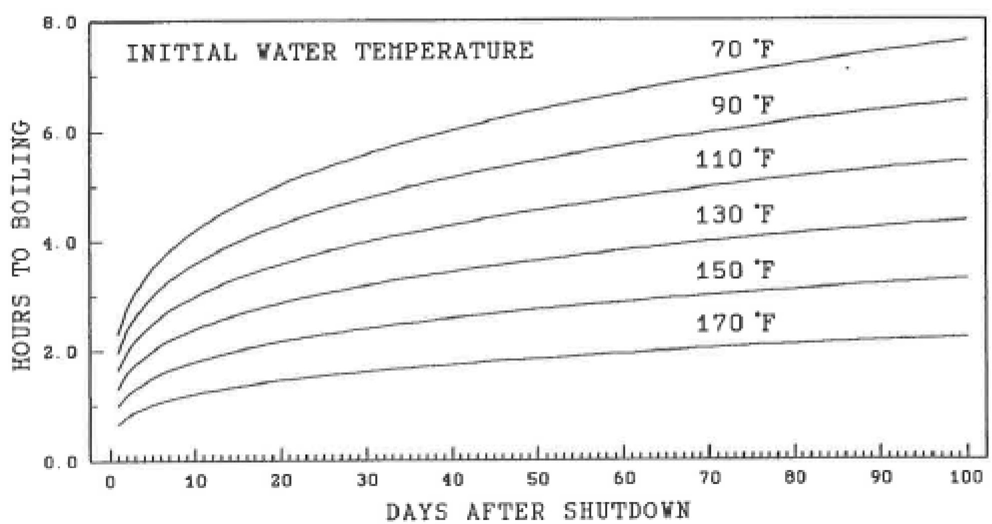


Figure 1 - TIME TO BOILING - WATER LEVEL AT HIGH LEVEL TRIP

Examination Outline Cross-Reference	Level	SRO
Revised to provide location of crack and ask the highest required emergency classification declaration. Changed foot to feet.	Tier#	1
	Group#	1
	K/A #	295023, G2.4.41
	Rating	4.6

295023 Refueling Acc

G2.4.41 Knowledge of the emergency action level thresholds and classifications.
(CFR: 41.10 / 43.5 / 45.11)

Question 81

Reference Provided

Refueling is in progress.

A spent fuel bundle is dropped just as it passes through the reactor cavity gate into the spent fuel pool resulting in:

- The fuel bundle is damaged.
- The fuel pool liner is cracked 3 feet above the storage racks.
- Refuel floor personnel report pool level is just below the skimmers and slowly lowering.
- RMA-RA-1 Fuel Pool Area Radiation Monitor reading has risen from 0.3 mRem/hr to 52 mRem/hr.
- RMP-RM-452 A-D Reactor Building Ventilation Exhaust Plenum Radiation Monitor readings are 50 mRem/hr.

Which of the following identifies the HIGHEST required Emergency Classification declaration for this condition and why?

A. Unusual Event based on pool water level

B. Unusual Event based on radiation level

C. Alert based on pool water level

D. Alert based on radiation level

Answer:
D. Alert based on radiation level
Explanation: This question requires execution of the EALs. It is not a direct look-up because it requires knowledge of the Reactor Building Ventilation Exhaust Plenum Radiation Monitor high-high trip setpoint. Answer D is correct because a fuel handling accident has occurred in Mode 5 and RMP-RM-452 A-D Reactor Building Ventilation Exhaust Plenum Radiation Monitor readings are 50 mr/hr, which is above the high-high setpoint of 49 mr/hr. Thus EAL AA2.1 is met due to radiation level. (Also note RM-RA-1 at 50 mr/hr is below the Alert threshold of 50 **R**/hr)
Distracters: Answer A is plausible because a less severe fuel handling accident might not cause high radiation levels. The unprepared candidate who does not know the Reactor Building Ventilation Exhaust Plenum Radiation Monitor high-high trip setpoint may believe rad levels are below the high-high setpoint, and since no information is given regarding rad monitor RMA-RA-2, concludes the only EAL exceeded is due to visual observation of lowered pool level, which is only a UE per EAL AU2.1. Answer B is plausible for the same reason as Answer A and also reflects EAL AU2.1. However he could be attracted to attributing an EAL entry to radiation level if he believes Reactor Building Ventilation Exhaust Plenum Radiation Monitor is elevated. He may consider rad level more significant than a slight change in pool level as reflected in the stem. Answer C is plausible because an unprepared student may extrapolate a reduction in pool level due to a liner leak into a condition that will inevitably result in fuel uncover, since there is no valve that can be closed to isolate the leak. There are mitigating actions in plant procedure to patch these types of leaks. There are also many methods of pool makeup to prevent the fuel from being uncovered. The implication in the stem is that the leak size is well within the capacity of makeup systems; therefore, EAL AA2.2 should not be chosen based on pool water level.
Technical References: Procedure 5.7.1, Emergency Classification: EPIPEALCOLD (EAL Wall Chart – Modes 4,5)
References to be provided to applicants during exam: EPIPEALCOLD (EAL Wall Chart – Modes 4,5)
Learning Objective:
GEN0030401C0C050E Concerning event classification: Given a copy of EPIP 5.7.1 and hypothetical abnormal plant symptoms, indications, or events, determine any and all EALs which have been exceeded and specify the appropriate emergency

Examination Outline Cross-Reference	Level	SRO
Eliminated references. Modified stem to include "FIRST" and C distractor to include IAW EOP 3A. Removed DW temperature and provided DWSIL is NOT met and reactor pressure is in the unsafe region of the RPV Saturation Curve to eliminate reference from the question. Change DW pressure to Torus pressure to support DW spray plausibility.	Tier#	1
	Group#	1
	K/A #	295028, EA2.02
	Rating	3.9

295028 High Drywell Temperature

EA2. Ability to determine and/or interpret the following as they apply to HIGH DRYWELL TEMPERATURE : (CFR: 41.10 / 43.5 / 45.13)

EA2.02 Reactor pressure

Question 82

The crew has entered EOP-1A (RPV Control) and EOP-3A (Containment Control).

The following conditions exist:

- RPV pressure is 30 psig and is in the Unsafe (RED) region of the RPV Saturation Temperature curve.
- Torus water temperature is 190°F.
- Torus pressure 15 psig.
- PC water level 15 ft.
- DWSIL is NOT met.
- All RPV level indications are erratic.

What is the CRS' FIRST required action at this time?

A. Transition to EOP-2A (Emergency Depressurization).

B. Transition to EOP-2B (RPV Flooding).

C. Initiate DW sprays IAW EOP-3A (Containment Control).

D. Transition to EOP-2A (Steam Cooling).

Answer:
B. Transition to EOP-2B (RPV Flooding).

Explanation:
Conditions are in the Unsafe (RED) region area of the RPVST. This, along with unstable level indication, makes RPV level unknown, which requires transition to EOP-2B.

Distracters:
Answer A is plausible since containment parameters are elevated, particularly DW pressure and SP level. It is wrong because emergency depressurization conditions for these parameters are not met.

Answer C is plausible because DW spray is required to be initiated when DW pressure exceeds 10 psig. C is wrong because the given condition is DWSIL is not met.

Answer D is plausible because conditions given reflect that a severe LOCA is in progress. D is wrong because transition to steam cooling is based on RPV water level, which is unavailable.

Technical References: EOP Caution 1; EOP-1A, RPV Control; EOP-3A, Containment Control

References to be provided to applicants during exam: NONE

Learning Objective:
INT00806050011000 Given plant conditions and EOP flowchart 1A, RPV CONTROL, determine required actions.

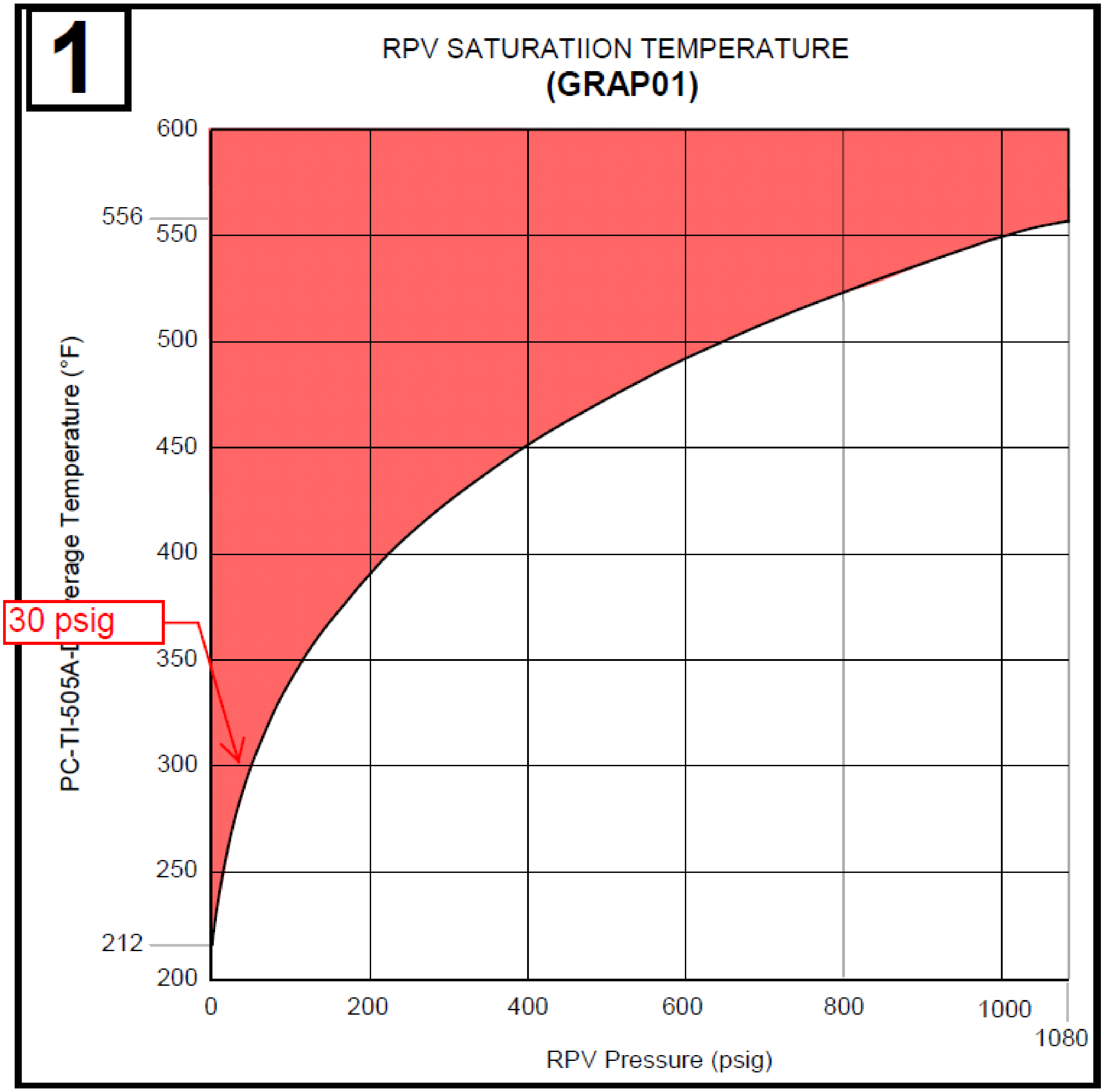
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:		
	55.43(b)(5)	
Level of Difficulty:	3	
SRO Only Justification:		

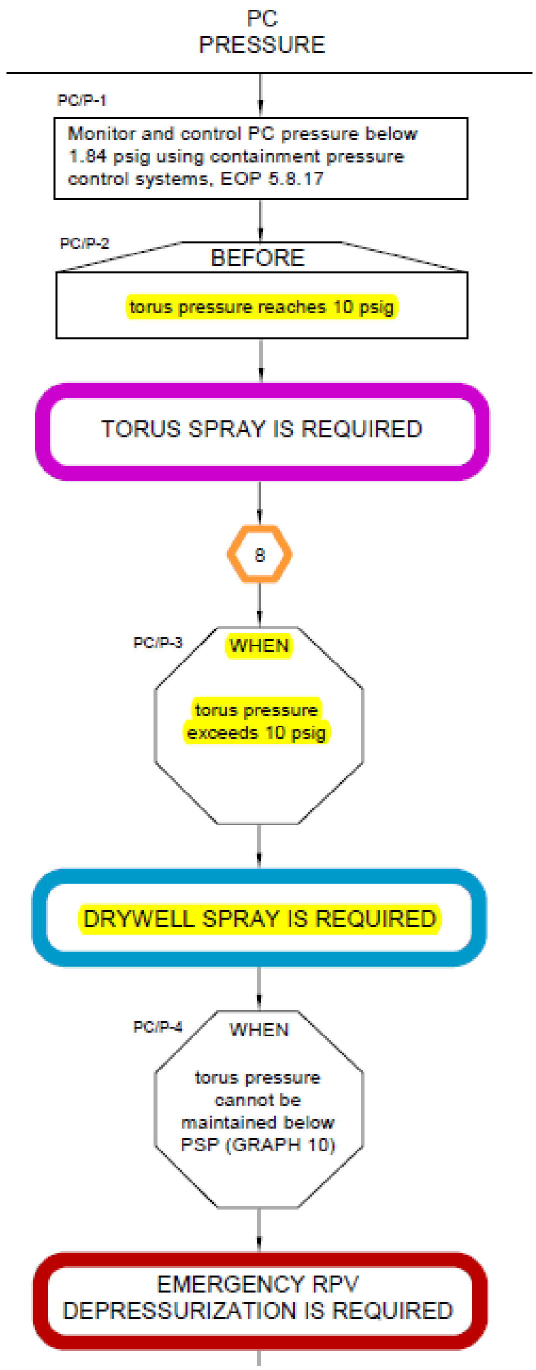
This requires interpretation of plant conditions and resulting procedure selection.

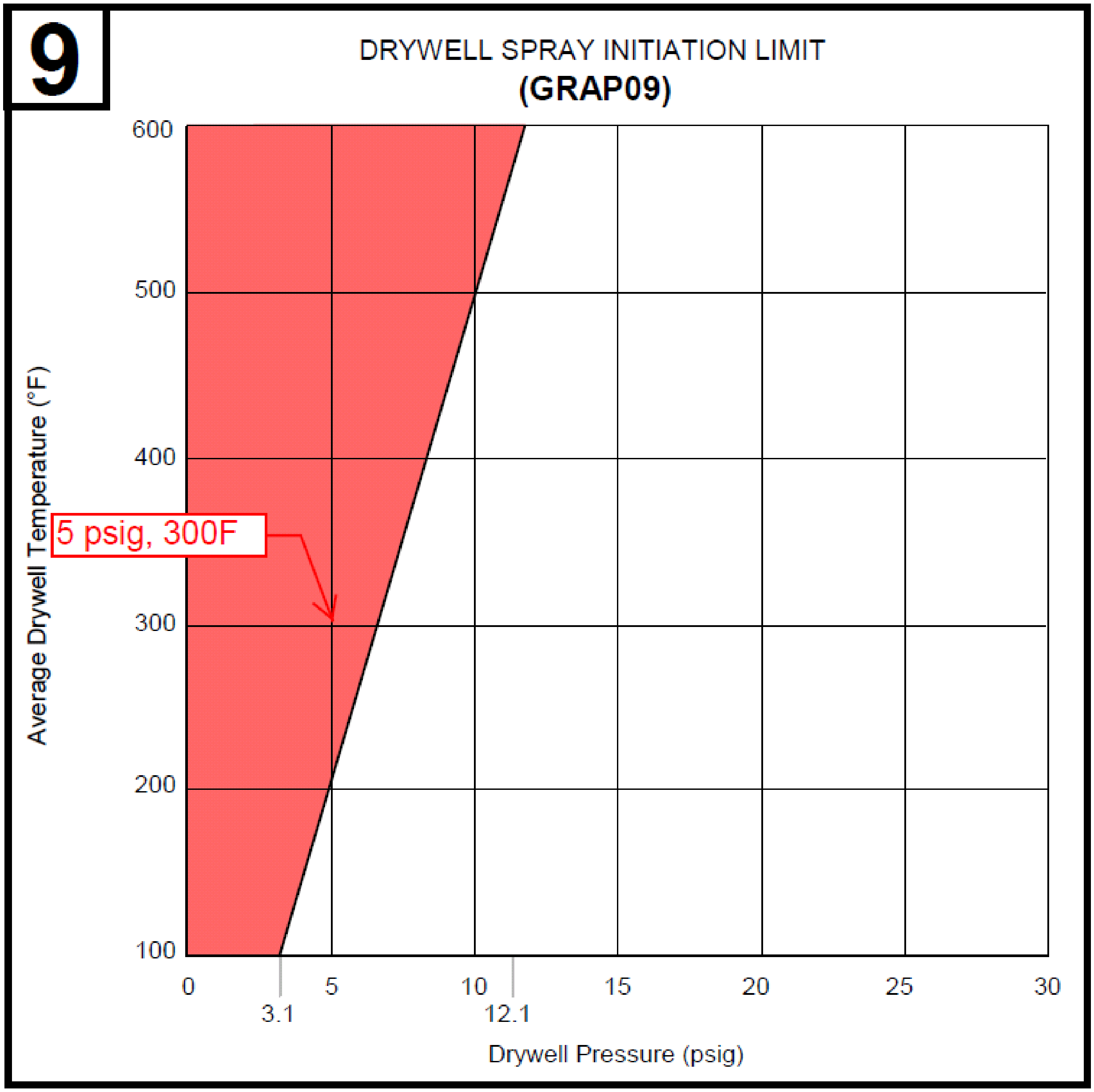
1 RPV water level indications are affected by instrument run temperatures and RPV pressure:

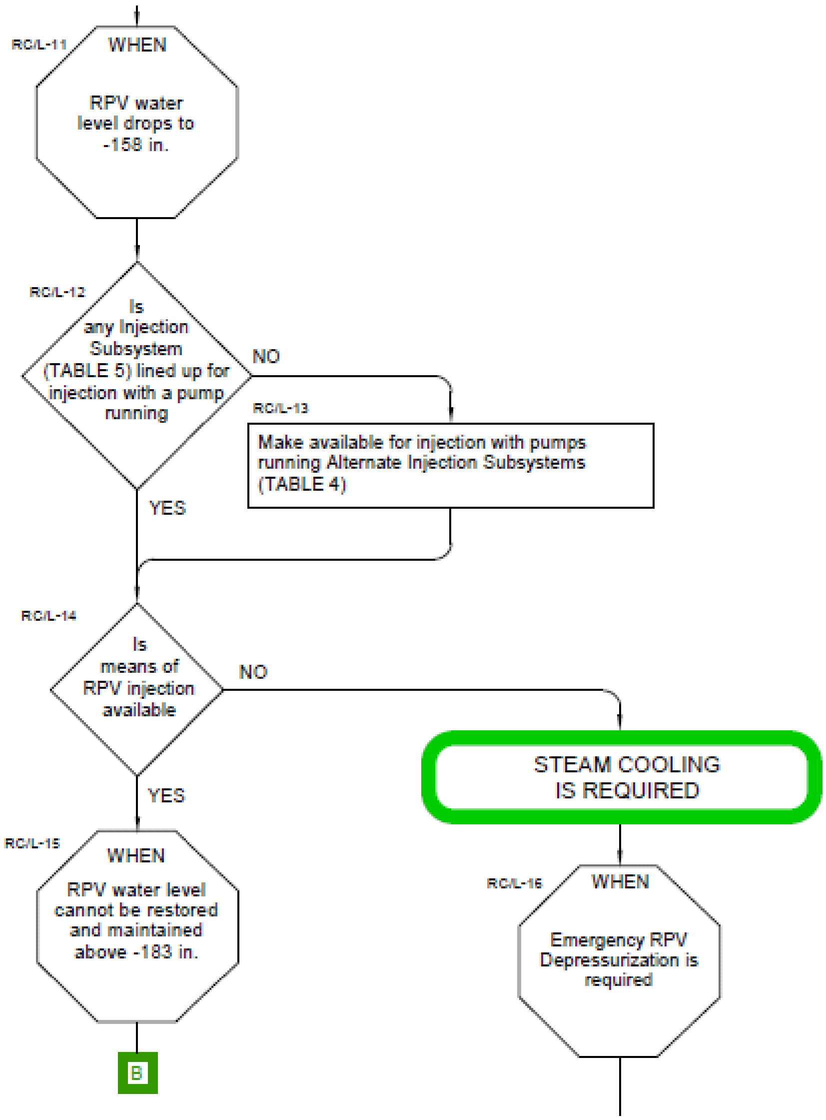
- If PC-TI-505A-D temperatures near any instrument run are above RPV Saturation Temperature (GRAPH 1), the instrument may be unreliable due to boiling in the run
- Water level indication must read above Minimum Indicated Level for instrument (GRAPH 15) to be usable

(apply level correction (GRAPH 14) for RPV pressure at or above 200 psig)




PC
PRESSURE
PC/P-1
Monitor and control PC pressure below 1.84 psig using containment pressure control systems, EOP 5.8.17
PC/P-2
BEFORE
torus pressure reaches 10 psig
TORUS SPRAY IS REQUIRED
8
PC/P-3
WHEN
torus pressure exceeds 10 psig
DRYWELL SPRAY IS REQUIRED
PC/P-4
WHEN
torus pressure cannot be maintained below PSP (GRAPH 10)
EMERGENCY RPV DEPRESSURIZATION IS REQUIRED


9
DRYWELL SPRAY INITIATION LIMIT
(GRAP09)
Average Drywell Temperature (°F)
600
500
400
300
200
100
5 psig, 300F
0
5
10
15
20
25
30
3.1
12.1
Drywell Pressure (psig)


RC/L-11
WHEN
RPV water level drops to -158 in.
RC/L-12
Is any Injection Subsystem (TABLE 5) lined up for injection with a pump running
NO
RC/L-13
Make available for injection with pumps running Alternate Injection Subsystems (TABLE 4)
YES
RC/L-14
Is means of RPV injection available
NO
STEAM COOLING IS REQUIRED
YES
RC/L-15
WHEN
RPV water level cannot be restored and maintained above -183 in.
B
RC/L-16
WHEN
Emergency RPV Depressurization is required

Examination Outline Cross-Reference	Level	SRO
	Tier#	1
	Group#	2
	K/A #	295015, G2.2.25
	Rating	4.2

295015 Incomplete SCRAM

G2.2.25 Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits. (CFR: 41.5 / 41.7 / 43.2)

Question 83

Consider the following Tech Spec REQUIRED ACTION:

Control Rod Scram Accumulators
3.1.5

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Two or more control rod scram accumulators inoperable with reactor steam dome pressure ≥ 900 psig.	B.1 Restore charging water header pressure to ≥ 940 psig.	20 minutes from discovery of Condition B concurrent with charging water header pressure < 940 psig

What is the Tech Spec Bases for the allowed COMPLETION TIME of 20 minutes for this action?

A. Allows time to start and align a CRD pump.

B. Allows time to place the standby CRD flow control valve into service.

C. Allows time to locally recharge the inoperable scram accumulators with nitrogen.

D. Allows time to fully insert and disarm the control rods associated with the inoperable scram accumulators.

Answer: .		
A. Allow time to start and align a CRD pump		
Explanation: TS 3.1.5 Action B.1 assumes no CRD pumps are running if charging water header pressure is <940 psig. The completion time for Action B starts when 2 or more scram accumulators for withdrawn control rods are declared inoperable concurrently with charging water header pressure < 940 psig. If the condition cannot be corrected within 20 minutes, Action B.1 is not met and Action D must be entered, which requires placing the Reactor Mode Switch in SHUTDOWN immediately. The TS Bases states the 20 minute completion time provided for Action B.1 should be adequate for starting a CRD pump, as reflected by Answer A. It is important for the SRO to know he should use the 20 minutes allowed by TS to focus directions on restarting a CRD pump.		
Distracters: Answer B is plausible because the CRD flow control valve, which is downstream of the charging water header, closes on a high CRD flow signal to divert CRD flow to the charging water header to recharge CRD HCU accumulators following a scram. It is wrong because TS 3.1.5 condition B assumes there is a problem with CRD flow causing low flow upstream of the flow control valves that would result in low charging water pressure, common to multiple CRD accumulators. Answer C is plausible because accumulator low nitrogen pressure would cause an individual accumulator to be inoperable, below 940 psig nitrogen pressure, but this would be addressed by TS 3.1.5 Actions A.2.1 and A.2.2. Answer D is plausible because inserting and disarming a control rod would be necessary if the rod was declared inoperable, which is an alternative to declaring a control rod "slow" when its accumulator is inoperable. However, answer D is wrong because TS 3.1.5 Action B does not address urgency for inserting/disarming a control rod, only for restoring a CRD pump to operation.		
Technical References: TS 3.1.5 and bases		
References to be provided to applicants during exam: none		
Learning Objective:		
INT00705020010300 Given a set of plant conditions that constitutes non-compliance with a Section 3.1 LCO, determine the ACTIONS that are required. INT00705020010200 Discuss the applicable Safety Analysis in the Bases associated with each Section 3.1 Specification.		
Question Source:	Bank #	

(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:		
	55.43(b)(2) & (6)	
Level of Difficulty:	2	
SRO Only Justification:		
This requires knowledge of TS bases.		

BASES

ACTIONS

A.1 and A.2 (continued)

Required Action A.1 is modified by a Note indicating that declaring the control rod "slow" only applies if the associated control scram time was within the limits of Table 3.1.4-1 during the last scram time test. Otherwise, the control rod would already be considered "slow" and the further degradation of scram performance with an inoperable accumulator could result in excessive scram times. In this event, the associated control rod is declared inoperable (Required Action A.2) and LCO 3.1.3 is entered. This would result in requiring the affected control rod to be fully inserted and disarmed, thereby satisfying its intended function, in accordance with ACTIONS of LCO 3.1.3.

The allowed Completion Time of 8 hours is reasonable, based on the large number of control rods available to provide the scram function and the ability of the affected control rod to scram only with reactor pressure at high reactor pressures.

B.1, B.2.1, and B.2.2

With two or more control rod scram accumulators inoperable and reactor steam dome pressure ≥ 900 psig, adequate pressure must be supplied to the charging water header. With inadequate charging water pressure, all of the accumulators could become inoperable, resulting in a potentially severe degradation of the scram performance. Therefore, within 20 minutes from discovery of charging water header pressure < 940 psig concurrent with Condition B, adequate charging water header pressure must be restored. The allowed Completion Time of 20 minutes is reasonable, to place a CRD pump into service to restore the charging water header pressure, if required. This Completion Time is based on the ability of the reactor pressure alone to fully insert all control rods.

The control rod may be declared "slow," since the control rod will still scram using only reactor pressure, but may not satisfy the times in Table 3.1.4-1. Required Action B.2.1 is modified by a Note indicating that declaring the control rod "slow" only applies if the associated control scram time is within the limits of Table 3.1.4-1 during the last scram time test. Otherwise, the control rod

(continued)

3.1 REACTIVITY CONTROL SYSTEMS

3.1.5 Control Rod Scram Accumulators

LCO 3.1.5 Each control rod scram accumulator shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

-----------------------------------NOTE-----------------------------------
Separate Condition entry is allowed for each control rod scram accumulator.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One control rod scram accumulator inoperable with reactor steam dome pressure $\geq$ 900 psig.	A.1 --------NOTE--------- Only applicable if the associated control rod scram time was within the limits of Table 3.1.4-1 during the last scram time Surveillance. ---------------------	
	Declare the associated control rod scram time "slow."	8 hours
	<u>OR</u>	
	A.2 Declare the associated control rod inoperable.	8 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Two or more control rod scram accumulators inoperable with reactor steam dome pressure ≥ 900 psig.	B.1 Restore charging water header pressure to ≥ 940 psig.	20 minutes from discovery of Condition B concurrent with charging water header pressure < 940 psig
	AND	
	B.2.1 --------NOTE--------- Only applicable if the associated control rod scram time was within the limits of Table 3.1.4-1 during the last scram time Surveillance. --------------------	
	Declare the associated control rod scram time "slow."	1 hour
	OR	
	B.2.2 Declare the associated control rod inoperable.	1 hour

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One or more control rod scram accumulators inoperable with reactor steam dome pressure < 900 psig.	C.1 Verify the associated control rods are fully inserted.	Immediately upon discovery of charging water header pressure < 940 psig
	AND	
	C.2 Declare the associated control rod inoperable.	1 hour
D. Required Action B.1 or C.1 and associated Completion Time not met.	D.1 --------NOTE--------- Not applicable if all inoperable control rod scram accumulators are associated with fully inserted control rods. --------------------	
	Place the reactor mode switch in the shutdown position.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.5.1 Verify each control rod scram accumulator pressure is ≥ 940 psig.	7 days

Examination Outline Cross-Reference	Level	SRO
Added Procedure 5.7.2 (Emergency Director EPIP) to Technical references (provided highlighted markup) which directs performing site evacuation IAW procedure 5.7.11 (which is not from memory). The MET data provided requires interpretation beyond the "wind direction from" arrows (elevation and actual degree wind direction). Reference is needed to correctly answer this question.	Tier#	1
	Group#	2
	K/A #	295017, AA2.05
	Rating	3.8

295017 High Off-site Release Rate

AA2. Ability to determine and/or interpret the following as they apply to HIGH OFF-SITE RELEASE RATE : (CFR: 41.10 / 43.5 / 45.13)
AA2.05 Meteorological data

Question 84

Reference Provided

An event has been declared due to high off-site release rates.

An evacuation of non-emergency response site personnel is required.

Which one of the following MET Information Displays depicts conditions that warrant directing the site evacuation using the SOUTH route?

A. Display A

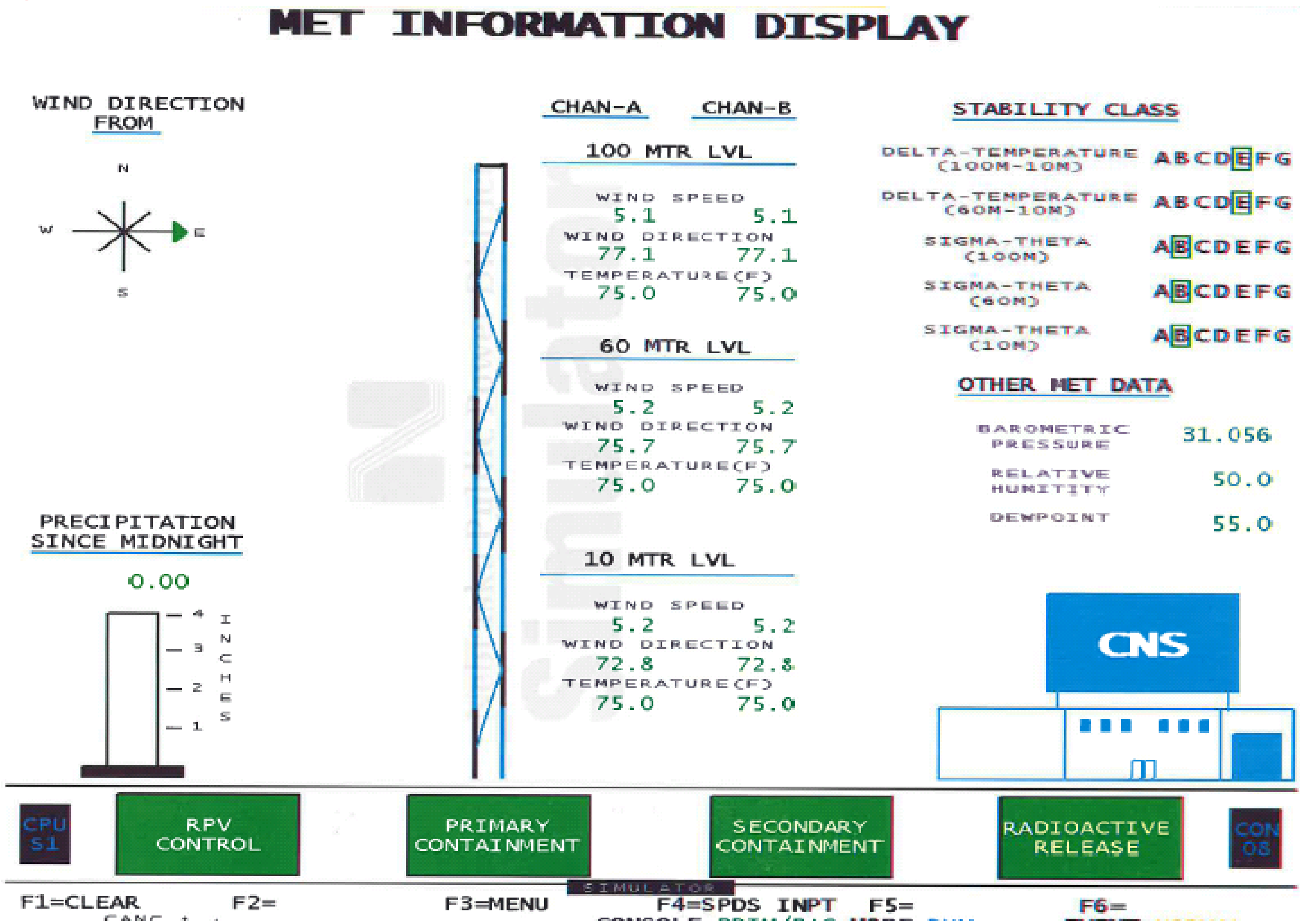
B. Display B

C. Display C

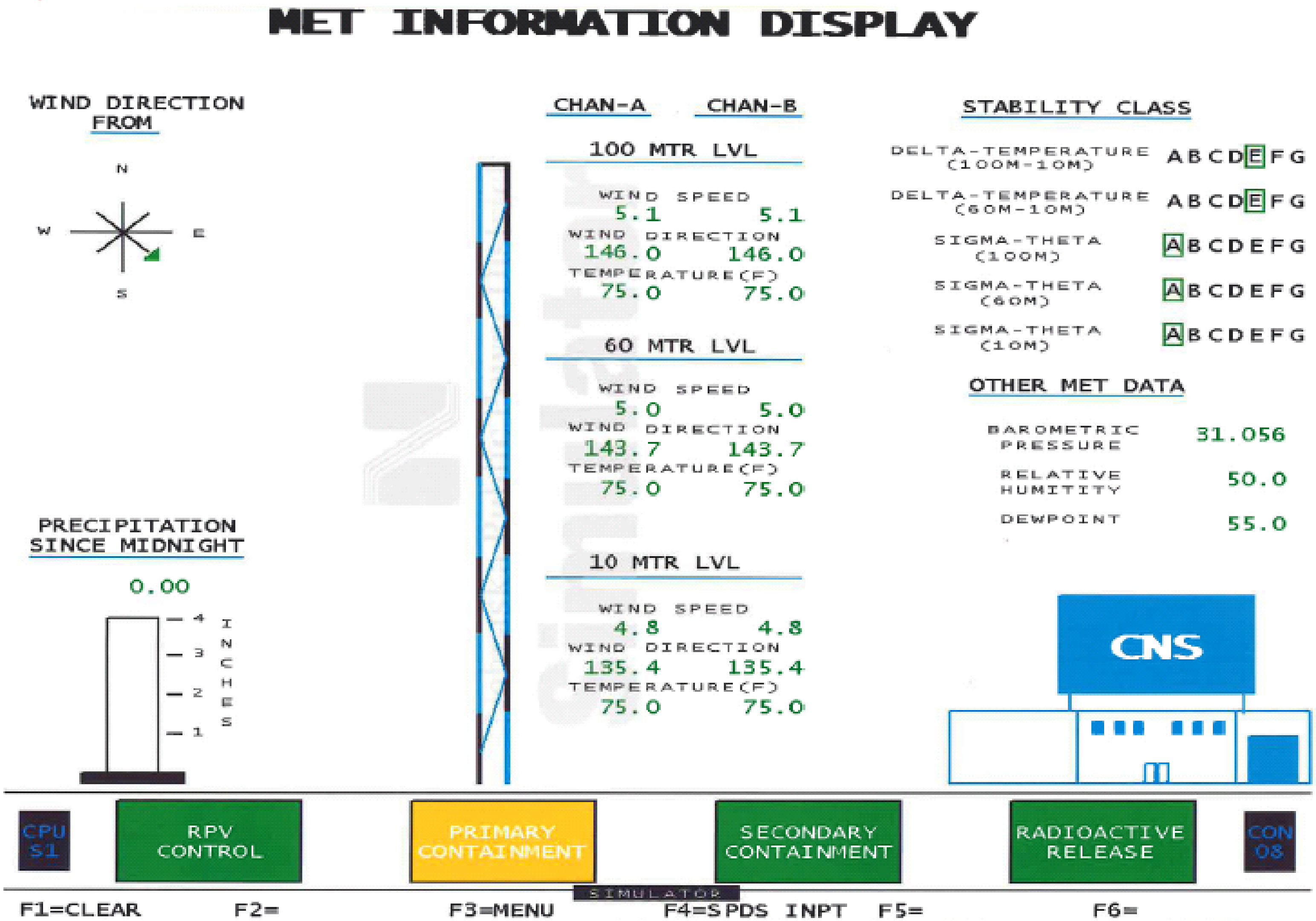
D. Display D

(Refer to the attached MET Information Displays)

DISPLAY A

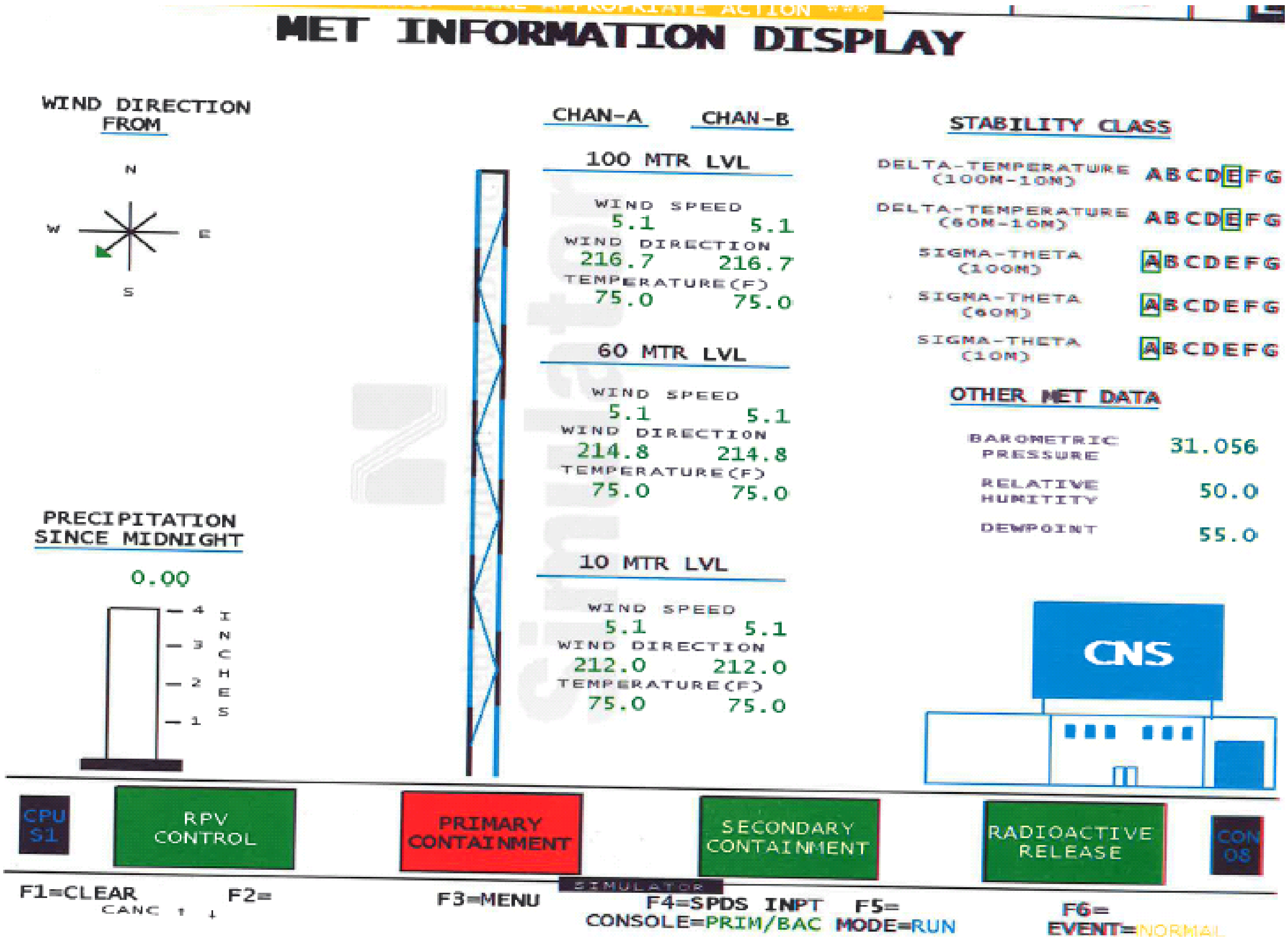

MET INFORMATION DISPLAY
WIND DIRECTION FROM
N
W
E
S
PRECIPITATION SINCE MIDNIGHT
0.00
4
3
2
1
INCHES
CHAN-A
CHAN-B
100 MTR LVL
WIND SPEED
5.1 5.1
WIND DIRECTION
77.1 77.1
TEMPERATURE(F)
75.0 75.0
60 MTR LVL
WIND SPEED
5.2 5.2
WIND DIRECTION
75.7 75.7
TEMPERATURE(F)
75.0 75.0
10 MTR LVL
WIND SPEED
5.2 5.2
WIND DIRECTION
72.8 72.8
TEMPERATURE(F)
75.0 75.0
STABILITY CLASS
DELTA-TEMPERATURE (100M-10M) A B C D E F G
DELTA-TEMPERATURE (60M-10M) A B C D E F G
SIGMA-THETA (100M) A B C D E F G
SIGMA-THETA (60M) A B C D E F G
SIGMA-THETA (10M) A B C D E F G
OTHER MET DATA
BAROMETRIC PRESSURE 31.056
RELATIVE HUMIDITY 50.0
DEWPOINT 55.0
CNS
CPU S1
RPV CONTROL
PRIMARY CONTAINMENT
SECONDARY CONTAINMENT
RADIOACTIVE RELEASE
CON 08
SIMULATOR
F1=CLEAR
F2=
F3=MENU
F4=SPDS INPT
F5=
F6=

DISPLAY B

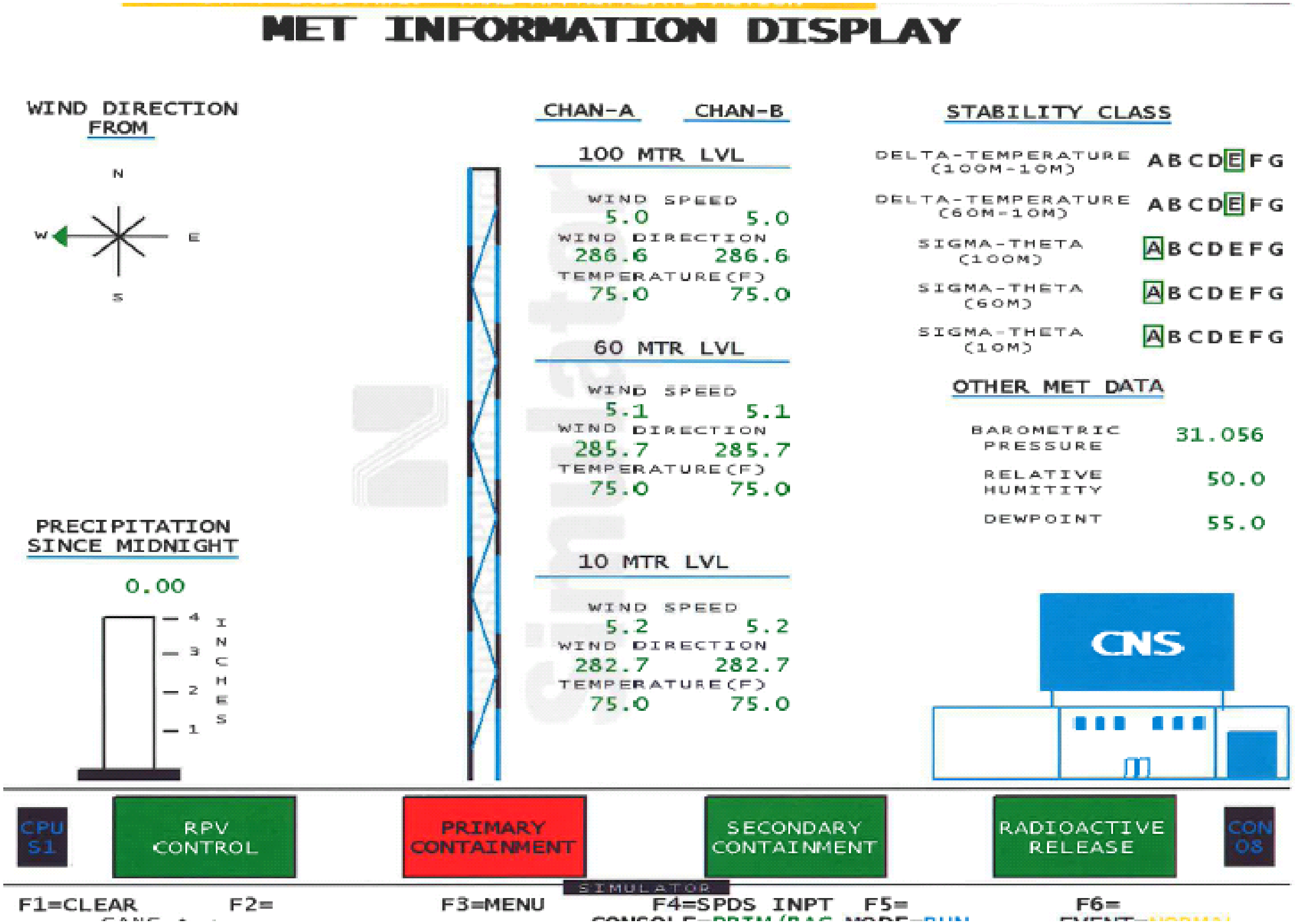


MET INFORMATION DISPLAY
WIND DIRECTION FROM
N
W
E
S
PRECIPITATION SINCE MIDNIGHT
0.00
4
3
2
1
INCHES
CHAN-A CHAN-B
100 MTR LVL
WIND SPEED 5.1 5.1
WIND DIRECTION 146.0 146.0
TEMPERATURE(F) 75.0 75.0
60 MTR LVL
WIND SPEED 5.0 5.0
WIND DIRECTION 143.7 143.7
TEMPERATURE(F) 75.0 75.0
10 MTR LVL
WIND SPEED 4.8 4.8
WIND DIRECTION 135.4 135.4
TEMPERATURE(F) 75.0 75.0
STABILITY CLASS
DELTA-TEMPERATURE (100M-10M) A B C D E F G
DELTA-TEMPERATURE (60M-10M) A B C D E F G
SIGMA-THETA (100M) A B C D E F G
SIGMA-THETA (60M) A B C D E F G
SIGMA-THETA (10M) A B C D E F G
OTHER MET DATA
BAROMETRIC PRESSURE 31.056
RELATIVE HUMITITY 50.0
DEWPOINT 55.0
CNS
CPU S1
RPV CONTROL
PRIMARY CONTAINMENT
SECONDARY CONTAINMENT
RADIOACTIVE RELEASE
CON 08
SIMULATOR
F1=CLEAR
F2=
F3=MENU
F4=SPDS INPT
F5=
F6=

DISPLAY C


MET INFORMATION DISPLAY
WIND DIRECTION FROM
N
W
E
S
CHAN-A
CHAN-B
100 MTR LVL
WIND SPEED
5.1 5.1
WIND DIRECTION
216.7 216.7
TEMPERATURE(F)
75.0 75.0
60 MTR LVL
WIND SPEED
5.1 5.1
WIND DIRECTION
214.8 214.8
TEMPERATURE(F)
75.0 75.0
10 MTR LVL
WIND SPEED
5.1 5.1
WIND DIRECTION
212.0 212.0
TEMPERATURE(F)
75.0 75.0
STABILITY CLASS
DELTA-TEMPERATURE (100M-10M) A B C D E F G
DELTA-TEMPERATURE (60M-10M) A B C D E F G
SIGMA-THETA (100M) A B C D E F G
SIGMA-THETA (60M) A B C D E F G
SIGMA-THETA (10M) A B C D E F G
OTHER MET DATA
BAROMETRIC PRESSURE 31.056
RELATIVE HUMIDITY 50.0
DEWPOINT 55.0
PRECIPITATION SINCE MIDNIGHT
0.00
4
3
2
1
INCHES
CNS
CPU S1
RPV CONTROL
PRIMARY CONTAINMENT
SECONDARY CONTAINMENT
RADIOACTIVE RELEASE
CON 08
F1=CLEAR
CANC ↑ ↓
F2=
F3=MENU
SIMULATOR
F4=SPDS INPT CONSOLE=PRIM/BAC
F5= MODE=RUN
F6= EVENT=NORMAL

DISPLAY D


MET INFORMATION DISPLAY
WIND DIRECTION FROM
N
W
E
S
CHAN-A
CHAN-B
100 MTR LVL
WIND SPEED
5.0 5.0
WIND DIRECTION
286.6 286.6
TEMPERATURE(F)
75.0 75.0
60 MTR LVL
WIND SPEED
5.1 5.1
WIND DIRECTION
285.7 285.7
TEMPERATURE(F)
75.0 75.0
10 MTR LVL
WIND SPEED
5.2 5.2
WIND DIRECTION
282.7 282.7
TEMPERATURE(F)
75.0 75.0
STABILITY CLASS
DELTA-TEMPERATURE (100M-10M) A B C D E F G
DELTA-TEMPERATURE (60M-10M) A B C D E F G
SIGMA-THETA (100M) A B C D E F G
SIGMA-THETA (60M) A B C D E F G
SIGMA-THETA (10M) A B C D E F G
OTHER MET DATA
BAROMETRIC PRESSURE 31.056
RELATIVE HUMIDITY 50.0
DEWPOINT 55.0
PRECIPITATION SINCE MIDNIGHT
0.00
4
3
2
1
INCHES
CNS
CPU S1
RPV CONTROL
PRIMARY CONTAINMENT
SECONDARY CONTAINMENT
RADIOACTIVE RELEASE
CON 08
SIMULATOR
F1=CLEAR F2= F3=MENU F4=SPDS INPT F5= F6=

Answer: B. Display B		
Explanation: Procedure 5.7.11 Attachment 2 states use the SOUTH route if wind is from 90º-180º. 140º, answer B, falls within this range.		
Distracters: All answers are (approximately) evenly spaced apart. All distracters are plausible since they fall within the total range of wind directions, 0º to 360º. Each distracter is wrong because it falls outside of the range for using the SOUTH route, 90º-180º, and each falls within the ranges given by procedure 5.7.11 attachment 2 for using the NORTH route, 0 º-89º and 180 º-360º. Answer A falls within 0 º-89º, and answers C and D fall within 180 º-360º Distracter A depicts wind direction ~70 º. Distracter C depicts wind direction ~215 º. Distracter D depicts wind direction ~285 º.		
Technical References: Procedure 5.7.11 (Early Dismiss/Evacuation of Site Personnel), Rev. 18 Procedure 5.7.2 (Emergency Director EPIP), Rev. 33		
References to be provided to applicants during exam: Procedure 5.7.11 Attachment 2		
Learning Objective:		
GEN0030401E0E0500 State the primary meteorological parameters used in dose assessment. GEN0030401D0D0800 Given plant conditions, procedure 5.7.2 and 5.7.20, determine appropriate protective action recommendations		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:		
	55.43(b)(5)	
Level of Difficulty:	2	
SRO Only Justification:		
The Emergency Director determines the route of evacuation based on wind direction		

and release rates.

Evacuation require ERO members to perform these evolutions.

1.3.8 Step N/A-9 allows entry into EPIP 5.7.11 when the ED determines that non-ERO personnel should be released. It is advisable to release non-ERO personnel if the event has the potential to upgrade to avoid having to perform Evacuation and monitoring (required at SAE or higher classification), as this will take away ERO resources.

1.4 Off-Site Notification leg, bases.

1.4.1 Step N/N-1 verifies that the communicator is available to perform notifications to off-site agencies within the required 15 minutes from declaration.

1.4.2 Step N/N-2 directs the completion of an Initial Notification Form per EPIP 5.7.6.

1.4.2.1 NOTE 1 states that the recommended PAR for a NOUE is "NONE".

1.4.3 Step N/N-3 ensures the initial notification is reviewed and signed to authorize transmittal to off-site agencies.

PROCEDURE 5.7.2	REVISION 33	PAGE 18 OF 36

ATTACHMENT 2 EVACUATION ANNOUNCEMENT

1. This attachment is to be used to announce the evacuation of non-ERO and/or ERO Relief Shift personnel (ERO personnel being evacuated and who are to return later to perform ERO relief duty). The announcement(s) can be performed individually or concurrently based on emergency conditions.

2. Fill in blanks for Step 2.3 using guidance below.

2.1 Determine the appropriate Evacuation Route and fill in blank (1).

2.1.1 Wind from 90° to 180°, use SOUTH Route.

2.1.2 Wind from 0° to 89° OR 181° to 360°, use NORTH Route.

2.2 IF there are hazards to be avoided, THEN complete blank (2) as needed. 'N/A' if there are no hazards.

- Locations to be avoided or specific routes to be taken due to emergency conditions.
- Any precautions needed for security events/severe weather conditions.

2.3 Announce following over station Gaitronics:

"Attention all NON-ERO and/or ERO Relief Shift personnel. The Emergency Director has directed that NON-ERO and/or ERO Relief Shift personnel be EVACUATED from the Site.

Non-ERO and/or ERO Relief Shift personnel are being evacuated to the Off-Site Assembly Area in Auburn using the (1) _____ Route. Proceed directly to the Off-Site Assembly Area and remain there until released.

Stay clear of (2) ____

____

____."

2.4 Repeat announcement.

Examination Outline Cross-Reference	Level	SRO
SRO determines which EOPs are applicable based upon conditions provided – no change required. Moved condensate information to bulleted conditions and add IAW EOP 1A to answers C & D.	Tier#	1
	Group#	2
	K/A #	295029, G2.4.6
	Rating	4.7
295029 High Suppression Pool Wtr Lvl G2.4.6 Knowledge of EOP mitigation strategies. (CFR: 41.10 / 43.5 / 45.13)		

Question 85

The following conditions exist during a LOCA:

- Reactor pressure is 150 psig, stable.
- Reactor water level is -160 inches, lowering 1 inch/min.
- Torus water level is 16.2 ft., rising 0.2 inch/min.
- Condensate Booster Pump A and Condensate Pump A are the only available injection source to the RPV.

Which of the following strategies is the CRS required to direct for the current conditions?

A. Secure Condensate system injection IAW EOP-3A.
Reduce Torus water level using RHR IAW 2.2.69.3.

B. Secure Condensate system injection IAW EOP-3A.
Emergency depressurize IAW EOP-2A.

C. Continue injection with the Condensate system IAW EOP-1A.
Reduce Torus water level using RHR IAW 2.2.69.3.

D. Continue injection with the Condensate system IAW EOP-1A.
Emergency depressurize IAW EOP-2A.

Answer: D Continue injection with the Condensate system IAW EOP-1A Emergency depressurize IAW EOP-2A.
Explanation:

This question requires prioritization of potentially contradictory EOP actions. When SP level cannot be maintained below 16.0 ft, EOP-3A step SP/L-5 directs securing injection systems that take suction from outside primary containment, if adequate core cooling can be assured. In this case, RPV level is below -160 inches and lowering; therefore, Condensate pump A injection is necessary for adequate core cooling. Adequate core cooling is maintained, for the present, by level above -183 inches with Condensate A injection. Since level is lowering and no other injection systems are available, EOP-1A step RC/L-15 should be answered NO, that level cannot be maintained above -183 inches, the point at which adequate core cooling will be lost, resulting in Emergency Depressurization is required. Thus, answer D is correct.

Distracters:
Distracters that include securing condensate pump A, which takes suction from the hotwell, are plausible, since EOP-3A step SP/L-5 directs securing injection systems that take suction from outside primary containment. This includes answers A and B. However, these answers are wrong because that injection should only be secured if other systems are available to assure adequate core cooling are available, but the stem states only Condensate pump A is available.

Answers that include reducing Torus water level using RHR are plausible since EOP-3A steps SP/L-1 and SP/L-3 direct using RHR to control SP level, in this case by rejecting water to radwaste or to the condenser. This includes answers A and C. These are wrong because for the given RPV level, a Group 2 isolation would be present, preventing opening of the RHR reject valves, and there is no provision to defeat that interlock.

Technical References: EOP-1A, RPV Control; EOP-3A, Primary Containment Control, EPGs rev 3

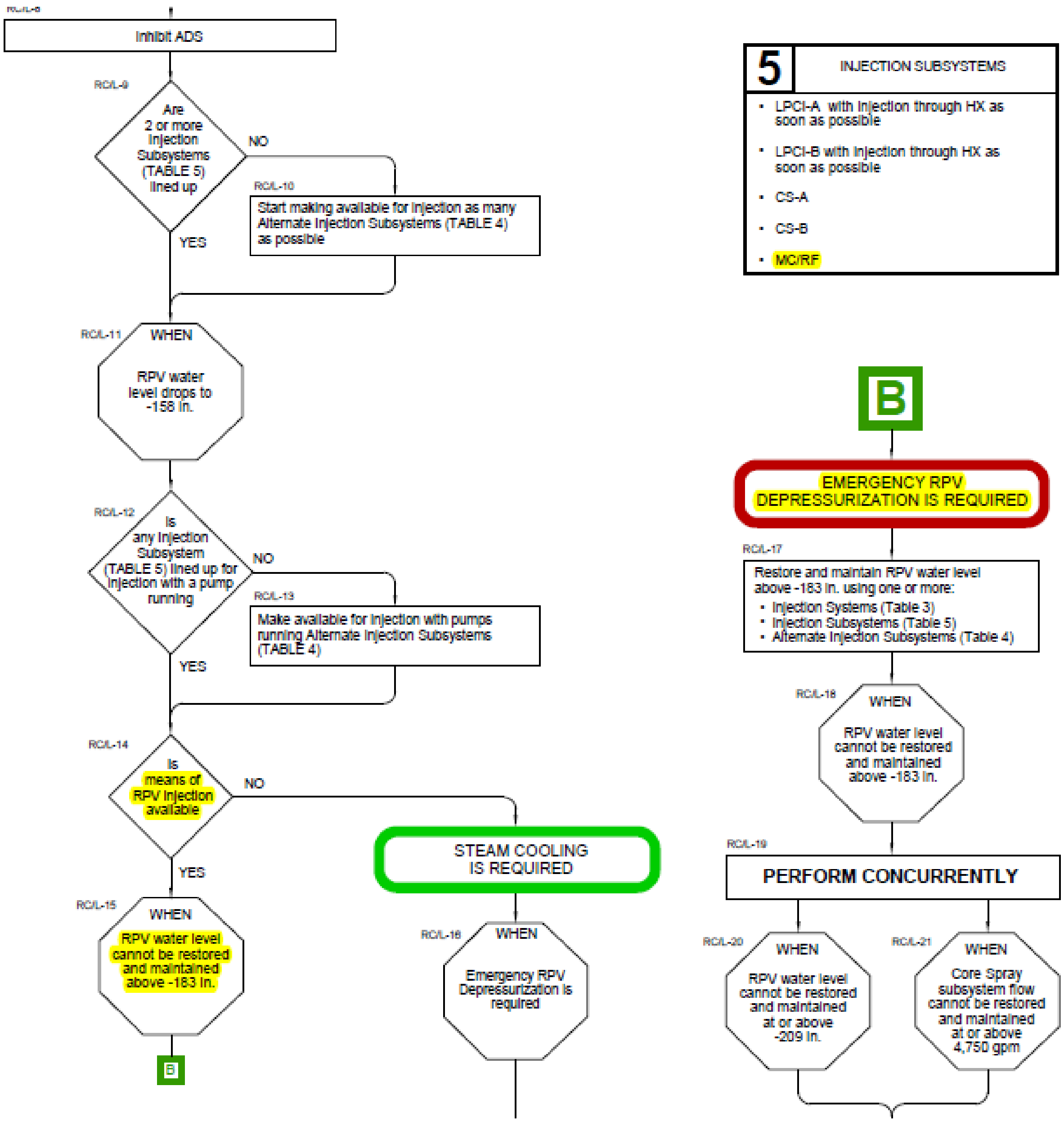
References to be provided to applicants during exam: none

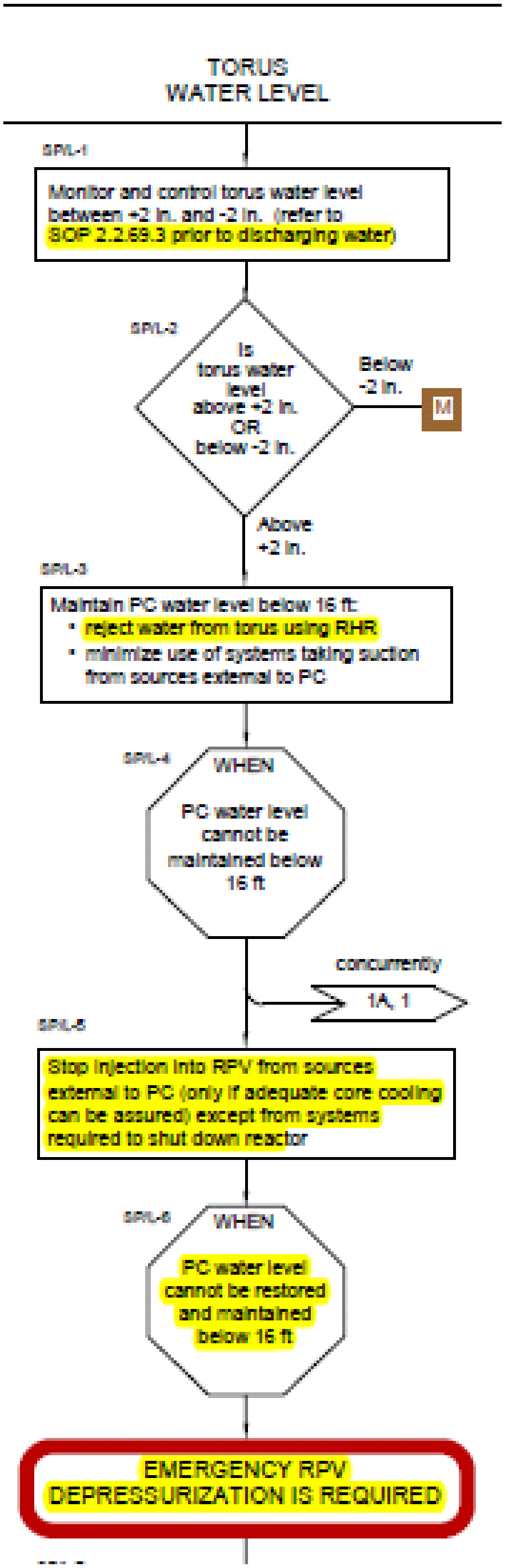
Learning Objective:
INT0320126Q0Q0100 Given plant condition(s), and the applicable Abnormal/Emergency Procedure, determine the correct subsequent actions required to mitigate the event(s).

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X

10CFR Part 55 Content:	
	55.43(b)(5)
Level of Difficulty:	3

SRO Only Justification:	
This involves SRO decision making regarding procedure selection and execution.	




TORUS
WATER LEVEL
SP/L-1
Monitor and control torus water level between +2 in. and -2 in. (refer to SOP 2.2.69.3 prior to discharging water)
SP/L-2
Is torus water level above +2 in. OR below -2 in.
Below -2 in.
M
Above +2 in.
SP/L-3
Maintain PC water level below 16 ft:
• reject water from torus using RHR
• minimize use of systems taking suction from sources external to PC
SP/L-4
WHEN
PC water level cannot be maintained below 16 ft
concurrently
1A, 1
SP/L-5
Stop injection into RPV from sources external to PC (only if adequate core cooling can be assured) except from systems required to shut down reactor
SP/L-6
WHEN
PC water level cannot be restored and maintained below 16 ft
EMERGENCY RPV DEPRESSURIZATION IS REQUIRED

EPG/SAG Step (SP/L-3.1, continued)

If suppression pool water level and RPV pressure cannot be maintained below the SRV Tail Pipe Level Limit but only if adequate core cooling is assured, terminate injection into the RPV from sources external to the primary containment except from systems required to shut down the reactor.

Discussion

A break in the RPV may be contributing to the high suppression pool water level condition being addressed in Step SP/L-3.1; water being injected into the RPV may be spilling out a break and accumulating in the suppression pool. Accordingly, injection from sources outside the primary containment is terminated to prevent any further increase in suppression pool water level that may occur through this mechanism. Assuring adequate core cooling takes precedence over terminating injection into the RPV from external sources since additional action can still be taken to prevent SRV system damage and containment failure. Operation of systems used to inject boron or insert control rods need not be terminated if the systems are being used to shut down the reactor.

Examination Outline Cross-Reference	Level	SRO
	Tier#	2
	Group#	1
	K/A #	211000, A2.03
	Rating	3.4

211000 SLC

A2. Ability to (a) predict the impacts of the following on the STANDBY LIQUID CONTROL SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5 / 45.6)
A2.03 A.C. power failures

Question 86

Reference Provided

The plant is at rated power.

SLC Pump A was declared inoperable to perform planned maintenance at 0000 on December 1.

On December 5 at 1200, DG2 is declared inoperable due to a blown control power fuse in its starting circuit.

What is the **LATEST** time that Mode 3 is **required** to be entered per Tech Specs for this condition?

A. December 6 at 0800

B. December 6 at 1200

C. December 8 at 1200

D. December 13 at 0000

Answer:
B. December 6 at 1200

Explanation:
With Div 2 DG inop, per TS 3.8.1.B.2, SLC B, which is redundant to SLC A, must be declared inop within 4 hours. Then, with both SLC A and B inop, TS 3.1.7 requires restoring one SLC operable within the next 8 hours, or being in HSD within the following 12 hours. (4hr + 8hr + 12hr = 24 hours from December 5 at 1200, thus December 6 at 1200).

Distracters:
Answer A is plausible because it reflects the time if SLC B is declared inop as soon as Div 2 DG is inop. It is the time for both SLCs being inop (8hr +12hr from Dec. 5 at 1200.) This would be an incorrect application of TS 3.0.6 since a specific action for this condition is given in TS, as described in TS 3.8.1 bases for Action B.2. It is wrong because it is not the maximum time allowed by TS.

Answer C is plausible if the candidate does not recognize the support/redundancy relationship between Div 2 DG, SLC A, and SLC B. This answer is wrong because it only reflects the time required by TS 3.1.7 for SLC A inop. (7d + 12hr from Dec 1 at 0000).

Answer D is plausible because it reflects the time if only Div 2 DG operability is considered (7d+12hr from Dec. 5 at 1200), and for that reason it is also wrong. The unprepared candidate would focus on the DG inoperability alone due to its importance and the familiar high level of focus DGs usually receive.

Technical References: TS 3.1.7, TS 3.8.1 and bases

References to be provided to applicants during exam: TS 3.1.7 and TS 3.8.1, (LCOs through Conditions/Actions only)

Learning Objective:
INT00705010010200 Given plant conditions and a Specification, apply the rules of Section 3.0 to determine appropriate actions.

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:		
	55.43(b)(5)	
Level of Difficulty:	3	
SRO Only Justification:		

This requires application of TS Action statements.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.3 Restore offsite circuit to OPERABLE status.	7 days AND 14 days from discovery of failure to meet LCO
B. One DG inoperable.	B.1 Perform SR 3.8.1.1 for OPERABLE offsite circuit(s). AND B.2 Declare required feature(s), supported by the inoperable DG, inoperable when the redundant required feature(s) are inoperable. AND	1 hour AND Once per 8 hours thereafter 4 hours from discovery of Condition B concurrent with inoperability of redundant required feature(s) (continued)

BASES

ACTIONS (continued)

entered concurrently. The "AND" connector between the 7 day and 14 day Completion Times means that both Completion Times apply simultaneously, and the more restrictive Completion Time must be met.

Similar to Required Action A.2, the second Completion Time of Required Action A.3 allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." This exception results in establishing the "time zero" at the time the LCO was initially not met, instead of at the time that Condition A was entered.

B.1

To ensure a highly reliable power source remains with one DG inoperable, it is necessary to verify the availability of the offsite circuits on a more frequent basis. Since the Required Action only specifies "perform," a failure of SR 3.8.1.1 acceptance criteria does not result in a Required Action being not met. However, if a circuit fails to pass SR 3.8.1.1, it is inoperable. Upon offsite circuit inoperability, additional Conditions must then be entered.

B.2

Required Action B.2 is intended to provide assurance that a loss of offsite power, during the period that a DG is inoperable, does not result in a complete loss of safety function of critical systems. These features are designed to be powered from redundant safety related divisions. Redundant required features failures consist of inoperable features associated with a division redundant to the division that has an inoperable DG.

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." In this Required Action the Completion Time only begins on discovery that both:

a. An inoperable DG exists; and

b. A redundant required feature on the other division is inoperable.

If, at any time during the existence of this Condition (one DG inoperable), a redundant required feature subsequently becomes inoperable, this Completion Time begins to be tracked.

3.1 REACTIVITY CONTROL SYSTEMS

3.1.7 Standby Liquid Control (SLC) System

LCO 3.1.7 Two SLC subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SLC subsystem inoperable.	A.1 Restore SLC subsystem to OPERABLE status.	7 days
B. Two SLC subsystems inoperable.	B.1 Restore one SLC subsystem to OPERABLE status.	8 hours
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 4.	12 hours 36 hours

AC Sources - Operating
3.8.1

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. (continued)	B.3.1	Determine OPERABLE DG is not inoperable due to common cause failure.	24 hours
	<u>OR</u>		
	B.3.2	Perform SR 3.8.1.2 for OPERABLE DG.	24 hours
	<u>AND</u>		
	B.4	Restore DG to OPERABLE status.	7 days <u>AND</u> 14 days from discovery of failure to meet LCO
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)
	<u>AND</u>		
	C.2	Restore one offsite circuit to OPERABLE status.	24 hours

(continued)

Examination Outline Cross-Reference	Level	SRO
TS 3.3.1.1 is no longer being provided as a reference. Conditions provided allow for determination of inoperable IRMs due to not achieving proper SRM/IRM overlap – No change.	Tier#	2
	Group#	1
	K/A #	215003, G2.2.42
	Rating	4.6
215003 IRM G2.2.42 Ability to recognize system parameters that are entry-level conditions for Technical Specifications. (CFR: 41.7 / 41.10 / 43.2 / 43.3 / 45.3)		

Question 87

Plant start up is in progress.

All SRMs are fully inserted and indicate between 1.8 E6 cps and 1.9 E6 cps, rising.

IRM status is as follows when SRM withdrawal begins:

- IRM A is bypassed.
- IRM B reads 2/40 on range 3
- IRM C reads 20/40 on range 3
- IRM D reads 25/40 on range 1
- IRM E reads 1/40 on range 1
- IRM F reads 40/125 on range 4
- IRM G reads 8/125 on range 4
- IRM H reads 70/125 on range 2

Which of the following identifies the Tech Spec requirements for this condition?

A. **Only** one Division 1 IRM is inoperable.
Initiate only a Potential LCO Tracking Report.

B. **Only** one Division 1 IRM is inoperable.
Initiate an Active TS LCO Tracking Report.

C. Two Division 1 IRMs are inoperable.
Initiate an Active TS LCO Tracking Report.

D. One Division 1 IRM and one Division 2 IRM are inoperable
Initiate only a Potential LCO Tracking Report.

Answer:
C. Two Division 1 IRMs are inoperable. Initiate an Active TS LCO Tracking Report.
Explanation: TS SR 3.3.1.1.5 requires SRM/IRM overlap to be verified while SRMs are fully inserted. TS bases for SR 3.3.1.1.5 states if overlap for a group of channels is not demonstrated (e.g., IRM/APRM overlap), the reason for the failure of the Surveillance should be determined and the appropriate channel(s) declared inoperable. As defined in TS Bases for SR 3.3.1.1.5, Overlap between SRMs and IRMs is the IRM channel is on scale before retracting SRMs from fully inserted. This would be the IRM reading greater than 1/40 of scale on range 1, since that is the bottom of the LOG scale. In the case given IRM E is downscale, so overlap does not exist, and it must be declared inop when SRMs are retracted from fully inserted. Also, IRM A is bypassed, which makes it inoperable. IRMs A and E are both Division 1. With less than three Division 1 IRMs operable, TS 3.3.1.1 Condition A applies. An Active LCO must be entered per 2.0.11, Entering and Exiting Technical Specification/TRM/ODCM LCO Condition(s), Attachement 6 step 1.7.2. Therefore, answer C is correct.
Distracters: . Answer A is plausible because IRM A is the only IRM in Bypass. The unprepared candidate may not recognize SR 3.3.1.1.5 requirements are not met, or he may not recognize that bypassed IRM A is inoperable. It is wrong because IRM E is also inoperable, and an Active LCO is required. Answer B is plausible because IRM A is the only IRM in Bypass. The unprepared candidate may not know that only three IRMs per division are required operable in Mode 2 per TS 3.3.1.1 and conclude an Active LCO is required. It is wrong because both IRM A and IRM E are inoperable. Answer D is plausible because IRM A is bypassed/inoperable and because Division 2 IRM F is reading higher than all other IRMs. The unprepared candidate may believe IRM H would not pass channel check criteria, but per 6.LOG.601, delta between IRM readings is N/A for acceptance criteria, so channel check criteria are met. It is wrong because IRM A and E only are inoperable, and because that requires an Active LCO.
Technical References: TS 3.3.1.1 and bases Procedure 2.0.11 {Entering And Exiting Technical Specification/TRM/ODAM LCO Condition(S}, Rev. 41 Procedure 6.LOG.601 (Daily Surveillance Log - Modes 1, 2, And 3), Rev. 117
References to be provided to applicants during exam: none
Learning Objective:

INT00705040010300 Given a set of plant conditions that constitutes non-compliance with a Section 3.3 LCO, determine the ACTIONS that are required. INT00705040010100 Given a set of plant conditions, recognize non-compliance with a Section 3.3 Requirement.		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:		
	55.43(b)(2)	
Level of Difficulty:	3	
SRO Only Justification:		
This requires knowledge of TS bases regarding IRM/SRM overlap criteria and determination of what type of LCO is required per administrative procedure.		

RPS Instrumentation
3.3.1.1

Table 3.3.1.1-1 (page 1 of 3)
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 3.3.1.1.4 SR 3.3.1.1.5 SR 3.3.1.1.6 SR 3.3.1.1.12(a,b) SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 121/125 divisions of full scale
	5(c)	3	H	SR 3.3.1.1.1 SR 3.3.1.1.3 SR 3.3.1.1.4 SR 3.3.1.1.12(a,b) SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 121/125 divisions of full scale
b. Inop	2	3	G	SR 3.3.1.1.3 SR 3.3.1.1.4 SR 3.3.1.1.13	NA
	5(c)	3	H	SR 3.3.1.1.3 SR 3.3.1.1.4 SR 3.3.1.1.13	NA
2. Average Power Range Monitors					
a. Neutron Flux — High (Startup)	2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.3 SR 3.3.1.1.4 SR 3.3.1.1.6 SR 3.3.1.1.8 SR 3.3.1.1.10(a,b) SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 14.5% RTP
b. Neutron Flux-High (Flow Biased)	1	2	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.4 SR 3.3.1.1.7 SR 3.3.1.1.8 SR 3.3.1.1.9 SR 3.3.1.1.10(a,b) SR 3.3.1.1.12(a,b) SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 0.75 W + 62.0% RTP(d)

(continued)

(a) 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.

(b) 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 Limiting Trip Setpoint and the methodologies used to determine the as-found and the as-left tolerances are specified in the Technical Requirements Manual

(c) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

(d) [0.75 W + 62.0% - 0.75 ΔW] RTP when reset for single loop operation per LCO 3.4.1, "Recirculation Loops Operating."

RPS Instrumentation
3.3.1.1

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
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	Adjust the channel to conform to a calibrated flow signal.	31 days
SR 3.3.1.1.8	Calibrate the local power range monitors.	1000 MWD/T average core exposure
SR 3.3.1.1.9	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.1.1.10	------------------------------NOTES------------------------------ 1. Neutron detectors and recirculation loop flow transmitters are excluded. 2. For Function 2.a, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. --- Perform CHANNEL CALIBRATION.	184 days

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

Surveillance Frequency extensions for RPS Functions, described in Reference 11, were not affected by the difference in configuration, since each automatic RPS channel has a test switch which is functionally the same as the manual scram switches in the generic model. As such, a functional test of each RPS channel test switch is required to be performed once every 7 days. The Frequency of 7 days is based on the reliability analysis of Reference 12.

SR 3.3.1.1.5 and SR 3.3.1.1.6

These Surveillances are established to ensure that no gaps in neutron flux indication exist from subcritical to power operation for monitoring core reactivity status.

The overlap between SRMs and IRMs is required to be demonstrated to ensure that reactor power will not be increased into a neutron flux region without adequate indication. This is required prior to withdrawing SRMs from the fully inserted position since indication is being transitioned from the SRMs to the IRMs.

The overlap between IRMs and APRMs is of concern when reducing power into the IRM range. On power increases, the system design will prevent further increases (by initiating a rod block) if adequate overlap is not maintained. Overlap between IRMs and APRMs exists when sufficient IRMs and APRMs concurrently have onscale readings such that the transition between MODE 1 and MODE 2 can be made without either APRM downscale rod block, or IRM upscale rod block. On controlled shutdowns, the IRM reading 121/125 of full scale will be set equal to or less than 45% of rated power. All range scales above that scale on which the most recent IRM calibration was performed will be mechanically blocked. Overlap between SRMs and IRMs similarly exists when, prior to withdrawing the SRMs from the fully inserted position, all operable IRM channels shall be on scale.

As noted, SR 3.3.1.1.6 is only required to be met during entry into MODE 2 from MODE 1. That is, after the overlap requirement has been met and indication has transitioned to the IRMs, maintaining overlap is not required (APRMs may be reading downscale once in MODE 2).

If overlap for a group of channels is not demonstrated (e.g., IRM/APRM overlap), the reason for the failure of the Surveillance should be determined and the appropriate channel(s) declared inoperable. Only those appropriate channels that are required in the current MODE or condition should be declared inoperable.

ATTACHMENT 6 INFORMATION SHEET

- 1.5 The Safety Function Determination Program ensures any LOSF is detected and appropriate actions taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if a LOSF exists. Additionally, other appropriate limitations and remedial or compensatory actions may be identified to be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. This procedure implements the requirements of Technical Specification Program 5.5.11.

  - 1.5.1 The applicability of LCO 3.0.6 includes all levels of supported systems (i.e., a system supports a system which, in turn, supports a system). These lower level relationships may transition from systems in TS to systems not in TS to systems in TS. For example, Service Water (a TS System) supports the ECCS Room coolers (non-TS System) which support the ECCS pumps (TS Systems). If an ECCS pump is inoperable solely because its room cooler is inoperable, and the room cooler is inoperable solely because Service Water is inoperable, then it is appropriate to apply LCO 3.0.6.

- 1.6 This procedure utilizes the user identification/password combination in SAP for electronic authentication as defined in Procedure 1.9.

- 1.7 DEFINITIONS

  - 1.7.1 OPERABLE - OPERABILITY - A system, subsystem, division, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, division, component, or device to perform its specified safety function(s) are also capable of performing their safety-related and/or important-to-safety function(s).

  - 1.7.2 ACTIVE LCO - An Active LCO shall be entered for those conditions where the SSC design function is required to be OPERABLE in the current mode as defined by Technical Specifications.

  - 1.7.3 POTENTIAL LCO - A POTENTIAL LCO shall be entered when any of the following conditions are met:

    - 1.7.3.1 A condition where any SSC is inoperable, but is not required to be OPERABLE, in the current mode.

    - 1.7.3.2 A condition where any SSC is inoperable, but the minimum required number of components is still met.

    - 1.7.3.3 A condition where any SSC is OPERABLE based on actions required to verify and/or compensate for the specific condition.

PROCEDURE 2.0.11 REVISION 40 PAGE 21 OF 25

ATTACHMENT 7 NEUTRON MONITORING INSTRUMENT CHECKS

LOC	INSTRUMENT	0700-1000 READING/ RANGE	1900-2200 READING/ RANGE	OPERABILITY LIMIT	MAX Δ	APPLICABLE MODES	ATT. 22 NOTES
PNL 9-5	NM-NR-46A IRM A	/	/	Channel Check	N/A	2	118
PNL 9-5	NM-NR-46A IRM C	/	/				
PNL 9-5	NM-NR-46C IRM E	/	/				
PNL 9-5	NM-NR-46C IRM G	/	/				
PNL 9-5	NM-NR-46B IRM B	/	/				
PNL 9-5	NM-NR-46B IRM D	/	/				
PNL 9-5	NM-NR-46D IRM F	/	/				
PNL 9-5	NM-NR-46D IRM H	/	/				
LOC	INSTRUMENT	0700-1000 SAT/UNSAT (a)	1900-2200 SAT/UNSAT (a)	OPERABILITY LIMIT	MAX Δ	APPLICABLE MODES	ATT. 22 NOTES
PNL 9-5	IRMs Detector Not Full In	MCU		SAT	N/A	2	75

(a) Satisfactory if:
1. Any non-bypassed IRM is not full in and a rod block results; or
2. All non-bypassed IRMs full in and no rod block from this function.

Section Break (Next Page)

ATTACHMENT 7 NEUTRON MONITORING INSTRUMENT CHECKS

LOC	INSTRUMENT	0700-1000 READING/ RANGE	1900-2200 READING/ RANGE	OPERABILITY LIMIT	MAX Δ	APPLICABLE MODES	ATT. 22 NOTES
PNL 9-12	IRM A	....../	....../	Channel Check	N/A	1, 2	8, 76, 77, 118
PNL 9-12	IRM C	....../	....../				
PNL 9-12	IRM E	....../	....../				
PNL 9-12	IRM G	....../	....../				
PNL 9-12	IRM B	....../	....../				
PNL 9-12	IRM D	....../	....../				
PNL 9-12	IRM F	....../	....../				
PNL 9-12	IRM H	....../	....../				

Determine range at Panel 9-5.

1. Readings taken except when instrument INOP.
2. IF IRM is INOP, THEN ensure companion APRM downscale trip OPERABILITY is addressed (i.e., IRM Mode switch out of operate).

LOC	INSTRUMENT	0700-1000 READING	1900-2200 READING	OPERABILITY LIMIT	MAX Δ	APPLICABLE MODES	ATT. 22 NOTES
PNL 9-12	SRM A			≥ 3.0 cps	N/A	2, 3	13, 14, 73, 74, 111, 113
PNL 9-12	SRM B						
PNL 9-12	SRM C						
PNL 9-12	SRM D						

Section Break (Next Page)

Examination Outline Cross-Reference	**Level**	SRO
Removed TS 3.3.1.1 reference and changed second part of question to determine APRM operability by evaluating whether RPS trip capability is maintained or NOT.	**Tier#**	2
	Group#	1
	K/A #	215005, A2.05
	Rating	3.6

215005 APRM / LPRM

A2. Ability to (a) predict the impacts of the following on the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5 / 45.6)

A2.05 Loss of recirculation flow signal

Examination Outline Cross-Reference	Level	SRO
Removed TS 3.3.1.1 reference and changed second part of question to determine APRM operability by evaluating whether RPS trip capability is maintained or NOT.	Tier#	2
	Group#	1
	K/A #	215005, A2.05
	Rating	3.6

215005 APRM / LPRM

A2. Ability to (a) predict the impacts of the following on the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5 / 45.6)

A2.05 Loss of recirculation flow signal

Question 88

The plant is operating at 85% power with Total Core Flow at 76%.

The signals from both Recirculation Drive Flow Transmitters (FT-110A and FT-110B) on Recirculation Loop A fail to zero.

(1) What is the direct effect of this condition?

AND

(2) What is the status of RPS Trip capability IAW TS 3.3.1.1 {Reactor Protection System (RPS) Instrumentation} due to this condition?

A. (1) Rod Block ONLY.
(2) Trip capability is MAINTAINED.

B. (1) Rod Block ONLY.
(2) Trip capability is NOT maintained.

C. (1) Rod Block and half scram.
(2) Trip capability is MAINTAINED.

D. (1) Rod Block and half scram.
(2) Trip capability is NOT maintained.

Answer: B. (1) Rod Block ONLY. (2) Trip capability is NOT maintained.
Explanation: Operation at 85% power on the 100% rod line places core flow approximately 70% on the Power-to-Flow Map. Each Recirc Loop discharge line has two flow transmitters that send flow signals to the APRM flow units. Each transmitter sends a signal to either the A Flow Unit or the B flow unit. Each Flow Unit receives a signal from one transmitter on the A Recirc Loop and one on the B Recirc Loop. Failure of both drive flow transmitters on either Recirc Loop to zero will result in both A and B Flow Units sensing a reduction of ~50%. The A Flow Unit feeds the Flow-Biased circuits for APRMs A, C, E and the B Flow Unit feeds APRMs B, D, F; therefore, all APRMs would be affected by this condition. In this case Recirc flow sensed by the flow units will reduce from ~76% to ~38%. APRMs will still sense 85% power since there is no actual change in Recirc flow. Plotting 85% power at 38% flow on the Power-to-Flow Map reveals the operating point sensed by the APRMs is above the Rod Block setpoint but below the Scram setpoint. An automatic rod block would be generated, both due to the APRM Flow-Biased function and due to the Flow Comparator function. Since the flow circuit is required for APRM operability, affected APRMs are inoperable per TS 3.3.1.1 bases. TS bases for TS 3.3.1.1 Action C.1 states for trip capability to be maintained, each RPS trip system must have at least one channel that is either OPERABLE or in trip. In this case for the RPS function, there are no channels that meet this requirement; therefore, trip capability is not maintained. Therefore, Answer B is correct. P/F Map is not needed due to APRM scram & Rod block setpoints being required to be known from memory. TS Scram is .75W+62% -> .75(38)+62= 90.5% 85% no scram. Rod Block is .75W+51% -> .75(38)+51=79.5% 85% rod block present
Distracters: Answer A is incorrect due to trip capability not being maintained. This choice is plausible due to confusing operability with being functional since the APRMs will still cause flow biased scrams. Since the APRMs are inoperable, functionality cannot be credited for maintaining trip function. The candidate correctly identifies the impact of this failure and confuses the Flow Unit Comparator function with respect to the APRM Flow-Biased Trip function required by TS 3.3.1.1 would select this answer. Answer C is incorrect due to the failure only causing a rod block AND trip capability not being maintained. This choice is plausible due to confusing flow bias scram setpoint for this power level AND confusing operability with being functional since the APRMs will still cause flow biased scrams. Since the APRMs are inoperable, functionality cannot be credited for maintaining trip function. The candidate confuses the impact of this failure and confuses the Flow Unit Comparator function with respect to the APRM Flow-Biased Trip

function required by TS 3.3.1.1 would select this answer.

Answer D is incorrect due to the failure only causing a rod block. This choice is plausible due to confusing flow bias scram setpoint for this power level. The candidate confuses the impact of this failure and recognizes the Flow Unit Comparator function with respect to the APRM Flow-Biased Trip function required by TS 3.3.1.1 would select this answer.

Technical References:
TS 3.3.1.1 & T3.3.1
P/F Map

References to be provided to applicants during exam: NONE

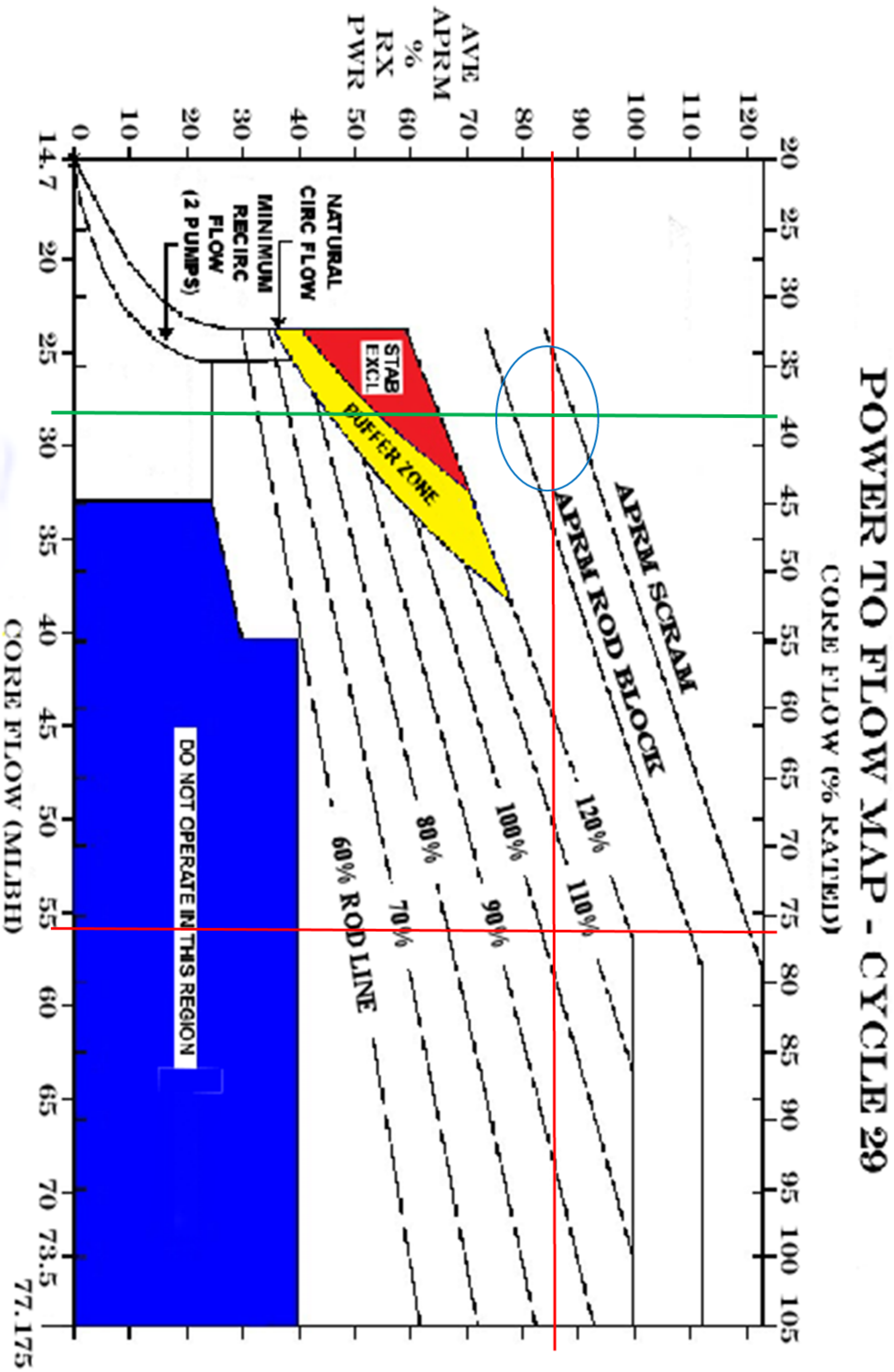
Learning Objective:

COR0020102001050G Describe the interrelationships between the Average Power Range Monitor System and the following: Flow converter/comparator network

INT00705040010300 Given a set of plant conditions that constitutes non-compliance with a Section 3.3 LCO, determine the ACTIONS that are required.

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:		
	55.43(b)(2)	
Level of Difficulty:	3	
SRO Only Justification:		

This question requires application of TS bases knowledge to determine operability and impact of inoperability.


POWER TO FLOW MAP - CYCLE 29
CORE FLOW (% RATED)
20 25 30 35 40 45 50 55 60 65 70 75 80 85 90 95 100 105
AVE APRM % RX PWR
0 10 20 30 40 50 60 70 80 90 100 110 120
CORE FLOW (MLBH)
14.7 20 25 30 35 40 45 50 55 60 65 70 73.5 77.175
NATURAL CIRC FLOW
MINIMUM RECIRC FLOW (2 PUMPS)
STAB EXCL
BUFFER ZONE
APRM SCRAM
APRM ROD BLOCK
120%
110%
100%
90%
80%
70%
60% ROD LINE
DO NOT OPERATE IN THIS REGION

BASES

APPLICABLE SAFETY ANALYSIS, LCO, and APPLICABILITY (continued)

specifically credited in the safety analyses, but is intended to provide protection against transients where THERMAL POWER increases slowly, and to provide protection for power oscillations which may result from reactor thermal hydraulic instability.

The APRM System is divided into two groups of channels with three APRM Channel inputs to each trip system. The system is designed to allow one channel in each trip system to be bypassed. Any one APRM channel in a trip system can cause the associated trip system to trip. Four channels of Average Power Range Monitor Neutron Flux-High (Flow Biased) with two channels in each trip system arranged in a one-out-of-two logic are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal. In addition, to provide adequate coverage of the entire core, at least 11 LPRM inputs are required for each APRM channel, with at least two LPRM inputs from each of the four axial levels at which the LPRMs are located. Each APRM channel receives a flow signal representative of total recirculation loop flow. The total recirculation loop drive flow signals are generated by two flow units, one of which supplies signals to the trip system A APRMs, while the other supplies signals to the trip system B APRMs. Each flow unit signal is provided by summing up the flow signals from the two recirculation loops. The instrumentation is an analog type with redundant flow signals that can be compared. Each required Average Power Range Monitor Neutron Flux-High (Flow Biased) channel requires an input from one OPERABLE flow unit. If a flow unit is inoperable, the associated Average Power Range Monitor Neutron Flux-High (Flow Biased) channels must be considered inoperable.

The terms for the Allowable Value of the APRM Neutron Flux-High (Flow Biased) trip are defined as follows: S is the setting in percent rated power; W is the two loop recirculation flow rate in percent rated flow (rated loop recirculation flow rate is that recirculation flow rate which provides 100% core flow at 100% power); ΔW is the difference between two loop and single loop effective drive flow at the same core flow. ΔW equals zero for two recirculation loop operation.

The Average Power Range Monitor Neutron Flux-High (Flow Biased) Function is required to be OPERABLE in MODE 1 when there is the possibility of generating excessive THERMAL POWER and potentially exceeding the SL applicable to high pressure and core flow conditions (MCPR SL). During MODES 2 and 5, other IRM and APRM Functions provide protection for fuel cladding integrity.

BASES

ACTIONS (continued)

C.1

Required Action C.1 is intended to ensure that appropriate actions are taken if multiple, inoperable, untripped channels within the same trip system for the same Function result in the Function not maintaining RPS trip capability. A Function is considered to be maintaining RPS trip capability when sufficient channels are OPERABLE or in trip (or the associated trip system is in trip), such that both trip systems will generate a trip signal from the given Function on a valid signal. For the typical Function with one-out-of-two taken twice logic and the IRM and APRM Functions, this would require both trip systems to have one channel OPERABLE or in trip (or the associated trip system in trip). For Function 5 (Main Steam Isolation Valve-Closure), this would require both trip systems to have each channel associated with the MSIVs in three main steam lines (not necessarily the same main steam lines for both trip systems) OPERABLE or in trip (or the associated trip system in trip). For Items 7.a and 7.b (Scram Discharge Volume Water Level - High, Level Transmitter and Level Switch), this would require both trip systems in each SDV to have one channel (either an Item 7.a or 7.b channel) OPERABLE or in trip (or the associated trip system in trip). For Function 8 (Turbine Stop Valve-Closure), this would require both trip systems to have two channels, each OPERABLE or in trip (or the associated trip system in trip). For Functions 10 (Reactor Mode Switch-Shutdown Position) and 11 (Manual Scram) this would require both trip systems to have one channel each OPERABLE or in trip (or the associated trip system in trip).

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. The 1 hour Completion Time is acceptable because it minimizes risk while allowing time for restoration or tripping of channels.

D.1

Required Action D.1 directs entry into the appropriate Condition referenced in Table 3.3.1.1-1. The applicable Condition specified in the Table is Function and MODE or other specified condition dependent and may change as the Required Action of a previous Condition is completed. Each time an inoperable channel has not met any Required Action of Condition A, B, or C and the associated Completion Time has expired, Condition D will be entered for that channel and provides for transfer to the appropriate subsequent Condition.

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.

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
	OR	
	A.2 Place associated trip system in trip.	12 hours
B. One or more Functions with one or more required channels inoperable in both trip systems.	B.1 Place channel in one trip system in trip.	6 hours
	OR	
	B.2 Place one trip system in trip.	6 hours
C. One or more Functions with RPS trip capability not maintained.	C.1 Restore RPS trip capability.	1 hour

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
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 < 29.5% 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

Table 3.3.1.1-1 (page 2 of 3)
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 (continued)					
c. Neutron Flux - High (Fixed)	1	2	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.4 SR 3.3.1.1.8 SR 3.3.1.1.9 SR 3.3.1.1.10(a,b) SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 120.0% RTP
d. Downscale	1	2	F	SR 3.3.1.1.4 SR 3.3.1.1.8 SR 3.3.1.1.9(a,b) SR 3.3.1.1.13	≥ 3.0% RTP
e. Inop	1,2	2	G	SR 3.3.1.1.4 SR 3.3.1.1.8 SR 3.3.1.1.9 SR 3.3.1.1.13	NA
3. Reactor Vessel Pressure — High	1,2	2	G	SR 3.3.1.1.4 SR 3.3.1.1.9 SR 3.3.1.1.12(a,b) SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 1050 psig
4. Reactor Vessel Water Level — Low (Level 3)	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.9 SR 3.3.1.1.12(a,b) SR 3.3.1.1.13 SR 3.3.1.1.15	≥ 3 inches
5. Main Steam Isolation Valve — Closure	1	4	F	SR 3.3.1.1.4 SR 3.3.1.1.9 SR 3.3.1.1.12 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 10% closed
6. Drywell Pressure — High	1,2	2	G	SR 3.3.1.1.4 SR 3.3.1.1.9 SR 3.3.1.1.12(a,b) SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 1.84 psig

(continued)

(a) 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.

(b) 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 Limiting Trip Setpoint and the methodologies used to determine the as-found and the as-left tolerances are specified in the Technical Requirements Manual.

T 3.3 INSTRUMENTATION

T 3.3.1 Control Rod Block Instrumentation

TLCO 3.3.1 The control rod block instrumentation for each Function in Table T3.3.1-1 shall be OPERABLE.

APPLICABILITY: According to Table T3.3.1-1.

ACTIONS

---NOTE---
Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with less than minimum channels OPERABLE.	A.1 Initiate Reactor Manual Control System rod withdrawal block.	1 hour

Table T3.3.1-1 (Page 2 of 3)
Control Rod Block Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	ACCEPTANCE LIMITS
2. IRM				
a. Detector Not Full In	2	6	TSR 3.3.1.1 TSR 3.3.1.3 TSR 3.3.1.8	NA
b. Upscale	2	6	TSR 3.3.1.1 TSR 3.3.1.3 TSR 3.3.1.4 TSR 3.3.1.8	≤ 108/125 of Full Scale
c. Inoperative	2	6	TSR 3.3.1.3 TSR 3.3.1.4	NA
d. Downscale	2[(d)]	6	TSR 3.3.1.1 TSR 3.3.1.3 TSR 3.3.1.4 TSR 3.3.1.8	≥ 2.5/125 of Full Scale
3. APRM				
a. Upscale (Flow Biased)	1	4	TSR 3.3.1.1 TSR 3.3.1.2 TSR 3.3.1.5 TSR 3.3.1.7[(e)]	≤ (0.75W + 51.0% - 0.75 ΔW) [(f)(g)]
b. Upscale (Startup)	2	4	TSR 3.3.1.1 TSR 3.3.1.4 TSR 3.3.1.7	≤ 11.5%
c. Inoperative	1, 2	4	TSR 3.3.1.1 TSR 3.3.1.5	NA
d. Downscale	1	4	TSR 3.3.1.1 TSR 3.3.1.5 TSR 3.3.1.7	≥ 3%
e. Upscale (Fixed)	1	4	TSR 3.3.1.1 TSR 3.3.1.2 TSR 3.3.1.5 TSR 3.3.1.7	≤ 109.5%

(continued)

(d) With IRMs on Range 2 or above.

(e) Calibration of the recirculation loop flow transmitters is only required once every 24 months.

(f) W is the two-loop recirculation flow rate in percent of rated. Trip level setting is in percent of rated power (2419 = 100% RTP).

(g) ΔW is the difference between two-loop and single-loop effective drive flow and is used for single recirculation loop operation. ΔW = 0 for two loop recirculation loop operation.

Table T3.3.1-1 (Page 3 of 3)
Control Rod Block Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	ACCEPTANCE LIMITS
4. Flow Bias Comparator	1, 2	2	TSR 3.3.1.4 TSR 3.3.1.8	≤ 10% Difference In Recirculation Flows
5. Flow Bias Upscale	1, 2	2	TSR 3.3.1.4 TSR 3.3.1.6[(e)]	≤ 110% Recirculation Flow
6. Scram Discharge Volume Water Level High	1, 2, 5	2	TSR 3.3.1.5 TSR 3.3.1.8	≤ 46 inches

(e) Calibration of the recirculation loop flow transmitters is only required once every 24 months.

Examination Outline Cross-Reference	Level	SRO
Modified stem to "The CRS has authorized I&C to start SR 3.3.2.2 (Feedwater and Main Turbine High Water Level Trip Instrumentation) Channel Calibration on NBI-LT-52C at 0800". Modified explanation to reflect CRS authorize to start at 0800.	Tier#	2
	Group#	1
	K/A #	259002, G2.2.12
	Rating	4.1
259002 Reactor Water Level Control G2.2.12 Knowledge of surveillance procedures. (CFR: 41.10 / 45.13)		

Question 89

The plant is at 30% power.

- The CRS has authorized I&C to start SR 3.3.2.2 (Feedwater and Main Turbine High Water Level Trip Instrumentation) Channel Calibration on NBI-LT-52C at 0800.
- I&C needs 12 hours to complete the surveillance.

There is a note before the applicable Tech Spec Surveillance Requirement that provides a Delayed Entry Time (DET) of 6 hours.

Regarding the provisions of Delayed Entry Time (DET) for performance of this surveillance IAW 0.26 (Surveillance Program):

(1) Is NBI-LT-52C considered to be operable or inoperable when the surveillance is started?

AND

(2) When is the associated Tech Spec Action statement **required** to be entered?

A. (1) Operable
(2) 0800

B. (1) Operable
(2) 1400

C. (1) Inoperable
(2) 0800

D. (1) Inoperable

(2) 1400

Answer: D. (1) Inoperable (2) 1400
Explanation: This question tests application rules for performing TS surveillances with respect to determining operability and entering TS action statements, SRO responsibilities. The note before SR 3.3.2.2.1 allows 6 hours before having to enter the respective condition/action statement provided sufficient channels are monitoring the parameter such that the trip function is maintained. This period is defined in procedure 0.26, Surveillance Program, as Delayed Entry Time (DET). Procedure 0.26 states use of DET does not circumvent having to declare the equipment inoperable. The stem states the CRS authorized SR start at 0800, therefore the affected channel must be considered inoperable at 0800. The remaining channels must be capable of causing the safety function if an actual reactor water level high condition was to occur. In this case, as described in TS bases, logic is any 2-out-of-3, so channels A and B maintain the trip function. Therefore, TS 3.3.2.2 Condition A is not required to be entered until 6 hours after 0800, i.e. 1400. Hence, answer D is correct.
Distracters: Answers that state in part (1) that the channel under test is considered operable are plausible, because the unprepared student that believes delayed entry into a TS action statement is appropriate may infer the reason is the channel is not inoperable during the DET. This applies for answers A and B. This is wrong, as clearly stated in procedure 0.26, Surveillance Program. Answers that state in part (2) that the TS action statement must be entered at 0800, immediately, are plausible because the unprepared student may believe removing one channel of the Level 8 trip defeats the safety function, since all 3 channels are required operable by the LCO. He may also remember per procedure 0.26 the channel must be considered to be inoperable as soon as it is removed from service, and confuse that to imply entry into a TS action cannot be delayed. He may also believe the DET allowance cannot be used since the stem states the surveillance is expected to take 12 hours, twice the 6 hour DET. This applies to answers A and C. This is wrong because with 2 channels remaining in service, trip capability is maintained, and the action statement is not **required**, as stated in the stem, to be entered for 6 hours.
Technical References: TS 3.3.2.2 note before SRs, TS 3.3.2.2 bases; procedure 0.26, Surveillance Program
References to be provided to applicants during exam: none

Learning Objective:		
SKL00801020010900 Briefly describe the administrative process for the application of DETs during surveillance testing.		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:		
	55.43(b)(2)	
Level of Difficulty:	3	
SRO Only Justification:		
This question tests knowledge of rules governing determining operability and entering TS action statements related to performing TS surveillances.		

3.3 INSTRUMENTATION

3.3.2.2 Feedwater and Main Turbine High Water Level Trip Instrumentation

LCO 3.3.2.2 Three channels of feedwater and main turbine high water level trip instrumentation shall be OPERABLE.

APPLICABILITY: THERMAL POWER $\geq$ 25% RTP.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One feedwater and main turbine high water level trip channel inoperable.	A.1 Place channel in trip.	7 days
B. Two or more feedwater and main turbine high water level trip channels inoperable.	B.1 Restore feedwater and main turbine high water level trip capability.	2 hours
C. Required Action and associated Completion Time not met.	C.1 Reduce THERMAL POWER to < 25% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

-----NOTE-----
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 feedwater and main turbine high water level trip capability is maintained.

SURVEILLANCE		FREQUENCY
SR 3.3.2.2.1	Perform CHANNEL CHECK.	24 hours
SR 3.3.2.2.2	Perform CHANNEL CALIBRATION. The Allowable Value shall be ≤ 54.0 inches.	24 months
SR 3.3.2.2.3	Perform LOGIC SYSTEM FUNCTIONAL TEST including valve actuation.	24 months

BASES

ACTIONS

A Note has been provided to modify the ACTIONS related to feedwater and main turbine high water level trip instrumentation channels. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent divisions subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable feedwater and main turbine high water level trip instrumentation channels provide appropriate compensatory measures for separate inoperable channels. As such, a Note has been provided that allows separate Condition entry for each inoperable feedwater and main turbine high water level trip instrumentation channel.

A.1

With one channel inoperable, the remaining two OPERABLE channels can provide the required trip signal. However, overall instrumentation reliability is reduced because a single failure in one of the remaining channels concurrent with feedwater controller failure, maximum demand event, may result in the instrumentation not being able to perform its intended function. Therefore, continued operation is only allowed for a limited time with one channel inoperable. If the inoperable channel cannot be restored to OPERABLE status within the Completion Time, the channel must be placed in the tripped condition per Required Action A.1. Placing the inoperable channel in trip would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to continue with no further restrictions. Alternately, if it is not desired to place the channel in trip (e.g., as in the case where placing the inoperable channel in trip would result in a feedwater or main turbine trip), Condition C must be entered and its Required Action taken.

The Completion Time of 7 days is based on the low probability of the event occurring coincident with a single failure in a remaining OPERABLE channel.

B.1

With two or more channels inoperable, the feedwater and main turbine high water level trip instrumentation cannot perform its design function (feedwater and main turbine high water level trip capability is not maintained). Therefore, continued operation is only permitted for a 2 hour period, during which feedwater and main turbine high water level trip capability must be restored. The trip capability is considered maintained

BASES

ACTIONS (continued)

when sufficient channels are OPERABLE or in trip such that the feedwater and main turbine high water level trip logic will generate a trip signal on a valid signal. This requires two channels to each be OPERABLE or in trip. If the required channels cannot be restored to OPERABLE status or placed in trip, Condition C must be entered and its Required Action taken.

The 2 hour Completion Time is sufficient for the operator to take corrective action, and takes into account the likelihood of an event requiring actuation of feedwater and main turbine high water level trip instrumentation occurring during this period. It is also consistent with the 2 hour Completion Time provided in LCO 3.2.2 for Required Action A.1, since this instrumentation's purpose is to preclude a MCPR violation.

C.1

With the required channels not restored to OPERABLE status or placed in trip, THERMAL POWER must be reduced to < 25% RTP within 4 hours. As discussed in the Applicability section of the Bases, operation below 25% RTP results in sufficient margin to the required limits, and the feedwater and main turbine high water level trip instrumentation is not required to protect fuel integrity during the feedwater controller failure, maximum demand event. The allowed Completion Time of 4 hours is based on operating experience to reduce THERMAL POWER to < 25% RTP from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

The Surveillances are modified by a Note to indicate that 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 feedwater and main turbine high water level trip capability. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken. This Note is based on the reliability analysis (Ref. 3) assumption that 6 hours is the average time required to perform channel Surveillance. That analysis demonstrated that the 6 hour testing allowance does not significantly reduce the probability that the feedwater pump turbines and main turbine will trip when necessary.

4.1.5 Prepare a schedule of surveillance tests that ensures:

4.1.5.1 Tests are performed within required frequency.

4.1.5.2 Weekly divisional separation of testing.

4.1.5.3 Coordination with plant conditions.

4.1.5.4 Minimizing duplicate testing.

5. SURVEILLANCE TEST AUTHORIZATION

NOTE – The authority to authorize performance of the test may be delegated to the CRS or the WCCA as the SM deems necessary.

5.1 Shift Manager shall:

5.1.1 Understand intent of test.

5.1.2 Understand direct and indirect actions of the test signals.

5.1.3 Ensure plant conditions are as required per the following:

5.1.3.1 If entire test is to be performed, ensure plant conditions are as required by the prerequisites of the test before authorizing test performance.

5.1.3.2 If only a portion of test is being performed (example, performing portions of a Surveillance Procedure for PWT), ensure applicable prerequisites are met prior to authorizing partial test performance.

5.1.4 If a scheduled divisional test, ensure it is authorized in its designated week.

5.1.5 Ensure performance of test will not compromise divisional protection of ECCS Systems.

5.1.6 Ensure no other tests are in-progress on the same system such that the combination of tests would result in erroneous signals or inadvertent initiation.

5.1.7 Be aware of any other systems affected by the test and how they are affected.

5.1.8 Ensure equivalent instrumentation, when used for Inservice Testing (IST) of pumps, meet the criteria as stated in the definition of equivalent instrument of this procedure.

NOTE – The use of delayed entry time (DET) does not circumvent the requirement to declare the SSC inoperable.

5.1.9 If a TS SR allows the use of delayed entry into an associated Condition and Required ACTION, use Attachment 1 or any similar documentation/electronic log, as defined in this procedure, to facilitate tracking the TS delayed entry time (DET).

ATTACHMENT 3	INFORMATION SHEET

b. AC that is not an associated SR and does not directly affect OPERABILITY of the associated TS component is addressed in one of the following ways:

1. An action is specified if the AC is not met; or

2. If an action is not specified, it is expected that corrective action processes and communication expectations to Management are used as applicable.

1.3.4 AC in TS, TRM, or ODAM Surveillance Procedures that are clearly not associated with the OPERABILITY of a system structure component (SSC) are identified with a ■Non-TS■ flag in Acceptance Criteria Section.

1.4 Setpoints listed in the Surveillance Procedure are the actual values the instruments are set at. Actual setpoints will vary from the TS Allowable Value or TRM Acceptance Limit due to application of elevation corrections and other conservative margins. Refer to Procedure 3.26 and the individual Instrument Setpoint Data Sheets for further clarification.

1.5 The designation of the first digit of the three digit follower on 6 series procedure numbers is:

100: Pump OPERABILITY/Surveillance
200: Valve OPERABILITY/Surveillance
300: CHANNEL CALIBRATION
400: IST Testing
500: ISI, ILRT, LLRT, or other leak tests
600: Miscellaneous
700: CHANNEL FUNCTIONAL

1.6 OPERABILITY DURING SURVEILLANCE TESTING GUIDELINES

1.6.1 Technical Specifications, Technical Requirements Manual, and Off-Site Dose Assessment Manual Surveillance Requirements may have notes that allow delayed entry into Conditions and Required ACTIONs for equipment made inoperable by performance of the surveillance. Even though delayed entry is allowed, the equipment/component is still considered inoperable while performing these surveillances. The delayed entry is only allowed if there is not a loss of function.

1.6.1.1 Restoration of an individual instrument within a Surveillance Procedure, including Independent Verification when required, shall be completed prior to testing another instrument within that Surveillance Procedure.Ⓟ[9]

ATTACHMENT 3 INFORMATION SHEET

1.9 DEFINITIONS

1.9.1 Delayed Entry Time (DET) - The TS allowed delay time for entry into an associated Condition and Required ACTION during performance of a required surveillance when the specified function is maintained.

1.9.2 Allowable Range, Calibration Tolerance, Tolerance, etc. - The allowable deviation from a specified or true value before re-adjustment is required. These are expected deviations between surveillance performances and are not considered Administrative Limits.

1.9.3 DET Tracking Log - Refers to Attachment 1 or any similar documentation/electronic log. If something other than Attachment 1 is used, it shall contain at a minimum: the date, procedure number, TS allowed delay time (DET), and DET start and stop time.

1.9.4 AS FOUND Data - Data taken prior to any adjustments.

1.9.5 AS LEFT Data - Data taken after adjustments are made or work is complete.

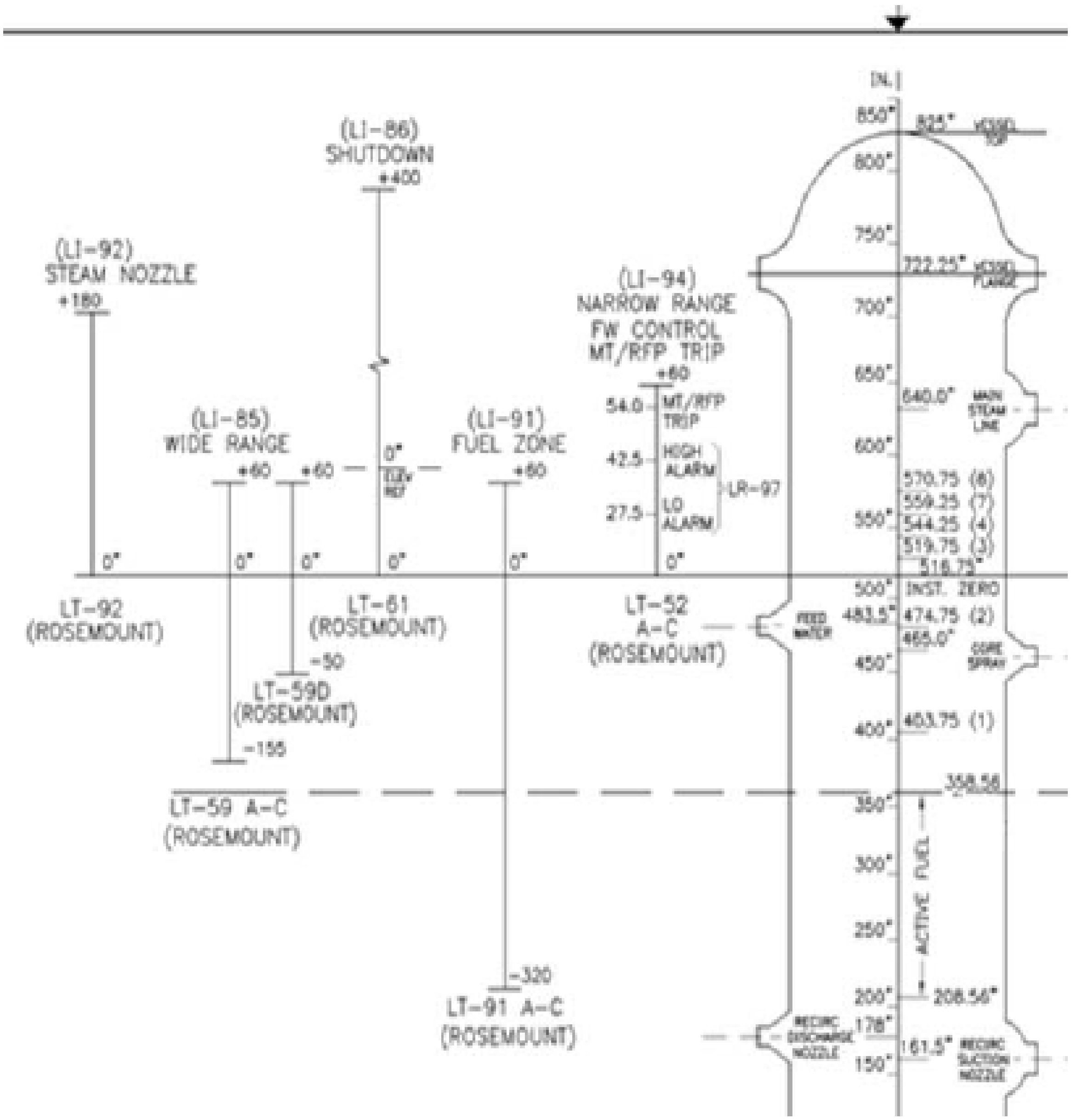
1.9.6 Current Plant Configuration - Includes Operating Mode, reactor power level, reactor temperature and pressure, equipment Out Of Service or Inoperable, etc.).Ⓟ[12]

1.9.7 Discrepancy - Any inconsistency between expected and observed responses during performance of procedure. Expected system or equipment response is defined by the Surveillance Procedure affecting it. If there is no conditional direction within the procedure for marking a step N/A, the N/A step is a discrepancy.

1.9.8 Discrepancy Sheet - Refers to Attachment 2 or any similar form. The Discrepancy Sheet may be used with non-surveillance program procedures for the purpose of recording discrepancies and resolutions. If a similar form is used, it shall contain at a minimum the procedure number as applicable, discrepancy, resolution, initial, and date.

1.9.9 Equivalent Instrument - For IST pump testing, when a different instrument other than specified by the Test Procedure is used or when temporary instrumentation is used to replace inoperable permanent instrumentation, the following criteria shall be applied to ensure equivalence:

1.9.9.1 Range - For analog instruments, the full scale range shall be less than or equal to that of the specified instrument or three times the reference value of the measured parameter. For digital instruments, the instrument shall be selected such that the reference value of the measured parameter does not exceed 70% of the calibrated range of the instrument.


(LI-92)
STEAM NOZZLE
+180
(LI-86)
SHUTDOWN
+400
(LI-94)
NARROW RANGE
FW CONTROL
MT/RFP TRIP
+60
54.0
MT/RFP TRIP
42.5
HIGH ALARM
27.5
LO ALARM
LR-97
(LI-85)
WIDE RANGE
+60
+60
0"
ELEV REF
(LI-91)
FUEL ZONE
+60
0"
LT-92
(ROSEMOUNT)
LT-61
(ROSEMOUNT)
LT-52
A-C
(ROSEMOUNT)
-50
LT-59D
(ROSEMOUNT)
-155
LT-59 A-C
(ROSEMOUNT)
-320
LT-91 A-C
(ROSEMOUNT)
IN.
850"
825" VESSEL TOP
800"
750"
722.25" VESSEL FLANGE
700"
650"
640.0" MAIN STEAM LINE
600"
570.75 (8)
559.25 (7)
550"
544.25 (4)
519.75 (3)
516.75"
500"
INST. ZERO
483.5"
474.75 (2)
FEED WATER
465.0"
CORE SPRAY
450"
400"
403.75 (1)
358.56
350"
300"
ACTIVE FUEL
250"
200"
208.56"
178"
RECIRC DISCHARGE NOZZLE
161.5"
RECIRC SUCTION NOZZLE
150"

Examination Outline Cross-Reference	Level	SRO
Revised question to predict ground indication and determine the correct procedure utilized to perform ground isolation on sensitive panels IAW procedure 2.0.1. Added "NO circuit analysis exists for this panel" to eliminate operational confusion.	Tier#	2
	Group#	1
	K/A #	263000, A2.01
	Rating	3.2

263000 DC Electrical Distribution

A2. Ability to (a) predict the impacts of the following on the D.C. ELECTRICAL DISTRIBUTION ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5 / 45.6)

A2.01 Grounds

Question 90

The plant is at rated power when the following annunciator alarms:

125V DC BUS 1A GROUND	PANEL/WINDOW: C-1/B-2

The ground is continuous (not intermittent).
The ground is on a SENSITIVE PANEL.
NO circuit analysis exists for this panel.

Which one of the following completes the statements below regarding the indication on 125 VDC Switchgear 1A ground detector (GRD DET) due to this ground and how ground isolation is required to be performed IAW Procedure 2.0.1 (Plant Operations Policy)?

Ground indicating lights indicate ____(1)____.
Ground isolation is accomplished by ____(2)____.

A. (1) equal brilliance
(2) creating and utilizing a work order IAW 0-CNS-WM-102 (Work Implementation and Closeout)

B. (1) equal brilliance
(2) temporarily de-energizing equipment IAW 2.2A_125DC.Div1 {125 VDC Power Checklist (Div 1)}. A work order is NOT required.

C. (1) one brighter than the other
(2) creating and utilizing a work order IAW 0-CNS-WM-102 (Work Implementation and Closeout)

D. (1) one brighter than the other
(2) temporarily de-energizing equipment IAW 2.2A_125DC.Div1 {125 VDC Power Checklist (Div 1)}. A work order is NOT required.

Answer:
C. (1) one brighter than the other
(2) creating and utilizing a work order IAW 0-CNS-WM-102 (Work Implementation and Closeout)

Explanation:
This question requires knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures by the SRO.
Ground Detection lights are installed on the switchgear. These lights are normally lit and do not have control switches like the ones on the battery charger. One light is connected between the positive bus and ground (labeled "GRD DET –") and the other light between negative bus and ground (labeled "GRD DET +"). If a ground occurs, the light associated with the grounded terminal will be bright and the other light will be dim or out. DC ground annunciator procedure provides guidance to attempt to locate and isolate ground per Procedure 2.0.1. Stem provides ground is continuous and located on a sensitive panel. IAW procedure 2.0.1, grounds located on sensitive panels need additional planning which requires utilizing the work order process IAW 0-CNS-WM-102 to totally understand, plan, and control the impacts of isolation process. Non-sensitive panel ground isolation is performed by temporarily de-energizing equipment IAW the applicable Operating Procedure Checklist (in this case 2.2A_125DC.Div1 {125 VDC Power Checklist (Div 1)}.
This is reflected by correct Answer C.

Distracters:
Answer A is incorrect due to ground indication having one light brighter than the other. This choice is plausible due to ground indications being easily confused. The candidate that confuses the ground indication and correctly identifies additional planning is required to perform ground isolation on sensitive panels would select this answer.
Answer B is incorrect due to ground indication having one light brighter than the other and ground isolation being required to be performed IAW procedure 0-CNS-WM-102. This choice is plausible due to ground indications being easily confused and utilizing the applicable operating procedure being required on other than sensitive panels. The candidate that confuses the ground indication and does not recognize additional planning is required to perform

ground isolation on sensitive panels would select this answer. Answer D is incorrect due to ground isolation being required to be performed IAW procedure 0-CNS-WM-102. This choice is plausible due to utilizing the applicable operating procedure being required on other than sensitive panels. The candidate that correctly identifies the ground indication and does not recognize additional planning is required to perform ground isolation on sensitive panels would select this answer.		
Technical References: Procedure 2.2A_125DC.Div1 {125 VDC Power Checklist (Div 1)}; Rev.3 Procedure 0-CNS-WM-102 (Work Implementation and Closeout), Rev. 4 Procedure 2.0.1 (Plant Operations Policy), Rev. 63 Procedure 2.3_C-1 (Panel C - Annunciator C-1), Rev. 30		
References to be provided to applicants during exam: none		
Learning Objective:		
COR0020702006E Describe the interrelationship between the DC Electrical Distribution System and the following: e. Ground detection INT032010300A010L Discuss the following as described in Conduct of Operations Procedure 2.0.1, Plant Operations Policy: Ground isolation guidelines		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:		
	55.43(b)(5)	
Level of Difficulty:	3	
SRO Only Justification:		
This question requires knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures by the SRO.		

125V DC BUS 1A GROUND	PANEL/WINDOW: C-1/B-2

1. OPERATOR OBSERVATION AND ACTION

1.1 Attempt to locate and isolate ground per Procedure 2.0.1.

1.2 IF 125 VDC 1A de-energized, THEN enter Procedure 5.3DC125.

1. PURPOSE ® [1,3]

1.1 This procedure establishes administrative controls for maintenance activities at Cooper Nuclear Station (CNS). It provides instructions to ensure maintenance at CNS is planned and implemented in a manner consistent with its importance to plant safety and its potential to impact unit availability.

PROCEDURE 0-CNS-WM-102	REVISION 4	PAGE 1 OF 32

CNS OPERATIONS MANUAL CONDUCT OF OPERATIONS PROCEDURE 2.0.1 PLANT OPERATIONS POLICY	USE: INFORMATION QUALITY: QAPD RELATED EFFECTIVE: 3/16/15 APPROVAL: ITR-RDM OWNER: OPS MANAGER DEPARTMENT: OPS

REV.	DATE	CHANGES
62	6/17/14	Replaced Procedure 0.40 with Procedure 0-CNS-WM-102.
63	3/16/15	Changes to Sections 15 and 16.

1. PURPOSE

1.1 The purpose of this procedure is to provide guidance and instruction regarding overall operation of Cooper Nuclear Station.

2. PRECAUTIONS AND LIMITATIONS

2.1 Radiation detection instrumentation scales that read in roentgen (R) units are equivalent to rem units. One R is equivalent to 1 rem. The R unit is no longer defined in revised 10CFR20 and present R readings should be expressed as rem.

11.2 If ground is continuous, perform following: Ⓟ[4]

11.2.1 Notify Electrical Maintenance and Engineering of ground as soon as possible.

11.2.2 Initiate a Condition Report documenting condition and evaluate OPERABILITY per Procedure 0.5.OPS, Operations Review of Notifications/Operability Determination.

11.2.3 Determine from available indications (alarms, ground lights, ground current meters, or relays) which switchgear, MCC, or distribution panel that ground is on.

11.2.4 Determine if changes in plant equipment lineup or work activity has occurred which may be cause (cycling equipment or work in-progress).

11.2.5 Obtain a breaker list from respective system operating procedure and note status of all breakers on load center.

11.2.6 Coordinate ground isolation activities through Control Room.

11.2.7 Temporarily de-energize equipment powered off suspected bus that is not needed for continued safe system or plant operation.

11.2.8 If equipment configuration allows and equipment transfer does not cause unacceptable risk to plant operation, place standby equipment in service and remove previously running equipment.

11.2.9 Actions taken to isolate ground should be documented with major equipment changes and cause of ground recorded in Control Room Log.

11.2.10 Work Order process per Procedure 0-CNS-WM-102 should be used to locate grounds on sensitive panels or equipment.

12. STATION TOURS IN THE RCA

12.1 Station tour requirements for RCA are established to allow easy access to areas where radiation protection measures are taken to minimize accumulation of radiation or for contamination control.

13. CONTROL OF PRE-PLANNED EVOLUTION

13.1 Before allowing any pre-planned complex evolution to start, the SM should ensure all members of operating crew have been briefed on actions each member is to take and expected results of those actions.

13.2 Outage or startup meetings should not hinder ability of SM to control operation of plant. Meeting should be scheduled such that SM can brief crew before evolution is to start.

Examination Outline Cross-Reference	Level	SRO
Requires interpreting 14 days form LCO entry vs. Time CONDITION D is entered which is a common misconception – no change.	Tier#	2
	Group#	2
	K/A #	216000, A2.12
	Rating	2.9

216000 Nuclear Boiler Inst.

A2. Ability to (a) predict the impacts of the following on the NUCLEAR BOILER INSTRUMENTATION ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5 / 45.6)
A2.12 Instrument isolation valve closures

Question 91

Reference Provided

The plant is at rated power.

A 1 drop/minute leak on a reference leg fitting downstream of the instrument isolation valve for NBI-LT-92 (Steam Nozzle Level Transmitter) was discovered.

The reference leg instrument isolation valve for NBI-LT-92 was closed exactly 7 days ago to isolate leakage.

(1) What is the impact of these conditions over time on Steam Nozzle Level Indicator?

AND

(2) How long from **NOW** is the associated report required to be submitted to the NRC IAW TS 5.6.6 {Post Accident Monitoring (PAM) Instrumentation Report}?

A. (1) Indicated level will lower.
   (2) 14 days.

B. (1) Indicated level will lower.
   (2) 7 days.

C. (1) Indicated level will rise.
   (2) 14 days.

D. (1) Indicated level will rise.
(2) 7 days.

Answer: C. (1) Indicated level will rise. (2) 14 days.
Explanation: This question requires knowledge of level transmitter operation and administrative TS 5.6.6. The reference leg for a level transmitter provides the high pressure input to the dp cell. If the reference leg is isolated, and a leak exists downstream of the isolation valve as stated in the stem, then sensed dp will lower. Indicated level is inversely proportional to sensed dp; therefore, indicated level will rise. The single Steam Nozzle level indication channel is classified as Post-Accident Monitoring (PAM) Instrumentation required operable per TS 3.3.3.1. With this instrument inoperable, TS .3.3.3.1 Condition C applies. If Condition C is not exited within 7 days, Condition D is entered, requiring immediate entry into Condition F. Condition F requires immediate compliance with TS 5.6.6. TS 5.6.6 outlines reporting requirements for inoperable PAM instrumentation. Reports to the NRC per TS 5.6.6 are due within 14 days. Therefore, answer C is correct.
Distracters: Answer A is plausible because if the same conditions stated in the stem were applied to the variable leg of the level transmitter, indicated level would lower. The unprepared student could easily confuse the two different effects. It is wrong because indicated level will rise. Answer B is plausible and wrong as described for answer A, above. Part 1 of the answer is plausible since the unprepared student may believe the 14 day requirement to submit the report to the NRC must be backdated to start from the time of the initial inoperability. This is incorrect since the required action to submit the report begins only after Condition B is entered: therefore, the associated completion time of 14 days begins when Condition B is entered, 14 days from **now** as stated in the stem. Answer D is plausible because of the same reasons stated for part 2 of distracter B. It is wrong because the special report per TS 5.6.6 is due to the NRC within 14 days.
Technical References: TS 3.3.3.1, PAM Instrumentation with Table 3.3.3.2-1 REQUIRED Channels blank.; TS 5.6.6, PAM Instrumentation Report
References to be provided to applicants during exam: TS 3.3.3.1, 5.6.6, PAM Instrumentation Report
Learning Objective:

INT00705040010300 Given a set of plant conditions that constitutes non-compliance with a Section 3.3 LCO, determine the ACTIONS that are required.		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.43(b)(1)	
Level of Difficulty:	3	
SRO Only Justification:		
This requires knowledge of Administrative Controls TS 5.6.6.		

3.3 INSTRUMENTATION

3.3.3.1 Post Accident Monitoring (PAM) Instrumentation

LCO 3.3.3.1 The PAM Instrumentation for each Function in Table 3.3.3.1-1 shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each Function. For Function 5, separate Condition entry is allowed for each penetration flow path.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one required channel inoperable.	A.1 Restore required channel to OPERABLE status.	30 days
B. Required Action and associated Completion Time of Condition A not met.	B.1 Initiate action in accordance with Specification 5.6.6.	Immediately
C. One or more Functions with two required channels inoperable. OR One Function 2.c channel inoperable.	C.1 Restore one required channel to OPERABLE status.	7 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. Required Action and associated Completion Time of Condition C not met.	D.1	Enter the Condition referenced in Table 3.3.3.1-1 for the channel.	Immediately
E. As required by Required Action D.1 and referenced in Table 3.3.3.1-1.	E.1	Be in MODE 3.	12 hours
F. As required by Required Action D.1 and referenced in Table 3.3.3.1-1.	F.1	Initiate action in accordance with Specification 5.6.6.	Immediately

Table 3.3.3.1-1 (page 1 of 1)
Post Accident Monitoring Instrumentation

FUNCTION	REQUIRED CHANNELS	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1
1. Reactor Pressure	2	E
2. Reactor Vessel Water Level		
a. Fuel Zone	2	E
b. Wide Range	2	E
c. Steam Nozzle	1	F
3. Suppression Pool Level (Wide Range)	2	E
4. Primary Containment Gross Radiation Monitors	2	F
5. PCIV Position	2 per penetration flow path(a)(b)	E
6. Primary Containment H_2 & O_2 Analyzer	2	E
7. Primary Containment Pressure		
a. Drywell Narrow Range	2	E
b. Drywell Wide Range	2	E
c. Suppression Chamber Wide Range	2	E
8. Suppression Pool Water Temperature	2(c)	E

(a) Not required for isolation valves whose associated penetration flow path is isolated by at least one closed and deactivated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured.

(b) Only one position indication channel is required for penetration flow paths with only one installed control room indication channel.

(c) A channel requires a minimum of four resistance temperature detectors (RTDs) to be OPERABLE with no two adjacent RTDs inoperable.

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

2. NEDE-23785-1-P-A, "The GESTR-LOCA and SAFER Models for the Evaluation of the Loss-of-Coolant Accident", Volume III, Revision 1, October 1984.

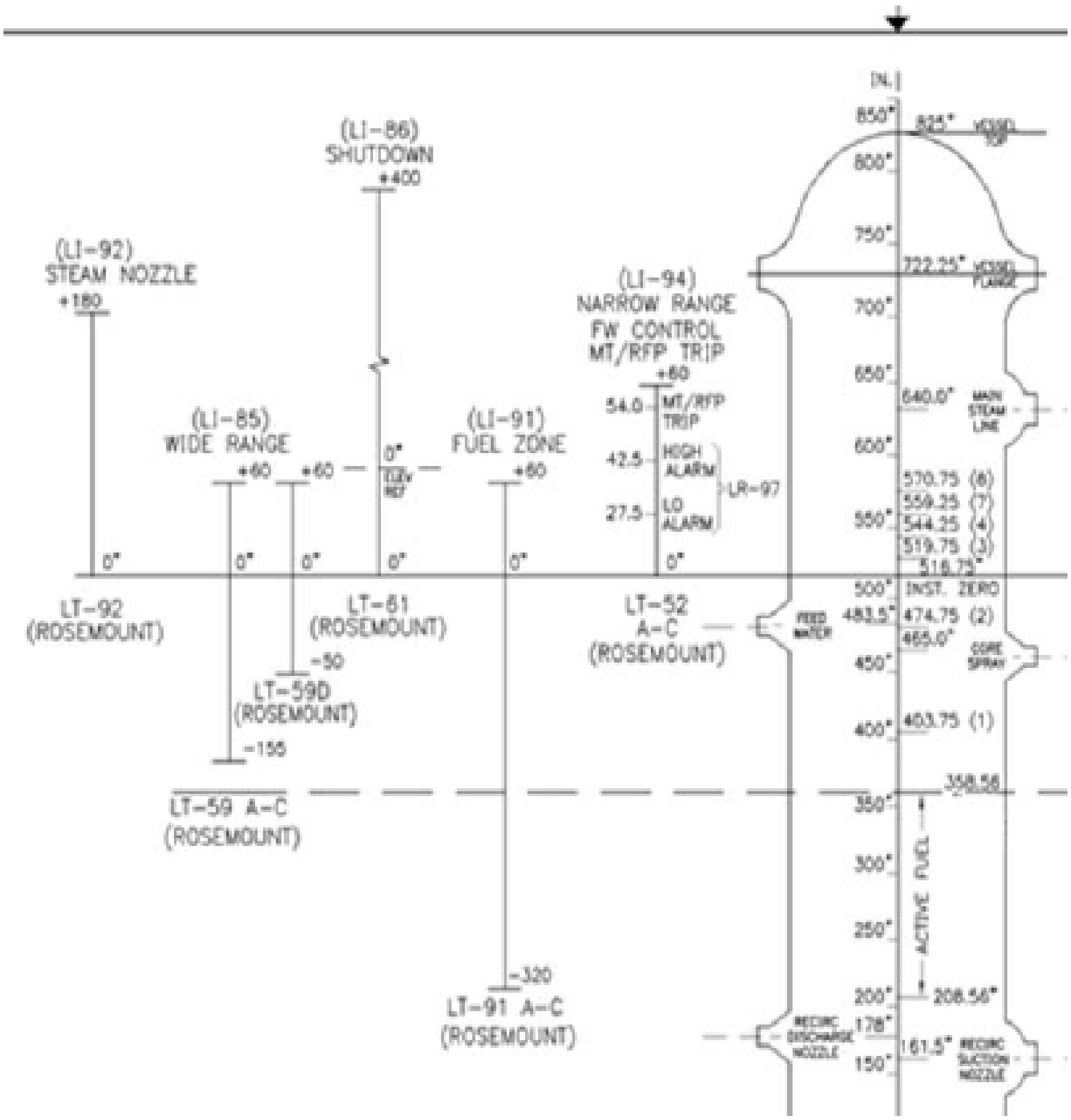
3. NEDO-31960 and NEDO-31960 Supplement 1, "BWR Owner's Group Long-Term Stability Solutions Licensing Methodology" (the approved Revision at the time the reload analysis is performed).

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.6 Post Accident Monitoring (PAM) Instrumentation Report

When a report is required by Condition B or F of LCO 3.3.3.1, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.


(LI-92)
STEAM NOZZLE
+180
(LI-86)
SHUTDOWN
+400
(LI-85)
WIDE RANGE
+60
+60
0"
(LI-91)
FUEL ZONE
+60
(LI-94)
NARROW RANGE
FW CONTROL
MT/RFP TRIP
+60
54.0
MT/RFP TRIP
42.5
HIGH ALARM
27.5
LO ALARM
LR-97
LT-92
(ROSEMOUNT)
LT-61
(ROSEMOUNT)
LT-52
A-C
(ROSEMOUNT)
-50
LT-59D
(ROSEMOUNT)
-155
LT-59 A-C
(ROSEMOUNT)
-320
LT-91 A-C
(ROSEMOUNT)
850"
825"
800"
750"
722.25"
700"
650"
640.0"
600"
570.75 (6)
559.25 (7)
544.25 (4)
550"
519.75 (3)
516.75"
500"
INST. ZERO
483.5"
474.75 (2)
465.0"
FEED WATER
CORE SPRAY
450"
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403.75 (1)
358.56
350"
300"
ACTIVE FUEL
250"
200"
208.56"
178"
161.5"
150"

Examination Outline Cross-Reference	Level	SRO
	Tier#	2
	Group#	2
	K/A #	268000, G2.4.47
	Rating	4.2

268000 Radwaste

G2.4.47 Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material. (CFR: 41.10 / 43.5 / 45.12)

Question 92

Reference Provided

Plant startup is in progress with Mode 1 scheduled to be entered in 7 hours.

Consider the attached Drywell Leakage data from 6.LOG.601 (Daily Surveillance Log - Modes 1, 2, and 3).

What Tech Spec 3.4.4 (RCS Operational LEAKAGE) ACTION is required now **and** is entry into Mode 1 allowed?

(Assume NO acceptable risk evaluation is performed.)

A. Perform ACTION A.1 within 4 hours.
Mode 1 may be entered while this condition exists.

B. Perform ACTION A.1 within 4 hours.
Entry into Mode 1 is prohibited while this condition exists.

C. Perform ACTION B.1 OR B.2 within 4 hours.
Mode 1 may be entered while this condition exists.

D. Perform ACTION B.1 OR B.2 within 4 hours.
Entry into Mode 1 is prohibited while this condition exists.

(Refer to the Attached Leak Rate Display)

ATTACHMENT 3 UNIDENTIFIED LEAK RATE CHECKS

Time Last Pumped (Previous Day): 1600

RW-FQ-527 [a]	0000 READING	0800 READING	1600 READING
Pump Used (F-1/F-2)	F-1	F-2	F-1
Present Grand Total	900178	900630	903188
(-) Previous Grand Total	900015	900178	900630
(=) Total Gallons	163	452	2558
(÷) Time Interval (minutes)	480	480	480
(=) Leak Rate (gpm) [b]	0.34	0.94	5.33
(-) Previous Day Leak Rate (gpm)	0.36	0.35	0.42
(=) Change In Leak Rate (gpm)	-0.02	0.59	4.91

Answer:
B. (1) Perform ACTION A.1 within 4 hours.
(2) Entry into Mode 1 is prohibited while this condition exists.

Explanation:
For the data provided, Unidentified leakage rate for the period at 1600 is greater than the TS 3.4.4 (b) limit of 5 gpm. This requires entry into TS 3.4.4 Condition A, Action A.1.

RW-FQ-527 [a]	0000 READING	0800 READING	1600 READING
Pump Used (F-1/F-2)	F-1	F-2	F-1
Present Grand Total	900178	900630	903188
(-) Previous Grand Total	900015	900178	900630
(=) Total Gallons	163	452	2558
(÷) Time Interval (minutes)	480	480	480
(=) Leak Rate (gpm) [b]	0.34	0.94	5.33
(-) Previous Day Leak Rate (gpm)	0.36	0.35	0.42
(=) Change In Leak Rate (gpm)	-0.02	0.59	4.91

Since mode change is not scheduled for another 7 hours, TS 3.0.4 does not allow entering Mode 1 For TS 3.4.4, since in the next 4 hours, TS 3.4.4 Actions C1 requires the plant to be in Mode 3 within the following 12 hours if the condition is not corrected. Therefore, Answer B is correct.

Distracters:
Answer A is plausible because the unprepared candidate might not know TS 3.0.4 prevents entering Mode 1. The candidate may believe since the plant is already in a Mode where the LCO is required, there are no additional restrictions for entering other Modes listed in the applicability. It is wrong because TS 3.0.4 prohibits entering Mode 1 in this case.

Answers C and D are plausible because there are limitations for the rate of increase of unidentified leakage, ≤2 gpm increase in a 24 hour period, and the data in the table provided results in increases above that limit. However, Answers C and D are both

wrong because TS 3.4.4 Condition B is only applicable in Mode 1: therefore, it is not required in Mode 2. In addition to being wrong for that reason, Answer C is also wrong because TS 3.0.4 prevents entering Mode 1 For TS 3.4.4, since TS 3.4.4 Actions ultimately require plant shutdown if the condition is not corrected.		
Technical References: TS 3.4.4; TS 3.0.4; 6.LOG.601, Daily Surveillance Log - Modes 1, 2, and 3		
References to be provided to applicants during exam: TS 3.4.4		
Learning Objective:		
COR00111020010300 Given conditions and/or parameters associated with Leak Detection, determine if related Technical Specification and Technical Requirements Manual Limiting Conditions for Operation are met.		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:		
	55.43(b)(2)	
Level of Difficulty:	3	
SRO Only Justification:		
Requires selection of the appropriate TS 3.4.4 action statement and application of TS 3.0.4.		

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.4 RCS Operational LEAKAGE

LCO 3.4.4 RCS operational LEAKAGE shall be limited to:

a. No pressure boundary LEAKAGE;

b. ≤ 5 gpm unidentified LEAKAGE;

c. ≤ 30 gpm total LEAKAGE averaged over the previous 24 hour period; and

d. ≤ 2 gpm increase in unidentified LEAKAGE within the previous 24 hour period in MODE 1.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Unidentified LEAKAGE not within limit. OR Total LEAKAGE not within limit.	A.1	Reduce LEAKAGE to within limits.	4 hours
B. Unidentified LEAKAGE increase not within limit.	B.1 OR	Reduce unidentified LEAKAGE increase to within limits.	4 hours (continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.2 Verify source of unidentified LEAKAGE increase is not service sensitive type 304 or type 316 austenitic stainless steel.	4 hours
C. Required Action and associated Completion Time of Condition A or B not met. OR Pressure boundary LEAKAGE exists.	C.1 Be in MODE 3. AND C.2 Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.4.1 Verify RCS unidentified and total LEAKAGE and unidentified LEAKAGE increase are within limits.	12 hours

3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

LCO 3.0.1 LCOs shall be met during the MODES or other specified conditions in the Applicability, except as provided in LCO 3.0.2 and LCO 3.0.7.

LCO 3.0.2 Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met, except as provided in LCO 3.0.5 and LCO 3.0.6.

If the LCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required, unless otherwise stated.

LCO 3.0.3 When an LCO is not met and the associated ACTIONS are not met, an associated ACTION is not provided, or if directed by the associated ACTIONS, the unit shall be placed in a MODE or other specified condition in which the LCO is not applicable. Action shall be initiated within 1 hour to place the unit, as applicable, in:

a. MODE 2 within 7 hours;

b. MODE 3 within 13 hours; and

c. MODE 4 within 37 hours.

Exceptions to this Specification are stated in the individual Specifications.

Where corrective measures are completed that permit operation in accordance with the LCO or ACTIONS, completion of the actions required by LCO 3.0.3 is not required.

LCO 3.0.3 is only applicable in MODES 1, 2, and 3.

LCO 3.0.4 When an LCO is not met, entry into a MODE or other specified condition in the Applicability shall only be made:

(continued)

3.0 LCO APPLICABILITY

LCO 3.0.4 (continued)

a. When the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time; or

b. After performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering the MODE or other specified condition in the Applicability, and establishment of risk management actions, if appropriate; exceptions to this Specification are stated in the individual Specifications; or

c. When an allowance is stated in the individual value, parameter, or other Specification.

This Specification shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

LCO 3.0.5 Equipment removed from service or declared inoperable to comply with ACTIONS may be returned to service under administrative control solely to perform testing required to demonstrate its OPERABILITY or the OPERABILITY of other equipment. This is an exception to LCO 3.0.2 for the system returned to service under administrative control to perform the testing required to demonstrate OPERABILITY.

LCO 3.0.6 When a supported system LCO is not met solely due to a support system LCO not being met, the Conditions and Required Actions associated with this supported system are not required to be entered. Only the support system LCO ACTIONS are required to be entered. This is an exception to LCO 3.0.2 for the supported system. In this event, additional evaluations and limitations may be required in accordance with Specification 5.5.11, "Safety Function Determination Program (SFDP)." If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

(continued)

Examination Outline Cross-Reference	Level	SRO
Revised question to allow SRO to make a decision on RCIC system operation based upon plant conditions.	Tier#	2
	Group#	2
	K/A #	272000, A2.10
	Rating	4.1

272000 Radiation Monitoring

A2. Ability to (d) predict the impacts of the following on the RADIATION MONITORING SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5 / 45.6)
A2.10 Loss of coolant accident

Question 93

The reactor is shutdown due a valid MSIV isolation with the following conditions present:

- HPCI is disassembled for maintenance.
- RPV pressure is 800 psig and slowly lowering (3 psig/minute).
- RPV water level is -160 inches and slowly lowering (1 inch/minute).
- RCIC, CRD, and SLC are injecting with flows maximized.
- Core Spray Pump A is tripped on overcurrent.
- RHR Loop B has been isolated due to a leak on the HX inlet.

A steam leak now occurs one foot upstream of the RCIC trip/throttle valve causing this area to exceed Maximum Normal Operating Temperature.

Which one of the following completes the statements below regarding the Area Radiation Monitor (ARM) indication which FIRST rises due to this steam leak and the required action for the RCIC system under the CURRENT plant conditions?

The ARM located in the ____(1)___ will rise FIRST.
RCIC is required to ____(2)____ while transitioning to EOP 2A (Emergency RPV Depressurization) under the current plant conditions.

A. (1) NE Quad
   (2) remain in service IAW EOP 1A (RPV Control)

B. (1) NE Quad
   (2) be isolated IMMEDIATELY IAW EOP 5A (Secondary Containment Control)

C. (1) SW Quad

(2) remain in service IAW EOP 1A (RPV Control)

D. (1) SW Quad
(2) be isolated IMMEDIATELY IAW EOP 5A (Secondary Containment Control)

Answer:
A (1) NE Quad
(2) remain in service IAW EOP 1A (RPV Control)

Explanation:
Requires knowledge of a LOCA (steam leak in secondary containment) on radiation monitors AND both evaluating plant conditions and selecting a procedure (specific steps) in which to proceed.
Per EOP-5A table 10, radiation monitor RMA-RA-13 monitors RCIC / Core Spray pump room in the NE quad. A steam leak near the RCIC turbine would first be detected by this radiation monitor. EOP-5A step SC-10 provides isolating all systems discharging into its area except systems required for damage control and systems required to support EOPs when radiation levels rise above the Maximum Normal Operating value. Requires determining actions based upon conflicting direction provided within EOP 1A & 5A. With all high pressure systems injecting at maximum flow and RPV water level still lowering will require emergency depressurization to allow low pressure systems to restore & maintain level greater than -183 inches and then isolating the RCIC system once it is determined it is not required to maintain adequate core cooling. Therefore, answer A is correct.

Distracters:
Answers that in part (1) list the SW quad are plausible because another steam driven system, HPCI is located in that area. They are wrong because for a steam leak at the RCIC turbine, radiation levels in the NE quad would be affected first. This applies to answers C and D.

Answers that in part (2) list be isolated IMMEDIATELY IAW EOP 5A are wrong because the RCIC system is currently being utilized to maintain Adequate Core Coling (ACC). Until low pressure ECCS (CS B and RHR Loop A) and Condensate are being utilized to restore and maintain RPV water level above -183 inches, RCIC is required to maintain ACC.. This applies to answers B and D.

Technical References:
EOP-5A (Secondary Containment Control), Rev. 15
EOP 1A (RPV Control), Rev. 18

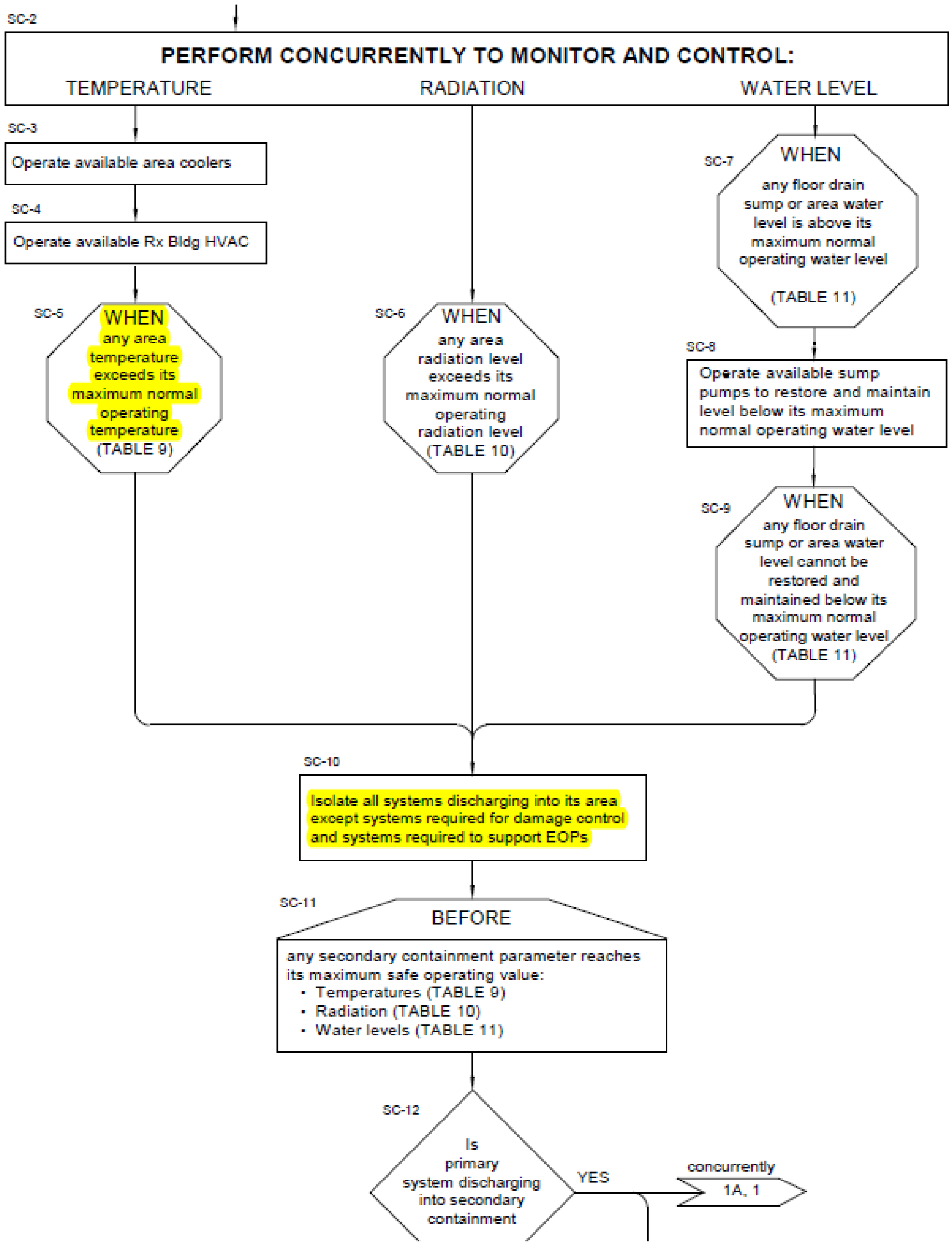
References to be provided to applicants during exam: none

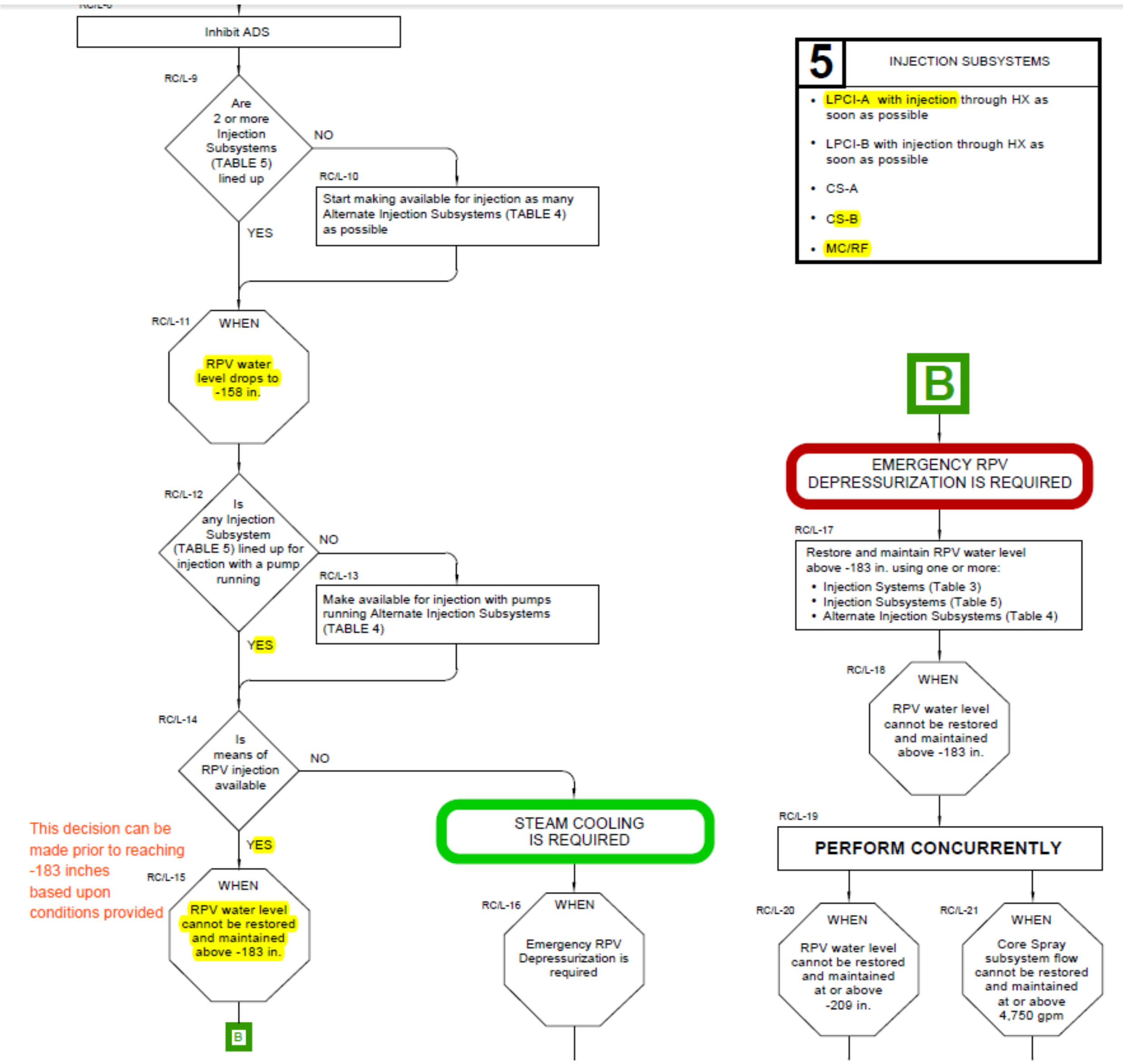
Learning Objective:

INT00806170010600, Given plant conditions and EOP flowchart 5A, SECONDARY CONTAINMENT CONTROL and RADIOACTIVITY RELEASE CONTROL, determine required actions.		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:		
	55.43(b)(2)	
Level of Difficulty:	3	
SRO Only Justification:		
This involves procedure selection for mitigating a specific event.		

10 SECONDARY CONTAINMENT RADIATION LEVELS SPDS 15 (5)

Maximum Normal Operating Value Area	Any ARM Alarmed	Range (mR/hr)	Maximum Safe Operating Value Area	Value (mR/hr)	Actual Value
FUEL POOL AREA	RMA-RA-1	100 - 10^6	1001' El.	1000	
FUEL POOL AREA	RMA-RA-2	.01 - 100	1001' El.		
RWCU PRECOAT AREA	RMA-RA-4	0.1 - 1000	958' El.		
RWCU SLUDGE AND DECANT PUMP AREA	RMA-RA-5	0.1 - 1000	931' El.	1000	
CRD HYDRAULIC EQUIP AREA (SOUTH)	RMA-RA-8	.01 - 100	903' El.		
CRD HYDRAULIC EQUIP AREA (NORTH)	RMA-RA-9	.01 - 100			
HPCI PUMP ROOM	RMA-RA-10	.01 - 100	HPCI Room		
RHR PUMP ROOM, (SOUTHWEST)	RMA-RA-11	.01 - 100	SW Quad	1000	
TORUS HPV AREA (SOUTHWEST)	RMA-RA-27	1.0 - 10000	SW Torus		
RHR PUMP ROOM, (NORTHWEST)	RMA-RA-12	.01 - 100	NW Quad	1000	
RCIC/CORE SPRAY PUMP ROOM, (NORTHEAST)	RMA-RA-13	.01 - 100	NE Quad	1000	
CORE SPRAY PUMP ROOM, (SOUTHEAST)	RMA-RA-14	.01 - 100	SE Quad	1000	


SC-2
PERFORM CONCURRENTLY TO MONITOR AND CONTROL:
TEMPERATURE
RADIATION
WATER LEVEL
SC-3
Operate available area coolers
SC-4
Operate available Rx Bldg HVAC
SC-5
WHEN
any area temperature exceeds its maximum normal operating temperature (TABLE 9)
SC-6
WHEN
any area radiation level exceeds its maximum normal operating radiation level (TABLE 10)
SC-7
WHEN
any floor drain sump or area water level is above its maximum normal operating water level
(TABLE 11)
SC-8
Operate available sump pumps to restore and maintain level below its maximum normal operating water level
SC-9
WHEN
any floor drain sump or area water level cannot be restored and maintained below its maximum normal operating water level (TABLE 11)
SC-10
Isolate all systems discharging into its area except systems required for damage control and systems required to support EOPs
SC-11
BEFORE
any secondary containment parameter reaches its maximum safe operating value:
• Temperatures (TABLE 9)
• Radiation (TABLE 10)
• Water levels (TABLE 11)
SC-12
Is primary system discharging into secondary containment
YES
concurrently
1A, 1


Inhibit ADS
RC/L-9
Are 2 or more Injection Subsystems (TABLE 5) lined up
NO
YES
RC/L-10
Start making available for injection as many Alternate Injection Subsystems (TABLE 4) as possible
5
INJECTION SUBSYSTEMS
LPCI-A with injection through HX as soon as possible
LPCI-B with injection through HX as soon as possible
CS-A
CS-B
MC/RF
RC/L-11
WHEN
RPV water level drops to -158 in.
B
EMERGENCY RPV DEPRESSURIZATION IS REQUIRED
RC/L-12
Is any Injection Subsystem (TABLE 5) lined up for injection with a pump running
NO
YES
RC/L-13
Make available for injection with pumps running Alternate Injection Subsystems (TABLE 4)
RC/L-17
Restore and maintain RPV water level above -183 in. using one or more:
Injection Systems (Table 3)
Injection Subsystems (Table 5)
Alternate Injection Subsystems (Table 4)
RC/L-18
WHEN
RPV water level cannot be restored and maintained above -183 in.
RC/L-14
Is means of RPV injection available
NO
YES
STEAM COOLING IS REQUIRED
This decision can be made prior to reaching -183 inches based upon conditions provided
RC/L-15
WHEN
RPV water level cannot be restored and maintained above -183 in.
B
RC/L-16
WHEN
Emergency RPV Depressurization is required
RC/L-19
PERFORM CONCURRENTLY
RC/L-20
WHEN
RPV water level cannot be restored and maintained at or above -209 in.
RC/L-21
WHEN
Core Spray subsystem flow cannot be restored and maintained at or above 4,750 gpm

Examination Outline Cross-Reference	Level	SRO
	Tier#	3
	Group#	N/A
	K/A #	G2.1.23
	Rating	4.4

1. Conduct of Operations

G2.1.23 Ability to perform specific system and integrated plant procedures during all modes of plant operation. (CFR: 41.10 / 43.5 / 45.2 / 45.6)

Question 94

Which one of the following completes the statement below regarding actions when the transition from Emergency Operating Procedures (EOPs) to Severe Accident Guidelines (SAGs) is required IAW 5.9SAMG (Severe Accident Management Guidance)?

Anytime SAG-1 (RPV, Containment and Radioactivity Release Control) is entered, the Shift Manager ensures EOP actions continue until ____(1)____ responsibilities have been transferred to the ____(2)____.

A. (1) decision-making
   (2) EOF

B. (1) decision-making
   (2) TSC

C. (1) communication
   (2) EOF

D. (1) communication
   (2) TSC

Answer:
B. (1) decision-making responsibilities
   (2) TSC

Explanation:
Requires knowledge of Shift Manager responsibilities during the transition of EOPs to SAGs. Until the Operations Coordinator is ready to assume decision-making

responsibility in the TSC, the Shift Manager in Control Room shall continue to direct plant response using EOP flowcharts.
Distracters: A. The answer is incorrect due to decision-making responsibilities being transferred to the TSC. This choice is plausible due to decision-making responsibilities for on-site and off-site being easily confused. The candidate that correctly identifies decision making vs. communication responsibilities and confuses the facility would select this answer. C. The answer is incorrect due to decision-making responsibilities being transferred to the TSC. This choice is plausible due to communication responsibilities do get transferred to the EOF. The candidate that confuses decision making vs. communication responsibilities and carries that error to the correct facility that is responsible for communications would select this answer. D. The answer is incorrect due to decision-making responsibilities being transferred to the TSC. This choice is plausible due to communication responsibilities do get transferred to the EOF and is easily confused with the TSC. The candidate that confuses decision making vs. communication responsibilities and the facility would select this answer.
Technical References: Procedure 5.9SAMG (Severe Accident Management Guidance), Rev. 12
References to be provided to applicants during exam: NONE
Learning Objective:
INT03501050041900 Identify the title of the individual who has decision making authority and responsibility for carrying out the SAG instructions once the ERO is fully manned and command authority established. INT03501050020000 Explain the transition into SAGs and how SAGs supplement the EOPs.

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.43(b)(5)	
Level of Difficulty:	2	
SRO Only Justification:		
Knowledge of SM responsibilities during transition from EOPs to SAGs.		

CNS OPERATIONS MANUAL SEVERE ACCIDENT PROCEDURE 5.9SAMG SEVERE ACCIDENT MANAGEMENT GUIDANCE	USE: REFERENCE QUALITY: QAPD RELATED EFFECTIVE: 10/14/13 APPROVAL: ITR-RDM OWNER: OSG SUPV DEPARTMENT: OPS

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1. PURPOSE©[1]

1.1 The CNS Severe Accident Management Guidance provides direction for the Emergency Response Organization in mitigating the consequences of a severe accident. A severe accident is defined to be an emergency event in which adequate core cooling cannot be restored and maintained. This guidance is a symptom-based extension of the accident mitigation strategies contained in the Emergency Operating Procedure (EOP) flowcharts. Therefore, the guidance is entered only when directed by the EOP flowcharts. The guidance is designed to be implemented by the Technical Support Center (TSC).

2. INSTRUCTIONS

2.1 Verify primary containment flooding is required, as specified, in one of following EOP flowcharts:

2.1.1 1A, RPV Control.

2.1.2 2B, RPV Flooding.

2.1.3 7A, RPV Level (Failure-To-Scram).

2.1.4 7B, RPV Flooding (Failure-To-Scram).

2.2 Transfer EOP/SAG decision-making responsibility from the Control Room Shift Manager to TSC Operations Coordinator using checklist in Attachment 4 or inside Operations Coordinator PIM as follows:

2.2.1 Ensure TSC activation is complete. Until Operations Coordinator is ready to assume decision-making responsibility, Shift Manager in Control Room shall continue to direct plant response using EOP flowcharts.

2.2.2 Ensure TSGs are in place and being reviewed by TSC Staff.

Examination Outline Cross-Reference	Level	SRO
Corrected explanations and modified stem to state "from initial applicability time".	Tier#	3
	Group#	N/A
	K/A #	G2.2.12
	Rating	4.1

2. Equipment Control

G2.2.12 Knowledge of surveillance procedures. (CFR: 41.10 / 45.13)

Question 95

Consider SR 3.0.2 requirements for scheduled performance of surveillances that are required to be performed "once per 8 hours".

Which of the following times for initial and subsequent surveillance completion reflect the **LONGEST ALLOWED** frequencies that will satisfy SR 3.0.2 requirements for "once per 8 hours" from an initial applicability time of 0000?

A. Initial performance at 0800;
   Next performance at 1600.

B. Initial performance at 1000;
   Next performance at 1600.

C. Initial performance at 0800;
   Next performance at 1800.

D. Initial performance at 1000;
   Next performance at 1800.

Answer:
C. Initial performance at 0800;
   Next performance at 1800

Explanation:
Per TS SR 3.0.2, surveillances with frequencies specified on a "once per" basis, the surveillance frequency extension of 1.25 times the specified frequency is allowed for subsequent performances following the initial performance, but not for the initial performance, which in the case given is 8 hours. The subsequent performance may

then be 8 X 1.25 = 10 hours later, at 1800. The bases SR 3.0.2 clearly states SR 3.0.2 is for operational flexibility.,		
Distracters: Distracter A is plausible if the candidate does not know the 25% frequency extension may be applied and since it reflects surveillance performance once every 8 hours. It is wrong because the question asks for the longest allowed frequency. If the frequency extension of 25% is utilized for the second performance, it would be allowed to be delayed an additional 2 hours from 1600. Distractor B is plausible if the candidate mistakes which performance, initial or subsequent, the 25% frequency extension may be applied to. Answer B is wrong because it reflects the 25% extension being applied to the first performance, which must be done by 0800, not 1000. Distractor D is plausible if the candidate mistakes how the 25% frequency extension may be applied. Answer D is wrong because it reflects the 25% extension being applied to both initial and subsequent performances.		
Technical References: SR 3.0.2 and bases Procedure 0.26 (Surveillance Program), Rev. 68		
References to be provided to applicants during exam: none		
Learning Objective:		
INT00705010010700 Explain the Frequency rules for periodic actions (both Required Actions and Surveillance Requirements) and apply these rules to determine when a periodic action must be performed.		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:		
	55.43(b)(2)	
Level of Difficulty:	3	
SRO Only Justification:		
This requires knowledge of application of TS 3.0.2, described in TS 3.0.2 bases.		

3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

SR 3.0.1	SRs shall be met during the MODES or other specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO. Failure to perform a Surveillance within the specified Frequency shall be failure to meet the LCO except as provided in SR 3.0.3. Surveillances do not have to be performed on inoperable equipment or variables outside specified limits.
SR 3.0.2	The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met. For Frequencies specified as "once," the above interval extension does not apply. If a Completion Time requires periodic performance on a "once per . . ." basis, the above Frequency extension applies to each performance after the initial performance. Exceptions to this Specification are stated in the individual Specifications.
SR 3.0.3	If it is discovered that a Surveillance was not performed within its specified Frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, whichever is greater. This delay period is permitted to allow performance of the Surveillance. A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed.

(continued)

SR 3.0.1 (continued)

Upon completion of maintenance, appropriate post maintenance testing is required to declare equipment OPERABLE. This includes ensuring applicable Surveillances are not failed and their most recent performance is in accordance with SR 3.0.2. Post maintenance testing may not be possible in the current MODE or other specified conditions in the Applicability due to the necessary unit parameters not having been established. In these situations, the equipment may be considered OPERABLE provided testing has been satisfactorily completed to the extent possible and the equipment is not otherwise believed to be incapable of performing its function. This will allow operation to proceed to a MODE or other specified condition where other necessary post maintenance tests can be completed. Some examples of this process are:

a. Control Rod Drive maintenance during refueling that requires scram testing at > 800 psi. However, if other appropriate testing is satisfactorily completed and the scram time testing of SR 3.1.4.3 is satisfied, the control rod can be considered OPERABLE. This allows startup to proceed to reach 800 psi to perform other necessary testing.

b. High pressure coolant injection (HPCI) maintenance during shutdown that requires system functional tests at a specified pressure. Provided other appropriate testing is satisfactorily completed, startup can proceed with HPCI considered OPERABLE. This allows operation to reach the specified pressure to complete the necessary post maintenance testing.

SR 3.0.2

SR 3.0.2 establishes the requirements for meeting the specified Frequency for Surveillances and any Required Action with a Completion Time that requires the periodic performance of the Required Action on a "once per..." interval.

SR 3.0.2 permits a 25% extension of the interval specified in the Frequency. This extension facilitates Surveillance scheduling and considers plant operating conditions that may not be suitable for conducting the Surveillance (e.g., transient conditions or other ongoing Surveillance or maintenance activities).

BASES

SR 3.0.2 (continued)

The 25% extension does not significantly degrade the reliability that results from performing the Surveillance at its specified Frequency. This is based on the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the SRs. The exceptions to SR 3.0.2 are those Surveillances for which the 25% extension of the interval specified in the Frequency does not apply. These exceptions are stated in the individual Specifications. The requirements of regulations take precedence over the TS. An example of where SR 3.0.2 does not apply is in the Containment Leakage Rate Testing Program. This program establishes testing requirements and Frequencies in accordance with the requirements of the regulations. The TS cannot in and of themselves extend a test interval specified in the regulations.

As stated in SR 3.0.2, the 25% extension also does not apply to the initial portion of a periodic Completion Time that requires performance on a "once per..." basis. The 25% extension applies to each performance after the initial performance. The initial performance of the Required Action, whether it is a particular Surveillance or some other remedial action, is considered a single action with a single Completion Time. One reason for not allowing the 25% extension to this Completion Time is that such an action usually verifies that no loss of function has occurred by checking the status of redundant or diverse components or accomplishes the function of the inoperable equipment in an alternative manner.

The provisions of SR 3.0.2 are not intended to be used repeatedly merely as an operational convenience to extend Surveillance intervals (other than those consistent with refueling intervals) or periodic Completion Time intervals beyond those specified.

SR 3.0.3

SR 3.0.3 establishes the flexibility to defer declaring affected equipment inoperable or an affected variable outside the specified limits when a Surveillance has not been completed within the specified Frequency. A delay period of up to 24 hours or up to the limit of the specified Frequency, whichever is greater, applies from the point in time that it is discovered that the Surveillance has not been

2.8 The MISSED TECHNICAL SPECIFICATION/TECHNICAL REQUIREMENTS MANUAL SURVEILLANCE Section applies to unintentionally missed surveillance tests AND shall NOT be used for operational convenience to extend surveillance frequencies. The missed surveillance requirements are expected to be completed at the First Reasonable Opportunity.Ⓟ[12]

2.8.1 The MISSED TECHNICAL SPECIFICATION/TECHNICAL REQUIREMENTS MANUAL SURVEILLANCE Section applies only to those components or features which have previously had a Satisfactory Completion of Surveillance Test Procedure. Discovery that the applicable Surveillance Procedure is inadequate or has never been successfully performed does not constitute a missed surveillance and component or feature not demonstrated to meet applicable Surveillance Requirements is inoperable.Ⓟ[13]

2.8.2 IF an IST Program test not tied to a TECHNICAL SPECIFICATION SURVEILLANCE is not performed within its required test frequency, THEN a Functional Assessment should be performed per Procedure 0.5.OPS. TS SR 3.0.3 should not be applied.Ⓟ[16]

2.9 If a surveillance was previously completed, reviewed, signed off, and later (e.g., could be days or weeks), the Control Room is notified of questionable validity of test data in the surveillance:

2.9.1 Contact Work Week Director to ensure appropriate organizational resources dedicated to resolving the concern.

2.9.2 If necessary, re-perform surveillance to determine or establish system functionality/operability.

2.10 10CFR72 Surveillance Requirements Applicability for Dry Fuel Storage (Standardized NUHOMS®):

2.10.1 In accordance with the Certificate of Compliance (CoC) No. 72-1004, Latest Amendment for the NUHOMS® Storage System, only Daily or 24 hour surveillances are controlled by this procedure. Periodicities described in Table 1.3.1 of the CoC Technical Specifications as prior to loading (PL), during loading (L), prior to movement (S), or as necessary (AN) are controlled by approved Dry Fuel Storage Loading Procedures. The surveillance limits and conditions for periodicities PL, L, S, and AN are met and tracked within those Dry Fuel Loading Procedures.

3. SURVEILLANCE TEST FREQUENCY

3.1 Tests shall be scheduled so actual performance of test will either be more conservative or equal to actual exact time requirement of TS, TRM, or ODAM. This ensures if a test is inadvertently delayed, TS, TRM, or ODAM frequency requirements will not be violated.

8. MISSED TECHNICAL SPECIFICATION/TECHNICAL REQUIREMENTS MANUAL SURVEILLANCE℗[12,13]

NOTE 1 – This section applies to unintentionally missed Technical Specification or technical requirements manual surveillance tests and shall not be used for operational convenience to extend surveillance frequencies. The missed surveillance requirements are expected to be completed at the First Reasonable Opportunity.

NOTE 2 – This section does not apply to missed surveillances when it is discovered that the equipment or feature has never been satisfactorily tested. Component or feature which has never been successfully tested does not constitute a missed surveillance. Component or feature is inoperable until successfully tested.

NOTE 3 – This section does not apply to Inservice Test (IST) Program requirements that are not tied to a TS SR. This type of condition should be addressed via a functionality assessment per Procedure 0.5.OPS.℗[16]

8.1 Continue in this section only if Surveillance Procedure not performed within its required frequency.

8.2 Review Technical Specification SR 3.0.3, Technical Requirements Manual TSR 3.0.3, and associated Bases as appropriate.

8.3 SM shall ensure a Condition Report is initiated to document the missed surveillance, unless one is already completed.

8.3.1 Ensure Condition Report identifies that a MISSED SURVEILLANCE occurred to improve record search capability. The preferred location is in the Short Text portion (first line) of the Condition Report's Condition Description field.

8.3.2 The Condition Report shall be identified as a CAP item and should include the following to the extent known:

- Identify the surveillance missed, including frequency.
- State the reason the surveillance was missed.
- Identify the components that have not had the surveillance requirements met.
- Identify plant conditions necessary to perform the missed surveillance.

8.4 SM shall ensure the missed surveillance is documented in the Operations Narrative Log.

NOTE – Work Week Director notification is to ensure that the appropriate planning can take place. Operations Management and Risk Management Supervisor notification is informational only.

8.5 SM shall ensure Work Week Director, Operations Management, and Risk Management Supervisor are notified of the missed surveillance.

Examination Outline Cross-Reference	**Level**	SRO
Revised question to test TS bases knowledge of what activities are NOT allowed to be performed with an instrument returned to service under TS 3.0.5. Revised to incorporate recommended changes (activity which is permissible to return an inoperable instrument to service for IAW LCO 3.0.5 Bases).	**Tier#**	3
	Group#	N/A
	K/A #	G2.2.25
	Rating	4.2

2. Equipment Control

G2.2.25 Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits. (CFR: 41.5 / 41.7 / 43.2)

Question 96

An instrument channel has been placed in the tripped condition to comply with Technical Specification ACTIONS due to an instrument being inoperable (failed downscale).

It is permissible to return the inoperable instrument to service to accomplish which one of the following activities IAW LCO 3.0.5 TS Bases?

A. Perform vendor-recommended preventative maintenance on another channel in the opposite trip system.

B. Perform corrective maintenance on another channel in the opposite trip system.

C. Perform a post-maintenance testing on the affected channel.

D. Perform a daily channel check on the affected channel.

Answer:
C. Perform a post-maintenance testing on the affected channel.

Explanation:
TS 3.0.5 provides an exception to the TS 3.0.2 requirement to comply with TS Action statements. TS 3.0.5 states equipment removed from service to comply with TS Actions may be returned to service soley to demonstrate its operability or to allow performance of a required surveillance on other TS equipment. The administrative controls ensure the time the equipment is returned to service in conflict with the requirements of the ACTIONS is limited to the time absolutely necessary to perform the allowed SRs. This Specification does not provide time to perform any other

preventive or corrective maintenance.

Distracters:

Answer A is incorrect due to performing preventive or correct maintenance not being allowed on another instrument within the same or opposite trip system. This choice is plausible due confusing whether preventive maintenance can be performed while applying TS 3.0.5. The candidate that confuses where preventive maintenance is allowed would select this answer.

Answer B is incorrect due to performing preventive or correct maintenance not being allowed on another instrument within the same or opposite trip system. This choice is plausible due confusing whether corrective maintenance can be performed while applying TS 3.0.5. The candidate that confuses where corrective maintenance is allowed would select this answer.

Answer D is incorrect due to surveillances are not required to be performed on inoperable equipment unless it is an effort to restore the equipment operable. And in this case, obtaining a reading would serve no purpose with respect to a channel check since the instrument has failed downscale. The candidate that confuses performing a channel check on inoperable equipment vs. to support instrument operability would select this answer.

Technical References: TS 3.0.5

References to be provided to applicants during exam: none

Learning Objective:

INT00705010010200 Given plant conditions and a Specification, apply the rules of Section 3.0 to determine appropriate actions.

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	

10CFR Part 55 Content:	
	55.43(b)(2)
Level of Difficulty:	3
SRO Only Justification:	

This question requires knowledge of TS 3.0.5 bases.

3.0 LCO APPLICABILITY

LCO 3.0.4 (continued)

a. When the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time; or

b. After performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering the MODE or other specified condition in the Applicability, and establishment of risk management actions, if appropriate; exceptions to this Specification are stated in the individual Specifications; or

c. When an allowance is stated in the individual value, parameter, or other Specification.

This Specification shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

LCO 3.0.5 Equipment removed from service or declared inoperable to comply with ACTIONS may be returned to service under administrative control solely to perform testing required to demonstrate its OPERABILITY or the OPERABILITY of other equipment. This is an exception to LCO 3.0.2 for the system returned to service under administrative control to perform the testing required to demonstrate OPERABILITY.

LCO 3.0.6 When a supported system LCO is not met solely due to a support system LCO not being met, the Conditions and Required Actions associated with this supported system are not required to be entered. Only the support system LCO ACTIONS are required to be entered. This is an exception to LCO 3.0.2 for the supported system. In this event, additional evaluations and limitations may be required in accordance with Specification 5.5.11, "Safety Function Determination Program (SFDP)." If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

(continued)

LCO 3.0.5

LCO 3.0.5 establishes the allowance for restoring equipment to service under administrative controls when it has been removed from service or declared inoperable to comply with ACTIONS. The sole purpose of this Specification is to provide an exception to LCO 3.0.2 (e.g., to not comply with the applicable Required Action(s)) to allow the performance of SRs to demonstrate:

a. The OPERABILITY of the equipment being returned to service; or

b. The OPERABILITY of other equipment.

The administrative controls ensure the time the equipment is returned to service in conflict with the requirements of the ACTIONS is limited to the time absolutely necessary to perform the allowed SRs. This Specification does not provide time to perform any other preventive or corrective maintenance.

An example of demonstrating the OPERABILITY of the equipment being returned to service is reopening a containment isolation valve that has been closed to comply with Required Actions and must be reopened to perform the SRs.

INFORMATION ONLY

INFORMATION ONLY

BASES

LCO 3.0.5 (continued)

An example of demonstrating the OPERABILITY of other equipment is taking an inoperable channel or trip system out of the tripped condition to prevent the trip function from occurring during the performance of an SR on another channel in the other trip system. A similar example of demonstrating the OPERABILITY of other equipment is taking an inoperable channel or trip system out of the tripped condition to permit the logic to function and indicate the appropriate response during the performance of an SR on another channel in the same trip system.

Examination Outline Cross-Reference	Level	SRO
	Tier#	3
	Group#	N/A
	K/A #	G2.3.4
	Rating	3.7

3. Radiation Control

G2.3.4 Knowledge of radiation exposure limits under normal or emergency conditions. (CFR: 41.12 / 43.4 / 45.10)

Question 97

Which one of the following completes the statement below regarding the emergency exposure allowed to rescue a mechanic that has sustained life-threatening injuries IAW Procedure 5.7.12 (Emergency Radiation Exposure Control)?

The Shift Manager acting as Emergency Director may authorize an emergency exposure …

A. up to a MAXIMUM 5 REM.

B. up to a MAXIMUM 10 REM.

C. up to a MAXIMUM 25 REM.

D. GREATER than 25 REM.

Answer:
D. Greater than 25 REM

Explanation: Procedure 5.7.12, Emergency Radiation Exposure Control, step 3.1.1.1 states the Emergency Director may authorize emergency exposures up to 25 REM or more for rescue/treatment of an individual with life-threatening injuries. This condition, along with corrective activities to avoid extensive exposures to large populations, are the only justifications for authorizing more than 25 REM exposure.

Distracters:
Distracter A is plausible since it is an emergency exposure value listed in Procedure 5.7.12 for emergency related activities. It is wrong because it is not the

maximum exposure allowed for direct, life-saving activities.

Distracter B is plausible since it is an emergency exposure value listed in Procedure 5.7.12 for emergency related activities. It is wrong because it is not the maximum exposure allowed for direct, life-saving activities.

Distracter C is plausible since it is an emergency exposure value listed in Procedure 5.7.12 for emergency related activities. It is wrong because it is not the maximum exposure allowed for direct, life-saving activities including rescue of the individual. 25 REM would be the maximum for non-direct support of life saving activities. Procedure 5.7.12 Attachment 1, Guide on Dose Limits for Workers Performing Emergency Services, only refers to life-saving activities in relation to the “>25R” limit.

Technical References:

Procedure 5.7.12 (Emergency Radiation Exposure Control), Rev. 13
Procedure 5.7.2 (Emergency Director EPIP), Rev. 33

References to be provided to applicants during exam: none

Learning Objective:

GEN0030401F0F1300 From memory, given conditions, determine if Emergency Exposures should be authorized, and if so, for whom.

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	

10CFR Part 55 Content:	
	55.43(b)(4)
Level of Difficulty:	2
SRO Only Justification:	

The SRO acting as Emergency Director may not delegate authorization of emergency exposure above 10CRF20 occupational dose limits.

- 2.6 Personnel authorized to receive emergency exposures should meet the following criteria:
  - 2.6.1 Personnel conducting corrective or protective actions or life-saving actions who may receive a TEDE in excess of occupational limits should be selected on a voluntary basis.
  - 2.6.2 Personnel shall be familiar with the hazards of any exposure received under emergency conditions.
  - 2.6.3 Declared pregnant woman shall not take part in these actions.
  - 2.6.4 Personnel should not have received previous emergency exposures. Emergency exposure should be limited to once in a lifetime.
  - 2.6.5 All occupational doses, including emergency doses, are required to be included as part of a worker's exposure history and hence can affect the worker's future allowable exposure.

3. REQUIREMENTS

NOTE 1—The examples listed below do not represent an absolute list. The existing situation may dictate additional conditions under which exceeding 10CFR20 limits may be warranted.

NOTE 2—The terms population and large population are not specifically defined in regulations; however, the following is listed as guidance in EPA 400: "Risks to populations in this context "substantial risks" means collective doses that are significantly larger than those incurred through the protective activities engaged in by the workers. In the context of this guidance, exposure of workers that is incurred for the protection of large populations may be considered justified for situations in which the collective dose avoided by the emergency operation is significantly larger than that incurred by the workers involved".

NOTE 3—Federal Register/Vol. 47, No. 205/Friday, October 22, 1982/Notices, lists the following guidance: "The ratio of total United States population to the maximum number of people in the vicinity of an operating reactor could be erroneously interpreted so that progressively smaller populations would be subject to progressively larger individual risks. This is not the intent of the recommendations". Therefore, the members of the public around CNS constitute populations or large populations.

- 3.1 The Emergency Director may authorize emergency exposures under the following conditions:
  - 3.1.1 LIFE-SAVING ACTIONS 25 REM OR MORE
    - 3.1.1.1 Rescue and/or treatment of personnel with life threatening injuries.
    - 3.1.1.2 Corrective activities to avoid extensive exposures to large populations.

≤ 25 rem

3.1.2→CORRECTIVE·OR·PROTECTIVE·ACTIONS·25·REM¶

3.1.2.1→Support·of·life·saving·activities·or·protection·of·large·populations·when·lower·dose·is·not·practical.¶

3.1.3→CORRECTIVE·OR·PROTECTIVE·ACTIONS·10·REM·¶

3.1.3.1→Protecting·valuable·property·when·lower·dose·is·not·practical.¶

3.1.4→ALL·OTHER·EMERGENCY·CONDITIONS·5·REM¶

3.1.4.1→Collection·of·in-plant·airborne·and·liquid·samples.®[1]·¶

3.1.4.2→Performing·personnel·decontamination.¶

3.1.4.3→Use·of·the·post-accident·sampling·system.®[1]¶

3.1.4.4→Radiological·monitoring·(teams).¶

4.→INSTRUCTIONS¶

4.1→Only·the·Emergency·Director·has·the·authority·to·authorize·exposures·in·excess·of·occupational·limits.¶

4.1.1→The·Emergency·Director·shall·evaluate·the·request·for·exposures·in·excess·of·the·occupational·limits·by·reviewing·the·scope·of·work,·travel·path,·dose·estimate,·and·justification·against·the·limits·specified.¶

4.2→PERSONNEL·EXPOSURE·CONTROL¶

4.2.1→Individuals·shall·not·enter·any·area·where·dose·rates·are·unknown·or·unmeasurable·with·instruments·immediately·available.¶

4.2.1.1→If·possible,·the·following·survey·instruments·should·be·used:¶

a.→High·range·portable·survey·instrument,·0·to·1000·rem/hr·(0·to·10·Sv/hr);·this·should·be·the·instrument·of·choice.¶

b.→Low·range·portable·survey·instrument,·0·to·50·rem/hr·(0·to·0.5·Sv/hr).¶

4.2.1.2→METER·USE¶

a.→Perform·a·battery·check.¶

b.→Allow·time·for·the·meter·to·warm·up,·if·required.¶

c.→Check·meter·response·with·a·check·source.¶

d.→Enter·suspected·radiation·areas·with·the·meter·set·on·appropriate·scale·and·switch,·as·necessary.¶

ATTACHMENT 1 EMERGENCY EXPOSURE LIMITS

GUIDE ON DOSE LIMITS FOR WORKERS PERFORMING EMERGENCY SERVICES

CONDITION	EMERGENCY LIMIT, up to (see NOTE)
Sampling, Surveys (on and off-site), Decontamination, PASS System use, etc.	5R
Restoration/repair of critical equipment during the emergency phase of a declared emergency.	10R
Protection of valuable equipment (ensure lubrication, prevent damage, fire-fighting, etc.)	10R
Isolation of a radiological release less than G.E. (as calculated by CNS-Dose).	10R
Isolation of a radiological release greater than G.E. (as calculated by CNS-Dose) <u>and</u> off-site protective actions (evacuation) have been completed.	10R
Providing first aid to less seriously injured personnel or in <u>support</u> of life-saving activities.	10R
Isolation of a radiological release greater than G.E. (as calculated by CNS-Dose) <u>and</u> off-site protective actions (evacuation) have not been completed.	25R
Prevent imminent core damage.	25R
Life-saving activities or protection of <u>large populations</u> (only on a voluntary basis).	> 25R

- **<u>NOTE</u>** – Sum of external effective dose equivalent and committed effective dose equivalent to non-pregnant adults from exposure and intake during an emergency situation. Workers performing services during emergencies should limit dose to the lens of the eye to 3 times the listed value and doses to any other organ (including skin and body extremities) to 10 times the listed value. These limits apply to all doses from an incident, except those received in unrestricted areas as members of the public during the intermediate phase of the incident (refer to Chapters 3 and 4 of EPA 400).

Section Break (Next Page)

CNS OPERATIONS MANUAL EPIP PROCEDURE 5.7.2 EMERGENCY DIRECTOR EPIP	USE: REFERENCE QUALITY: QAPD RELATED EFFECTIVE: 1/6/15 APPROVAL: ITR-RDM OWNER: T. J. RIENTS DEPARTMENT: EP

1. PURPOSE

1.1 This procedure prescribes actions to be taken by the Emergency Director during any declared emergency. It is organized in a logical sequence to provide guidance for the consistent implementation of the appropriate portions of the CNS Emergency Plan. These portions of the Emergency Plan are implemented via use of various Emergency Plan Implementing Procedures (EPIPs).

2. PRECAUTIONS AND LIMITATIONS

2.1 The Emergency Director may not delegate the following:®[4]

- Event Declaration.
- The decision to notify authorities responsible for off-site emergency measures.
- Recommend protective actions to authorities responsible for off-site emergency measures.
- Authorize emergency workers to receive dose in excess of 10CFR20 occupational limits.
- Authorize KI for emergency workers.

ATTACHMENT 2 INFORMATION SHEET

ATTACHMENT 2 INFORMATION SHEET

1. DISCUSSION

1.1 Under emergency conditions, it may become necessary for emergency workers to receive exposures in excess of occupational limits established by 10CFR20. Emergency dose exposure limits (guidance) are defined for emergency workers performing several activities. These exposure limits are listed on Attachment 1.

1.2 Only the Emergency Director has the authority to authorize exposures in excess of occupational limits. These exposures are only justifiable if it is determined that benefits to be achieved are commensurate with the projected dose and every reasonable effort is being made to maintain emergency workers doses As Low As Reasonably Achievable (ALARA).

1.3 Justification of any such exposure must include the presence of conditions that prevent the rotation of workers or other commonly-used dose reduction methods such as protective clothing or respiratory protection. For the risks to be judged "acceptable", the net benefit to society needs to be positive.

2. REFERENCES

2.1 TECHNICAL SPECIFICATIONS

2.1.1 Section 5.4, Procedures.

2.2 CODES AND STANDARDS

2.2.1 10CFR20.

2.2.2 Environmental Protection Agency EPA 400-4-92-001, Manual of Protective Action Guides and Protective Actions for Nuclear Incidents, May 1992.

2.2.3 Federal Register/Vol. 47, No. 205/Friday, October 22, 1982/Notices.

2.2.4 ICRP Report 59, Permissible Dose for Internal Radiation Working Breathing Rate.

2.2.5 NCRP Report 39, 1971, Basic Radiation Protection Criteria.

2.2.6 NPPD Emergency Plan for CNS.

2.2.7 NUREG 0654/FEMA-REP-1, Revision 1, Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants.

2.2.8 NUREG 0737, November 1980, Emergency Exposure Limits.

Examination Outline Cross-Reference	Level	SRO
Revised to state "is/is NOT" permitted to.	Tier#	3
	Group#	N/A
	K/A #	G2.3.12
	Rating	3.7

3. Radiation Control

G2.3.12 Knowledge of radiological safety principles pertaining to licensed operator duties, such as containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc. (CFR: 41.12 / 45.9 / 45.10)

Question 98

Which one of the following completes the statements below identifying a condition which warrants Potassium Iodide (KI) distribution and who can authorize KI distribution to CNS employees?

KI shall be authorized for emergency workers when a calculated dose of ____(1)____ is likely to be received.

The Emergency Director ____(2)____ permitted to delegate this authorization to the Radiation Protection Manager.

A. (1) ≥ 1 REM TEDE
   (2) is

B. (1) ≥ 1 REM TEDE
   (2) is NOT

C. (1) ≥ 5 REM CEDE thyroid
   (2) is

D. (1) ≥ 5 REM CEDE thyroid
   (2) is NOT

Answer:

D (1) ≥ 5 REM CEDE thyroid
(2) is NOT

Explanation:
5.7.14 section 3.1 states the Emergency Director (non-delegable) shall authorize KI for emergency workers when any of the following conditions are present:

- Fuel cladding has been determined to be lost per EPIP 5.7.1, Attachment 3.
- A calculated dose of ≥ 5 rem (0.05 Sv) to the thyroid (CDE) is likely to be received;

Therefore answer D is correct.

Distracters:
Distractors that in Part 1 list ≥ 1 REM TEDE are plausible because along with ≥ 5 REM CEDE, this is an offsite dose value at which a General Emergency must be declared. It is wrong because only thyroid absorption of Iodine is a concern for KI issuance, and Iodine absorption is related in CEDE, or internal dose. This includes answers A and B.

Distractors that in Part 2 list authorization of KI may be delegated by the Emergency Director to the Radiation Protection Manager are plausible because the RP Manager is the primary person responsible for KI distribution in the EOF. It is wrong because 5.7.14 section 3.1 states this duty is non-delegable. This includes answers A and C.

Technical References: 5.7.14, Stable Iodine Thyroid Blocking (KI).

References to be provided to applicants during exam: none

Learning Objective:
GEN0030401B0B020A Emergency Measures: State two types of protective actions and who is responsible for directing each. (On site - Off site).

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank	
	New	X
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.43(b)(4)	
Level of Difficulty:	2	
SRO Only Justification:		

This requires knowledge Emergency Director duties pertaining to Emergency Plan procedures.

CNS·OPERATIONS·MANUAL¶ EPIP·PROCEDURE·5.7.14¶ STABLE·IODINE·THYROID·BLOCKING·(KI)¤	USE:··REFERENCE¶ QUALITY:··QAPD·RELATED¶ EFFECTIVE:··7/1/15¶ APPROVAL:··ITR-RDM¶ OWNER:··S.·C.·REZAB¶ DEPARTMENT:··EP¤

1. PURPOSE

1.1 This procedure defines who has the authority to distribute Potassium Iodide (KI), under what emergency conditions KI should be distributed, and method used to distribute KI.©[1]

2. PRECAUTIONS AND LIMITATIONS

2.1 KI should not be taken by persons allergic to iodine.

2.2 Do not take more than the recommended dose of KI.

2.3 KI shall only be taken on a voluntary basis.

2.4 KI should not be taken for more than 10 days.

2.5 Individuals suspected of having a KI sensitivity should contact their medical provider for evaluation and if warranted, be placed on medical restriction for KI. These individuals should be assigned to ERFs beyond the 10 mile EPZ or off-site ERFs with emergency ventilation systems and charcoal filtering.

3. REQUIREMENTS

3.1 Emergency Director (non-delegable) shall authorize KI for emergency workers when any of the following conditions are present:

3.1.1 Fuel cladding has been determined to be lost per EPIP 5.7.1, Attachment 3.

3.1.2 A calculated dose of ≥ 5 rem (0.05 Sv) to the thyroid (CDE) is likely to be received.

ATTACHMENT 2 KI DISTRIBUTION INSTRUCTIONS

ATTACHMENT 2 KI DISTRIBUTION INSTRUCTIONS

1. Obtain completed Attachment 1 from the Emergency Director.

1.1 Notify the affected non-NPPD emergency response organizations that the Emergency Director has authorized KI and it is available.

NOTE – Administration of KI to non-NPPD personnel shall be the responsibility of the organizations to which these personnel belong.

1.2 Designate an individual in each of the affected ERO facilities to distribute KI by contacting the individual.

FACILITY	PRIMARY PERSON FOR DISTRIBUTION	NAME OF INDIVIDUAL CONTACTED TO PERFORM DISTRIBUTION
CR	SHIFT CHEM/RP	
OSC	CHEM/RP LEAD**	
TSC	CHEM/RP COORDINATOR	
SECURITY	SECURITY COORDINATOR***	
EOF	RAD CONTROL MANAGER	

** Primary for distribution to West Warehouse addition if required. Distribution may require team dispatch.

*** Assisted by the TSC Chem/RP Coordinator. SAS covered by Shift Chem/RP.

1.3 Immediately communicate or distribute a copy of Attachment 1 to the individuals responsible for KI distribution.

1.4 Instruct individuals distributing KI in the facilities to perform Steps 1.4.1 through 1.10 of this attachment.

1.4.1 Obtain the following materials:

1.4.1.1 Containers of 130 mg KI tablets (14 tablets/container).

1.4.1.2 Patient Package Inserts.

1.4.1.3 Completed Attachment 1.

1.4.1.4 Copies of Attachments 3 and 4.

1.4.1.5 Lists of the individuals authorized to receive KI as required by the assigned distribution:

a. Current Emergency Response Facility Staffing Board (ALL).

b. A list of current Operations Watchstanders (CR).

Examination Outline Cross-Reference	Level	SRO
Revised to determine whether the EOP 3A is required to be re-entered/continued and if all legs/SP temp leg is required to be addressed. Provide actual SP temperature to meet KA and support “continue in EOP 3A” plausibility.	Tier#	3
	Group#	N/A
	K/A #	G2.4.1
	Rating	4.8
4. Emergency Procedures / Plan G2.4.1 Knowledge of EOP entry conditions and immediate action steps. (CFR: 41.10 / 43.5 / 45.13)		

Question 99

EOPs have been entered due to high Drywell Pressure from a small steam leak.

- All legs of EOP-1A have been addressed by the CRS.
- All legs of EOP-3A have been addressed by the CRS except for Torus Water Temperature.

Average Torus Water Temperature is now 96°F.

Which one of the following completes the statement below regarding the required CRS actions?

The CRS is required to ____(1)____ EOP-3A and address ____(2)____.

A. (1) re-enter
(2) the Torus Water Temperature leg ONLY

B. (1) re-enter
(2) ALL legs

C. (1) continue in
(2) the Torus Water Temperature leg ONLY

D. (1) continue in
(2) ALL legs

Answer:
B. (1) re-enter

(2) ALL legs

Explanation:
Guidance for EOP execution given in the PSTG (ref. B-4-4) states anytime an EOP entry condition is exceeded, any associated EOP is required to be re-entered at the beginning, even if it had been entered previously due to another parameter. Another parameter exceeding its EOP entry condition may be indicative of degrading conditions where specific EOP action may not have been required earlier but is now. Changes in one parameter may also be indicative that degradation is occurring in the another parameter/function that needs to be re-addressed. Therefore, Answer B is correct.

Distracters:
Answer A is incorrect due to all legs of EOP 3A being required to be addressed. This choice is plausible due to EOP entry condition being the primary focus for execution. The candidate that focuses on the re-entry condition or does not recognize the entry condition would select this answer.
Answer C is incorrect due to all legs of EOP 3A being required to be addressed following re-entry. This choice is plausible due to not recognizing the EOP entry condition making this a correct answer by only requiring continuing and addressing the leg that has not been addressed. The candidate that focuses on the re-entry condition OR does not recognize the EOP entry would select this answer.
Answer D is incorrect due to EOP 3A re-entry being required. This choice is plausible due to not recognizing the EOP entry condition making this a partially correct answer. The candidate that does not recognize EOP re-entry requirement and confuses whether all vs. the applicable leg is required to be addressed would select this answer.

Technical References: PSTG (ref. B-4-4), , EOP-3A

References to be provided to applicants during exam: none

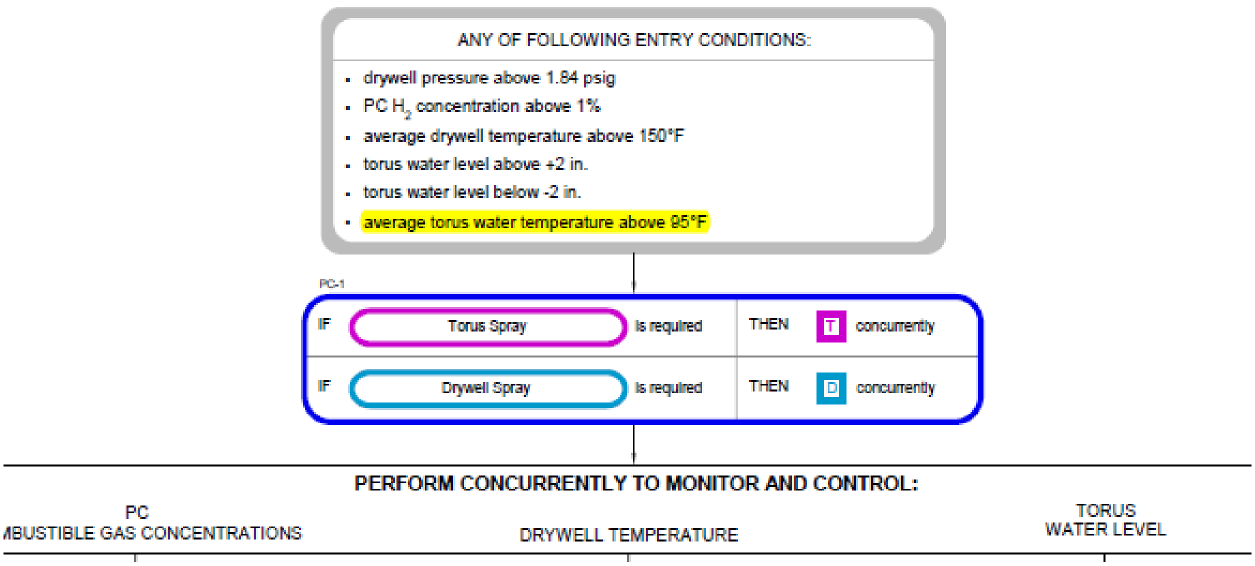
Learning Objective:
INT00806040010200 Discuss the method used to track progress in the flowcharts (place keeping).
INT00806130011100 Given plant conditions and EOP Flowchart 3A, PRIMARY CONTAINMENT CONTROL, determine required actions.

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X

Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	

10CFR Part 55 Content:	
	55.43(b)(5)
Level of Difficulty:	2
SRO Only Justification:	
This involves EOP procedure selection given successive entry conditions for different procedures. It relates to proper EOP execution, which is an SRO function.	

PRIMARY CONTAINMENT CONTROL



EPG/SAG Step (Introduction, continued)

Each emergency operating procedure (EOP) developed from the EPGs should be entered whenever a defined entry condition occurs or an explicit direction to do so is encountered, even if the procedure has already been entered, unless instructions developed from the severe accident guidelines are being executed. EOPs may be exited when it has been determined that an emergency no longer exists.

Plant-specific accident management guidelines (PSAMGs) developed from the SAGs should be entered when required by the EOPs. Additional EOP entry conditions may then be disregarded until the PSAMGs are exited. Like EOPs, PSAMGs may be exited when it has been determined that an emergency no longer exists.

The EPG entry conditions are symptomatic of both emergencies and events which may degrade into emergencies. The existence of an entry condition is therefore not necessarily indicative of an emergency. Nor does the absence of entry conditions, by itself, necessarily signify that an emergency does not exist.

Discussion

The EPG entry conditions are based upon the values of key plant parameters rather than the existence of certain events. The conditions have been defined so as to be simple, operationally significant, unambiguous, readily identifiable, and familiar to control room operators. The specified setpoints also provide advance warning of potential emergency conditions, allowing action to be taken sufficiently early to prevent more severe consequences. The low RPV water level entry condition setpoint specified in the RPV Control guideline is an example: although RPV water level at the low level scram setpoint does not itself constitute an emergency condition, correct and prompt operator action may be required when this condition occurs to prevent core uncovery.

When an entry condition occurs, the corresponding procedure must be entered. If another entry condition for the same procedure occurs, the procedure must be reentered at the beginning, even if it is already being executed. If entry conditions for more than one procedure occur at the same time, the procedures must be executed concurrently.

Examination Outline Cross-Reference	Level	SRO
	Tier#	3
	Group#	N/A
	K/A #	G2.4.30
	Rating	4.1
4. Emergency Procedures / Plan G2.4.30 Knowledge of events related to system operation/status that must be reported to internal organizations or external agencies, such as the State, the NRC, or the transmission system operator. (CFR: 41.10 / 43.5 / 45.11)		

Question 100

Which one of the following completes the statements below regarding EAL declaration and notifications?

When an EAL has been exceeded, the Shift Manager must officially declare an event within 15 minutes from when ____(1)____.

The state agencies must be notified within ____(2)____ of declaration.

A. (1) the EAL wall chart assessment begins
(2) 15 minutes

B. (1) the EAL wall chart assessment begins
(2) 60 minutes

C. (1) indication of exceeding the EAL became available to plant Operators
(2) 15 minutes

D. (1) indication of exceeding the EAL became available to plant Operators
(2) 60 minutes

Answer:
C. (1) indication of exceeding the EAL became available to plant Operators (2) 15 minutes
Explanation:
When an EAL is exceeded, the Shift Manager has 15 minutes from the time the

indication that the EAL had been exceeded manifests itself to the operating crew to assume the role of Emergency Director, assess the EALs, and make the emergency declaration. Then, state and local agencies must be notified within 15 minutes of the emergency declaration.

Distracters:

Answer A is plausible because 15 minutes is such a short duration for the crew to recognize EAL related conditions and that communication to route to the Shift Manager and then for the Shift Manager to review EALs per procedure 5.7.1, Emergency Classifications and identify the highest EAL that has been exceeded. Since when the Shift Manager first receives information from the operators is beyond his control, the unprepared candidate might assume the 15 minutes starts after he has received knowledge of the event. It is wrong because the 15 minute clock starts as soon as that information becomes available to the operators, whether they recognize it or not.

Answer B is plausible for the same reason as stated above for Answer A. 60 minutes is plausible because that is the time limit for notifying the NRC. It is wrong because of the reason stated above for answer A and because the time limit for notifying State and local agencies is 15 minutes for protection of the near site public.

Answer D, 60 minutes is plausible because that is the time limit for notifying the NRC. It is wrong because the time limit for notifying State and local agencies is 15 minutes for protection of the near site public.

Technical References: 5.7.1, Emergency Classifications, step 2.2; 5.7.6, Notifications, steps 2.1, 2.4

References to be provided to applicants during exam: none

Learning Objective:

GEN0030401B0B030B Emergency Notifications and Communications Systems: State the time requirements for initial and/or follow-up notifications to offsite agencies.

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	

10CFR Part 55 Content:	
	55.43(b)(5)
Level of Difficulty:	2

SRO Only Justification:	
The SRO is responsible for assuming the duties of Emergency Director, including emergency declaration and notification requirements.	

CNS OPERATIONS MANUAL EPIP 5.7.1 EMERGENCY CLASSIFICATION	USE: REFERENCE QUALITY: QAPD RELATED EFFECTIVE: 4/8/15 APPROVAL: ITR-RDM OWNER: L. J. DUBOIS DEPARTMENT: EP

1. PURPOSE[2]

1.1 This procedure provides the formal set of threshold conditions necessary to classify an event at CNS into one of the four emergency classifications described in NUREG-0654, NEI-99-01, Revision 5, and the CNS Emergency Plan.

2. PRECAUTIONS AND LIMITATIONS

2.1 The steps required by this procedure are in addition to the steps required to maintain or restore the station to a safe condition.

2.2 Assessment, classification, and declaration an emergency condition shall be completed within 15 minutes after the initial availability of indications to plant Operators that an EAL has been exceeded provided that:

2.2.1 Implementation of response actions required to protect public health and safety are not delayed.

AND

2.2.2 Any delay in declaration does not deny the State and Local authorities the opportunity to implement measures necessary to protect the public health and safety.

CNS OPERATIONS MANUAL EPIP PROCEDURE 5.7.6 NOTIFICATION	USE: REFERENCE QUALITY: QAPD RELATED EFFECTIVE: 5/27/15 APPROVAL: ITR-RDM OWNER: L. J. DUBOIS DEPARTMENT: EP

1. PURPOSE

1.1 Provide instructions for initial, follow-up, and termination notifications to responsible State and Local governmental agencies, NRC notifications, initial generation of news releases to the Media, and notifications to other off-site support agencies.©[5,7]

2. PRECAUTIONS AND LIMITATIONS

2.1 Initial notifications to State/Local agencies shall be performed within 15 minutes of each declaration of an Emergency and/or change in Protective Action Recommendations (PARs).©[2,4]

2.2 Initial notifications to State/Local agencies take priority over notifications to the NRC except as provided by notifications required by Hostile Action Procedures 5.5AIRCRAFT and 5.5SECURITY.

2.3 Initial notifications to State/Local agencies take priority over any follow-up notifications to these agencies.

2.4 NRC notification shall be performed immediately following notification of responsible State and Local governmental agencies, and <u>not</u> later than 1 hour after the time of declaration of one of the emergency classes.

2.5 At a NOUE, follow-up notifications should be made for significant changes in plant status related to the unusual event or other classifiable conditions.

- 2.6 At an ALERT or higher classification, follow-up notifications to responsible State and Local governmental agencies shall be performed approximately every 60 minutes or sooner if there is a significant change in the status of the emergency.©[2]

- 2.7 IF a release of Airborne Radioactive material greater than NOUE limits is occurring or has occurred during the event, THEN a follow-up notification, including Section 8, "Release Information" of the Notification Report should be completed immediately once initial notifications to State and Local Authorities and the NRC have been completed, when performing manual Notification Reports. IF CNSDOSE includes the Section 8 Information on the initial notification, THEN a follow-up notification is not required.

- 2.8 Notification of termination to off-site agencies shall be performed within 1 hour after the termination of the emergency.

- 2.9 If the Control Room must be evacuated and off-site notification responsibilities have not been transferred to the EOF, the Shift Communicator shall perform off-site notifications over the state notification telephone from any accessible CNS PBX telephone.

- 2.10 The Emergency Director shall be immediately notified of any difficulties or delays in completing this procedure.

- 2.11 The Emergency Director may not delegate the decision to notify authorities responsible for off-site emergency measures.©[3]

- 2.11.1 Facility Directors may sign initial Notification Forms with Emergency Director verbal approval and appropriate log entries.©[1]

- 3. ENTRY CONDITIONS

- 3.1 An Emergency has been declared per Procedure 5.7.1.

- 4. NOTIFICATIONS FROM CONTROL ROOM

- 4.1 Emergency Director (ED) directs Shift Communicator (Communicator) to prepare Notification Report.

- 4.2 IF PMIS is not available, THEN Communicator perform following:

- 4.2.1 Complete Notification Report using Attachments 1 and 2.

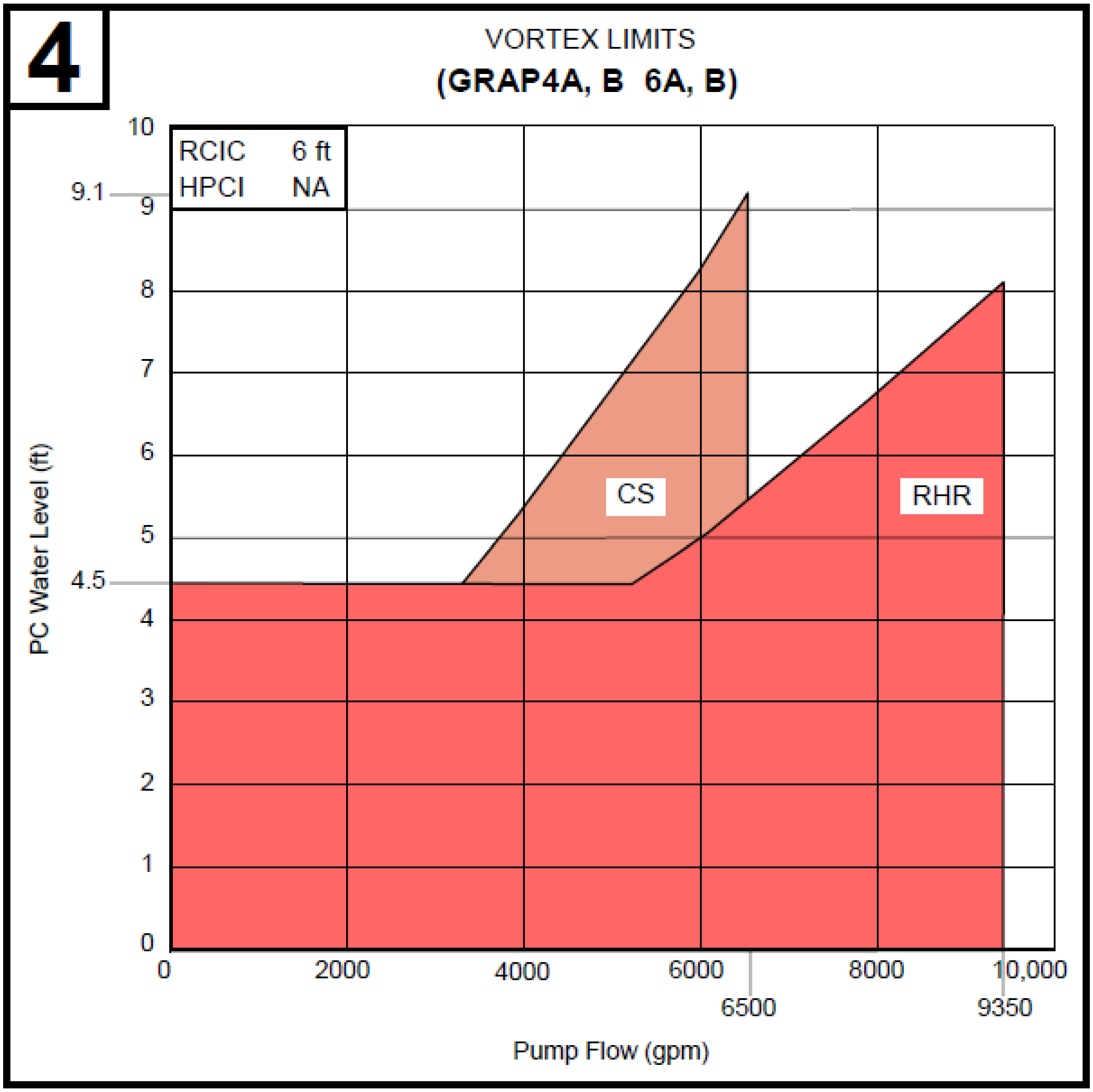
- 4.2.2 Proceed to Step 4.4.

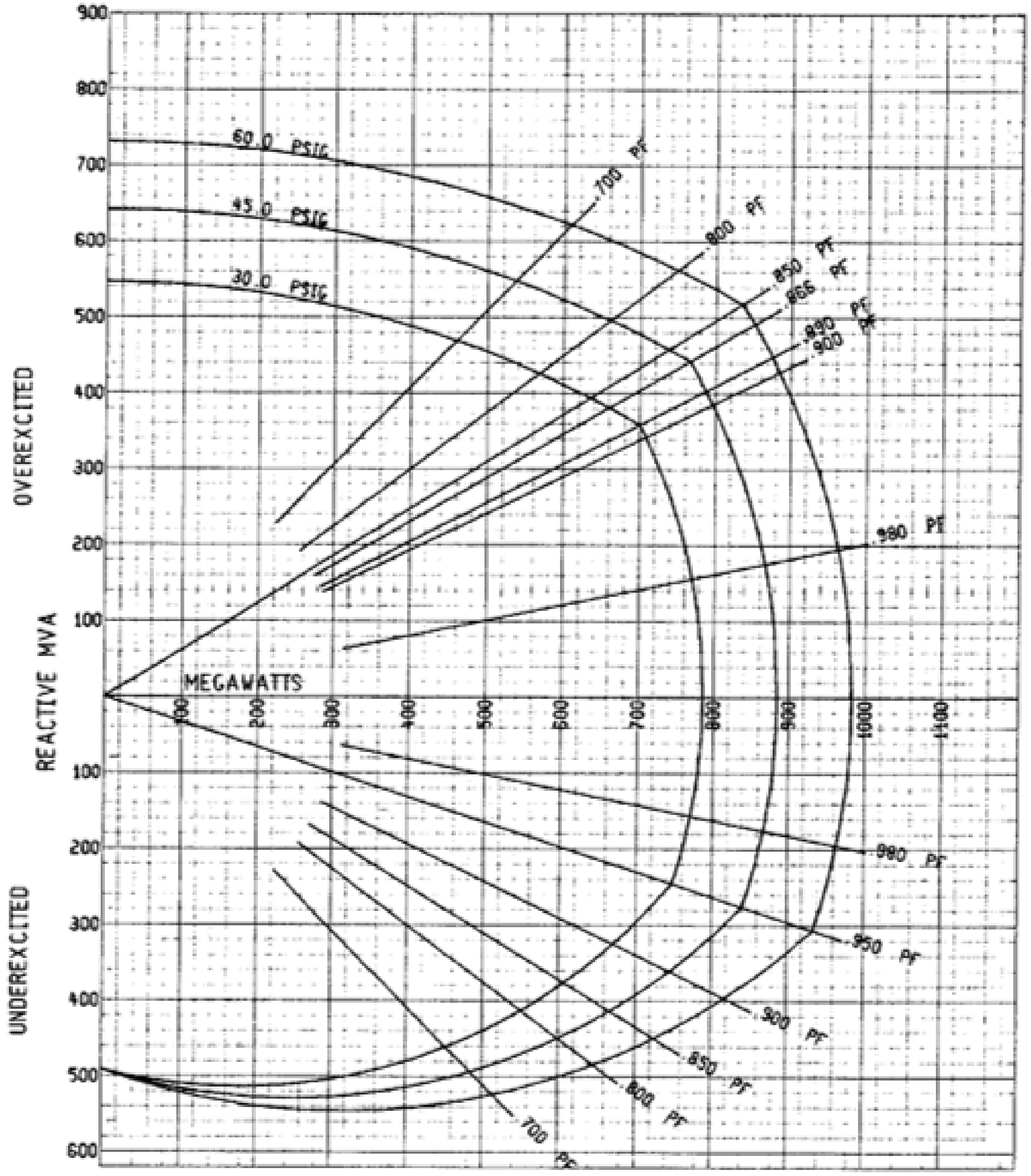
- 4.3 Communicator fills out Notification Report via a PMIS terminal.

- 4.3.1 Determine if it is raining via MET display.

- 4.3.2 Complete or replicate dose assessment using PMIS Dose Program following PMIS on-line instructions or Procedure 5.7.17.

RO References


4
VORTEX LIMITS
(GRAP4A, B 6A, B)
RCIC 6 ft
HPCI NA
CS
RHR
PC Water Level (ft)
Pump Flow (gpm)
10
9.1
9
8
7
6
5
4.5
4
3
2
1
0
0
2000
4000
6000
6500
8000
9350
10,000

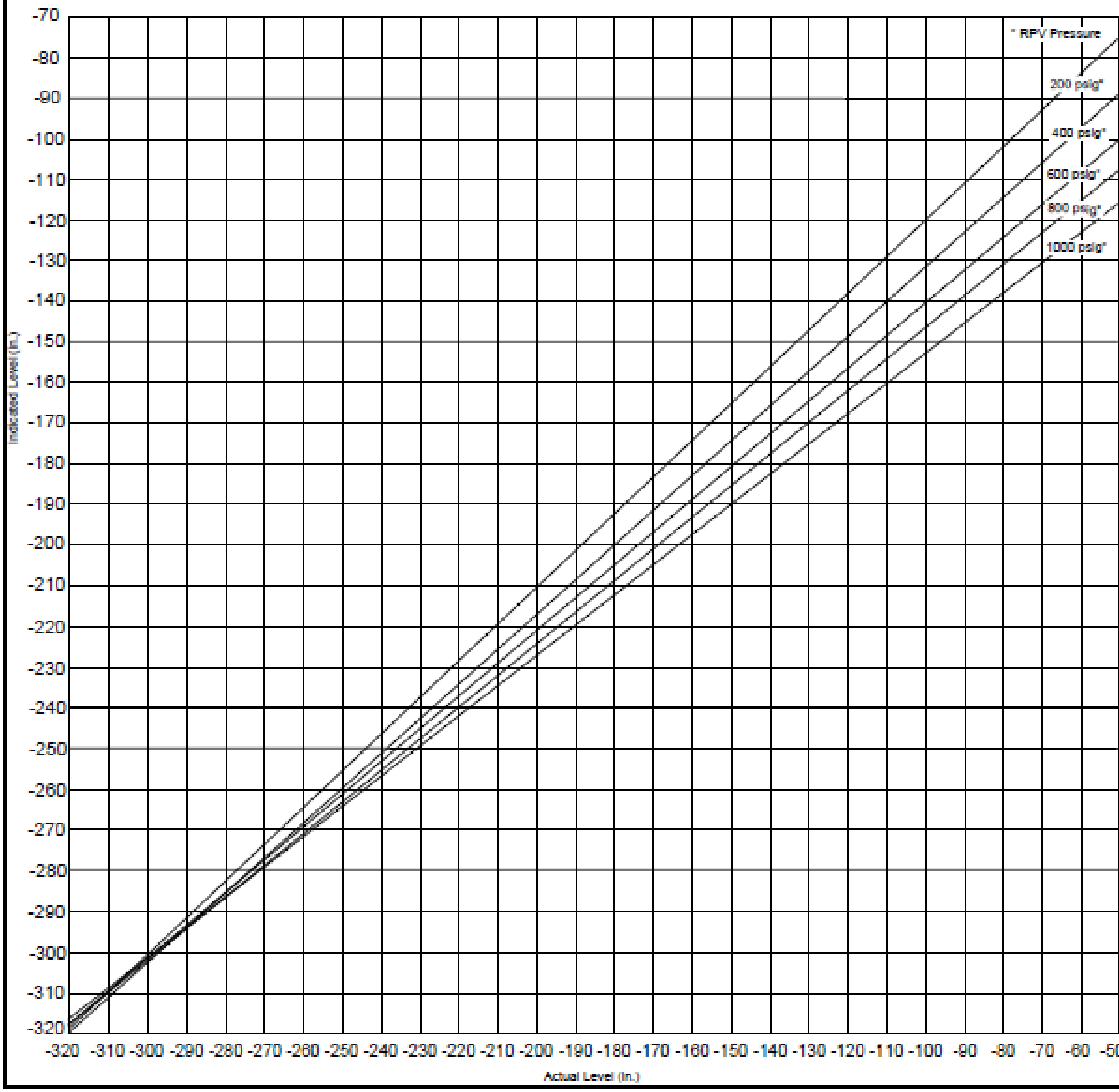


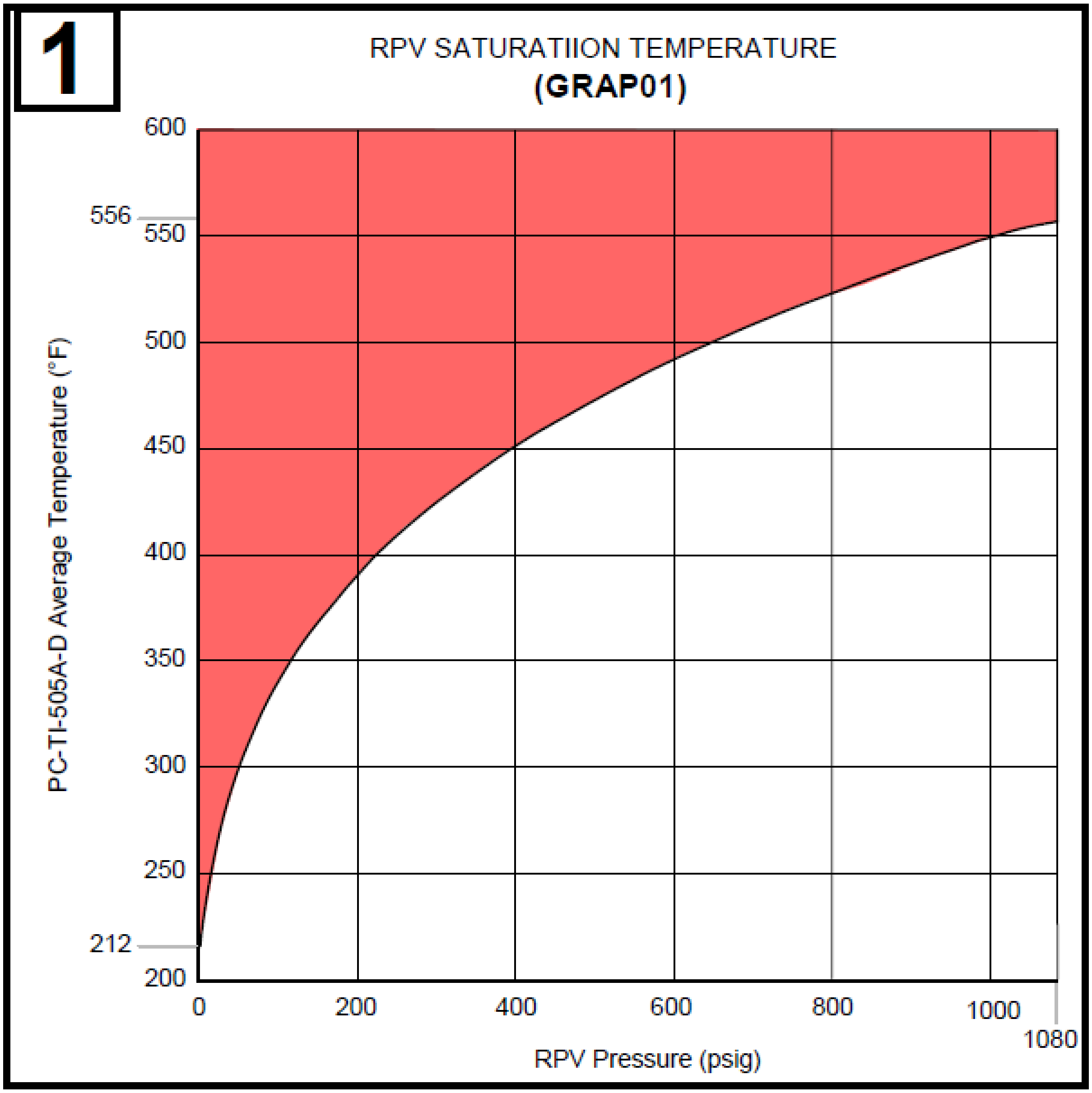
HYDROGEN INNER-COOLED TURBINE GENERATOR
983.000 MVA .85 PF 22.0 KV 25797 AMPERES
3 PHASE 60 CYCLES 1800 RPM .58 SCR 60 PSIG

Figure 1

14

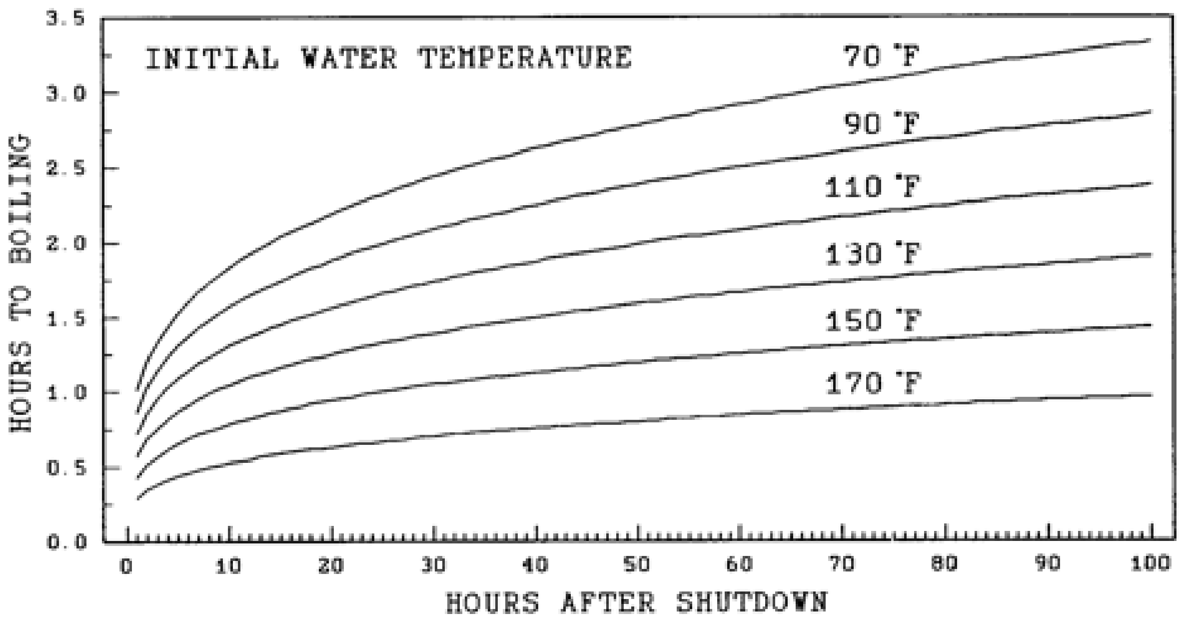
FUEL ZONE (FZ) RANGE CORRECTION
(GRAP14A, B)




1
RPV SATURATIION TEMPERATURE
(GRAP01)
PC-TI-505A-D Average Temperature (°F)
600
556
550
500
450
400
350
300
250
212
200
0
200
400
600
800
1000
1080
RPV Pressure (psig)

SRO References

ATTACHMENT 5 TIME TO CORE BOILING/TIME TO CORE UNCOVERY



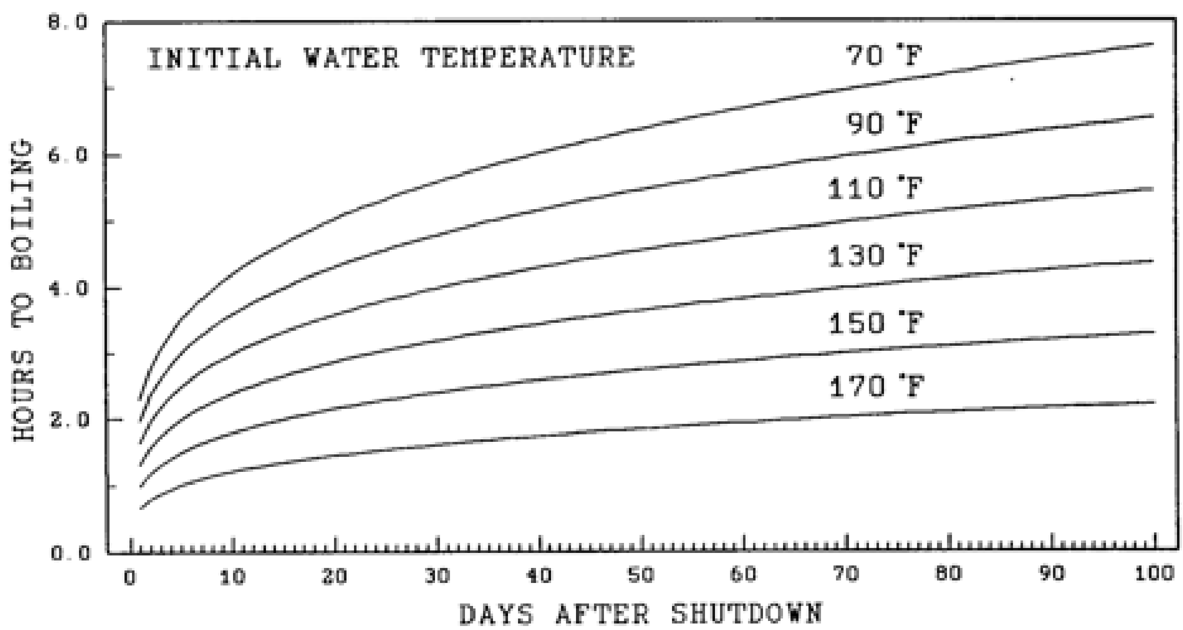


Figure 1 - TIME TO BOILING - WATER LEVEL AT HIGH LEVEL TRIP

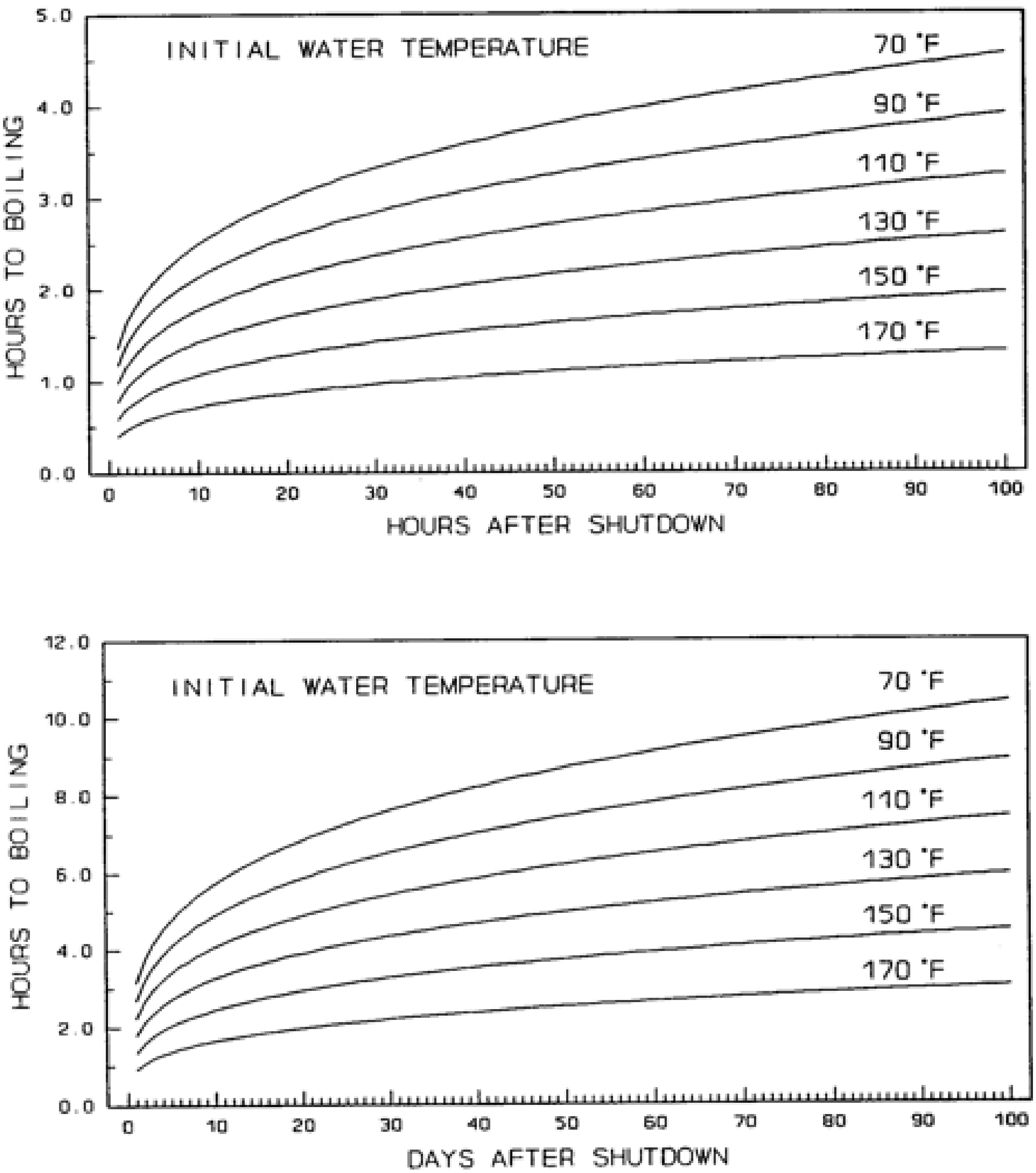
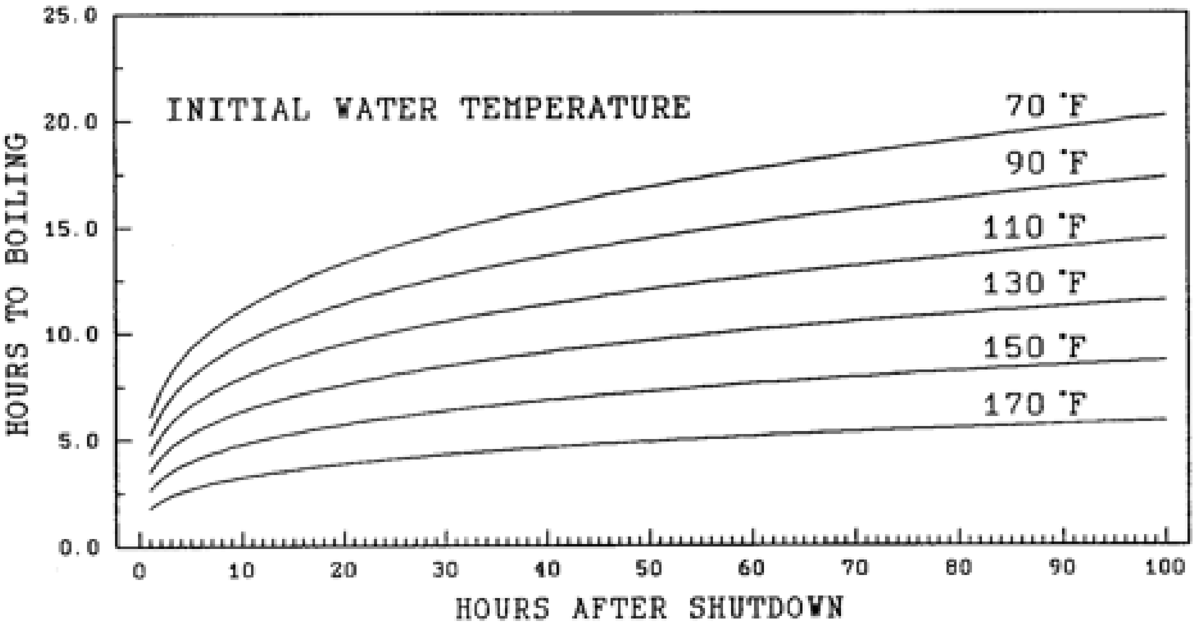


Figure 2 - TIME TO BOILING - WATER LEVEL AT FLANGE



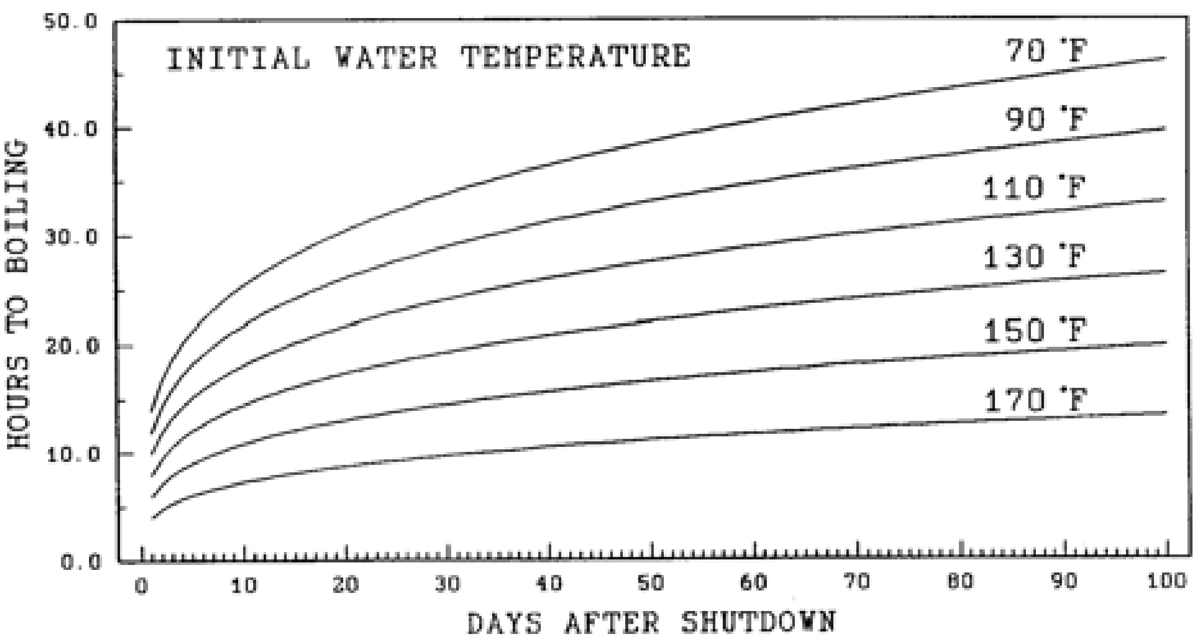
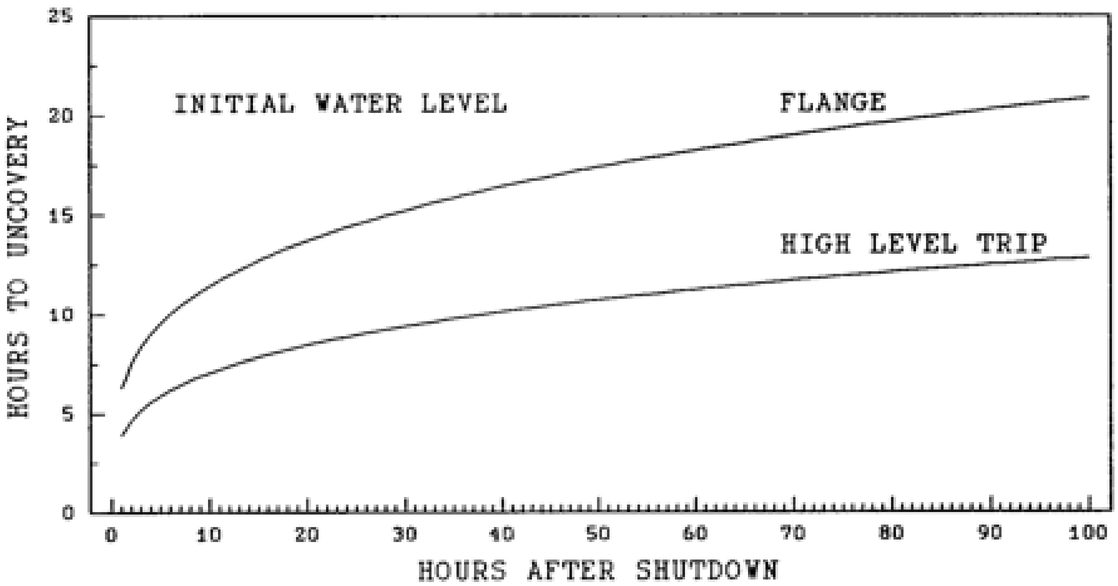


Figure 3 - TIME TO BOILING - WATER TO LEVEL FLOODED TO 1001'



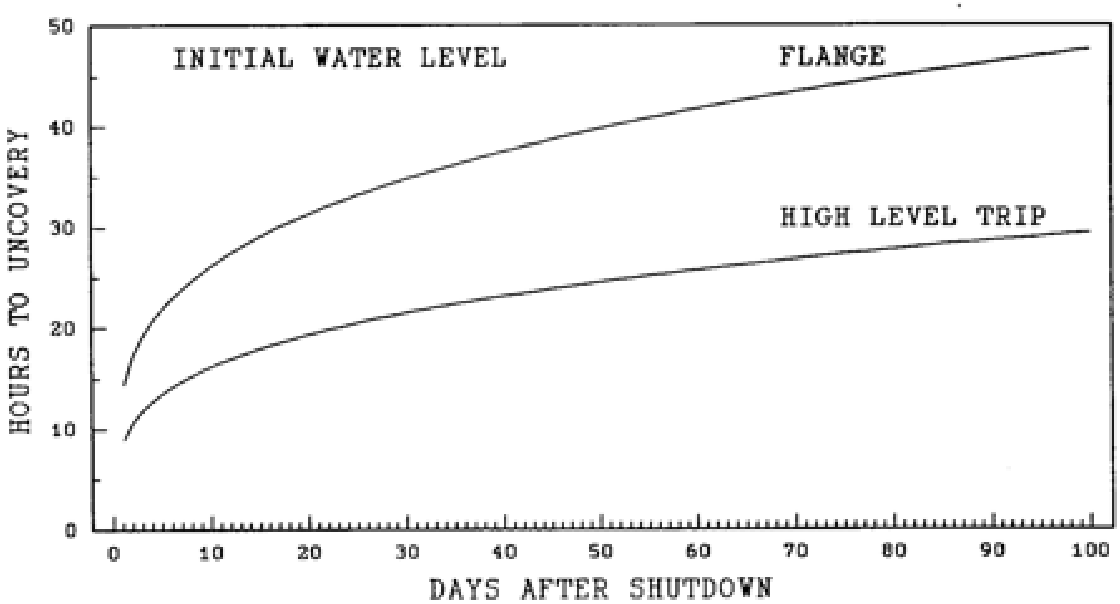
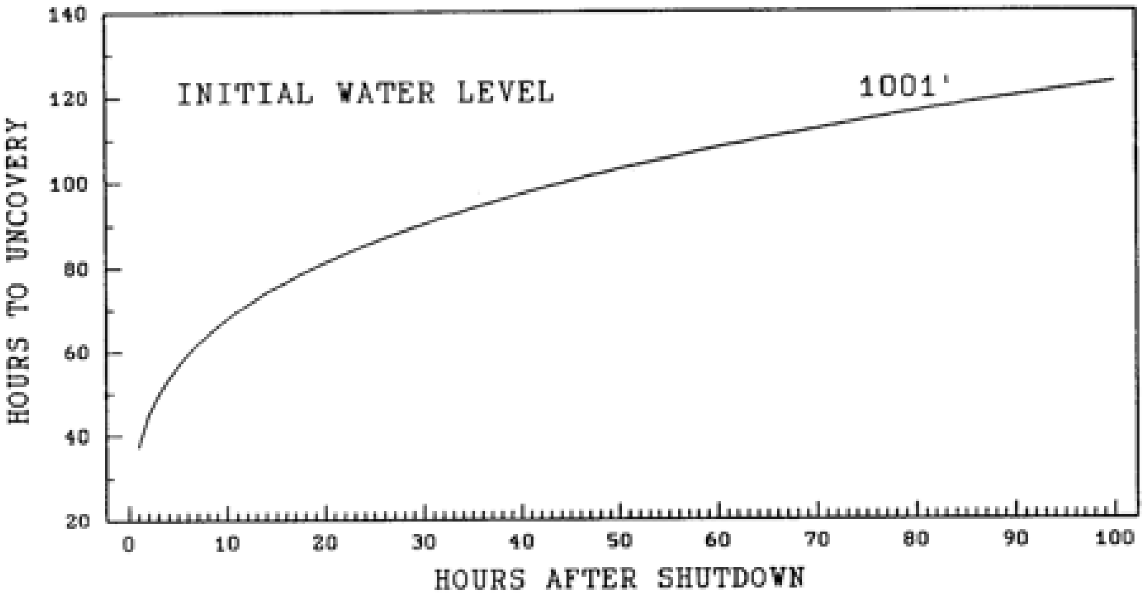


Figure 4 - TIME TO CORE UNCOVERY



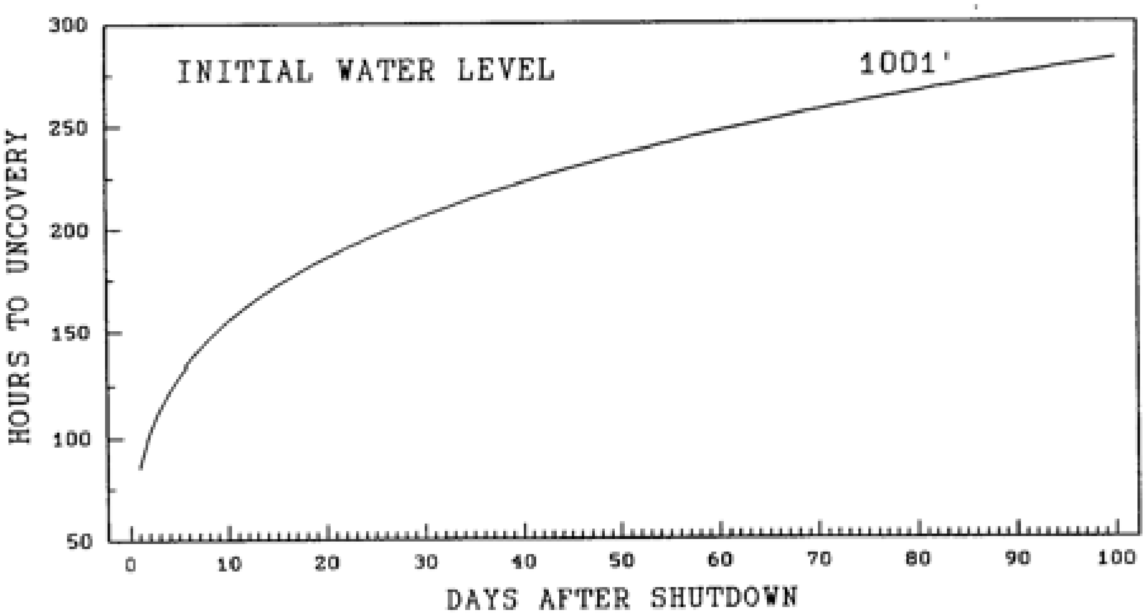


Figure 5 - TIME TO CORE UNCOVERY

INFORMATION ONLY

3.1 REACTIVITY CONTROL SYSTEMS

3.1.7 Standby Liquid Control (SLC) System

LCO 3.1.7

APPLICABILITY:

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SLC subsystem inoperable.	A.1 Restore SLC subsystem to OPERABLE status.	7 days
B. Two SLC subsystems inoperable.	B.1 Restore one SLC subsystem to OPERABLE status.	8 hours
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3. <u>AND</u> C.2 Be in MODE 4.	12 hours 36 hours

3.8 ELECTRICAL POWER SYSTEMS

3.8.1 AC Sources — Operating

LCO 3.8.1

APPLICABILITY:

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.	1 hour AND Once per 8 hours thereafter
	AND 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)
	AND	(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.3	Restore offsite circuit to OPERABLE status.	7 days AND 14 days from discovery of failure to meet LCO
B. One DG inoperable.	B.1	Perform SR 3.8.1.1 for OPERABLE offsite circuit(s).	1 hour AND Once per 8 hours thereafter
	AND		
	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)
	AND		
			(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. (continued)	B.3.1	Determine OPERABLE DG is not inoperable due to common cause failure.	24 hours
	OR		
	B.3.2	Perform SR 3.8.1.2 for OPERABLE DG.	24 hours
	AND		
	B.4	Restore DG to OPERABLE status.	7 days AND 14 days from discovery of failure to meet LCO
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

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. One offsite circuit inoperable. AND One 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 either division. ---------------------------- D.1 Restore offsite circuit to OPERABLE status. OR D.2 Restore DG to OPERABLE status.	 24 hours 24 hours
E. Two DGs inoperable.	E.1 Restore one DG to OPERABLE status.	2 hours
F. Required Action and associated Completion Time of Condition A, B, C, D, or E not met.	F.1 Be in MODE 3. AND F.2 Be in MODE 4.	12 hours 36 hours
G. Three or more required AC sources inoperable.	G.1 Enter LCO 3.0.3.	Immediately

3.3 INSTRUMENTATION

3.3.3.1 Post Accident Monitoring (PAM) Instrumentation

LCO 3.3.3.1

APPLICABILITY:

ACTIONS

---NOTE---

Separate Condition entry is allowed for each Function. For Function 5, separate Condition entry is allowed for each penetration flow path.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one required channel inoperable.	A.1 Restore required channel to OPERABLE status.	30 days
B. Required Action and associated Completion Time of Condition A not met.	B.1 Initiate action in accordance with Specification 5.6.6.	Immediately
C. One or more Functions with two required channels inoperable. <u>OR</u> One Function 2.c channel inoperable.	C.1 Restore one required channel to OPERABLE status.	7 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. Required Action and associated Completion Time of Condition C not met.	D.1	Enter the Condition referenced in Table 3.3.3.1-1 for the channel.	Immediately
E. As required by Required Action D.1 and referenced in Table 3.3.3.1-1.	E.1	Be in MODE 3.	12 hours
F. As required by Required Action D.1 and referenced in Table 3.3.3.1-1.	F.1	Initiate action in accordance with Specification 5.6.6.	Immediately

INFORMATION ONLY

Table 3.3.3.1-1 (page 1 of 1)
Post Accident Monitoring Instrumentation

FUNCTION	REQUIRED CHANNELS	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1
		E
2. Reactor Vessel Water Level		
a. Fuel Zone		E
b. Wide Range		E
c. Steam Nozzle		F
		E
		F
		E
		E
		E
		E
		E
		E

(a) Not required for isolation valves whose associated penetration flow path is isolated by at least one closed and deactivated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured.

(b) Only one position indication channel is required for penetration flow paths with only one installed control room indication channel.

(c) A channel requires a minimum of four resistance temperature detectors (RTDs) to be OPERABLE with no two adjacent RTDs inoperable.

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

2. NEDE-23785-1-P-A, "The GESTR-LOCA and SAFER Models for the Evaluation of the Loss-of-Coolant Accident", Volume III, Revision 1, October 1984.

3. NEDO-31960 and NEDO-31960 Supplement 1, "BWR Owner's Group Long-Term Stability Solutions Licensing Methodology" (the approved Revision at the time the reload analysis is performed).

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.6 Post Accident Monitoring (PAM) Instrumentation Report

When a report is required by Condition B or F of LCO 3.3.3.1, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.4 RCS Operational LEAKAGE

LCO 3.4.4 RCS operational LEAKAGE shall be limited to:

APPLICABILITY:

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Unidentified LEAKAGE not within limit. <u>OR</u> Total LEAKAGE not within limit.	A.1 Reduce LEAKAGE to within limits.	4 hours
B. Unidentified LEAKAGE increase not within limit.	B.1 Reduce unidentified LEAKAGE increase to within limits. <u>OR</u>	4 hours (continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. (continued)	B.2	Verify source of unidentified LEAKAGE increase is not service sensitive type 304 or type 316 austenitic stainless steel.	4 hours
C. Required Action and associated Completion Time of Condition A or B not met. <u>OR</u> Pressure boundary LEAKAGE exists.	C.1	Be in MODE 3.	12 hours
	<u>AND</u>		
	C.2	Be in MODE 4.	36 hours