Facility: Hope Creek

Vendor: GE

Exam Date: 2019

Exam Type: RO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	295001 AA2.06	
Importance Rating	3.2	

K/A Statement: Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION: Nuclear boiler instrumentation

Question: RO #1

Given:

- The plant is operating at 100% rated power.

When:

- A transient caused by a fault in the reactor recirculation control circuitry occurs.

Immediately following the transient, the plant stabilizes with the following parameters:

- Reactor Power is at 50% rated power.
- "A" Recirc pump is tripped.
- "B" Recirc pump speed is at 52%.
- Loop "A" total jet pump flow (FI-R611A-B21) is at 2 Mlbm/hr.
- Loop "B" total jet pump flow (FI-R611B-B21) is at 43.8 Mlbm/hr.
- Jet Pump Flow Recorder (FR-R613-B21) 42.1 Mlbm/hr.

What is actual core flow (WT)?

A. 41.8 Mlbm/hr.

B. 42.1 Mlbm/hr.

C. 45.8 Mlbm/hr.

D. 43.8 Mlbm/hr.

Proposed Answer: **B**

Explanation (Optional): HC.OP-AB.RPV-0003 (attached)

IF Operating Recirc Loop flow **>** 23 Kgpm. THEN:
A. **DETERMINE** Actual Core Flow by
SUBTRACTING 85% of Idle Loop Jet Pump Flow
FROM Operating loop Jet Pump Flow

[FI-R611A(B)-B21 **–** (0.85 x FI-R611B(A)-B21)]
B. **VERIFY** proper function of the subtraction
circuit by checking that calculated core flow (step
A6) is the same as Total Jet Pump Flow (FR-
R613-B21 OR A190).

A: **Incorrect-**.With flow <23Kgpm (approx. 48%speed) **43.8 – 2 = 41.8.** Recirc speed is at 52% (>23 Kgpm).

B: **Correct-** Recirc. Speed at 52% (>23Kgpm), [FI-R611A(B)-B21 **–** (0.85 x FI-R611B(A)-B21)] **43.8 – (.85)2 = 43.8-1.7 = 42.1** which matches the Jet Pump Recorder (FR-R613-B21) 42.1 Mlbm/hr.

C: **Incorrect- 43.8 + 2 = 45.8**, Recirc. Speed at 52% (>23Kgpm), [FI-R611A(B)-B21 **–** (0.85 x FI-R611B(A)-B21)] **43.8 – (.85)2 = 43.8-1.7 = 42.1.**

D: **Incorrect-**. Jet Pump Recorder (FR-R613-B21) 42.1 Mlbm/hr. not (FI-R611B-B21) which is at **43.8 Mlbm/hr.**

Technical Reference(s): HC.OP-AB.RPV-0003 (Attach if not previously provided)

Recirculation System/Power Oscillations

Proposed References to be provided to applicants during examination: none

Learning Objective: Interpret and apply charts, graphs and tables contained within Recirculation System/Power Oscillations.

Question Source: Bank #86019

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(2)

Comments:

RETAINMENT OVERRIDE	
CONDITION	**ACTION**
I. Affected Recirc Pump seal cavity temperature greater than 200° F Date/Time: ______________	___ **I.a** **REDUCE** the affected Recirc pump speed to minimum, **TRIP** the affected Recirc pump and enter Condition A

CAUTIONS:

1. **With the operable loop drive flow near 23k gpm (≈48% Pump Speed), the flow through the idle loop is close to zero and will swap back and forth between forward and reverse flow. Operation in this region can cause excessive Jet Pump vibration and the potential for consequent stress fatigue of the riser brace welds to the vessel. Operation near this value should be minimized.**

NOTES:

2. Core Flow >40% during single loop operation prevents excessive cool down of the idle loop.
13. If WTFLAG = 1 or 2, AND the operable loop drive flow ≤23k gpm (≈48% Pump Speed), the PPC WT (Core Flow) will not be accurate. In this case, add the Jet Pump Loop Flows to determine WT.
    If WTFLAG = 3, WT is calculated from the core plate DP and is accurate.
    If WTFLAG = 4, WT is calculated from the recirculation drive flows and is accurate.

ADDITIONAL INFORMATION:

Procedures:

- HC.OP-DL.ZZ-0026(Q), Surveillance Log.
- HC.OP-IO.ZZ-0006(Q), POWER CHANGES DURING OPERATION.
- HC.SE-PR.ZZ-0003(Q), Thermal Cycle Monitoring.

Valves:

- 1-HV-F031A (B), Recirculation Pump Discharge Valve.

Indications:

- 1BBFR-R613-B21, Jet Pump Flow Recorder
- 1BBFI-R611A (B)-B21, Jet Pump Loop A (B) Flow
- A190, Reactor Jet Pump Total Flow

SUBSEQUENT OPERATOR ACTIONS

CONDITION			ACTION
A. Single Reactor Recirc Pump Tripped. **[T/S 3.4.1.1]** **[T/S 4.4.1.2]** Date/Time: ____________	___	A.1	**INSERT** Control Rods to clear APRM Upscale Alarms.
	___	A.2	**CONTINUE** actions in this condition while monitoring for Power Oscillations. **[IER L2 15-34, Rec. 1a]**
	___	A.3	**ENSURE** that the Recirc VFD 7.2Kv Input Breaker has TRIPPED for the tripped Pump
	___	A.4	**CLOSE** HV-F031A (B) for ≈ 5 minutes, THEN **RE-OPEN** HV-F031A (B). **UNLESS** isolated IAW condition E "Seal Failure" **[CD-976B]**
	___		★ **CAUTION 1** ★
	___		****NOTE 13****
	___	A.5	IF Operating Recirc Loop flow ≤ 23 Kgpm. **DETERMINE** Actual Core Flow by **ADDING** Idle Loop Jet Pump Flow AND Operating Loop Jet Pump Flow. (FI-R611A-B21 and FI-R611B-B21)
	___	A.6	IF Operating Recirc Loop flow **>** 23 Kgpm. THEN:
	___		A. **DETERMINE** Actual Core Flow by SUBTRACTING 85% of Idle Loop Jet Pump Flow FROM Operating loop Jet Pump Flow [FI-R611A(B)-B21 – (0.85 x FI-R611B(A)-B21)]
	___		B. **VERIFY** proper function of the subtraction circuit by checking that calculated core flow (step A6) is the same as Total Jet Pump Flow (FR-R13-B21 OR A190).
	___	A.7	**IMPLEMENT** DL.ZZ-0026 Att. 3v AND IO-6 Requirements for Single Loop operations.
	___		****NOTE 2****
	___	A.8	IF core flow cannot be raised to **>** 40%, THEN **DIRECT** System Engineering to evaluate Single Loop operation IAW SE-PR.ZZ-0003. **[950919568]**

Facility: Hope Creek

Vendor: GE

Exam Date: 2019

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295003 G2.2.39	
	Importance Rating	3.9	

K/A Statement: Knowledge of less than or equal to one hour Technical Specification action statements for systems. Partial or Complete Loss of AC Power / 6

Question: RO #2

Given:

- The plant is at 100% rated power with all systems in a normal lineup.

Then:

- At 1330 the 'A' Emergency Diesel Generator is declared inoperable and tagged out for emergent work.
- At 1400 the surveillance, HC.OP-ST.ZZ-0001 "Power Distribution Lineup-Weekly", was completed satisfactorily to demonstrate the OPERABILITY of the remaining AC sources.
- At 1415 the 500 Kv Bus 10X lockout relay actuates and the 10X Bus is locked out and declared inoperable.

All other plant equipment is OPERABLE.

What is the LATEST required time by Technical Specifications to perform HC.OP-ST.ZZ-0001 "Power Distribution Lineup-Weekly" to demonstrate the OPERABILITY of the remaining AC sources?

A. 1430

B. 1500

C. 1515

D. 2200

Proposed Answer: **C**

Explanation (Optional):

A: **Incorrect-** Based on the initial time of when the EDG went Inoperable, however the ST.ZZ-0001 was completed prior to 1430 and the additional loss of the 10X bus would require one hour from 1415.

B: **Incorrect-** IAW TS 3.8.1.1 (see attached) the surveillance must be completed within 1 hour of the offsite source becoming inoperable. The additional loss of the 10X bus would require one hour from 1415.

C: **Correct-** IAW TS 3.8.1.1 (see attached) the surveillance must be completed within 1 hour of the offsite source becoming inoperable (Bus 10X) given that the EDG is already inoperable.

D: **Incorrect-** IAW TS 3.8.1.1 (see attached) the surveillance must be completed within 1 hour of the offsite source becoming inoperable and at least once per 8 hours thereafter. If the student does not recognize that the surveillance has to be taken due to the additional loss of the 10X bus.

Technical Reference(s): TS 3.8.1.1

A.C. Sources

HC.OP-ST.ZZ-0001 Power Distribution Lineup-Weekly

Proposed References to be provided to applicants during examination: None

Learning Objective: Given specific plant operating conditions which require operator actions within 1 hour From Memory select the correct Technical Specification action(s) for the following: A.C. Sources TS 3.8.1.1

Question Source: Bank #

Modified Bank #

New **X**

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41 (10)

Comments:

3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.1 A.C. SOURCES

A.C. SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.8.1.1 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. Two physically independent circuits between the offsite transmission network and the onsite Class 1E distribution system, and

b. Four separate and independent diesel generators, each with:

   1. A separate fuel oil day tank containing a minimum of 360 gallons of fuel,

   2. A separate fuel storage system consisting of two storage tanks containing a minimum of 44,800 gallons of fuel, and

   3. A separate fuel transfer pump for each storage tank.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

Note: LCO 3.0.4.b is not applicable to DGs.

a. With one offsite circuit of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter. Restore the inoperable offsite circuit to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With one diesel generator of the above required A.C. electrical power sources inoperable,

   1. Demonstrate the OPERABILITY of the above required A.C. offsite sources by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter. If the diesel generator became inoperable due to any cause other than an inoperable support system, an independently testable component, or preplanned preventive maintenance or testing, demonstrate the OPERABILITY of the remaining diesel generators by performing Surveillance Requirement 4.8.1.1.2.a.4 separately for each diesel generator within 24 hours* unless the absence of any potential common mode failure for the remaining diesel generators is demonstrated.

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* This test is required to be completed regardless of when the inoperable diesel generator is restored to OPERABILITY.

ELECTRICAL POWER SYSTEMS

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

2. For the inoperable A or B diesel generator, if continued operation is permitted by LCO 3.7.1.3:

   a) Restore the inoperable diesel generator to OPERABLE status within 72 hours, or

   b) Verify the Salem Unit 3 gas turbine generator (GTG) is available within 72 hours and once per 12 hours thereafter#, and restore the inoperable diesel generator to OPERABLE status within 14 days.

   Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

3. For the inoperable C or D diesel generator, if continued operation is permitted by LCO 3.7.1.3, restore the inoperable diesel generator to OPERABLE status within 14 days, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. With one offsite circuit of the above required A.C. sources and one diesel generator of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter. If a diesel generator became inoperable due to any causes other than an inoperable support system, an independently testable component, or preplanned preventive maintenance or testing, demonstrate the OPERABILITY of the remaining OPERABLE diesel generators separately for each diesel generator by performing Surveillance Requirement 4.8.1.1.2.a.4 within 16 hours unless the absence of any potential common mode failure for the remaining diesel generators is demonstrated*. If continued operation is permitted by LCO 3.7.1.3, restore at least two offsite circuits and all four of the above required diesel generators to OPERABLE status within 72 hours from time of the initial loss or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours. A successful test(s) of diesel generator OPERABILITY per Surveillance Requirement 4.8.1.1.2.a.4 performed under this ACTION statement for the OPERABLE diesel generators satisfies the diesel generator test requirements of ACTION Statement b.

d. With both of the above required offsite circuits inoperable, restore at least one of the above required offsite circuits to OPERABLE status within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours. With only one offsite circuit restored to OPERABLE status, restore at least two offsite circuits to OPERABLE status within 72 hours from time of initial loss or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

---

* This test is required to be completed regardless of when the inoperable diesel generator is restored, to OPERABILITY.

\# After the initial verification period, the GTG may be unavailable for a single period of up to 24-hours and the once-per 12-hour requirement to verify that the GTG is available may be suspended during this period.

Facility: Hope Creek

Vendor: GE

Exam Date: 2019

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295004 AK1.05	
	Importance Rating	3.3	

K/A Statement: Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER: Loss of breaker protection

Question: RO #3

Given:

- The plant is at 100% rated power.

Then:

- A loss of 125 VDC occurs to the normal in-feed breaker for the 7.2 Kv Bus 10A110.

Which describes the effect of this loss, if any?

The breaker ____ (1) ____ trip on a bus lockout. The ability to open and/or close the breaker from the control room ____ (2) ____ remain functional.

A. (1) will NOT ; (2) will NOT

B. (1) will NOT ; (2) will

C. (1) will ; (2) will NOT

D. (1) will ; (2) will

Proposed Answer: **A**

Explanation (Optional): Conditions which cause an INOP 7.2 KV infeed breaker include:

Breaker not in the operate position

Loss of control power

Loss of trip coil continuity

Local breaker hand switch in PULL-TO- LOCK

Breaker closing springs not charged

Breaker fails to close in 2.5 seconds after the CLOSE PB is depressed.

A: **Correct:** The 125 VDC Power System supplies DC power as breaker control power for: 7.2 KV, 4.16 KV, 480 V Unit Substation & 250 VDC breakers

B: **Incorrect:** Breaker protection and control room control function is lost.

C: **Incorrect:** Breaker protection and control room control function is lost.

D: **Incorrect:** The 125 VDC supplies must be swapped manually.

Technical Reference(s): E-0109 Sh. 2 (Attach if not previously provided)

7.2 Kv Bus 10A110 Schematic

Proposed References to be provided to applicants during examination: none

Learning Objective: Summarize/Identify the interrelationship between the Non 1E Power System and each of the following: Non 1E 125VDC System.
Summarize/Identify the source of control power for the Non 1E AC Distribution System. (As available)

Question Source: Bank # 120369

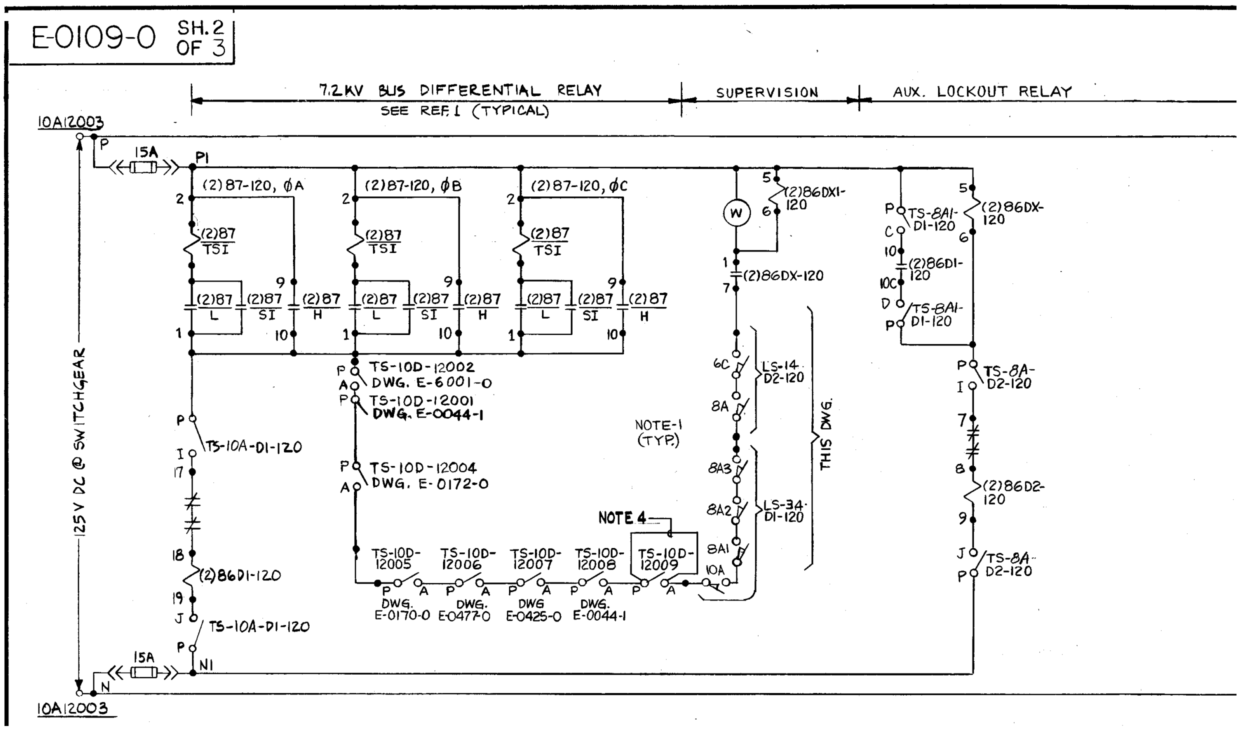
Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41 (7)


E-0109-0 SH. 2 OF 3
7.2KV BUS DIFFERENTIAL RELAY
SEE REF. 1 (TYPICAL)
SUPERVISION
AUX. LOCKOUT RELAY
10A12003
15A
(2)87-120, ØA
(2)87-120, ØB
(2)87-120, ØC
(2)87 TSI
(2)87 L
(2)87 SI
(2)87 H
TS-10D-12002
DWG. E-6001-0
TS-10D-12001
DWG. E-0044-1
TS-10D-12004
DWG. E-0172-0
TS-10A-D1-120
(2)86D1-120
125 V DC @ SWITCHGEAR
NOTE 4
TS-10D-12005
TS-10D-12006
TS-10D-12007
TS-10D-12008
TS-10D-12009
DWG. E-0170-0
DWG. E-0477-0
DWG E-0425-0
DWG. E-0044-1
(2)86DX1-120
(2)86DX-120
LS-14-D2-120
LS-34-D1-120
NOTE-1 (TYP.)
THIS DWG.
TS-8A1-D1-120
(2)86D1-120
TS-8A-D2-120
(2)86D2-120

Facility: Hope Creek

Vendor: GE

Exam Date: 2019

Exam Type: RO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	295005 AK2.05	
Importance Rating	2.6	

K/A Statement: Knowledge of the interrelations between MAIN TURBINE GENERATOR TRIP and the following: Extraction steam system

Question: RO #4

Given:

- The plant is operating at 100% rated power.

When:

- A Main Turbine trip occurs.

How will the Extraction Steam System be affected?

The Air Relay Dump Valves (ARDVs) will reposition to isolate _______________.

A. the Feedwater Heater Extraction Steam Isolation Valve to prevent water induction into the main turbine.

B. the Feedwater Heater Bleeder Trip Valve (BTV) to prevent a main turbine overspeed.

C. the Feedwater Heater Extraction Steam Isolation Valve to prevent a main turbine overspeed.

D. the Feedwater Heater Bleeder Trip Valve (BTV) to prevent water induction into the main turbine.

Proposed Answer: B

Explanation (Optional): The ARDVs reposition to allow the Bleeder trip valves (AOVs) to isolate to prevent any steam or energy back to the main turbine to prevent an overspeed condition. The Extraction Steam Isolation valves isolate on HI HI level in the associated heater to protect from water induction in the main turbine. However, the extraction steam isolations are MOVs and are not associated with the ARDVs.

A: **Incorrect-**.The Feedwater heater extraction steam isolation valves are in line with the bleeder trip valves, however they are MOVs and do not close on ARDVs repositioning. The extraction Steam isolation valves isolate on a HI HI level in the associated heater to protect the main turbine from water induction.

B: **Correct-** See above explanation. This is the purpose and function of the Bleeder Trip Valves.

C: **Incorrect-** Purpose of the bleeder trip valves.

D: **Incorrect-** Prevent flashing in the heater which prevents the steam from flowing into the main turbine causing an overspeed condition.

Technical Reference(s):	HC.OP-SO.AF-0001	(Attach if not previously provided)
	Extraction Steam Operation	
	M-02-1 FWH P&ID	
	NOH04FWHEATC Feedwater Heater Extraction LP	

Proposed References to be provided to applicants during examination: none

Learning Objective:	From memory, summarize the operation of the bleeder trip valves following a turbine trip.	(As available)

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

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41 (7)

Comments:

The Extraction Steam System utilizes motor operated isolation valves and bleeder trip valves to protect the main turbine.

1. The motor operated isolation valves automatically close on "Hi-Hi" water level in their associated heater, **protecting the turbine from water induction**.

2. The bleeder trip valves automatically close on a turbine trip to prevent condensate in the heater shell from **flashing to steam and overspeeding the turbine.**

When a turbine trip occurs, the air relay dump valves (XV-2100, 2101) reposition due to a loss of ETS header pressure. This causes the following actions to occur:

XV-2100

Bleeder trip valves for FWHTR's 3 (XV-1374 A, B, C) and 4 (XV-1375 A, B, C) fail closed.

Bleeder trip valves for the SSE (XV-2012 A, B) fail closed.

XV-2101

Bleeder trip valves for FWHTR 6 (XV-1369 A, B, C) fail closed.

All extraction line drain valves for FWHTR's 3,4,5 and 6 (A, B, C) fail open.

HC.OP-SO.AF-0001(Z)

3.3 **Interlocks**

3.3.1. Feedwater Heater Extraction Steam Header Drain AOV will open when the associated Feedwater Heater Shell Side Extraction Steam Isolation MOV is closed. ___

3.3.2. LV-1506A (B, C) #6 Feedwater Heater Drain Vlv, will fail closed if EITHER HV-1613A (B, C) FWH Inlet Valve, OR HV-1600A (B, C) FWH Outlet Valve, for the #3, #4 and #5 FWH are NOT 100% open. ___

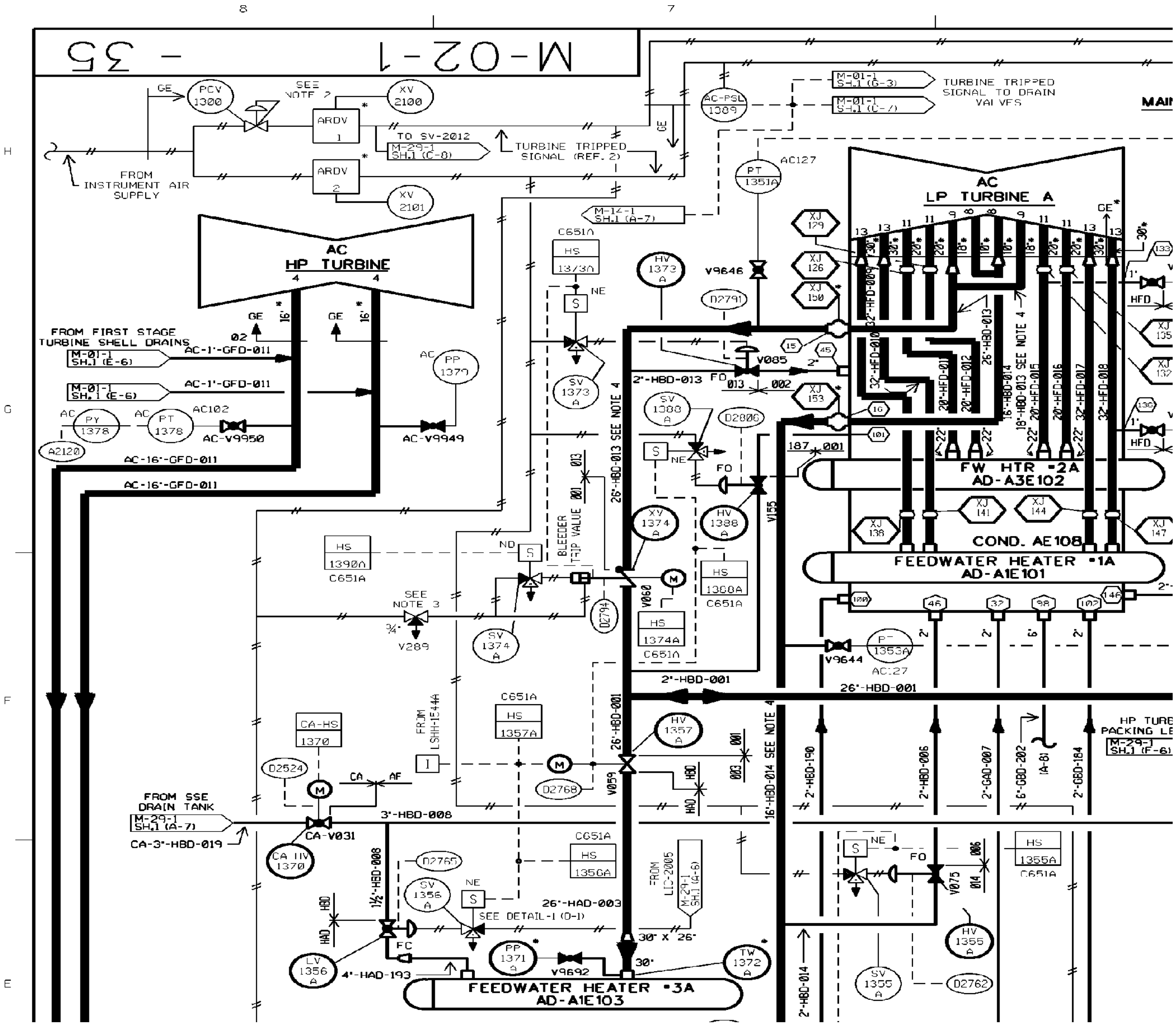
3.3.3. Feedwater Heater Bleeder Trip Valves will trip close on any of the following conditions:

- Hi Hi Level (29 inches) in the associated Feedwater Heater (#6 FWH only) ___
- Associated Feedwater Heater Shell Side Extr Stm Isln MOV is < 100% open (#6 FWH only) ___
- Main Turbine trip ___

HC.OP-SO.AF-0001(Z)

3.3.5. Feedwater Heater Hi Hi Level (29" on #2, 3, 4, 5, 6) will cause the following:

- Feedwater Heater Drain Valve on the next highest pressure heater closes. (LV-1532A, B, OR C) FWH Drain to #2 FWH closes after a 10 second time delay. ___
- Feedwater Heater Extr Stm Isln MOV closes. ___
- #1 OR #2 (29") Feedwater Heaters Hi Hi Level, after a 10 second time delay, will close the associated Feedwater Heater Inlet AND Outlet valves for the Drain Cooler, #1 AND #2 Feedwater Heaters (HV-1633 A, B, OR C AND HV-1620 A, B, OR C). ___
- Steam Seal Evaporator Drain Tank to #3 Feedwater Heater (LV-1356 A, B, OR C) will fail closed on the associated #3 Feedwater Heater Hi Hi Level (29 inches). ___


M-02-1 - 35
AC HP TURBINE
AC LP TURBINE A
FW HTR #2A AD-A3E102
COND. AE108
FEEDWATER HEATER #1A AD-A1E101
FEEDWATER HEATER #3A AD-A1E103
FROM INSTRUMENT AIR SUPPLY
FROM FIRST STAGE TURBINE SHELL DRAINS
TURBINE TRIPPED SIGNAL (REF. 2)
TURBINE TRIPPED SIGNAL TO DRAIN VALVES
FROM SSE DRAIN TANK
BLEEDER TRIP VALVE
26"-HBD-013 SEE NOTE 4
26"-HBD-001
2"-HBD-001
CA-V031
V059
V060
V085
V155
V075
V9644
V9692
V289
AC-V9950
AC-V9949
AC-16"-GFD-011
AC-1"-GFD-011
CA-3"-HBD-019
3"-HBD-008
26"-HAD-003
4"-HAD-193

Facility: Hope Creek

Vendor: GE

Exam Date: 2019

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295006	AK3.04
	Importance Rating	3.1	

K/A Statement: Knowledge of the reasons for the following responses as they apply to SCRAM: Reactor water level setpoint setdown: Plant-Specific

Question: RO #5

Following a reactor scram with reactor water level below +12.5 inches, the Setpoint Setdown Logic will ___________.

A. lower the DFCS Startup Level Controller setpoint to prevent vessel overfeed.

B. lower total feed flow signal, to match actual steam flow to the turbine, which will now be 'zero'.

C. lower the DFCS Master Level Controller setpoint to prevent a vessel overfeed.

D. lower the total steam flow signal so feed flow will vary due to any deviation between actual and desired level only.

Proposed Answer: C

Explanation (Optional): Setpoint Setdown - The setpoint setdown circuit at HCGS provides a means to help prevent **excessive inventory make-up** in Automatic Level control following a Reactor Scram. Activated by Median Narrow range level detector sensing +12.5" (level 3) with a 0.5 second time delay. Designed to electrically lower the Auto Level setpoint on the Master Controller to +14"

A: **Incorrect-** The S/U Controller is normally in manual with power turned off.

B: **Incorrect-**. The level input remains so water level will be controlled automatically by DFCS.

C: **Correct-** Automatically lower the DFCS Master Level Controller setpoint to prevent a vessel overfeed - The Master Controller Setpoint is electrically lowered to +14".

D: **Incorrect-**. DFCS automatically shifts to single element when Steam Flow <30% but the setpoint is changed to +14" not normal level.

Technical Reference(s): H-1-AE-ECS-0128 Sheets 3A-3Z (Attach if not previously provided)

DFCS Functional and Logic Diagram

Proposed References to be provided to applicants during examination: none

Learning Objective: From memory, state the purpose of the setpoint setdown unit and describe how it accomplishes its purpose. (As available)

Question Source: Bank # 72532

Modified Bank # (Note changes or attach parent)

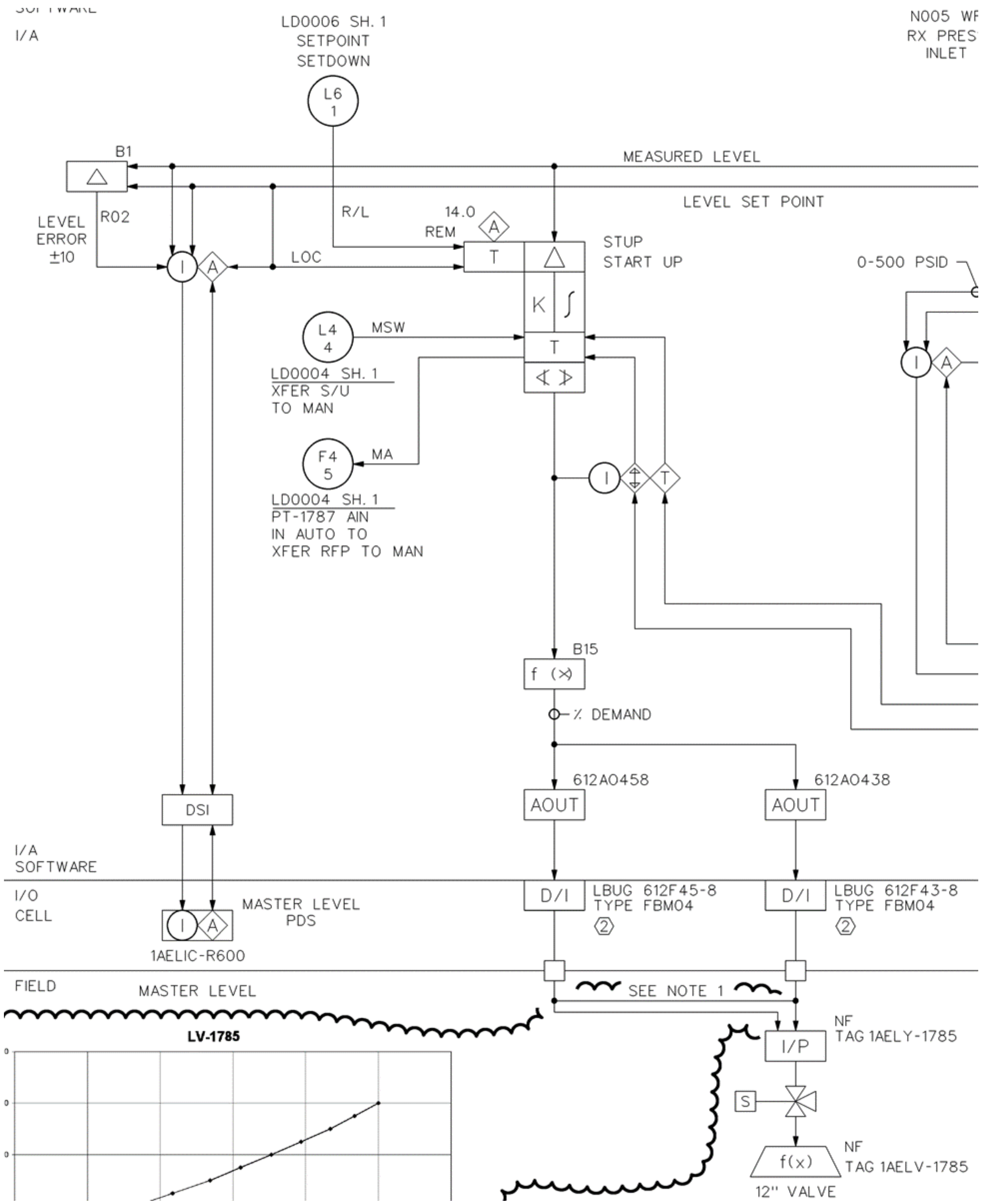
New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41 (7)

Comments:



I/A
LD0006 SH. 1
SETPOINT
SETDOWN
L6
1
N005 WF
RX PRES
INLET
B1
MEASURED LEVEL
LEVEL SET POINT
LEVEL
ERROR
±10
R02
R/L
14.0
REM
A
T
STUP
START UP
I
LOC
0-500 PSID
K
L4
4
MSW
LD0004 SH. 1
XFER S/U
TO MAN
F4
5
MA
LD0004 SH. 1
PT-1787 AIN
IN AUTO TO
XFER RFP TO MAN
B15
f (x)
% DEMAND
612A0458
612A0438
DSI
AOUT
I/A
SOFTWARE
I/O
CELL
D/I
LBUG 612F45-8
TYPE FBM04
LBUG 612F43-8
TYPE FBM04
2
MASTER LEVEL
PDS
1AELIC-R600
FIELD
MASTER LEVEL
SEE NOTE 1
LV-1785
NF
TAG 1AELY-1785
I/P
S
NF
TAG 1AELV-1785
12" VALVE

Facility: Hope Creek

Vendor: GE

Exam Date: 2019

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295016 AA1.05	
	Importance Rating	2.8	

K/A Statement: Ability to operate and/or monitor the following as they apply to CONTROL ROOM ABANDONMENT: D.C. electrical distribution

Question: RO #6

Given:

- Smoke in the control room has caused the control room to be evacuated.
- All actions of HC.OP-AB.HVAC-0002, CONTROL ROOM ENVIRONMENT have been performed.
- RPV level is stable with RCIC injection.
- HPCI is no longer required for injection.

IAW HC.OP-AB.HVAC-0002, HPCI would be shutdown by opening the circuit breaker for HPCI RELAY VERT BD 10C620 at which one of the following 1E D.C. electrical distribution sources?

A. 10D251

B. 1AD417

C. 10D261

D. 1CD417

Proposed Answer: **B**

Explanation (Optional): IAW HC.OP-AB.HVAC-0002 subsequent operator action C.12 (see attached) When HPCI is no longer required, then HPCI is shutdown IAW Attachment 2 (see attached).

A: Incorrect- 10D251 is the 1E 250 VDC electrical distribution for HPCI. The 10C620 panel is not powered from 10D251.

B: Correct- (see attached)

C: Incorrect- 10D261 is the 1E 250VDC electrical distribution for RCIC.

D: Incorrect- CD417 is the 1E 125 VDC for the “C” 1E channel. HPCI is an “A” 1E 125 VDC channel.

Technical Reference(s): HC.OP-AB.HVAC-0002 (Attach if not previously provided)

Control Room Environment

Proposed References to be provided to applicants during examination: none

Learning Objective: Analyze plant conditions and parameters to determine if plant operation is in accordance with the SHUTDOWN FROM OUTSIDE THE CONTROL ROOM Integrated Operating Procedure. (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41 (10)

Comments:

SUBSEQUENT OPERATOR ACTIONS (continued)

CONDITION	ACTION
	Continued from Page 11
C Evacuation of the Control Room is required.	___ C.11 IF emergency takeover is successful, **NOTIFY** the Shift Manager that control has been established from the Remote Shutdown Panel.
	___ ★ **CAUTION 5** ★
	___ C.12 WHEN HPCI is no longer required, **SHUTDOWN** HPCI IAW Attachment 2.
	___ C.13 IF RCIC operation is required, THEN **VERIFY** RCIC System auto initiation OR **PLACE** RCIC in service as follows:
	___ a. **ENSURE** RCIC Flow Controller is set to 600 gpm AND AUTO is ON.
	___ b. **START** 0P219.
	___ c. **OPEN** HV-F046.
	___ d. **ENSURE** FC-HV-4282 is OPEN.
	___ ★ **CAUTION 6** ★
	___ ★ **CAUTION 7** ★
	e. SIMULTANEOUSLY **PERFORM** the following:
	___ • **PRESS** AND **HOLD** HV-F045 OPEN PB UNTIL OPEN.
	___ • **OPEN** HV-F013.
	f. **ENSURE** the following:
	___ • HV-F025 is CLOSED
	___ • HV-F004 is CLOSED
	___ • HV-F022 is CLOSED
	___ • RCIC Turbine Speed of 2150 - 4500 rpm

Continued on Page 15

ATTACHMENT 2
HPCI SHUTDOWN

1.0 HPCI SHUTDOWN

NOTE

IF HPCI is in-service, opening the following circuit breaker will defeat all automatic and process trips. HPCI will NOT trip from the 10C620 panel. The Aux oil pump will not start when HPCI coasts down which could result in bearing damage.

Opening the 1AD417-10 breaker will shutdown HPCI by closing the Turbine Stop Valve, Governor Valve and HPCI Pump Discharge Valve. HPCI should be secured prior to opening this breaker.

IF HPCI is NOT in-service, opening the following circuit breaker will prevent the Aux oil pump from starting and HPCI will remain out of service. ___

1.1 IF the Main Control Room has been evacuated due to a FIRE event AND HPCI is in service, THEN **PERFORM** Step 1.3 only. ___

1.2 IF HPCI is in service OR as directed by the CRS, **DIRECT** I&C to perform the following:

1.2.1. **CONNECT** an ECCS Logic Tester to test jack E41A-J2, RPV HI WATER LVL SIGNAL SEAL-IN (PNL H11-P620). ___

1.2.2. **PLACE** A-B switch/jumper of ECCS Logic Tester connected to E41A-J2 in TEST. (PNL H11-P620) ___

1.2.3. **PLACE** B-C switch/jumper of ECCS Logic Tester connected to E41A-J2 in TEST. (PNL H11-P620) ___

1.3 **OPEN** the following circuit breaker at Class 1E 125VDC Distribution Panel 1AD417 (El 130', 'A' 1E Switchgear Room): **[CD-012Z]**

1.3.1. Circuit Breaker 10, HPCI RELAY VERT BD 10C620. ___

1.4 **DISCONNECT** the ECCS Logic Tester from Test jack E41A-J2, RPV HI WATER LVL SIGNAL SEAL-IN. (PNL H11-P620) ___

Facility: Hope Creek

Vendor: GE

Exam Date: 2019

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295018 AA2.03	
	Importance Rating	3.2	

K/A Statement: Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER: Cause for partial or complete loss

Question: RO #7

Given:

- The plant is operating at 100% rated power.
- “A” and “B” Reactor Auxiliary Cooling System (RACS) pumps are in-service.

When:

- The RACS head tank begins to lower rapidly.

Which of the following identifies the potential cause of the low level in the RACS head tank?

A. Tube rupture on the in-service RACS Heat Exchanger.

B. Tube rupture in the Reactor Water Clean-Up (RWCU) Regenerative Heat Exchanger.

C. Broken tube inside the Reactor Recirculation Pump Seal Cooler Heat Exchanger.

D. Tube rupture in the Reactor Water Clean-Up (RWCU) Non-Regenerative Heat Exchanger.

Proposed Answer: **A**

Explanation (Optional): Any leakage in the RACS Hx will be from higher pressure RACS (46 psig) to lower pressure SSW (25.4 psig). The indication of such a leak **would be a lowering expansion tank level**. Rising expansion tank level could be caused by Non-regenerative Hxs for the RWCU System or Seal Cooler Hxs on the Reactor Recirculation Pumps.

A: Correct. See above.

B: Incorrect. RACS does not supply the NRHX just the RHX.

C: Incorrect. .RACS system pressure is the lower pressure in the Recirc pump seal cooler heat exchanger.

D: Incorrect. RACS system pressure is the lower pressure in the RWCU NRHX.
.

Technical Reference(s): NOH01RACS00C (Attach if not previously provided)

Reactor Auxiliary Cooling System

HC.OP-AB.COOL-0003
RACS

Proposed References to be provided to applicants during examination: none

Learning Objective: Assess the interrelationship between RACS and any of the following loads: RWCU and Reactor Recirc (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41 (7)

Comments:

SUBSEQUENT OPERATOR ACTIONS (continued)

CONDITION	ACTION
E. Suspected in-leakage from another system. **[ODCM 3/4.11.1.1]** Date/Time: ______________	___ E.1 **DIRECT** Chemistry to sample the following to identify the source of in leakage, the contaminants and if a release is in progress: ● RACS. ● Service Water. E.2 **DETERMINE** if in leakage is from RWCU as follows: ___ a. **CHECK** RWCU differential flows at NUMAC's. ___ b. **CHECK** RACS CRIDS display RMS for rising radiation levels.

Facility: Hope Creek

Vendor: GE

Exam Date: 2019

Exam Type: RO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	295018 G2.4.50	
Importance Rating	4.2	

K/A Statement: Ability to verify system alarm setpoints and operate controls identified in the alarm response manual. Partial or Complete Loss of Instrument Air.

Question: RO #8

Given:

- The plant is operating at 70% rated power.

When:

- INSTRUMENT AIR HEADER A PRESSURE LO annunciator alarms.
- INSTRUMENT AIR HEADER B PRESSURE LO annunciator alarms.
- Main Condenser Vacuum is at 4.0 inches HgA and slowly degrading.
- 1-KAHV-7595 Service Air Supply Header Isolation Valve isolates.
- Instrument Air header pressure is still lowering.
- The PO reports that the air dryers have malfunctioned.
- Abnormal procedure AB.COMP-0001, Instrument and/or Service Air has been entered.
- Abnormal procedure AB.BOP-0006, Main Condenser Vacuum has been entered.

What is the current Instrument Air header pressure status and what action(s) is (are) required?

Instrument Air header pressure is ______ and _______________.

A. $\leq$ 70 psig; reduce Recirc pump speed to minimum, lock the mode switch in Shutdown.

B. $\leq$ 70 psig; reduce Reactor power, a reactor scram is NOT required.

C. > 70 psig; reduce Recirc pump speed to minimum, lock the mode switch in Shutdown.

D. > 70 psig; reduce Reactor power, a reactor scram is NOT required.

Proposed Answer: **A**

Explanation (Optional): HC.OP-AB.COMP-0001 Retainment Override (see attached)

A: **Correct-** The stem states that 1-KAHV-7595 Service Air Supply Header Isolation Valve is closed. The valve closes at **70 psig** instrument air header pressure. With the header pressure continuing to lower, IAW AB.COMP-0001 retainment override, Recirc to min and the MS to shutdown (see attached).

B: **Incorrect-** Lowering power would be IAW AB.BOP-0006, however the OHA for MAIN CONDENSER VACUUM LO is NOT currently in (see attached). The crew would still enter AB.BOP-0006 due to the degraded vacuum condition. The loss of instrument air would cause the degraded vacuum condition. The actions of AB.COMP-0001 would have to be taken (see attached). The multiple control rod drifts would require locking the M.S. in shutdown IAW AB.IC-0001, Control Rod abnormal.

C: **Incorrect-** 1-KAHV-7595 Service Air Supply Header Isolation Valve closes at **70 psig** instrument air header pressure. The pressure is continuing to lower.

D: **Incorrect-** 1-KAHV-7595 Service Air Supply Header Isolation Valve closes at **70 psig** instrument air header pressure. The pressure is continuing to lower.

Technical Reference(s): HC.OP-AB.COMP-0001 (Attach if not previously provided)

Instrument and/or Service Air

HC.OP-AB.BOP-0006
Main Condenser Vacuum

HC.OP-AB.IC-0001
Control Rod

Proposed References to be provided to applicants during examination: none

Learning Objective: Given plant conditions and plant procedures, determine required actions of the retainment override(s) and subsequent operator actions in accordance with Instrument and/or Service Air. (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New **X**

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41 (10)

Comments:

RETAINMENT OVERIDE	
CONDITION	**ACTION**
I. 1-KA-HV-7595 is CLOSED AND Instrument Air Header Pressure ≤ 70 psig. (PI-7603A(B) or A3016) AND Instrument Air Header Pressure continues to Lower Date/Time: ____________	___ **I.a** **REDUCE** Recirc. Pump speed to MINIMUM ___ **I.b** **LOCK** the Mode Switch in SHUTDOWN.

ADDITIONAL INFORMATION:

Valve Descriptions:

- 1-KBHV-7620 Instrument Air Dryer 00-F-104 Isolation Valve
- 1-KBHV-7619 Instrument Air Dryer 10-F-104 Isolation Valve
- 1-KBHV-11416 Instrument Air Dryer 1A-F-104 Isolation Valve
- 1-KAHV-7595 Service Air Supply Header Isolation Valve

Procedures:

- HC.OP-AB.ZZ-0001(Q), Transient Plant Conditions

Indications:

- PI-7603A, A Instrument Air Header Pressure
- PI-7603B, B Instrument Air Header Pressure
- A3016, CRD Rod Pilot Air Header Pressure

HC.OP-AB.COMP-0001(Q)
INSTRUMENT AND/OR SERVICE AIR

IMMEDIATE OPERATOR ACTIONS

CONDITION	ACTION
Trip of the Inservice Service Air Compressor Date/Time: _______________	**PLACE** the out of service Service Air Compressor in service. (AB.ZZ-0001)

AUTOMATIC ACTIONS

IF	THEN
< 92 psig Service Air Pressure.	Standby Service Air Compressor Auto Start
≤ 85 psig Emergency Instrument Air Receiver Pressure	• EIAC Auto Start • RACS Demineralizers Isolate
≤ 85 psig Instrument Air Pressure <u>OR</u> Loss of Power to 00-F-104	1-KBHV-7620 OPENS
≤ 85 psig Instrument Air Pressure <u>OR</u> Loss of Power to 10-F-104	1-KBHV-7619 OPENS
≤ 85 psig Instrument Air Pressure <u>OR</u> Loss of Power to 1A-F-104	1-KBHV-11416 OPENS
≤ 70 psig Instrument Air Pressure	1-KAHV-7595 AUTO CLOSES

PSEG Internal Use Only

HC.OP-AB.BOP-0006(Q)
MAIN CONDENSER VACUUM

IMMEDIATE OPERATOR ACTIONS

CONDITION	ACTION
Degraded Main Condenser Vacuum Date/Time: ______________	___ **REDUCE** Reactor Power to maintain MAIN CONDENSER A(B,C) VACUUM LO Overhead Alarm Clear.

HC.OP-AB.IC-0001(Q)
CONTROL ROD

IMMEDIATE OPERATOR ACTIONS

CONDITION	ACTION
Multiple Control Rods are Drifting Simultaneously. (Drifting Does NOT include Scrammed or Scramming Rods) Date/Time: ______________	___ **LOCK** the Mode Switch in SHUTDOWN. **[CM-HC-1982-108]**

Facility: Hope Creek

Vendor: GE

Exam Date: 2019

Exam Type: RO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	295021 AK1.02	
Importance Rating	3.3	

K/A Statement: Knowledge of the operational implications of the following concepts as they apply to LOSS OF SHUTDOWN COOLING: Thermal stratification

Question: RO #9

Given:

- The plant is in Operational Condition 4, preparing for plant startup.
- "B" RHR Loop is in Shutdown Cooling IAW HC.OP-SO.BC-0002, "Decay Heat Removal Operation".

Then:

- The Reactor Operator inadvertently opens the Reactor Recirc Pump 'B' Discharge valve, BB-HV-F031B.

Which of the following is the effect of opening the BB-HV-F031B?

A. RPV level constant and Head Vent temperature rising.

B. RPV level lowering and Head Vent temperature constant.

C. RPV level rising and Head Vent temperature rising.

D. RPV level rising and Head Vent temperature constant.

Proposed Answer: **C**

Explanation (Optional): When the RHR System is operating in the Shutdown Cooling Mode, Rx Recirc Loop suction and/or discharge valves for any non-running pumps must be closed to prevent bypassing the reactor core. **Establishing a bypass flow path could result in coolant temperature stratification** and/or inadvertent heat up and pressurization.

A: Incorrect- HV-F031B opening would significantly bypass the SDC flow to the jet pumps and reactor core. Heating up the vessel inventory and cause level to rise due to expansion.

B: Incorrect- Bypass flow path could result in coolant temperature stratification and/or inadvertent heat up and pressurization. The student could think that the vessel is being drained (valve lineup to drain the vessel to the torus).

C: Correct- HV-F031B opening would significantly bypass the SDC flow to the jet pumps and reactor core. Heating up the vessel inventory and cause level to rise due to expansion.

D: Incorrect-. Bypass flow path could result in coolant temperature stratification and/or inadvertent heat up and pressurization.

Technical Reference(s): HC.OP-SO.BC-0002 (Attach if not previously provided)

Decay Heat Removal Operation

HC.OP-AB.RPV-0009
Shutdown Cooling

Proposed References to be provided to applicants during examination: none

Learning Objective: Given procedure HC.OP-SO.BC-0001, "Residual Heat Removal System Operation" and HC.OP-SO.BC-0002, "Decay Heat Removal Operation", explain the listed prerequisites, precautions, and/or limitations during operation (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New **X**

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41 (10)

Comments:

3.2 Limitations

3.2.1. Independent Verification may be performed following completion of the procedure activity or upon completion of each step. If performance of an "**ENSURE**" step which has an IV signoff results in NO component manipulation then the IV signoff should be N/A'd upon completion of the step. ___

3.2.2. If valve or electrical lineup CANNOT be completed as required, then, the SM/CRS will determine whether the deviations are such that the system should NOT be placed in service or standby, as required. ___

3.2.3. The Reactor Coolant System cooldown limitations of T/S 3/4.4.6 shall be observed. ___

3.2.4. While the RHR System is operating in the Shutdown Cooling mode, any valve manipulations which would prevent ANY of the rated shutdown cooling flow (≈ 10,000 gpm) from returning to the reactor vessel via the respective Recirculation System discharge piping and jet pumps should NOT be performed. For example, recirc suction and discharge valves being open simultaneously in the loop seeing Shutdown Cooling return flow would result in a portion of the return flow being diverted back through the recirculation loop in the reverse direction, as opposed to into the respective jet pumps, where forced circulation through the core would occur. This limitation does NOT preclude reducing shutdown cooling flow to satisfy Limitations 3.2.21, 3.2.22 or 3.2.24 provided the Cavity is flooded, reducing reactor cavity level using Steps 5.2.43 or 5.2.44, or when intentionally reducing Shutdown Cooling flow to support Noble Metals Chemical Application. **[CD-122H]** ___

3.2.5. While the RHR System is operating in the shutdown cooling mode, maintaining the rated shutdown cooling flow of approximately 10,000 GPM to the Reactor vessel via the respective Recirculation System discharge piping and Jet Pumps is essential to assure that the RHR Heat Exchanger inlet temperature is representative of actual bulk coolant temperature. **[CD-122H]** ___

3.2.6. During the transition from Shutdown Cooling operations to establishment of normal Reactor Recirculation System operations, the RHR System may be left in Shutdown Cooling, ONLY if BP202 is the RHR Pump in service AND AP201 is the Reactor Recirculation Pump to be started. ___

CATEGORY II

SHUTDOWN COOLING

ALARMS

• RHR LOGIC A OUT OF SERVICE	**A6 – A1**
• RHR LOGIC B OUT OF SERVICE	**A7 – A1**
• RHR LOOP A TROUBLE	**A6 – B1**
• RHR LOOP B TROUBLE	**A7 – B1**
• RHR HX CLG WTR OUTLET TEMP HI	**A6 – D5**
• COMPUTER PT IN ALARM	**A4 – F5**
• APRM/RBM FLOW REF OFF NORMAL	**C6 – D1**

INDICATIONS

- Trip of RHR pump in Shutdown Cooling
- Isolation of Shutdown Cooling Flowpath
- Reduced or stopped RHR Shutdown Cooling flow to the Jet Pumps
- Lowering Core flow
- Rising Reactor coolant temperature/pressure
- Rising Recirc pump loop flow

Facility: Hope Creek

Vendor: GE

Exam Date: 2019

Exam Type: RO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	295023 AK2.03	
Importance Rating	3.4	

K/A Statement: Knowledge of the interrelations between REFUELING ACCIDENTS and the following: Radiation monitoring equipment.

Question: RO #10

Given:

- The plant is in OPCON 1.
- Irradiated fuel is being shuffled in the Spent Fuel Pool in preparation for new fuel arrival.

Then:

- The following VALID alarms are received in the Control Room:

- ➢ RADIATION MONITORING ALARM/TRBL
- ➢ NEW FUEL CRITICALITY RAD HI
- ➢ REFUEL FLR EXH RAD ALARM/TRBL
- ➢ RB EXH RADIATION ALARM/TRBL

Which of the following is the initial control room operator action for the conditions above IAW HC.OP-AB.CONT-0005, IRRADIATED FUEL DAMAGE?

A. Suspend the handling of irradiated Fuel.

B. Ensure Primary Containment Instrument Gas System isolates.

C. Verify the Drywell Integrity Airlock surveillance test is current.

D. Ensure the start of the ‘A’ AND ‘C’ SACS Pumps if not already running.

Proposed Answer: **A**

Explanation (Optional): Knowledge of the Refuel Floor and Reactor Building Exhaust radiation monitors and the affected equipment IAW HC.OP-AB.CONT-0005 Irradiated Fuel Damage (see attached).

A: **Correct -** Hi alarms on Rx Bldg and RF Floor indicate an isolation signal and IAW Cont-0005 I.O.A suspension of fuel handling.

B: **Incorrect** Hi alarms on Rx Bldg and RF Floor indicate an isolation signal and the compressor would isolate, however the first action would be to suspend fuel handling.

C: **Incorrect -** the concern is secondary NOT primary containment

D: **Incorrect -** The A and B SACS pumps only will start if not running (see attached).

Technical Reference(s): HC.OP-AB.CONT-0005 Irradiated Fuel Damage (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: Explain the reasons for how plant/system parameters respond when implementing Irradiated Fuel Damage. (As available)

Question Source: Bank # 122640

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41 (7)

Comments:

CATEGORY II

IRRADIATED FUEL DAMAGE

ALARMS

• REFUEL FLR EXH RAD ALARM/TRBL	**E6-A3**
• RB EXH RADIATION ALARM/TRBL	**E6-A5**
• NEW FUEL CRITICALITY RAD HI	**E6-A4**
• RADIATION MONITORING ALARM/TRBL	**C6-C1**

INDICATIONS

- Rising Refuel Floor Exhaust Radiation Levels.
- Rising Refuel Floor Area Radiation Levels.

TERMINATED Date/Time: ______________________

IMMEDIATE OPERATOR ACTIONS

CONDITION	ACTION
Irradiated fuel damage Date/Time: ____________	___ **SUSPEND** the handling of Irradiated Fuel/Components.

AUTOMATIC ACTIONS

IF	THEN
Reactor Building Exh (RBE) Hi Rad (1.0×10^{-3} μCi/cc)	• Reactor Building Ventilation System isolates and fans trip • Filtration, Recirculation and Ventilation System (FRVS) automatic start • Safety Auxiliary Cooling pump automatic start (A&B only) • Station Service Water pumps automatic start • Radwaste Ventilation Supply and Exhaust fans trip • Emergency Instrument Air Compressor trips • Primary Containment Instrument Gas System isolates and compressors trip
Refuel Floor Exh (RFE) Hi Rad (2.0×10^{-3} μCi/cc)	• Reactor Building Ventilation System isolates and fans trip • Filtration, Recirculation and Ventilation System (FRVS) automatic start • Safety Auxiliary Cooling pump automatic start (A&B only) • Station Service Water pumps automatic start • Radwaste Ventilation Supply and Exhaust fans trip • Emergency Instrument Air Compressor trips

Facility:	Hope Creek
Vendor:	GE
Exam Date:	2019
Exam Type:	RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295024 EK3.02	
	Importance Rating	3.5	

K/A Statement: Knowledge of the reasons for the following responses as they apply to HIGH DRYWELL PRESSURE: Suppression pool spray operation: Plant-Specific

Question: RO #11

The reason(s) for initiating Suppression Chamber sprays under high drywell pressure conditions is (are).

1. To prevent chugging of the downcomer openings.
2. To condense steam in the suppression chamber airspace.
3. To prevent the cycling of the Torus to Drywell Vacuum Breakers.

A. 1 and 3 ONLY.

B. 2 ONLY.

C. 1 ONLY.

D. 1, 2, and 3.

Proposed Answer: **B**

Explanation (Optional): See attached EOP-102 BASES
Although spraying the suppression chamber will not prevent chugging, it can reduce primary containment pressure by condensing any steam that may be present in the suppression chamber airspace and by absorbing heat energy from the enclosed atmosphere through the processes of evaporative and convective cooling.

A: Incorrect – Drywell sprays are initiated to preclude the possibility of chugging—the cyclic condensation of steam at the downcomer openings of the drywell vents. The suppression chamber sprays effect the suppression chamber airspace not the downcomers which communicate to the water side of the suppression pool. The Torus to Drywell vacuum breaker would not be effected by the suppression chamber sprays, however the Reactor Bldg. to Torus vacuum breakers could be effected (understanding how the vacuum breakers communicate between Rx Bldg. to Torus to Drywell).

B: **Correct-** Suppression chamber sprays reduce primary containment pressure by condensing any steam that may be present in the suppression chamber airspace.

C: Incorrect - The suppression chamber sprays effect the suppression chamber airspace not the downcomers which communicate to the water side of the suppression pool.

D: Incorrect - The suppression chamber sprays effect the suppression chamber airspace not the downcomers which communicate to the water side of the suppression pool. The Torus to Drywell vacuum breaker would not be effected by the suppression chamber sprays, however the Reactor Bldg. to Torus vacuum breakers could be effected (understanding how the vacuum breakers communicate between Rx Bldg. to Torus to Drywell).

Technical Reference(s): HC.OP-EO.ZZ-0102-BASES (Attach if not previously provided)

Primary Containment Control Bases

Proposed References to be provided to applicants during examination: none

Learning Objective: Given any step of the procedure, determine the reason for performance of that step and/or predict expected system response to control manipulations prescribed by that step. (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New **X**

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41 (10)

Comments:

7.4 DW/P-4 Before Supp Chamber Press exceeds 9.5 psig

DW/P-5 Is Supp Pool Level below 180 in

DW/P-6 Initiate Supp Chamber Sprays, use only those RHR pumps not required to assure adequate core cooling by continuous operation in the LPCI mode

Suppression chamber spray is the initial mitigation strategy employed by EOP-102 in preference to drywell spray as it does not affect electrical components in the drywell and it can be used prior to reaching 9.5 psig, which is the Suppression Chamber Spray Initiation Pressure (SCSIP). Drywell sprays cannot be initiated until suppression chamber pressure exceeds SCSIP to preclude the possibility of chugging—the cyclic condensation of steam at the downcomer openings of the drywell vents. When a steam bubble collapses at the exit of the downcomers, the rush of water drawn into the downcomers to fill the void induces stresses at the junction of the downcomers and the vent header. Repeated application of such stresses could cause fatigue failure of these joints, thereby creating a direct path between the drywell and suppression chamber. Steam discharged through the downcomers could then bypass the suppression pool and directly pressurize the primary containment

Although spraying the suppression chamber will not prevent chugging, it can reduce primary containment pressure by condensing any steam that may be present in the suppression chamber airspace and by absorbing heat energy from the enclosed atmosphere through the processes of evaporative and convective cooling. In evaporative cooling the water spray undergoes a change of state from liquid to vapor, whereas convective cooling involves no change of state. For suppression pool sprays, convective cooling is the expected significant cooling mechanism, as it 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 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. Conversely, evaporative cooling is not likely to be a significant factor as the atmosphere in the suppression pool is considered to be at a saturated state for the given pressure.

If steam is bypassing the suppression pool and entering the suppression chamber directly, initiation of suppression pool sprays may obviate the need for drywell sprays. If the pressure increase is due to the transfer of non-condensible gases from the drywell to the suppression chamber, however, suppression chamber pressure will be relatively unaffected by operation of suppression pool sprays and use of drywell sprays may be required.

Initiation of suppression pool spray is permitted only if suppression pool water level is below the elevation of the suppression pool spray nozzles. At Hope Creek, the value is conservatively set at 180 inches, the maximum indicated value for suppression pool level. If the suppression pool spray nozzles are submerged, no spray action will occur and therefore there is no benefit in initiating the system. As a result, if step DW/P- 5 is answered in the negative, direction to spray the suppression pool is not provided and the operator is directed to continue onward in the procedure.

Suppression pool spray is permitted only if the sources 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 suppression pool sprays since catastrophic failure of the primary containment is not expected under the conditions for which spray requirements are established. The wording of this override does, however, permit alternating between RPV injection and suppression pool spray modes as the need for each occurs, provided adequate core cooling can be maintained.

Facility: Hope Creek

Vendor: GE

Exam Date: 2019

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295025 EA1.04	
	Importance Rating	3.8	

K/A Statement: Ability to operate and/or monitor the following as they apply to HIGH REACTOR PRESSURE: HPCI: Plant-Specific.

Question: RO #12

Given:

- The plant was operating at 100% rated power.

When:

- An inadvertent MSIV isolation occurred.
- Reactor water level reached -60 inches before recovering and is now at 0 inches and rising.
- The CRS orders HPCI placed into full flow test mode for pressure control IAW HC.OP-AB.ZZ-0001, “Transient Plant Conditions”.

In order to do this, the operator must first:

A. ensure the HPCI VAC TK Vacuum Pump is running.

B. press the HPCI Manual Initiation PB.

C. start the HPCI Auxiliary Oil Pump.

D. reset the HPCI Initiation Logic.

Proposed Answer: D

Explanation (Optional): See attached Attachment 6(HPCI) of HC.OP-AQB.ZZ-0001

A: Incorrect- This step is performed if HPCI is being started for full flow test and is not injecting (page 6 step 1A)

B: Incorrect- This step is performed if HPCI is in the full flow test mode of operation, step 1A, page 5. HPCI is currently injecting.

C: Incorrect – HPCI is currently injecting The Aux. Oil Pump is running.

D: Correct- HPCI auto initiated on -38 inches, so step 2A of page 6 must be completed.

Technical Reference(s): HC.OP-AB.ZZ-0001 (Attach if not previously provided)
Transient Plant Conditions
Attachment 6 (HPCI)

Proposed References to be provided to applicants during examination: none

Learning Objective: Given plant conditions and a drawing of the controls, instrumentation and/or alarms located in the main control room, assess the status of the HPCI System by evaluation of the controls/instrumentation/alarms. (As available)

Question Source: Bank # 110943

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41 (7)

Comments:

ATTACHMENT 6

HPCI Injection

/
Date/Time

NOTE

Loss of 1AD481, 1CD481, 1AD482, or 1CD482 may affect controls and indication. ___

1.0 IF HPCI is in the Full Flow Test mode of operation
THEN **PROCEED** to step 2.0,
OTHERWISE, **PERFORM** the following:

A. **ARM,** THEN **PRESS** HPCI MAN INIT Pb. ___

B. IF System initiation does NOT occur,
THEN MANUALLY **START** the system as follows:

1. **ENSURE** HV-F008 is CLOSED. ___
2. **ENSURE** OP216 VAC TK VACUUM PUMP is running. ___
3. **ENSURE** HV-F059 is OPEN. ___
4. SIMULTANEOUSLY **PERFORM** the following:
   - **START** AUXILIARY OIL PUMP ___
   - **OPEN** FD-HV-F001. ___

C. **ENSURE** HV-F006 is OPEN. ___

D. **ENSURE** HV-8278 is OPEN. ___

E. **ENSURE** FD-HV-4922 is CLOSED. ___

F. **ADJUST** FIC-R600 HPCI FLOW setpoint, as necessary, to achieve desired flow. ___

G. IF HPCI flow controller oscillations are encountered,
THEN **OPERATE** controller in manual. ___

2.0 IF HPCI is in the Full Flow Test Mode of Operation,
THEN **PERFORM** the following:

A. **ADJUST** FIC-R600 HPCI FLOW setpoint to zero gpm. ___

B. **CLOSE** AP-HV-F011. ___

C. **CLOSE** HV-F008. ___

D. **OPEN** HV-F006. ___

E. **OPEN** HV-8278. ___

F. **ADJUST** FIC-R600 setpoint, as necessary, to achieve desired flow. ___

G. IF HPCI flow controller oscillations are encountered,
THEN **OPERATE** controller in manual. ___

3.0 At CRS discretion, **PLACE** suppression pool cooling in service. ___

END

ATTACHMENT 6

HPCI Full Flow Test Operation

/
Date/Time

1.0 IF HPCI is NOT in the Injection mode of operation
PERFORM the following:

A. **ENSURE** OP216 VAC TK VACUUM PUMP is RUNNING. ___

B. **ENSURE** HV-F059 is OPEN. ___

C. **ENSURE** HPCI AND RCIC Suctions are lined up to the CST. ___

D. **PRESS** HV-F008 INCR PB for ≈ 20 seconds. ___

E. **ADJUST** FIC-R600 HPCI FLOW setpoint to 1000 gpm. ___

F. SIMULTANEOUSLY **PERFORM** the following:

1. **START** AUXILIARY OIL PUMP ___
2. **PRESS** FD-HV-F001 OPEN Pushbutton ___

G. IMMEDIATELY **OPEN** AP-HV-F011. ___

H. WHEN Discharge Pressure turns **ADJUST** FIC-R600 setpoint to 3000 gpm. ___

I. **THROTTLE** HV-F008
AND **ADJUST** FIC-R600 setpoint, as necessary, up to and including full flow rate, to control HPCI pump parameters/reactor pressure. ___

2.0 IF HPCI is in the Injection Mode of Operation,
THEN **PERFORM** the following:

A. IF necessary, **RESET** HPCI INITIATION LOGIC. ___

B. **ADJUST** FIC R600 HPCI FLOW setpoint (STPT) to zero % ___

C. WHEN FLOW indicates zero gpm
THEN **CLOSE** HV-F006. ___

D. **CLOSE** HV-8278. ___

E. **ENSURE** HPCI AND RCIC Suctions are lined up to the CST. ___

F. **PRESS** HV-F008 INCR PB for ≈ 20 seconds ___

G. **OPEN** AP-HV-F011. ___

H. **ADJUST** FIC-R600 setpoint to achieve 3000 gpm flow. ___

J. **THROTTLE** HV-F008
AND **ADJUST** FIC-R600 setpoint, as necessary, up to and including full flow rate, to control HPCI pump parameters/reactor pressure. ___

3.0 At CRS discretion, **PLACE** suppression pool cooling in service. ___

END

Facility: Hope Creek
Vendor: GE
Exam Date: 2019
Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295026 EA1.01	
	Importance Rating	4.1	

K/A Statement: Ability to operate and/or monitor the following as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE: Suppression pool cooling.

Question: RO #13

Given:

- The plant is operating at 50% rated power.
- Suppression Pool cooling is in service.
- High Pressure Coolant Injection (HPCI) is operating in the CST to CST mode of operation IAW HC.OP-IS.BJ-0001(Q) - HPCI Main and Booster Pump Set – 0P204 and 0P217- In-service Test.
- Suppression pool temperature is 88°F and rising.

What are the requirements for entry into HC.OP-EO.ZZ-0102 “Primary Containment Control”?

ONLY when Suppression Pool temperature reaches _____________ and continues to rise.

A. 90°F

B. 100°F

C. 105°F

D. 120°F

Proposed Answer: **C**

Explanation (Optional):

A: Incorrect- The surveillance allows the suppression pool temperature of 105°F.

B: Incorrect- The surveillance allows the suppression pool temperature of 105°F.

C: Correct- Entry into HC.OP-EO.ZZ-0102 is **NOT** required at this time. The surveillance allows the suppression pool temperature of 105°F at which time the test is terminated and if 105°F cannot be maintained then entry into EOP-102 is required (see attached surveillance/tech spec).

D: Incorrect- OPCON 3 max temperature with the MSIV closed following a scram (T.S. 3.6.2.1.3). The surveillance only allows a suppression pool temperature of 105°F.

Technical Reference(s): HC.OP-IS.BJ-0001 (Attach if not previously provided)

T/S 3.6.2.1

HC.OP-EO.ZZ-0102

Proposed References to be provided to applicants during examination: none

Learning Objective: Given plant conditions, recognize the five (5) entry conditions for the Primary Containment Control Emergency Operating Procedure IAW HC.OP-EO.ZZ-0102. (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New **X**

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41 (7)

Comments:

.   ___

NOTE
Suppression Pool Average Water temperature requirements of T/S 3.6.2.1 should be complied with during HPCI operations. ___

5.14 PRIOR TO AND DURING HPCI operations,
IMPLEMENT Suppression Pool Average Water Temperature monitoring requirements of T/S 3.6.2.1 by performing HC.OP-DL.ZZ-0026(Q), Attachment 3m. ___

5.14.1. IF Suppression Pool Average Water Temperature reaches or exceeds 95°F,
IMPLEMENT Suppression Pool Average Water Temperature monitoring requirements of T/S 3.6.2.1 by performing HC.OP-DL.ZZ-0026(Q), Attachment 3f. ___

HC.OP-IS.BJ-0001(Q)

5.15 During the performance of this test the following actions shall be taken IF the following Suppression Pool Average Water Temperature(s) are reached:

5.15.1. 95°F, **ENSURE** RHR is in the Suppression Pool Cooling mode. (Entry into HC.OP-EO.ZZ-0102(Q) is not required at this time). ___

5.15.2. 105°F; **IMMEDIATELY TERMINATE** this test
AND **STOP** all testing that adds heat to the Suppression Pool. ___

5.15.3. IF Suppression Pool Average Water Temperature can not be maintained below 105°F
THEN **ENTER** HC.OP-EO.ZZ-0102(Q). ___

CONTAINMENT SYSTEMS

3/4.6.2 DEPRESSURIZATION SYSTEMS

SUPPRESSION CHAMBER

LIMITING CONDITION FOR OPERATION

3.6.2.1 The suppression chamber shall be OPERABLE with:

a. The pool water:

1. With an indicated water level between 74.5" and 78.5" and a

2. Maximum average temperature of 95°F during OPERATIONAL CONDITION 1 or 2, except that the maximum average temperature may be permitted to increase to:

a) 105°F during testing which adds heat to the suppression chamber.

b) 110°F with THERMAL POWER less than or equal to 1% of RATED THERMAL POWER.

3. Maximum average temperature of 95°F during OPERATIONAL CONDITION 3, except that the maximum average temperature may be permitted to increase to 120°F with the main steam line isolation valves closed following a scram.

b. A total leakage between the suppression chamber and drywell of less than the equivalent leakage through a 1-inch diameter orifice at a differential pressure of 0.80 psig.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

Facility: Hope Creek

Vendor: GE

Exam Date: 2019

Exam Type: RO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	295028 G2.1.23	
Importance Rating	4.3	

K/A Statement: Ability to perform specific system and integrated plant procedures during all modes of operation. High Drywell Temperature.

Question: RO #14

Given:

- The plant is operating at 100% rated power.

When:

- Turbine Building Chiller AK111 suffers an evaporator tube break.
- All Turbine Building Chilled Water pumps trip on low flow from Freon in the pump casings.
- Attempts to crosstie Chilled Water have failed.
- Drywell temperature is at 120°F and rising.
- Drywell pressure is at 1.0 psig and rising.

Which one of the following actions is required at this time IAW HC.OP-AB.CONT-0001, Drywell Pressure?

A. Immediately trip the Recirc pumps and maximize Drywell Cooling.

B. Reduce recirc pump speed to minimum and Lock the Mode Switch in Shutdown at 135°F Drywell temperature.

C. Immediately trip the Recirc pumps and initiate Drywell Sprays.

D. Reduce recirc speed to minimum and Lock the Mode Switch in Shutdown at 1.5 psig Drywell pressure.

Proposed Answer: **D**

Explanation (Optional): HC.OP-AB.CONT-0001 (see attached)

A. **Incorrect-**.IAW Subsequent action A. of CONT-0001 the operators would maximize drywell cooling, however IAW the retainment override they would not immediately trip the Recirc. Pumps until drywell pressure reached $\geq$ or = 1.68 psig. When drywell pressure reaches 1.5 psig, Reduce recirc speed to minimum and Lock the Mode Switch in Shutdown

B. **Incorrect-**.IAW EOP-102 a Reactor scram would not be required until reaching 340°F. (See attached).

C. **Incorrect-** IAW the retainment override they would not immediately trip the Recirc. Pumps until drywell pressure reached $\geq$ or = 1.68 psig. Under the current conditions the Drywell Sprays would not be initiated. (See attached DSIL)

D. **Correct-**. Reduce recirc speed to minimum and Lock the Mode Switch in Shutdown at 1.5 psig Drywell pressure. - Retainment override step of HC.OP-AB.CONT-0001 Drywell Pressure.

Technical Reference(s): HC.OP-AB.CONT-0001 (Attach if not previously provided)

Drywell Pressure

HC.OP-EO.ZZ-0102 BASES

Proposed References to be provided to applicants during examination: none

Learning Objective: Given any step of the procedure, determine the reason for performance of that step and/or predict expected system response to control manipulations prescribed by that step IAW available control room references. (As available)

Question Source: Bank #35468

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41 (10)

Comments:

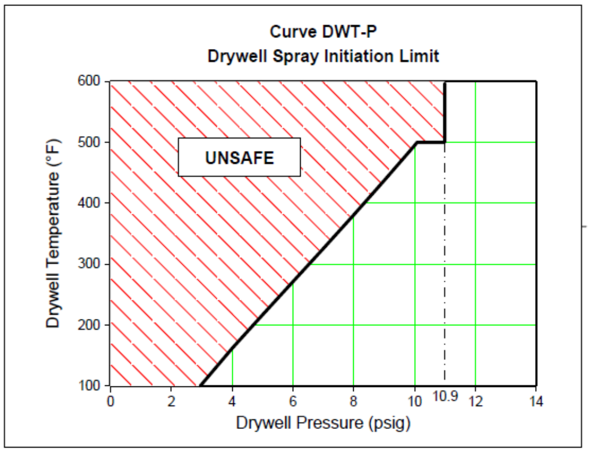
PSEG Internal Use Only HC.OP-AB.CONT-0001(Q)
DRYWELL PRESSURE

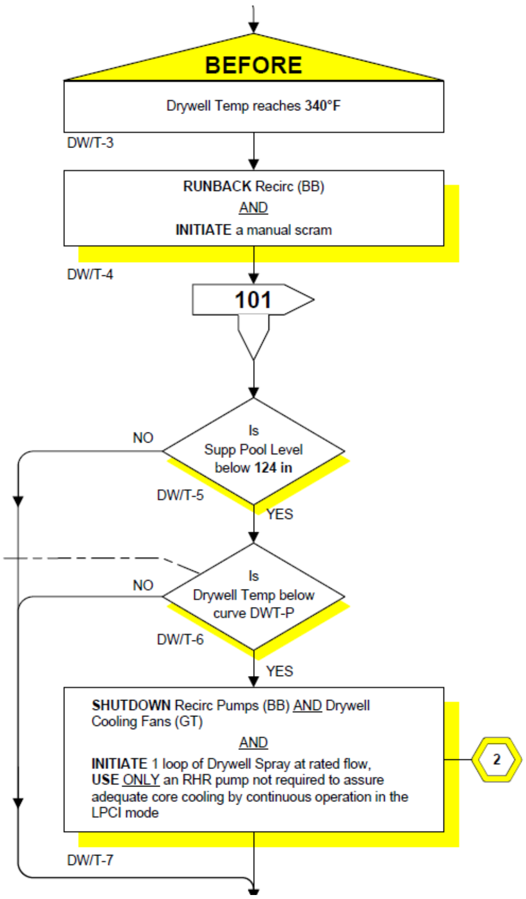
RETAINMENT OVERRIDE			
CONDITION			**ACTION**
I. Drywell Pressure is ≥ 1.5 psig and rising.	___	**I.a**	**REDUCE** Recirc. Pump Speed to MINIMUM.
Date/Time: ___________	___	**I.b**	**LOCK** the Mode Switch in Shutdown.
II. Drywell Pressure is ≥ 1.68 psig.	___	**II.a**	**TRIP** the Recirc Pumps within 10 minutes.
Date/Time: ___________			

PSEG Internal Use Only HC.OP-AB.CONT-0001(Q)
DRYWELL PRESSURE

SUBSEQUENT OPERATOR ACTIONS

CONDITION			ACTION
A. Unexpected rise in Drywell Pressure.	___	A.1	**TERMINATE** Containment Makeup AND Inerting.
		A.2	**MAXIMIZE** Drywell Cooling by **ENSURING**:
	___		• All Drywell Fan Cooling Coils are Open.
Date/Time: ___________	___		• All Drywell Fans are running in Fast Speed.
	___		****NOTE 1****
	___		• Proper TBCW system operation
	___	A.3	**PERFORM** the following:
	___		• Check Reactor Recirc. Pump Seals.
	___		• Check SRV Tailpipe Temperatures.
	___		• Drywell Leakage Source Detection IAW GP.ZZ-0005.


Curve DWT-P
Drywell Spray Initiation Limit
UNSAFE
Drywell Temperature (°F)
Drywell Pressure (psig)
600
500
400
300
200
100
0
2
4
6
8
10
10.9
12
14


BEFORE
Drywell Temp reaches 340°F
DW/T-3
RUNBACK Recirc (BB)
AND
INITIATE a manual scram
DW/T-4
101
NO
Is
Supp Pool Level
below 124 in
DW/T-5
YES
NO
Is
Drywell Temp below
curve DWT-P
DW/T-6
YES
SHUTDOWN Recirc Pumps (BB) AND Drywell Cooling Fans (GT)
AND
INITIATE 1 loop of Drywell Spray at rated flow,
USE ONLY an RHR pump not required to assure adequate core cooling by continuous operation in the LPCI mode
2
DW/T-7

Facility: Hope Creek

Vendor: GE

Exam Date: 2019

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295030 EK1.03	
	Importance Rating	3.8	

K/A Statement: Knowledge of the operational implications of the following concepts as they apply to LOW SUPPRESSION POOL WATER LEVEL: Heat capacity

Question: RO #15

Which ONE (1) of the following is the bases for the Suppression Pool level at which the primary containment pressure allowable limits could be exceeded and steam may not be adequately condensed?

A. HPCI exhaust line becomes uncovered.

B. Vent header drain lines become uncovered.

C. Suppression Pool Technical Specification minimum water level value.

D. Downcomers become uncovered.

Proposed Answer: D

Explanation (Optional): See attached HC.OP-EO.ZZ-0102 BASES

A: Incorrect- If suppression pool level cannot be maintained above 26 inches, and adequate core cooling is assured the operator is directed to secure HPCI. Operation of the HPCI turbine with its exhaust unsubmerged will tend to directly pressurize the suppression chamber. Action is already taken at the 38.5 inch level based on downcomers becoming uncovered and loosing suppression capabilities of the Suppression pool.

B: Incorrect- The threshold of 55 inches was selected as there is a 1¼ inch drain pipe attached to the low point of each of the eight vent pipes located in the torus. These drain pipes open into the torus at an indicated level of 50 IN; this level is between the low level LCO and the level at which the downcomers become uncovered. It is prudent to take the anticipatory actions to shutdown the reactor prior to the uncovery of these drain pipes

C: Incorrect- When suppression pool level lowers to below the Technical Specification lower limit, EOP-102 provides direction to use ECCS and safety-related service water systems and alignments not normally used to maintain suppression pool water level in general plant procedures.

D: Correct- Suppression pool water level must be maintained above the elevation of the downcomer vent openings to ensure that steam discharged from the drywell into the suppression pool following a primary system break will be adequately condensed. If suppression pool water level cannot be maintained above the specified minimum value, **steam may not be adequately condensed and primary containment pressure could exceed allowable limits.**

Technical Reference(s): HC.OP-EO.ZZ-0102-BASES (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: Given any step of the procedure, determine the reason for performance of that step and/or predict expected system response to control manipulations prescribed by that step (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New **X**

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41 (10)

Comments:

10.1 SP/L-1 Monitor and control Supp Pool Level between 74.5 and 78.5 in

SP/L-2 Is Supp Pool Level between 74.5 and 78.5 in

The initial action taken to control suppression pool water level employs the same methods typically used during normal plant operations: monitoring its status, and filling or draining the suppression pool as required to maintain water level within Technical Specification limits. Step SP/L-1 thus provides a smooth transition from general plant procedures to emergency operating procedures, and assures that the normal method of suppression pool water level control is attempted in advance of initiating more complex actions to correct out of limit conditions.

As long as suppression pool water level remains within Technical Specification high and low limits no further operator action is required in this subsection of the guideline other than continuing to monitor and control suppression pool water level using normal methods.

If outside the Technical Specification limits, further action is required and the operator is directed to SP/L-3

10.2 SP/L-3 Is Supp Pool Level Low or High

Separate steps in the suppression pool water level control subsection of EOP-102 provide appropriate instructions for responding to a high water level condition and a low water level condition.

10.3 SP/L-4 (Low Level) Restore Supp Pool Level to between 74.5 and 78.5 in using 1 or more of the following…

When suppression pool level lowers to below the Technical Specification lower limit, EOP-102 provides direction to use ECCS and safety-related service water systems and alignments not normally used to maintain suppression pool water level in general plant procedures. While typically not prudent for use during normal plant operation, these systems and alignments are used in an effort to maintain primary containment in its normal configuration and to prevent level from lowering to the point where the more severe actions of reactor scram and emergency RPV depressurization will be required.

10.4 SP/L-5 Can Supp Pool Level be maintained above 55 in

SP/L-6 Runback Recirc (BB)

And

Initiate a manual scram

If Suppression Pool level can be maintained above 55 inches, the operator is directed back to SP/L-1 to monitor and control suppression pool water level within Technical Specification high and low limits.

If suppression pool level cannot be maintained above 55 inches, direction is provided in Step SP/L-6 to run back the recirculation pumps to minimum speed, initiate a manual scram, and enter EOP-101 RPV Control at Step RC-1. Consistent with the definition of "can / cannot be maintained," a scram may be performed at any time with no particular margin to the limiting suppression pool water level intended. The appropriate timing of the action is event-dependent and requires an evaluation of system performance and availability in relation to parameter values and trends.

The threshold of 55 inches was selected as there is a 1¼ inch drain pipe attached to the low point of each of the eight vent pipes located in the torus. These drain pipes open into the torus at an indicated level of 50 IN; this level is between the low level LCO and the level at which the downcomers become uncovered (Hope Creek Drawing C-0932-0, Rev. 16). It is prudent to take the anticipatory actions to shutdown the reactor prior to the uncovery of these drain pipes; therefore, a suppression pool water level of 55 inches is conservatively used as the action limit.

This is performed so that the reactor is shut down in anticipation of emergency RPV depressurization if suppression pool level cannot be maintained above 38.5 inches.

- While the increase in core void fraction following emergency RPV depressurization would temporarily shut down the reactor, a potential for subsequent core damage exists and sudden in-surges of cold water could result in power spikes as RPV pressure decreases below the shutoff head of low pressure injection systems. Emergency depressurization with the reactor at power should therefore be avoided.
- EOP-101, Step RC/P-2 second override permits using the Main turbine bypass valves in anticipation of emergency RPV depressurization, reducing the heat input into primary containment.
- If emergency RPV depressurization is required EOP-101, Step RC/P-2 fourth override directs the use of EOP-202 Emergency RPV Depressurization.

The scram is effected indirectly, through entry of EOP-101, rather than through an explicit direction in EOP-102 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 EOP-101A if it is already in use

An explicit direction to enter EOP-101 must be provided since conditions requiring entry to EOP-102 do not necessarily require entry into EOP-101. A scram may have therefore not yet been initiated even if suppression pool level is low.

10.5 SP/L-7 Can Supp Pool Level be maintained above 38.5 in

SP/L-8 Emergency RPV Depressurization is required

If Suppression Pool level can be maintained above bottom of the downcomer vent openings, 38.5 inches, the operator is directed back to SP/L-1 to monitor and control suppression pool water level within Technical Specification high and low limits.

If suppression pool level cannot be maintained above 38.5 inches, emergency RPV depressurization is required.

Suppression pool water level must be maintained above the elevation of the downcomer vent openings to ensure that steam discharged from the drywell into the suppression pool following a primary system break will be adequately condensed. (Results of the Bodega Bay Mark I containment tests indicate 95% steam condensation may be achieved from a vertical downcomer vent that discharges at a level six inches above the suppression pool surface.) If suppression pool water level cannot be maintained above the specified minimum value, steam may not be adequately condensed and primary containment pressure could exceed allowable limits. Since the RPV may not be kept at pressure when pressure suppression capability is unavailable, Emergency RPV Depressurization is required.

The previous step that directs entry into EOP-101 at Step RC-1 when suppression pool water level cannot be maintained above 55 inches ensures that, if possible, the reactor is scrammed before RPV depressurization is initiated. Consistent with the definition of "can / cannot be maintained," a scram may be performed at any time with no particular margin to the limiting suppression pool water level intended. After the scram is performed, a second judgment is required to determine the need for emergency RPV depressurization. Again, consistent with the definition of "can / cannot be maintained," emergency RPV depressurization may be performed immediately following the scram if it is apparent that suppression pool water level will ultimately drop below the limiting elevation or delayed until the limit is actually reached. The appropriate timing of the two actions is event-dependent and requires an evaluation of system performance and availability in relation to parameter values and trends.

The effects of low suppression pool water level on suppression pool heat capacity are addressed through actions taken to control suppression pool temperature and RPV pressure below the Heat Capacity Temperature Limit (HCTL). Refer to the discussion of actions to maintain plant parameters below the HCTL in the section devoted to the SP/T Leg of EOP-102 earlier in this document, and in the EOP-101 BASES document within the section devoted to the RC/P Leg of EOP-101.

10.6 SP/L-9 Maintain Supp Pool Level above 26 in

SP/L-10 Can Supp Pool Level be maintained above 26 in

SP/L-11 Secure HPCI (AB.ZZ-0001) only if adequate core cooling is assured

If Suppression Pool level can be maintained above the elevation of the top of the HPCI exhaust (Calculation #4, PR 970926170 #31), 26 inches, the operator is directed back to SP/L-1 to monitor and control suppression pool water level within Technical Specification high and low limits.

If suppression pool level cannot be maintained above 26 inches, and adequate core cooling is assured the operator is directed to secure HPCI.

Facility:	Hope Creek
Vendor:	GE
Exam Date:	2019
Exam Type:	RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295031 EK2.05	
	Importance Rating	4.2	

K/A Statement: Knowledge of the interrelations between REACTOR LOW WATER LEVEL and the following: Low pressure coolant injection (RHR).

Question: RO #16

Given:

- The reactor is shutdown.
- All control rods fully inserted.
- Reactor pressure is 420 psig and stable.
- RPV level is being held relatively constant.

Then:

- 'C' RHR Loop receives a spurious LOCA level 1 signal.
- The OHA A6-A4 RHR LPCI LOOP C INITIATED is received.

Which of the following describes the status of "C" RHR?

A. BC-HV F017C (LPCI Injection valve) indicates open and the 'C' RHR minimum flow control valve (HV-F007C) will indicate open.

B. BC-HV F017C (LPCI Injection valve) indicates open and the 'C' RHR minimum flow control valve (HV-F007C) will indicate closed.

C. BC-HV F017C (LPCI Injection valve) indicates closed and the 'C' RHR minimum flow control valve (HV-F007C) will indicate open.

D. BC-HV F017C (LPCI Injection valve) indicates closed and the 'C' RHR minimum flow control valve (HV-F007C) will indicate closed.

Proposed Answer: A

Explanation (Optional):

A: **Correct-** The F017C will be open as the reactor pressure permissive **of < 450 psig** is satisfied with an injection signal present. The F007C will still be open as system flow is < 1270 gpm, because reactor pressure is above the **shutoff head of the pump (366 psig)** and will not be injecting to the vessel.

B: **Incorrect-**. The F017C will be open as the reactor pressure permissive of < 450 psig is satisfied with an injection signal present. The F007C will still be open as system flow is < 1270 gpm, because reactor pressure is above the shutoff head of the pump (366 psig) and will not be injecting.

C: **Incorrect-**. The F017C will be open as the reactor pressure permissive of < 450 psig is satisfied with an injection signal present.

D: **Incorrect-**. The F017C will be open as the reactor pressure permissive of < 450 psig is satisfied with an injection signal present.

Technical Reference(s): HC.OP-SO.BC-0001 Sect 5.3 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: Given a set of conditions and a drawing of the controls, instrumentation and/or alarms located in the main control room, assess the status of the Residual Heat Removal System or its components by evaluation of the controls/instrumentation/alarms (As available)

Question Source: Bank #135403

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41 (7)

Comments:

HC.OP-SO.BC-0001(Q)

5.3 **LPCI Initiation Observation**

NOTE

IF the RHR System is aligned for a manual mode of operation and LPCI protection is required, the system will automatically realign and initiate LPCI operation without operator action.

Upon receipt of LPCI initiation signal, RHR Pumps A and B start without delay on normal auxiliary power and RHR Pumps C and D start after a 5 second time delay. All four RHR Pumps start simultaneously on Standby Diesel power (LOP). ___

5.3.1. **ENSURE** the following occur upon receipt of LPCI System Initiation signal (as indicated by A(B,C,D) INIT AND SEALED IN on):

A. RHR Pumps A, B, C AND DP202 start as indicated by the following:

- PUMP A(B,C,D) RUNNING ON ___
- AI-6358 A(B,C,D) PUMP A(B,C,D) MTR AMPS ___

B. IF closed, THEN BC-HV-F048A(B), A(B) RHR HX SHELL BYP MOV opens AND CANNOT be closed for 3 minutes. ___

C. IF conducting RHR Surveillance Tests at the time of auto initiation, THEN, the following valves will close:

- BC-HV-F024A(B) RHR LOOP A(B) TEST RET MOV ___
- BC-HV-F010A(B) RHR LOOP C(D) TEST RET MOV ___

D. WHEN Reactor pressure is < 450 psig, THEN BC-HV-F017A(B,C,D), RHR LOOP A(B,C,D) LPCI INJ MOV will open. ___

E. WHEN associated RHR Pump flow increases to > 1400 gpm (as indicated on FI-R603A(B,C,D) or FR-R608A(B) - CRIDS A3137(A3139), LOOP A(B,C,D) FLOW), AND pump has been running for > 4 seconds, THEN BC-HV-F007A(B,C,D), RHR PMP A(B,C,D) MIN FLOW MOV will close. ___

Valve	Power	Auto Actuations	Interlocks
F007A * B C D (Minflow)	10B212 10B222 10B232 10B242	Open after 10 sec TD with Flow less than 1250 GPM - Auto close when RHR Pump running > 4 seconds and Loop flow exceeds 1270 GPM	To enable: (1) Respective pump bkr must be closed and loop flow less than 1250 GPM (2) 10 sec time delay - enable operator to estab. flowpath (ie. SDC) prior to auto open. (3) Respective pump bkr must be closed > 4 seconds and loop flow exceeds 1270 GPM for F007A(B,C,D) to auto close.

Valve	Power	Auto Actuations	Interlocks
F017A B C D (LPCI Injection)	10B212 10B222 10B232 10B242	Auto open on LPCI init. if following condition exist: (1) LPCI init. present in respective RHR loop logic (2) Power is avail. on respective pump bus (3) Reactor press. less than 450 psig (may be overridden by "AUTO OPEN OVRD")	Rx press. must be less than 450 psig to open valve either MAN or AUTO. Placing Ch. B RSP to EMERG will close F017B & inhibit all associated automatic & OVLD protection Features. The LPCI injection valve must be 100% closed (in the respective loop with a LPCI initiation signal present) to OPEN: F027A(B), F024A(B), F016. F021

Note: Technical Specifications requires LPCI mode flow of at least 10,000 gpm against a test line pressure corresponding to a reactor vessel to containment differential pressure of >20 psid.) Pump shutoff head is approx. **366 psig (875 ft)**

Facility: Hope Creek

Vendor: GE

Exam Date: 2019

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295037 EK3.01	
	Importance Rating	4.1	

K/A Statement: Knowledge of the reasons for the following responses as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN: Recirculation pump trip/runback: Plant-Specific.

Question: RO #17

Given:

- A failure to scram has occurred.

Current conditions:

- Reactor power is at 65% rated power.
- The main turbine is on line.

IAW the EOP Bases, the reactor recirculation pumps are required to be runback to minimum speed before tripping them in order to ___________________.

A. maintain the largest margin to the APLHGR power distribution limit.

B. prevent power instabilities due to operating at high power without adequate core flow following a Main Turbine trip.

C. prevent a Main Turbine trip and additional heat loading of the suppression pool from SRVs if power remains above the bypass valve capacity.

D. prevent a LOCA level 2 actuation.

Proposed Answer: C

Explanation (Optional): The most rapid flow rate reduction and, consequently, the most rapid power reduction, is achieved by tripping the recirculation pumps. However, if the recirculation pump trip is initiated from a high power level, the resulting rapid changes in steam flow, RPV pressure, and RPV water level may cause a trip of the main turbine-generator and a trip of RPV injection systems. **If the main turbine-generator trips and reactor power exceeds the turbine bypass valve capacity, RPV pressure will increase until one or more SRVs open. Heatup of the suppression pool then begins and RPV level lowering may be required.** If RPV injection systems trip, the resultant RPV water level transient may require emergency depressurization of the RPV and operation of less desirable RPV injection sources.
To effect a more controlled reduction in reactor power and thereby avoid main turbine-generator and RPV injection system trips and their associated complications, **a recirculation flow runback is performed prior to tripping the recirculation pumps.**
(See attached 101A-BASES)

A: **Incorrect-** Removing RPV flow will rely on natural circulation to prevent approaching fuel failure limits during an ATWS, it will certainly not lessen it

B: **Incorrect-** Actions taken will remove all forced circulation, and lower RPV level to lower power, lowering power takes precedent over instabilities.

C: **Correct-** See above explanation.

D: **Incorrect-** With this transient a High RPV water level will be the concern not low level.

Technical Reference(s): HC.OP-EO.ZZ-0101A-BASES (Attach if not previously provided)

ATWS

Proposed References to be provided to applicants during examination: None

Learning Objective: Given any step of the procedure, explain the reason for performance of that step and/or evaluate the expected system response to control manipulations prescribed by that step. (As available)

Question Source: Bank #33988

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41 (10)

Comments:

7.6 RC/Q-6 Is reactor power above 4% or unknown

RC/Q-7 Initiate SLC and verify RWCU isolates

RC/Q-8 Verify Recirc runback to minimum

RC/Q-9 Trip the Recirc pumps

If reactor power remains above the APRM downscale setpoint, the recirculation pumps are tripped to effect a prompt reduction in power. To minimize the RPV water level transient, the recirculation pumps are verified to be at minimum speed prior to tripping the pumps.

While tripping the pumps may place the plant in a high power-to-flow condition and thereby contribute to neutronic and thermal-hydraulic instabilities, continued recirculation pump operation may not be desirable or even possible:

- If RPV water level is lowered in accordance with EOP-101A, LP (level / power control), the pumps will trip automatically when the low RPV water level trip setpoint is reached.
- Allowing reactor power to remain high would increase the flow demand on RPV injection systems and the heat load on the primary containment.

9.1 RC/P-1 Monitor and Control RPV Pressure

The RPV pressure control subsection stabilizes RPV pressure below the high RPV pressure scram setpoint and, if necessary, depressurizes and cools down the RPV to cold shutdown conditions. Since changes in RPV pressure may affect the reactor shutdown margin and availability of injection sources, RPV depressurization and cooldown must be coordinated with core cooling and reactor power control strategies.

The main turbine bypass valves and the main condenser comprise the preferred mechanism for discharging and condensing steam from the RPV, but alternate methods are identified should the preferred method not be available.

Facility: Hope Creek

Vendor: GE

Exam Date: 2019

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295038 EA1.04	
	Importance Rating	2.8	

K/A Statement: Ability to operate and/or monitor the following as they apply to HIGH OFF-SITE RELEASE RATE: SPDS/ERIS/CRIDS/GDS: Plant-Specific

Question: RO #18

Given:

- An Unusual Event is declared due to a radiological release.
- The Meteorological Tower link to Hope Creek on SPDS is malfunctioning.
- The link to Salem Generating Station is working properly.

Which one of the following sets of data must be requested from Salem Station to be communicated to the States of New Jersey and Delaware with the Initial Contact Message Form (ICMF)?

A. Wind Direction TO; Wind Speed 33 ft elevation.

B. Wind Direction TO; Wind Speed 300 ft elevation.

C. Wind Direction FROM; Wind Speed 33 ft elevation.

D. Wind Direction FROM; Wind Speed 300 ft elevation.

Proposed Answer: **C**

Explanation (Optional): See attached EP-HC-325-F1

A: **INCORRECT-** Wind direction FROM not TO.

B: **INCORRECT-** Wind direction FROM not TO. 33 ft. elevation not 300 ft.

C: **CORRECT-** IAW ICMF section IV (see attached)

D: **INCORRECT-**.33 ft. elevation not 300 ft.

Technical Reference(s): EP-HC-325-F1 (Attach if not previously provided)

ICMF ATT. 1

Proposed References to be provided to applicants during examination: none

Learning Objective: ECG/E-Plan/Fire & Medical Questions (As available)

Question Source: Bank #35528

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41 (11)

Comments:

INITIAL CONTACT MESSAGE FORM

I. THIS IS ______________________, COMMUNICATOR IN THE CONTROL ROOM
(NAME)

AT THE **HOPE CREEK** NUCLEAR GENERATING STATION.

II. THIS IS NOTIFICATION OF AN **UNUSUAL EVENT** WHICH WAS

DECLARED AT ______________________ ON ______________
(Time - 24 HR CLOCK) (DATE)

EAL # ____________ DESCRIPTION OF EVENT ______________

____

____

____

____

III. ☐ THERE **IS** A RELEASE IN PROGRESS DUE TO THE EVENT
☐ THERE IS **NO** RELEASE IN PROGRESS DUE TO THE EVENT

Any release above normal, attributable to the event. See Basis for examples.

IV. 33 FT. LEVEL WIND DIRECTION (**From**): ____________ WIND SPEED: ____________
(From MET Computer /SPDS) (DEGREES) (MPH)

V. **NO PROTECTIVE ACTIONS ARE RECOMMENDED AT THIS TIME**

____
EC Initials
(Approval to Transmit ICMF)

Facility: Hope Creek

Vendor: GE

Exam Date: 2019

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	600000 AA2.17	
	Importance Rating	3.0	

K/A Statement: Ability to determine and interpret the following as they apply to PLANT FIRE ON SITE: Systems that may be affected by the fire.

Question: RO #19

Which of the following plant ventilation system(s) automatically shutdown in the event of a fire to assure that the fire dampers will close?

A. Service Area Supply AND Service Area Exhaust ONLY.

B. RBVS AND FRVS ONLY.

C. Service Area Supply, Service Area Exhaust AND RBVS ONLY.

D. Service Area Supply, Service Area Exhaust, RBVS AND FRVS.

Proposed Answer: **A**

Explanation (Optional): Some dampers were identified that would not close against the system design airflow. To assure that these dampers would close, in-duct ionization detectors were installed that cause automatic shutdown of the associated ventilation system. Refer to Table 2 (attached) for a summary of the ventilation systems that shutdown automatically to assure fire dampers will close. (Table 2 does not include RBVS OR FRVS)

A: **Correct-** See above explanation.

B: **Incorrect-**.RBVS and /or FRVS do not have in-duct ionization detectors, therefore no automatic shutdown of the systems due to a fire condition. FRVS has a deluge system for the charcoal beds. However, the deluge system has to be lined up manually with no automatic trips.

C: **Incorrect-**.RBVS has no in-duct ionization detectors and will not trip on a fire.

D: **Incorrect-** RBVS and /or FRVS do not have in-duct ionization detectors, therefore no automatic shutdown of the systems due to a fire condition. FRVS has a deluge system for the charcoal beds. However, the deluge system has to be lined up manually with no automatic trips.

Technical Reference(s): NOH01FIREPRO (Attach if not previously provided)

Table 2

Proposed References to be provided to applicants during examination: none

Learning Objective: Given a labeled diagram/drawing of the Auxiliary Building Service Area Ventilation System, summarize/identify the controls and/or indications available for the following: (As available)
a. Service Area Supply System (SAS)
b. Service Area Exhaust System (SAE)
c. Chemical Laboratory Exhaust System (CLE)
d. Remote Shutdown Panel (RSP) Supply Unit.

Question Source: Bank #119071

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41 (10)

Comments:

TABLE NO. 2

IN-DUCT SMOKE DETECTORS AND ASSOCIATED FIRE DAMPERS

In-Duct Smoke Detector	Fan	System	HVAC Control Panel	Inoperable Fire Dampers
0XSH-9018	0AV308 0BV308	SAE	00C381	0FPGLD-345D5, 0FPGLD-345D5
0XSH-9022	0AVH131 0BVH131	SAS	00C181	0FPGLD-334D6, 0FPGLD-338D10 0FPGLD-338D15, 0FPGLD-338D12 0FPGLD-338D13, 0FPGLD-345D2 0FPGLD-345D3, 0FPGLD-355D2 0FPGLD-355D4, 0FPGLD-356D2 0FPGLD-359D3, 0FPGLD-496D2
0XSH-9023	0AVH131 0BVH131	SAS	00C181	0FPGLD-334D6, 0FPGLD-338D10 0FPGLD-338D11, 0FPGLD-338D12 0FPGLD-338D13, 0FPGLD-345D2 0FPGLD-345D3, 0FPGLD-355D2 0FPGLD-355D4, 0FPGLD-356D2 0FPGLD-359D3, 0FPGLD-496D2
0XSH-9024	0AVH318 0BVH318	SRWE	00C380	0FPGHD-778D2, 0FPGHD-775D1 0FPGHD-775D2, 0FPGHD-356D3 0FPGHD-778D1, 0FPGHD-778D3
0XSH-9028	0AVH318 0BVH318	SRWE	00C380	0FPGHD-778D2, 0FPGHD-775D1 0FPGHD-775D2, 0FPGHD-356D3 0FPGHD-778D1, 0FPGHD-778D3

Facility: Hope Creek

Vendor: GE

Exam Date: 2019

Exam Type: RO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	700000 G2.4.47	
Importance Rating	4.2	

K/A Statement: Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material - Generator Voltage and Electric Grid Disturbances.

Question: RO #20

Given:

- The plant is at 100% rated power.

T=0:

- A SMD Alert of K7 is declared.
- ESOC Excess MVARS is in alarm.
- DC Neutral ground current is at 100 amps.
- HC.OP-AB.BOP-0004, Grid Disturbance has been entered.

T= 5 minutes:

- A fire is reported, burning directly below the 5015 line.
- The fire department is on scene of the fire.
- Yard NEO reports that the BX500 Main Power Transformer oil temperature is 105°C and rising at 2°C/minute.

TABLE 1

Temperature Indicator	Alarm Setpoint	Max Normal	Max Peak
Oil Temperature	105°C	110°C	120°C

T= 15 minutes:

Assuming the temperature trend above is constant, which of the following describes the plant status after operator action(s) is (are) completed?

A. Reactor scrammed, turbine tripped. All 500 kv lines available.

B. Reactor scrammed, turbine tripped. 5015 line removed from service.

C. Main Gen online at a lower power level. 5015 line removed from service.

D. Main Gen online at a lower power level. All 500 kv lines in-service.

Proposed Answer: **B**

Explanation (Optional): The candidate calculates the trend of temperature rise (T= 15 minutes at 125 and responds IAW HC.OP-AB.BOP-0004 Condition C with the oil temperature above Max Peak (the reactor and turbine must come off line (> or equal to 18% power). The fact that the fire is directly below the 5015 line this would be an immediate threat to the line, therefore IAW Condition D the line must be removed from service (see attached).

A: **Incorrect.** The fire poses an immediate threat to the 5015 line.

B: **Correct.** See above explanation.

C: **Incorrect.** With power > or equal to 18% power and exceeding the Table 1 Max Peak oil temperature the reactor and turbine has to come off line.

D: **Incorrect.** With power > or equal to 18% power and exceeding the Table 1 Max Peak oil temperature the reactor and turbine has to come off line. The fire poses an immediate threat to the 5015 line

Technical Reference(s): HC.OP-AB.BOP-0004 (Attach if not previously provided)

Grid Disturbance

Proposed References to be provided to applicants during examination: Table 1 of HC.OP-AB.BOP-0004

Learning Objective: Given plant conditions and plant procedures, determine required actions of the retainment override(s) and subsequent operator actions in accordance with Grid Disturbances. (As available)

Question Source: Bank #

Modified Bank # 56081 (Modified stem for both temperature and the threat of the fire. Different answer from original)

New

Question History: Original used on 2018 NRC Exam

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41 (10)

Comments: Original Question

Given:

- The plant is at 100% power.
- SMD Alert of K7.
- ESOC Excess MVARS in alarm.
- DC Neutral ground current of 100 amps
- A small marsh fire is observed 1 mile from the 5015 Red Lion Line burning away from the line.
- The fire department is monitoring the fire.

B Main Power Transformer is reading:

- Oil Temp is 107°C
- Winding Hot Spot Temp is 117°C

Which of the following describes the plant status after action(s) is(are) complete?

A. Main Gen online at a lower power level. All 500 kv lines inservice.

B. Reactor scrammed, turbine tripped. All 500 kv lines inservice.

C. Reactor scrammed, turbine tripped. 5015 line removed from service.

D. Main Gen online at a lower power level. 5015 line removed from service.

Answer: A

C. Geomagnetic Induced Current Alarm (at Main Transformer Local Alarm Panel) AND/OR DC Neutral Ground Current Alarm (via Salem Control Room)

Date/Time: ___________

___ **NOTE 5**

___ C.1 **NOTIFY** Salem SM/CRS of elevated Hope Creek DC Ground Current and to initiate abnormal operating procedures as necessary.

___ C.2 **MONITOR** Main Transformer Temperatures. (**REFER** to Table 1)

___ C.3 IF any temperatures are trending upward, THEN **CONTACT** System Engineering AND Load Dispatcher.

___ ✮ **CAUTION 1** ✮

___ C.4 IF approaching any Alarm Setpoint in Table 1 THEN **REDUCE** Generator Load to maintain temperature below the Alarm Setpoint.

___ **NOTE 6**

___ **NOTE 7**

___ C.5 IF oil temperature exceeds a Max Normal Setpoint of Table 1 WITH Engineering Concurrence THEN **PERFORM** one of the following:

___ • IF Reactor Power ≥18% THEN **PERFORM** the following:

___ a. **REDUCE** Recirc Pump speed to MINIMUM.

___ b. **LOCK** the Mode Switch in SHUTDOWN.

___ c. **TRIP** the Main Turbine.

___ • IF Reactor Power <18% THEN **TRIP** the Main Turbine.

___ C.6 IF oil temperature exceeds Max Peak Setpoint of Table 1 THEN **PERFORM** one of the following:

___ • IF Reactor Power ≥18% THEN **PERFORM** the following:

___ a. **REDUCE** Recirc Pump speed to MINIMUM.

___ b. **LOCK** the Mode Switch in SHUTDOWN.

___ c. **TRIP** the Main Turbine.

___ • IF Reactor Power <18% THEN **TRIP** the Main Turbine.

___ C.7 **DIRECT** System Engineering to initiate on-going Dissolved Gas Analyzer (Serveron) readings for the Main Transformers throughout the event.

SUBSEQUENT OPERATOR ACTIONS (continued)

CONDITION		ACTION
D. A Fire is burning near a Transmission Line that poses an IMMEDIATE threat to the line. Date/Time: ________	___	★ **CAUTION 2** ★
	___D.1	**NOTIFY** the Salem SM and the System Operator of the fire. The System Operator, HC and Salem Shift Manager assess the situation.
	___D.2	IF the fire is determined to be an immediate threat, **TAKE** measures to remove the line from service expeditiously.
	___D.3	**ENSURE** the stability guidelines described in A-5-500-EEE-1686 are followed.
E. A Fire is burning near a Transmission Line that MAY pose a FUTURE threat to a line. Date/Time: ________	___E.1	**NOTIFY** the Salem SM and the System Operator of the fire. The System Operator, HC and Salem Shift Manager assess the situation.
	___E.2	IF the fire is determined to be a future threat, **TAKE** measures to prepare to remove the line from service.
	___E.3	WHEN the System Operator removes the line from service, **ENSURE** the stability guidelines described in A-5-500-EEE-1686 are followed.
F. 500 kV Switchyard Voltage falls below 493 kV OR Predicted event may result in Switchyard Voltage falling below 493 kV. **[70049498]** Date/Time: ________ **[T/S 3.8.1.1]**	___F.1	IF the actual 500 kV switchyard voltage has fallen below 493 kV OR Notified that with a Loss of the Hope Creek Main Generator the predicted 500 kV Switchyard Voltage may fall below 493 kV THEN **DECLARE** ALL Offsite Power Sources INOPERABLE per T/S 3.8.1.1 OR 3.8.1.2.
	___F.2	IF Notified of event OTHER THAN a trip of the Hope Creek Main Generator, the predicted 500 kV switchyard voltage may fall below 493 kV OR ESOC CANNOT predict the effect on 500 kV Switchyard Voltages THEN **ENTER** a Heightened State of Awareness AND **USE** risk assessment tools, (e.g., Equipment Out Of Service), for assessing the increase in risk associated with a potential Loss of Offsite Power. (OP-AA-101-112-1002 and WC-AA-101, Att. 7)
	___F.3	**CONTACT** the Risk Assessment Group AND ESOC to determine the present grid operating conditions and further actions to take.
	___F.4	**NOTIFY** ESOC of risk changes that emerge during ongoing nuclear maintenance at Salem and Hope Creek that could impact generation.

Facility: Hope Creek

Vendor: GE

Exam Date: 2019

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295002 AA2.02	
	Importance Rating	3.2	

K/A Statement: Ability to determine and/or interpret the following as they apply to LOSS OF MAIN CONDENSER VACUUM: Reactor power: Plant-Specific.

Question: RO #21

Given:

- A plant startup is in progress IAW HC.OP-IO.ZZ-0003, Startup from Cold Shutdown to Rated Power.
- Reactor power is at 11.5% rated power.

Then:

- The Common Offgas Recombiner train performance starts to degrade and needs to be removed from service.
- The Unit 1 Offgas Recombiner train is cleared and tagged and NOT available.

Which one of the following describes the action required to allow placing the Mechanical Vacuum Pumps (MVP) in service IAW HC.OP-AB.BOP-0006, Main Condenser Vacuum abnormal and the reason for the power level reduction?

A. Lower reactor power by 7%; Offsite radiological release may be above allowable limits at the North Plant Vent.

B. Lower reactor power by 6%; Combustible gas concentrations may cause an explosion in the SJAE (Steam Jet Air Ejector) after condenser.

C. Lower reactor power by 6%; Offsite radiological release may be above allowable limits at the South Plant Vent.

D. Lower reactor power by 7%; Combustible gas concentrations may cause an explosion at the MVP.

Proposed Answer: D

Explanation (Optional): The mechanical vacuum pumps must be taken out of service before reaching 5% of rated thermal power. When the reactor is operating above 5% thermal power, significant amounts of hydrogen and oxygen are present in the main condenser. Operation of the mechanical vacuum pump under these circumstances could result in the presence of a **detonable concentration of hydrogen and oxygen at atmospheric pressure in the pump** and/or its associated water separator. This condition could result in combustion in the pump or the water separator and such an event could pose a hazard to any personnel in the vicinity of the pump since neither the pump nor the water separator are designed to be detonation resistant. The operation of the Mechanical Vacuum Pumps when the reactor is operating >5% power will also result in high radioactive levels at the **South Plant Vent.**

A: **Incorrect**- Right power reduction for <5% requirement, however the flowpath for the MVPs exhaust is through the South Plant Vent.

B: **Incorrect**- The mechanical vacuum pumps must be taken out of service before reaching 5% of rated thermal power.

C: **Inorrect-** The mechanical vacuum pumps must be taken out of service before reaching 5% of rated thermal power.

D: **Correct**- Subsequent actions of HC.OP-AB.BOP-0006 **(see attached)** direct power reduction to less than 5 percent. A 7 percent reduction for the given conditions will result in < 5 percent power. The explosion concern is the MVP pump (see above).

Technical Reference(s):	HC.OP-AB.BOP-0006 Main Condenser Vacuum	(Attach if not previously provided)
	HC.OP-IO.ZZ-0003 Startup from Cold Shutdown to Rated Power	
	NOH01 AIRREMC-11 Condenser Air Removal	

Proposed References to be provided to applicants during examination: none

Learning Objective:	Explain the reasons for how plant/system parameters respond when implementing Main Condenser Vacuum.	(As available)
	Apply Precautions, Limitations and Notes while executing the STARTUP FROM COLD SHUTDOWN TO RATED POWER Integrated Operating Procedure.	
	Explain the purpose for not operating the mechanical vacuum pumps above 5% of rated power	

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

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41 (10)

Comments:

SUBSEQUENT OPERATOR ACTIONS (continued)

CONDITION	ACTION
C. Offgas Malfunction. Date/Time: ________	___ C.1 **DISPATCH** Radwaste to the local Offgas panels to determine the cause of the malfunction. C.2 **ENSURE** one of the following valves is OPEN to supply Main Steam or Aux Steam to the Pre-heater: ___ • HV-5640 Main Steam supply (Control Room) ___ • HV-5660 Aux Steam supply (Local Panel) C.3 **PERFORM** EITHER of the following: ___ • **DIRECT** Radwaste to place the Standby Offgas train in service IAW RW.HA-0001. ___ • **PRESS** RECOMB TRAIN SELECT OPEN for the Standby Offgas train. ___ C.4 IF the Common train was placed in service from the Control Room, **DIRECT** Radwaste to open HV-5647. C.5 IF two RACS Pumps are in service, THEN **ENSURE** the following valves are OPEN: ___ • HV-2577 ___ • HV-7712A ****NOTE 1**** ___ C.6 IF HV-2016A(B) ISOLATES, THEN **RE-OPEN** HV-2016A(B) as follows: **PRESS** AND **HOLD** HV-2016A(B) OPEN Pushbutton UNTIL D3587(8) Low Flow Alarm CLEARS.
D. Reactor power is >5% and the Offgas System cannot be re-established. Date/Time: ________	D.1 **PERFORM** EITHER of the following: ★ **CAUTION 2** ★ ★ **CAUTION 6** ★ ___ • **LOWER** Reactor power to < 5%. THEN **PLACE** the Mechanical Vacuum Pumps I/S. (CG) **[CD-015B]** ___ • **REDUCE** Recirc Pump speed to MINIMUM, THEN **LOCK** the Mode Switch in SHUTDOWN.

NOTE
With Bypass Valve(s) open, 5% Reactor Power Shall **NOT** be exceeded with the Mechanical Vacuum Pump in service. [CD-015B]

____

Facility: Hope Creek

Vendor: GE

Exam Date: 2019

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295010 G2.4.31	
	Importance Rating	4.2	

K/A Statement: Knowledge of annunciator alarms, indications, or response procedures.- High Drywell Pressure / 5

Question: RO #22

Given:

- The plant is operating at 100% rated power.

When:

- DRYWELL PRESSURE HI/LO annunciator alarm illuminates.
- Drywell pressure is 1.0 psig and slowly rising.
- HC.OP-AB.CONT-0001, Drywell Pressure abnormal is entered.

Which of the following actions are required for the current conditions and why?

A. Reduce Recirc pump speed to minimum, lock the mode switch in Shutdown IAW the retainment override.

B. Ensure all Drywell fan cooling coils are open to the Drywell Coolers to maximize drywell cooling.

C. Align RACS to supply a backup to the Chilled Water for Drywell Cooling Units to ensure maximum heat removal capability.

D. Place Feedwater Sealing System in operation to eliminate Containment leakage to the Feedwater System.

Proposed Answer: B

Explanation (Optional):

A: **Incorrect.** (See attached) which directs reducing recirc to minimum speed and lock the mode switch in shutdown IAW the retainment override at drywell pressure of ≥ 1.5 psig.

B: **Correct**. Open both Cooling Coil Chilled water inlet valves to Drywell Cooler CVH212 to maximize drywell cooling. – (see attached subsequent action A.2).

C: **Incorrect.** (See attached subsequent action B), RACS typically has a 10 to 30 degree higher temperature and would not maximize heat removal. Chilled water is a better source and RACS is only directed if Chilled water is lost.

D: **Incorrect.** This is only performed post LOCA when Containment pressure is expected to be higher than Feedwater pressure and potential leakage could occur.

Technical Reference(s): HC.OP-AB.CONT-0001 (Attach if not previously provided)

Drywell Pressure

Proposed References to be provided to applicants during examination:

Learning Objective: Recognize abnormal indications/alarms and/or procedural requirements for implementing Drywell Pressure. (As available)

Question Source: Bank #33800

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41 (10)

Comments:

SUBSEQUENT OPERATOR ACTIONS

CONDITION	ACTION
A. Unexpected rise in Drywell Pressure. Date/Time: ____________	___ A.1 **TERMINATE** Containment Makeup AND Inerting. A.2 **MAXIMIZE** Drywell Cooling by **ENSURING**: ___ • All Drywell Fan Cooling Coils are Open. ___ • All Drywell Fans are running in Fast Speed. ___ ****NOTE 1**** ___ • Proper TBCW system operation ___ A.3 **PERFORM** the following: ___ • Check Reactor Recirc. Pump Seals. ___ • Check SRV Tailpipe Temperatures. ___ • Drywell Leakage Source Detection IAW GP.ZZ-0005.
B. Turbine Bldg. Chill Water System is lost to the Drywell. Date/Time: ____________	___ ★ **CAUTION 1** ★ B.1 **ALIGN** RACS to the Chill Water System for Drywell Cooling as follows: a. **ENSURE** RACS to the out of service Off-Gas Train is ISOLATED as follows: ___ • IF the Common Off-Gas Train is in service, THEN **CLOSE** HV-2577. ___ • IF Unit 1 Off-Gas Train is in service, THEN **CLOSE** HV-7712A1. ___ b. **CLOSE** HV-9532-1 AND HV-9532-2. ___ c. **PRESS** LOOP A SPLY/RTN OPEN RACS PB. ___ d. **PRESS** LOOP B SPLY/RTN OPEN RACS PB. e. **OBSERVE** the following indications: ___ • HV-9530A1/A3 CLOSED ___ • HV-9530B1/B3 CLOSED ___ • HV-9530A2/A4 OPEN ___ • HV-9530B2/B4 OPEN ___ f. **OPEN** HV-9532-1 AND HV-9532-2.

RETAINMENT OVERRIDE			
CONDITION		**ACTION**	
I. Drywell Pressure is ≥ 1.5 psig and rising.	___	**I.a**	**REDUCE** Recirc. Pump Speed to MINIMUM.
Date/Time: ___________	___	**I.b**	**LOCK** the Mode Switch in Shutdown.
II. Drywell Pressure is ≥ 1.68 psig.	___	**II.a**	**TRIP** the Recirc Pumps within 10 minutes.
Date/Time: ___________			

Facility:	Hope Creek
Vendor:	GE
Exam Date:	2019
Exam Type:	RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295014 AK1.03	
	Importance Rating	3.7	

K/A Statement: Knowledge of the operational implications of the following concepts as they apply to INADVERTENT REACTIVITY ADDITION: Shutdown margin.

Question: RO #23

Given:

- The plant is shutdown.
- The reactor head is removed but NO fuel has been removed from the vessel.
- The "B" Residual Heat Removal loop is operating in Shutdown Cooling mode.
- Reactor coolant temperature is 82°F and lowering at 10°F/hr.

Which of the following would be the result if reactor coolant temperature is allowed to continue lowering at the same rate for another 90 minutes?

A. The reactor vessel flange thermal stress limits will be exceeded.

B. The RPV administrative cooldown rate limit will be exceeded.

C. The calculated shutdown margin will be invalid.

D. All the conditions required for brittle fracture of the RPV Belt line will be present.

Proposed Answer: C

Explanation (Optional): Precaution 3.6 states "To prevent affecting the Reactor Shutdown Margin DO NOT allow Reactor temperatures to go below 70°F(when fuel is in the vessel). Cooldown below 68°F could result in an invalidation of Shutdown Margin calculations which are based in part on the Reactor being in the shutdown condition; cold i.e. 68°F AND Xenon free."

A: **Incorrect -** There are no flange temperature limits with the head removed and with the head studs tensioned the limit is 79°F.

B: **Incorrect –** 15°F cooldown in 90 minutes does not violate any applicable limitations.

C: **Correct –** See above explanation.

D: **Incorrect –** All the conditions required for brittle fracture of the RPV Belt line have to be present. Brittle fracture requires a stress be present and with the Reactor vented to atmosphere, the pressure element does not exist.

Technical Reference(s): HC.OP-IO.ZZ-0009 (Attach if not previously provided)

Refueling Operations

Proposed References to be provided to applicants during examination: none

Learning Objective: Apply Precautions, Limitations and Notes while executing the REFUELING OPERATIONS Integrated Operating Procedure. (As available)

Question Source: Bank #34270

Modified Bank # (attached parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41 (7)

Comments:

HC.OP-IO.ZZ-0009(Q)

3.6 To prevent affecting the Reactor Shutdown Margin DO NOT allow Reactor temperatures to go below 70°F(when fuel is in the vessel). Cooldown below 68°F could result in an invalidation of Shutdown Margin calculations which are based in part on the Reactor being in the shutdown condition; cold i.e. 68°F AND Xenon free. ___

Facility: Hope Creek

Vendor: GE

Exam Date: 2019

Exam Type: RO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	1	
Group #	2	
K/A #	295015	AK2.11
Importance Rating	3.5	

K/A Statement: Knowledge of the interrelations between INCOMPLETE SCRAM and the following: Instrument air.

Question: RO #24

Given:

- The plant experiences a spurious trip.
- All control rods did NOT fully scram.
- Reactor power is at 18% rated power.
- Scram valves opened.
- Scram pilot valve air header is at 0 psig
- SDV (Scram Discharge Volume) is full.

Which one of the following actions is directed by HC.OP-EO.ZZ-0101A, ATWS-RPV Control using supplemental EOP procedures to perform control rod insertion with the above conditions?

A. De-energize the scram solenoids IAW HC.OP-EO.ZZ-0102.

B. Individually scram control rods IAW HC.OP-EO.ZZ-0303.

C. Isolate and vent the scram air header IAW HC.OP-EO.ZZ-0306.

D. Defeat RPS interlocks and initiate a manual scram IAW HC.OP-EO.ZZ-0320.

Proposed Answer: D

Explanation (Optional): Alternate methods of inserting control rods following a failure-to-scram are listed in the table contained within Step RC/Q-20. The list contains all methods available to insert control rods, and the applicable plant procedures to execute these actions. There is no priority implied by the table order; **information available during the event dictate the sequence** in which the alternate control rod insertion methods are to be performed. (See attached EOP-101A RC/Q-5, 19,20)

A: **Incorrect**- With scram valves opened the scram solenoids are already de-energized.

B: **Incorrect-** With the scram valves opened the scram solenoids are already de-energized. The SRI switches for individual rod scrams de-energize the scram solenoids.

C: **Incorrect-** Scram air header is at 0 psig.

D: **Correct-** With the above conditions and IAW EOP-101A RC/Q-20 Defeating RPS interlocks and resetting the scram with draining of the SDV allows the operator to insert a manual scram to insert the control rods that are still out.

Technical Reference(s): HC.OP-EO.ZZ-0101A-BASES and FC (Attach if not previously provided)

ATWS-RPV Control

Proposed References to be provided to applicants during examination: none

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New **X**

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41 (10)

Comments:

EXECUTE the

Control Rod Insertion

RC/Q-5

RESET ARI, **DEFEAT** ARI logic trips
USE OP-EO.ZZ-320, if necessary

RC/Q-19

INSERT control rods using 1 OR more of the following methods as applicable
(Refer to RE-AB.ZZ-001 for rod insert sequence)

Plant Conditions	Available Methods
Scram valves failed to open	• **ISOLATE** AND **VENT** the scram air header **USE** OP-EO.ZZ-306
Scram valves opened but SDV is full	• **RESET** the scram, **DRAIN** the SDV, AND **INITIATE** a Manual Scram, **DEFEAT** RPS interlocks **USE** OP-EO.ZZ-320, if necessary • **RESTORE** Instrument Air **USE** OP-EO.ZZ-319
All OR some of the scram solenoids failed to deenergize	• **DEENERGIZE** scram solenoids **USE** OP-EO.ZZ-302 • Individually **SCRAM** control rods **NOT** inserted to OR beyond position 02 **USE** OP-EO.ZZ-303
Manual control rod insertion methods	• **BYPASS** RWM AND manually **INSERT** control rods **USE** HC.OP-AB.ZZ-0001 • **RESTORE** Instrument Air **USE** OP-EO.ZZ-319 • **VENT** control rod over piston volumes **USE** OP-SO.BF-002 • **INCREASE** CRD cooling water header differential pressure. It is not necessary to restore differential pressure for each control rod

1

RC/Q-20

7.12 RC/Q-19 Reset ARI, defeat ARI logic trips. Use OP-EO.ZZ-320, if necessary

RC/Q-20 Insert control rods using 1 or more of the following methods as applicable (refer to RE-AB.ZZ-001 for rod insert sequence)

Reactor shutdown on control rod insertion alone is preferable to injecting boron for the following reasons:

- Boron injection contaminates the primary system requiring extensive cleanup and subsequent inspection before continued plant operation is possible.
- If a leak occurs below the elevation of the RPV water level being maintained, boron injection may not be successful in shutting down the reactor.
- A reactor shutdown on boron is not necessarily a stable condition; if boron is subsequently diluted or displaced by a leak or an operational error, the reactor could return to criticality.

If control rods can be inserted sufficiently to shut down the reactor, boron injection may be terminated or avoided altogether. Since initiation of ARI restricts other alternate methods of inserting control rods, it is appropriate to defeat initiation signals which would otherwise prevent resetting ARI.

Alternate methods of inserting control rods following a failure-to-scram are listed in the table contained within Step RC/Q-20. The list contains all methods available to insert control rods, and the applicable plant procedures to execute these actions. There is no priority implied by the table order; information available during the event dictate the sequence in which the alternate control rod insertion methods are to be performed.

Facility: Hope Creek

Vendor: GE

Exam Date: 2019

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295017 AK3.02	
	Importance Rating	3.3	

K/A Statement: Knowledge of the reasons for the following responses as they apply to HIGH OFF-SITE RELEASE RATE: Plant ventilation.

Question: RO #25

Given:

- A gaseous radioactive release at a rate above the ALERT level has occurred.
- HC.OP-EO.ZZ-0103/4, Reactor Building & Rad Release directs restarting the Turbine Building Ventilation if shutdown.

What is the BASES behind this action?

A. To provide an elevated, monitored release point.

B. To provide filtering to reduce radioactive releases.

C. To prevent the radioactive gaseous release.

D. This action is solely for personnel habitability.

Proposed Answer: A

Explanation (Optional): Continued personnel access to the turbine building may be essential for responding to transients which may degrade into emergencies. This building is not an airtight structure, and radioactivity release inside the building would not only limit personnel access, but would eventually lead to an unmonitored ground level release. Operating HVAC preserves building accessibility and discharges radioactivity through an **elevated, monitored release point.**

A: **Correct-** Operating HVAC preserves building accessibility and discharges radioactivity through an elevated, monitored release point.

B: **Incorrect-** The TB HVAC is operated to direct the gases to a path that is elevated and can be monitored.

C: **Incorrect-** The TB is not an airtight building, and operating the ventilation system will not prevent a release.

D: **Incorrect-** While operating the HVAC preserves building accessibility, it also discharges radioactivity through an elevated, monitored release point.

Technical Reference(s): HC.OP-EO.ZZ.0103/4-BASES (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: Given any step in the procedure, describe the reason for performance of that step and/or expected system response to control manipulations prescribed by the step. (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New **X**

Question History:

Question Cognitive Level: Comprehension or Analysis

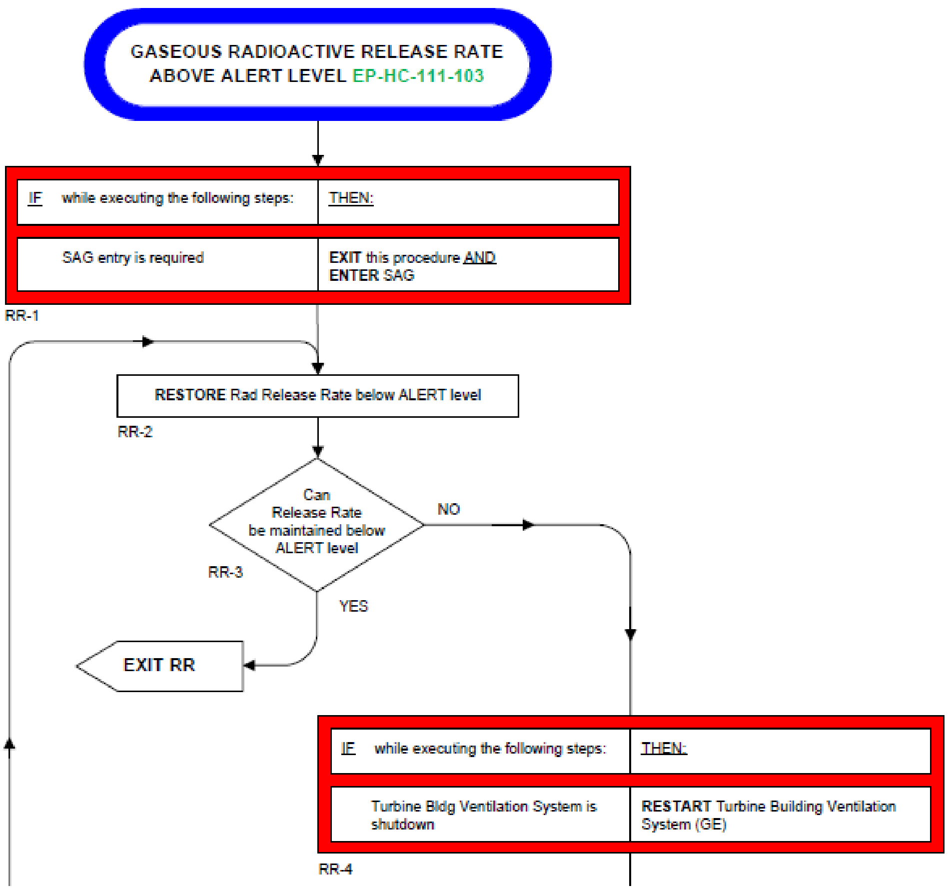
10 CFR Part 55 Content: 55.41 (10)

Comments:

10.3 RR-4 OVERRIDE

If while executing the following steps: Turbine Building Ventilation System is shutdown, then restart Turbine Building Ventilation System (GE).

Continued personnel access to the turbine building may be essential for responding to transients which may degrade into emergencies. This building is not an airtight structure, and radioactivity release inside the building would not only limit personnel access, but would eventually lead to an unmonitored ground level release. Operating HVAC preserves building accessibility and discharges radioactivity through an elevated, monitored release point.



103/4

Facility: Hope Creek

Vendor: GE

Exam Date: 2019

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295029 EA1.03	
	Importance Rating	2.9	

K/A Statement: Ability to operate and/or monitor the following as they apply to HIGH SUPPRESSION POOL WATER LEVEL: RHR/LPCI.

Question: RO #26

Given:

- The plant experiences a transient.

Current plant conditions:

• Reactor power	Reactor shutdown
• Reactor pressure	25 psig and stable
• Reactor level	+40 inches and slowly lowering
• Drywell pressure	1.53 psig and slowly rising
• Suppression pool level	87" and stable
• Suppression pool temp	102°F and slowly rising

Which of the following actions must be taken to control Suppression Pool Level IAW EOP's?

A. Lower suppression pool level using RCIC.

B. Lower suppression pool level using Core Spray.

C. Lower suppression pool level using RHR Loop 'B' to Radwaste.

D. Terminate injection to the RPV from Feedwater.

Proposed Answer: C

Explanation (Optional): When suppression pool level rises above the Technical Specification upper limit, EOP-102 provides direction to use ECCS and / or alignments not normally used to maintain suppression pool water level in general plant procedures. While typically not prudent for use during normal plant operation, these systems and alignments are used in an effort to maintain primary containment in its normal configuration and to prevent level from rising to the point where the more severe actions of reactor scram, termination of drywell sprays, termination of external injection sources, and emergency RPV depressurization will be required. (See attached EOP-102 FC)

A: **Incorrect-** Reactor pressure is too low. RCIC is isolated (64.5 psig).

B: **Incorrect-** Level reduction is required because level is > 78.5". Core Spray is used if suppression pool level is low.

C: **Correct-** Lower suppression pool level using RHR B to Radwaste IAW SP/L -12.(See attached)

D: **Incorrect-** SPL is below 124"

Technical Reference(s): HC.OP-EO.ZZ-0102FC (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: Given any step of the procedure, determine the reason for performance of that step and/or predict expected system response to control manipulations prescribed by that step (As available)

Question Source: Bank #34294

Modified Bank # (Note changes or attach parent)

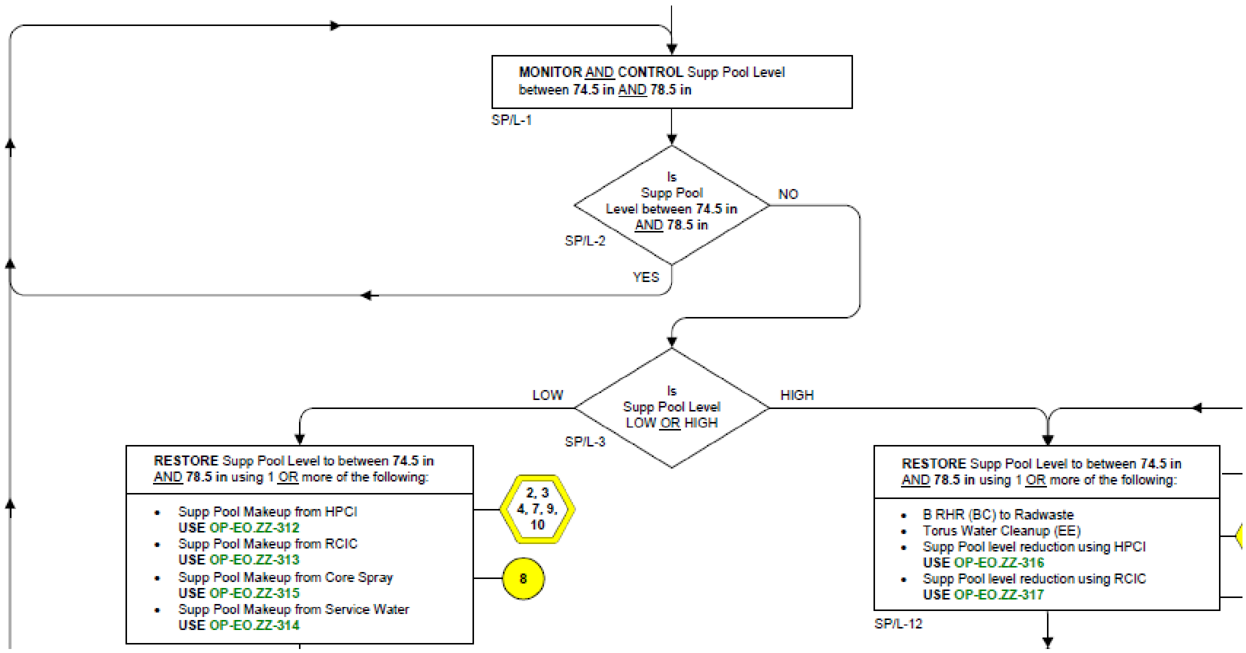
New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41 (10)

Comments:



MONITOR AND CONTROL Supp Pool Level between 74.5 in AND 78.5 in
SP/L-1
Is Supp Pool Level between 74.5 in AND 78.5 in
SP/L-2
NO
YES
Is Supp Pool Level LOW OR HIGH
SP/L-3
LOW
HIGH
RESTORE Supp Pool Level to between 74.5 in AND 78.5 in using 1 OR more of the following:
Supp Pool Makeup from HPCI USE OP-EO.ZZ-312
Supp Pool Makeup from RCIC USE OP-EO.ZZ-313
Supp Pool Makeup from Core Spray USE OP-EO.ZZ-315
Supp Pool Makeup from Service Water USE OP-EO.ZZ-314
2, 3 4, 7, 9, 10
8
RESTORE Supp Pool Level to between 74.5 in AND 78.5 in using 1 OR more of the following:
B RHR (BC) to Radwaste
Torus Water Cleanup (EE)
Supp Pool level reduction using HPCI USE OP-EO.ZZ-316
Supp Pool level reduction using RCIC USE OP-EO.ZZ-317
SP/L-12

Facility: Hope Creek

Vendor: GE

Exam Date: 2019

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295036 EA2.02	
	Importance Rating	3.1	

K/A Statement: Ability to determine and/or interpret the following as they apply to SECONDARY CONTAINMENT HIGH SUMP/AREA WATER LEVEL: Water level in the affected area.

Question: RO #27

Given:

- The reactor is operating at 100% rated power.
- OHA "CORE SPRAY PUMP ROOM FLOODED" is illuminated.
- An investigation reveals that the "D" Core Spray pump room floor level is 2.5 inches and stable.
- All RPV parameters are normal.

TABLE 2		
Area Description & Room Number	**Column 1 Max Normal Op Floor Level**	**Column 2 Max Safe Op Floor Level**
CRD Pump Room (4202)	1 in	4 1/2 in (25 min continuous running)
HPCI (4111)	1 in	4 1/2 in (30 min continuous running)
Core Spray Pump Rooms A(4118) & C(4116)	1 in	4 1/2 in (15 min continuous running)
RHR Pump Rooms A(4113) & C(4114)	1 in	4 1/2 in (20 min continuous running)
SACS A & C (4309)	1 in	4 1/2 in (INVESTIGATE)
RCIC Pump Room (4110)	1 in	4 1/2 in (17 min continuous running)
Core Spray Pump Rooms B(4104) & D(4105)	1 in	4 1/2 in (15 min continuous running)
RHR Pump Rooms B(4109) & D(4107)	1 in	4 1/2 in (20 min continuous running)
SACS B & D (4307)	1 in	4 1/2 in (INVESTIGATE)

IAW HC.OP-EO.ZZ-0103/4, Reactor Building & Rad Release, the operators must ___________.

A. immediately commence a normal reactor shutdown.

B. ensure all available sump pumps are in operation.

C. emergency depressurize the reactor.

D. runback reactor recirculation and initiate a manual scram.

Proposed Answer: B

Explanation (Optional): EOP-103 Table 2 provides a listing of areas that have been defined per safety function within the Reactor Building that are of concern for flooding. Column 1 values represent Maximum Normal Operating Floor Levels which coincide with the area room flooded alarm levels. The initial action taken to control reactor building floor water levels employs the same method typically used during normal plant operations: monitoring its status, isolating the source of flooding if possible, and using the building sumps to restore reactor building floor water levels to below area alarm thresholds. (See attached EOP-103 BASES RB-12)

A: **INCORRECT –** only required if above a Max Safe and the unable to isolate a system that is not a reactor coolant system. (See attached EOP-103 RB-14)

B: **CORRECT –** IAW step RB-12. (See attached EOP-103)

C: **INCORRECT-** only required if unable to isolate a reactor coolant leak into the building and two or more areas exceed the max safe parameter. (See attached EOP-103 RB-18)

D: **INCORRECT** - only required if unable to isolate a reactor coolant system leak and approaching a max safe parameter. (See attached EOP-103 RB-16 and 17)

Technical Reference(s): HC.OP-EO.ZZ-0103/4 BASES and FC (Attach if not previously provided)

Proposed References to be provided to applicants during examination: Table 2 of EOP-103/4 in the stem of question

Learning Objective: Given any step in the procedure, describe the reason for performance of that step and/or expected system response to control manipulations prescribed by the step. (As available)

Question Source: Bank #34565

Modified Bank # (Note changes or attach parent)

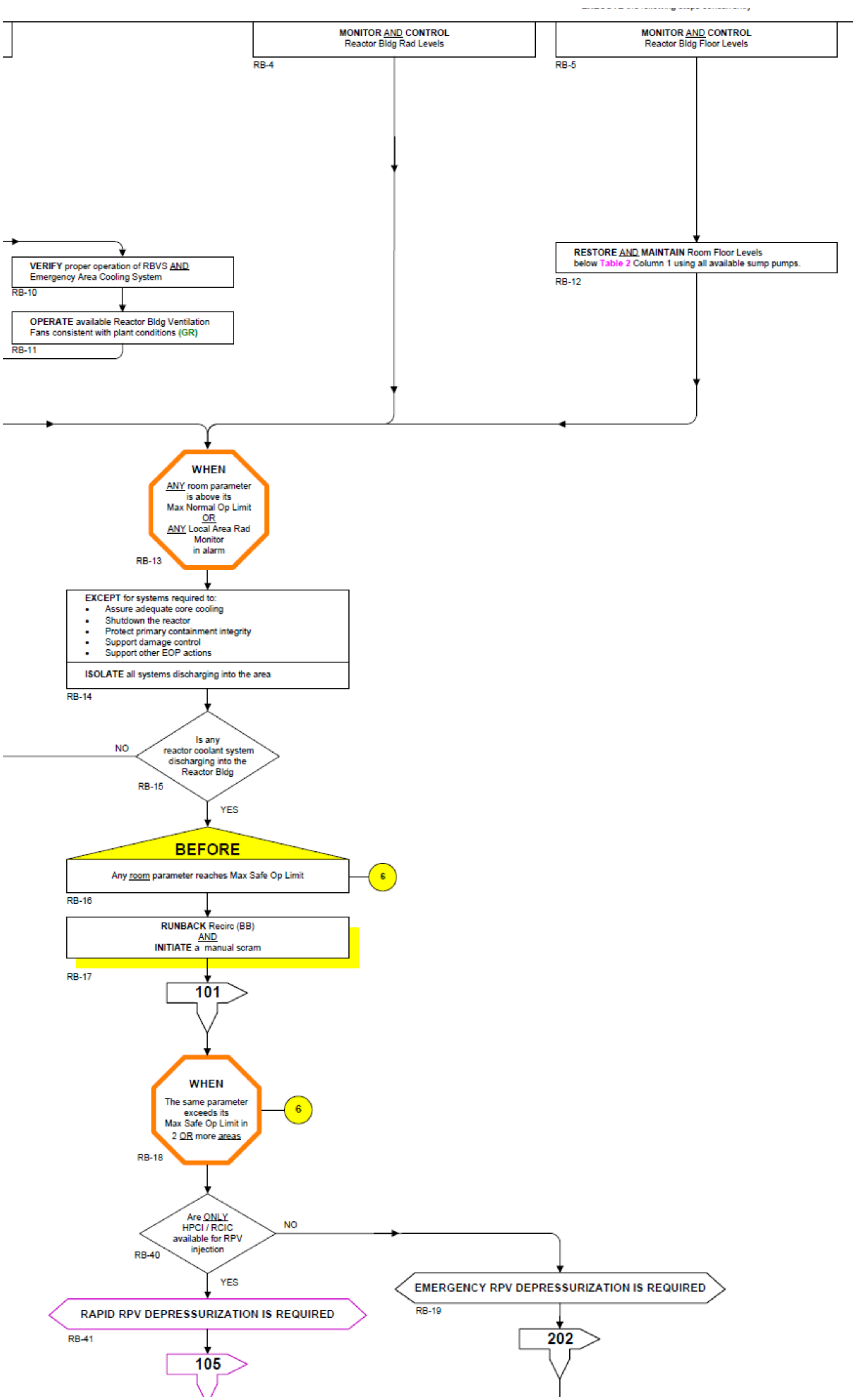
New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41 (10)

Comments:


MONITOR AND CONTROL
Reactor Bldg Rad Levels
RB-4
MONITOR AND CONTROL
Reactor Bldg Floor Levels
RB-5
VERIFY proper operation of RBVS AND Emergency Area Cooling System
RB-10
OPERATE available Reactor Bldg Ventilation Fans consistent with plant conditions (GR)
RB-11
RESTORE AND MAINTAIN Room Floor Levels below Table 2 Column 1 using all available sump pumps.
RB-12
WHEN
ANY room parameter is above its Max Normal Op Limit
OR
ANY Local Area Rad Monitor in alarm
RB-13
EXCEPT for systems required to:
• Assure adequate core cooling
• Shutdown the reactor
• Protect primary containment integrity
• Support damage control
• Support other EOP actions
ISOLATE all systems discharging into the area
RB-14
Is any reactor coolant system discharging into the Reactor Bldg
NO
YES
RB-15
BEFORE
Any room parameter reaches Max Safe Op Limit
6
RB-16
RUNBACK Recirc (BB)
AND
INITIATE a manual scram
RB-17
101
WHEN
The same parameter exceeds its Max Safe Op Limit in 2 OR more areas
6
RB-18
Are ONLY HPCI / RCIC available for RPV injection
NO
YES
RB-40
EMERGENCY RPV DEPRESSURIZATION IS REQUIRED
RB-19
202
RAPID RPV DEPRESSURIZATION IS REQUIRED
RB-41
105

7.5 RB-5 Monitor and control reactor building floor levels

RB-12 Restore and maintain room floor levels below Table 2 Column1 using all available sump pumps.

EOP-103 Table 2 provides a listing of areas that have been defined per safety function within the Reactor Building that are of concern for flooding. Column 1 values represent Maximum Normal Operating Floor Levels which coincide with the area room flooded alarm levels.

The initial action taken to control reactor building floor water levels employs the same method typically used during normal plant operations: monitoring its status, isolating the source of flooding if possible, and using the building sumps to restore reactor building floor water levels to below area alarm thresholds. Step RB-5 thus provides a smooth transition from general plant procedures to emergency operating procedures, and assures that the normal method of reactor building water level control is attempted in advance of initiating more complex actions to terminate increasing reactor building water levels.

As long as reactor building floor water levels remain below maximum normal operating limits no further operator action is required in this subsection of the guideline other than continuing to monitor and control reactor building radiation levels using available non-emergency methods.

Further action to scram the reactor is based on an inability to isolate a leak into the RB from a primary system, and action to depressurize is based on loss of safe shutdown function as defined by "areas"

Facility:	Hope Creek
Vendor:	GE
Exam Date:	2019
Exam Type:	RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	203000 G2.1.25	
	Importance Rating	3.9	

K/A Statement: Ability to interpret reference materials, such as graphs, curves, tables, etc.- RHR/LPCI: Injection Mode.

Question: RO #28

Given:

- A severe transient has occurred.

Current plant conditions:

- Suppression Chamber pressure: 10 psig
- Suppression Pool temperature: 240 °F
- Suppression Pool level at 38"
- Reactor pressure: 100 psig
- RPV Water Level is –100 inches and rising.
- RHR "A" pump flow: 10,000 gpm
- Core Spray "B" pump Flow: 1500 gpm
- All other low pressure ECCS pump are NOT in service.

Determine if Net Positive Suction Head (NPSH) requirements are being met.
[Reference attached]

A. There is sufficient NPSH for both the "A" RHR pump and the "B" Core Spray Pump.

B. There is sufficient NPSH for the "A" RHR pump ONLY.

C. There is sufficient NPSH for the "B" Core Spray Pump ONLY.

D. There is NOT sufficient NPSH for any pump.

Proposed Answer: A

Explanation (Optional): The limiting temperature for RHR pump at **10 psig is ~244°F** (see attached graph), since Given temperature is 240°F this puts the pump in the region of **ACCEPTABLE** operation. The limiting temperature for CS pump at **10 psig and 1500 gpm = ~248°F** (see attached graph). Since given temperature = 240°F this puts the B CS pump in the area of **ACCEPTABLE** operation.

A: **Correct-** In the acceptable region of EOP caution 2 (see above and attached EOP caution 2).

B: **Incorrect-** Both RHR and Core Spray.

C: **Incorrect-** Both RHR and Core Spray.

D: **Incorrect-** In the acceptable region of EOP caution 2

Technical Reference(s): HC.EOP Caution 2 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: EOP Caution 2

Learning Objective: Given plant conditions involving a Degraded ECCS Performance/Loss of NPSH, summarize required actions to mitigate the condition. (As available)

Question Source: Bank #115458

Modified Bank # (Note changes or attach parent)

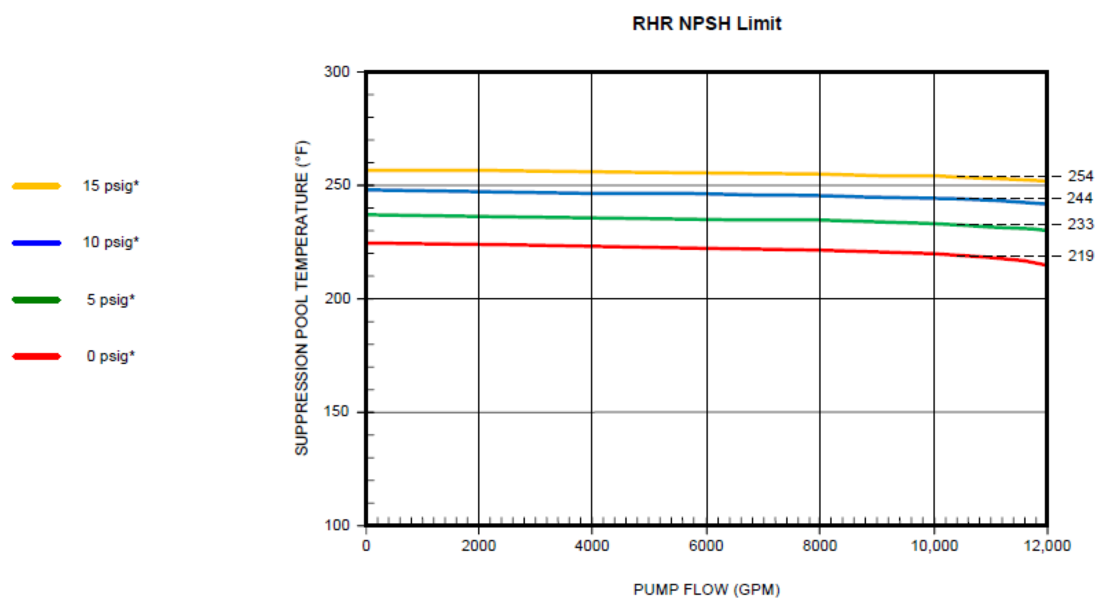
New

Question History:

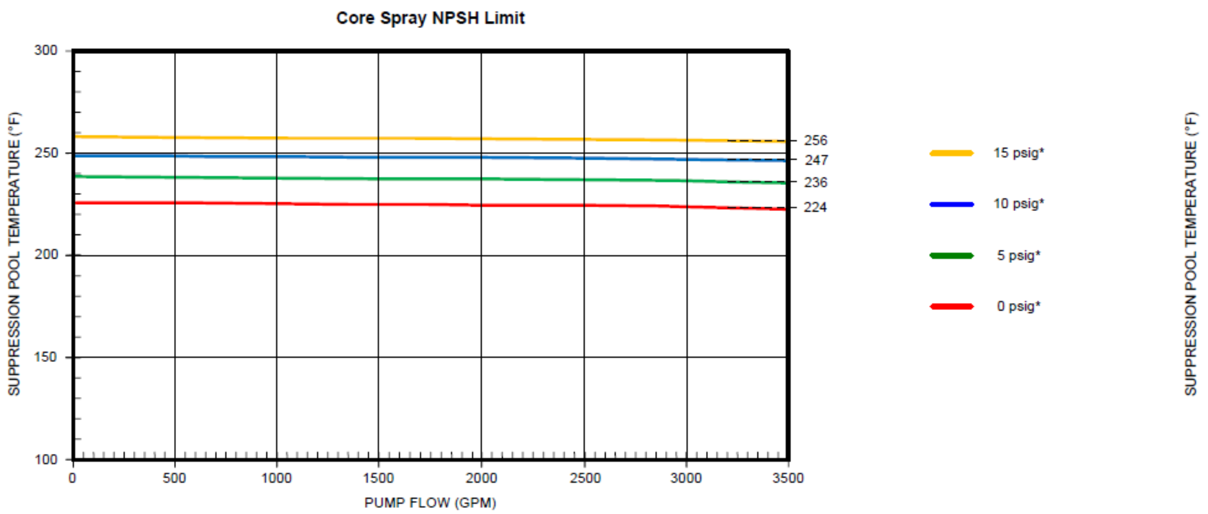
Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41 (10)

Comments:



chamber overpressure with supp pool level @ 38"



** supp chamber overpressure with supp pool level @ 38"*

Facility: Hope Creek

Vendor: GE

Exam Date: 2019

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	205000 G2.4.47	
	Importance Rating	4.2	

K/A Statement: Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference materials.- Shutdown Cooling.

Question: RO #29

Given:

- A reactor cooldown was in progress with ‘B’ RHR in the Shutdown Cooling (SDC) mode of operation.
- SACS cooling to ‘B’ RHR HX valve EG-HV-2512B just inadvertently closed.

At 1500:

- Reactor coolant pressure is at 52 psig.
- Reactor coolant temperature is at 300˚F.
- Reactor coolant temperature is rising at 2˚F/min.

At which one of the following times will SDC automatically isolate assuming NO operator action and constant heat up rate?

A. 1506

B. 1513

C. 1523

D. 1526

Proposed Answer: B

Explanation (Optional): Using Steam Tables

A: **Incorrect** - ~4˚F/min.

B: **Correct** - 300˚F ~67 psia~ 52 psig. The Isolation **setpoint is 82 psig** ~ 97 psia ~ 326˚F. 326-300 = 26 ˚F rise at 2˚F/min = 13 minutes = 1513

C: **Incorrect** – 97 psia -52 psia = 45 psi/2 = 22.5 min

D: **Incorrect-** 1˚F/min.

Technical Reference(s): HC.OP-AB.RPV-0009 (Attach if not previously provided)

Shutdown Cooling

Steam Tables

Proposed References to be provided to applicants during examination: Steam Tables (part of exam utensils package)

Learning Objective: Interpret and apply charts, graphs and tables contained within Shutdown Cooling. (As available)

Question Source: Bank #119069

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41 (10)

Comments:

IMMEDIATE OPERATOR ACTIONS

NONE

AUTOMATIC ACTIONS

IF	THEN
Reactor Pressure > 82 psig	The following valves cannot be opened from the Control Room OR their Remote Shutdown controls: • HV-F008 • HV-F009 • HV-F015A • HV-F015B
Reactor Pressure > 82 psig OR Reactor Level < 12.5" OR Loss of EITHER RPS Bus AND RSP Takeover Switches in NORMAL	The following valves will isolate: • HV-F008* • HV-F009* • HV-F015A* • HV-F015B*

*If GP.SM-0001 has been performed, these isolations are bypassed.

Facility: Hope Creek

Vendor: GE

Exam Date: 2019

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	205000 K1.08	
	Importance Rating	3.9	

K/A Statement: Knowledge of the physical connections and/or cause/effect relationships between SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE) and the following: LPCI.

Question: RO #30

Given:

- The plant is currently in Operational Condition 4
- RHR loop "B" aligned in the shutdown cooling mode.
- The RHR A(B) FPC MODE SUCT VLV CLOSURE TRIP OVERRIDE keylock switch is in the 'NORM' position.
- HC.OP-GP.SM-0001, Defeating Isolation Signals During Shutdown Cooling Operations has NOT been completed.

Then:

- A crack develops in the suction piping at the "B" reactor recirculation loop connection causing reactor water level to lower.
- Reactor water level begins to lower from normal water level to -135” and lowering.

Assuming NO operator action, which of the following statements describes the automatic response of the "B" RHR loop?

A. F008, F009 (SDC Suction) and F015B (SDC Return) remain open.
F017B (LPCI Injection) and F048B (HX Bypass) open.
The "B" RHR pump remains running and injects to the RPV.

B. F008, F009 (SDC Suction) and F015B (SDC Return) shut.
The "B" RHR pump trips.
F017B (LPCI Injection) and F048B (HX Bypass) open.
The "B" RHR pump restarts and injects to the RPV.

C. F008, F009 (SDC suction) and F015B (SDC Return) remain shut.
The “B” RHR pump trips.
F017B (LPCI Injection) and F048B (HX Bypass) open.
The "B" RHR pump does not attempt to restart.

D. F008, F009 (SDC Suction) and F015B (SDC Return) shut.
The "B" RHR pump trips.
F017B (LPCI Injection) and F048B (HX Bypass) open.
The "B" RHR pump attempts to restart and immediately trips.

Proposed Answer: D

Explanation (Optional): Since RPV Level 3 +12.5 inches would not isolate the leak because of the leak location, RPV level would lower until -129" is reached. B RHR would trip when F008 and F009 close (see attached figure 1). Because of the SDC lineup, the B pump would attempt to start and immediately trip. (see attached figure 1). SM-0001 not being completed at this time allows the NSSSS isolations to occur.

A: **Incorrect** – F008,F009, and F015 will isolate on the low level (+12.5”)

B: **Incorrect** – The “B” RHR pump will trip due to no suction path and will trip on restart due to no override in place.

C: **Incorrect** – The “B” RHR pump will attempt to restart however it will immediately trip.

D: **Correct-** See above explanation along with the attached SOP BC-0001 and 0002.

Technical Reference(s): HC.OP-SO.BC-0001 RHR SOP (Attach if not previously provided)

HC.OP-SO.BC-0002 Decay Heat Removal

Proposed References to be provided to applicants during examination: none

Learning Objective: From memory, state the purpose and summarize the different operational modes of the Residual Heat Removal System, system including the following components: (As available)

RHR Pumps

Given procedure HC.OP-SO.BC-0001, "Residual Heat Removal System Operation" and HC.OP-SO.BC-0002, "Decay Heat Removal Operation", explain the listed prerequisites, precautions, and/or limitations during operation.

Question Source: Bank #35900

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41 (7)

Comments:

3.2.15. WHEN the Reactor Vessel temperature is below 200°F, the time the RHR System is operating in the Shutdown Cooling mode WITH the Reactor Vessel level ≤ 80 inches should be minimized. ___

3.2.16. The RHR Shutdown Cooling operability requirements of T/S 3.4.9 and 3.9.11 shall be observed. ___

3.2.17. The Depressurization Systems (Suppression Pool) operability requirements of T/S 3/4.6.2 shall be observed. ___

3.2.18. The valves listed in Table BC-001 will isolate upon receipt of ANY of the following signals: ___

- RPV Level Low (Level 3, + 12.5").
- RPV High Pressure (82 psig).
- Manual Isolation.

TABLE BC-001		
EQUIPMENT NUMBER	**DESCRIPTION**	**POSITION**
BC-HV-F008	SD COOLING OUTBD ISLN MOV	CLOSE
BC-HV-F009	SD COOLING INBD ISLN MOV	CLOSE
BC-HV-F015A(B)	RHR LOOP A(B) RET TO RECIRC	CLOSE

3.2.19. The valves listed in Table BC-002 will isolate upon receipt of ANY of the following signals: ___

- RPV Level Low (Level 3, + 12.5").
- Drywell Pressure High (≥ 1.68 psig).
- Manual Isolation.

TABLE BC-002		
EQUIPMENT NUMBER	**DESCRIPTION**	**POSITION**
BC-HV-F049	RHR LOOP B DSCH TO LIQ RW ISLN MOV	CLOSE
BC-HV-F040	RHR LOOP B DSCH TO LIQ RW ISLN MOV	CLOSE

3.2.20. All pump operations should be performed IAW OP-HC-108-106-1001, Equipment Operational Control. ___

3.2.15. The following RHR Pump flow limitations shall be complied with at all times: **[70044148]** ___

Continuous Duty - Pump flow should be >1,800 gpm under continuous duty. Continuous duty is defined as operation of > 3 hours in a 24 hour period.

Low Flow Duty - Pump flow between 500 - 1,800 gpm is limited to 3 hours in a 24 hour period.

Low Low Flow Duty - Pump flow between 400 - 500 gpm is limited to 1.5 hours in a 24 hour period.

Starts/Stops - During pump starts and stops, flow can be very low. Pump flow < 400 gpm is limited to ½ hour in a 24 hour period.

3.3 Interlocks

3.3.1. WHEN in AUTO, LPCI will initiate upon receipt of EITHER of the following signals:

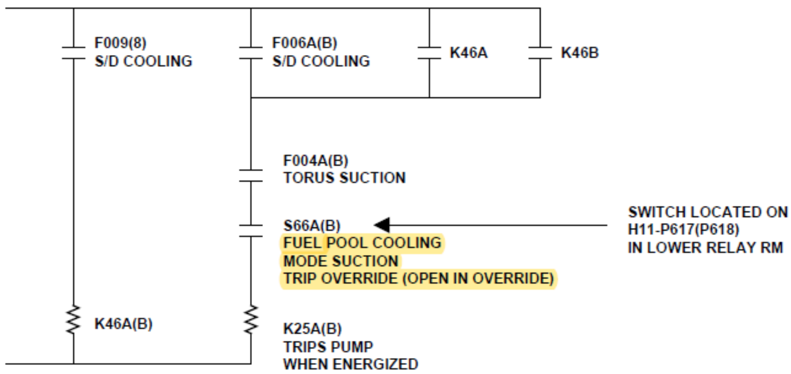
- High Drywell pressure (≥ 1.68 psig) ___
- Low Low Reactor water level (Level 1, -129") ___

TABLE 1
VALVE TABLE

Valve	Power	Auto Actuations	Interlocks
F017A B C D (LPCI Injection)	10B212 10B222 10B232 10B242	Auto open on LPCI init. if following condition exist: (1) LPCI init. present in respective RHR loop logic (2) Power is avail. on respective pump bus (3) Reactor press. less than 450 psig (may be overridden by "AUTO OPEN OVRD")	Rx press. must be less than 450 psig to open valve either MAN or AUTO. Placing Ch. B RSP to EMERG will close F017B & inhibit all associated automatic & OVLD protection Features. The LPCI injection valve must be 100% closed (in the respective loop with a LPCI initiation signal present) to OPEN: F027A(B), F024A(B), F016, F021
F047A B (HX Inlet)	10B212 10B222	None	Placing Ch. B RSP to EMERG transfers F047B to RSP
F003 A B (HX Inlet)	10B212 10B222	Throttle Valve	Placing Ch. B RSP to EMERG transfers F003B to RSP
F048A B (HX Bypass)	10B212 10B222	OPEN upon receipt of a LPCI initiation signal	Valve operation is inhibited for 3 minutes upon receipt of LPCI init. signal, valve interlocked open Ch. B RSP to EMERG transfers F048B to the RSP. OVLD protection can be bypassed in the OPEN direction (BYP IN OPEN)

HC.OP-SO.BC-0002(Q)

FIGURE 1
RHR PUMP SUCTION VALVE TRIP INTERLOCKS



A(B) RHR PUMP SUCTION VALVE TRIP INTERLOCK

NOTE: ALL VALVE CONTACTS ARE FROM LIMIT SWITCH POSITIONS AND CLOSE WHEN THE VALVES ARE NOT FULLY OPEN.
IF ANY VALVES IN THE SUCTION PATH ARE NOT FULLY OPEN AND THE FPC MODE SUCTION IS NOT OVERRIDDEN (A&B ONLY),
THEN THE RESPECTIVE RHR PUMP BREAKER WILL TRIP IMMEDIATELY UPON BREAKER CLOSURE.

HC.OP-SO.BC-0002(Q)

3.3.7. To ensure an adequate suction path and allow the RHR pump to operate the following must be satisfied: (**REFER** to Figure #1)

- For RHR pump C or D: ___
  F004C(D) must be 100% OPEN.

- For RHR pump A or B EITHER: ___
  (Note: may be overridden)

  HV-F004A(B) must be open
  OR
  HV-F006A(B), HV-F008 and HV-F009 must ALL be OPEN

If this valve alignment is not satisfied, the respective pump breaker will trip immediately upon breaker closure, or if the pump is running and the valves are < 100% open the pump breaker will trip. ___

Facility: Hope Creek

Vendor: GE

Exam Date: 2019

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	206000 K2.01	
	Importance Rating	3.2	

K/A Statement: Knowledge of electrical power supplies to the following: System Valves: BWR-2,3,4.

Question: RO #31

What is the power supply to the HPCI Inboard Steam Supply Isolation Valve (FD-HV-F002) and the HPCI Outboard Steam Supply Isolation Valve (FD-HV-F003)?

	HV-F002	HV-F003
A.	10B212	10B222
B.	10B232	10B212
C.	10B222	10B212
D.	10B212	10B232

Proposed Answer: B

Explanation (Optional) See attached load list for 10B212 and 10B232.

A: **Incorrect:** HPCI steam isolation valves are ‘A’ and ‘C’ channel 480 1E VAC.

B: **Correct:** F002 is ‘C’ channel 10B232203 breaker; F003 is ‘A’ channel 10B212053 breaker.

C: **Incorrect:** HPCI steam isolation valves are ‘A’ and ‘C’ channel 480 1E VAC.

D: **Incorrect:** F002 is ‘C’ channel 10B232203 breaker; F003 is ‘A’ channel 10B212053 breaker.

Technical Reference(s): E-0021-1 sheet 1 and 6 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: From memory summarize/identify the interrelationship(s) between the HPCI System and any of the following: (As available)

a. 480 VAC/120VAC Class 1E Distribution System.

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New **X**

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41 (7)

Comments:

480 VOLT MCC
10B212

REACTOR AREA (43421) MOTOR CONTROL CENTER 10B212 LOC

2-350 KCMIL (T)
830
E-0018-1(Q) SH. 1
S/U-PH
43421042
47410
47411

MCC UNIT NO.	NAMEPLATE (STYLE 2) LINE 1 & LINE 2 { SERVICE DESCRIPTION; LINE 3: EQUIP. NO. CKT. BRKR. NO.	SCHEME NO. / S/U SYS. NO.	LOAD HP KW OR KVA	FLA / LRA	CIRCUIT BREAKER TYPE (SEE ABBREV-IATIONS)	FRAME AMPS	CONT AMPS	MAGNETIC TRIP ② RANGE AMP	SETTING AMP	SETTING POS	STARTER NEMA SIZE	TYPE (SEE ABBREV-IATIONS)	OVERLOAD HEATER ④
011	INCOMING LINE	1B0155 / PH											
012	CORE SPRAY PMP SUCT 1BE-HV-F001A 52-212012	1Q0612 / BE	0.7	2.3 / 11.9	IM	150	7	21-70	28	B	1	FVR	FH25
013	RHR HX SHELL SIDE OUTLET 1BC-HV-F003A 52-212013	1Q0820 / BC	9.9	19.5 / 130	IM	150	50	150-500	250	C	2	FVR	FH48
014	MOV OVERLOAD BYPASS AUXILIARY CIRCUITS 10B212014	1B0210 / PH		–					–	–			
021	RWCU INBD SUCT ISLN 1BG-HV-F001 52-212021	1Q2013 / BG	1.6	4.0 / 25.3	IM	150	15	45-150	45	A	1	FVR	FH30
022	SPARE				TM	150	15	NON ADJUSTABLE	–	–	–	–	–
023	SPARE			–	TM	150	70	NON-ADJUSTABLE	–	–	3	FVNR	C-H TO SHORT THIS HTR
031	RHR PMP 1AP202 SUCT 1BC-HV-F004A 52-212031	1Q0821 / BC	1.0	2.8 / 16	IM	150	7	21-70	35	C	1	FVR	FH27
032	AVH213 INLET VANES 1GU-ED-9377A 52-212032	1V0691 / GU	0.17	0.5 / 2.5	IM	150	7	21-70	21	A	1	FVNR	FH09
033	SPARE	1V0689 / GU			TM	150	50	NON-ADJUSTABLE	–	–	–	–	–
034	CORE SPRAY OUTBOARD INJECT 1BE-HV-F004A 52-212034	1Q0616 / BE	12.8	18.4 / 143	IM	150	50	150-500	200	B	2	FVR	FH47
041	RHR PMP 1AP202 MIN FL BYP 1BC-HV-F007A 52-212041	1Q0823 / BC	0.33	0.75 / 5.25	IM	150	7	21-70	21	A	1	FVR	FH13
042	208Y/120V AC XFMR TO DIST PNL 10Y201 10X201 52-212042	1B0315 / PN	15 KVA	—	TM	150	25	NON-ADJUSTABLE	–	–	–	–	–
043	SPACE												
044	CORE SPRAY INBOARD INJECT 1BE-HV-F005A 52-212044	1Q0620 / BE	12.8	18.4 / 143	IM	150	50	150-500	200	B	2	FVR	FH47
051	CORE SPRAY TEST RTN 1BE-HV-F015A 52-212051	1Q0622 / BE	2.6	5.9 / 38.0	IM	150	15	45-150	75	C	1	FVR	FH35
052	RHR INJECTION 1BC-HV-F017A 52-212052	1Q0827 / BC	7.8	11.4 / 94.3	IM	150	30	90-300	120	B	1	FVR	FH42
053	HPCI ST SPLY OUTBD ISLN 1FD-HV-F003 52-212053	1Q0714 / FD	3.9	7.0 / 48.0	IM	150	30	90-300	90	A	1	FVR	FH37
	FRVS VENT SYSTEM												

NODE NODE NODE THE MULTIPLE NODE SYMBOLS INDICATE THE SUPPLYING FORCE AND/OR PRIMARY OR SECONDARY POINTS OF REFERENCE. THESE NODES ARE REFERENCED IN THE LOAD MANAGEMENT DATA BASE.

REACTOR AREA (CONTINUED) (43621) MOTOR CONTROL CENTER 10B232 LOCATION: REACTOR BLDG., EL. 102'-0"

203	HPCI STEAM SPLY ISLN 1FD-HV-F002 52-232203	1Q0720 / BJ	3.9	7.0 / 48.0	IM	150	30	90-300	90	A	1	FVR	FH36	#12	HS-F002 / 10C650	E-2076-0	431636

Facility:	Hope Creek
Vendor:	GE
Exam Date:	2019
Exam Type:	RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	209001 K3.01	
	Importance Rating	3.8	

K/A Statement: Knowledge of the effect that a loss or malfunction of the LOW PRESSURE CORE SPRAY SYSTEM will have on following: Reactor water level

Question: RO #32

Given:

- 1E 125 VDC bus 1AD417 is out of service due to an electrical fault.

Then:

- A leak develops inside the drywell from an unidentified source.
- Drywell pressure is at 7.3 psig and rising.
- RPV level is at –118" and lowering.
- RPV pressure is at 300 psig and slowly lowering.

What Core Spray system pumps are injecting into the RPV?

A. 'A' and 'C' ONLY.

B. 'B' and 'D' ONLY.

C. 'B', 'C', and 'D' ONLY.

D. 'A', 'B', 'C', and 'D'.

Proposed Answer: **B**

Explanation (Optional): 125 VDC Class 1E System supplies power to Divisions I, II, III, and IV CSS logic circuits: Division I 1AD417 breaker 9 (HII 617 logic panel) (see attached drawing). The "A" CS Logic would not initiate with a loss of AD417 preventing the 'A' CS pump from starting and **the injection valve from opening. RPV pressure is below shutoff head (380 psig) and RPV injection will be from the B Core Spray loop with the 'B' and 'D' pumps.**

A: Incorrect- RPV injection will be from both of the B Core Spray loop pumps 'B' and 'D'. **The 'C' pump would be running, however the injection valve would be closed and the 'A' pump would not be running. Therefore no injection from the "A" Core Spray loop.**

B: Correct-. See above explanation.

C: Incorrect-. Three Core Spray pumps would be running for this total injection. However, no injection would occur from the 'A' loop ('A' and 'C' Core Spray pumps) **with the injection valve closed.**

D: Incorrect-. The "A" CS Logic would not initiate with a loss of AD417 preventing the 'A' CS pump from starting and **the injection valve from opening**.

Technical Reference(s): PN1-E21-1040-0383 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: Given a labeled diagram/drawing of the Core Spray System controls/indication bezel:(NCO and Above) (As available)
Explain the function of each indicator.
Assess plant conditions that will cause the indicators to light or extinguish.
Determine the effect of each control switch on the Core Spray System.
Assess plant conditions or permissives required for the control switches to perform their intended functions.

Question Source: Bank #70492

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41 (7)

Comments:

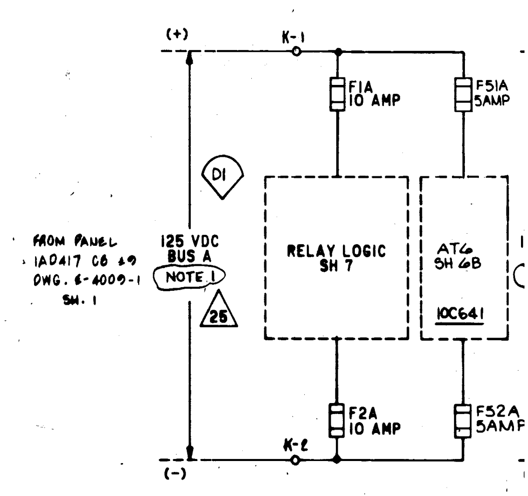


FIG. 3
H11-P617

Facility: Hope Creek

Vendor: GE

Exam Date: 2019

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	209001 K4.07	
	Importance Rating	2.8	

K/A Statement: Knowledge of LOW PRESSURE CORE SPRAY SYSTEM design feature(s) and/or interlocks which provide for the following: Pump operability testing.

Question: RO #33

Given:

- ‘A’ Core Spray Pump is running in the test return mode for a scheduled surveillance.

Then, a plant transient results in the following conditions:

- A leak into the drywell with pressure at 2.15 psig and slowly rising.
- Reactor water level at -50 inches and stable.
- Reactor pressure at 690 psig and stable.

Select the system response for these conditions.

A. The ‘A’ Core Spray Pump continues to run on minimum flow, the CSS Full Flow Test Valve (HV-F015A) closes and the CS Loop Injection Valve (HV-F005A) opens.

B. The ‘A’ Core Spray Pump trips, the CSS Full Flow Test Valve (HV-F015A) closes, then the pump restarts and CS Loop Injection Valve (HV-F005A) opens.

C. The ‘A’ Core Spray Pump continues to run on minimum flow, the CSS Full Flow Test Valve (HV-F015A) closes and the CS Loop Injection Valve (HV-F005A) does not reposition.

D. The ‘A’ Core Spray Pump trips, the CSS Full Flow Test Valve (HV-F015A) closes, then the pump restarts and runs on minimum flow and CS Loop Injection Valve (HV-F005A) does not reposition.

Proposed Answer: C

Explanation (Optional): Refer to attached HC.OP-SO.BE-0001 Core Spray SOP interlocks section.

A: **Incorrect.** Injection valve (F005) doesn’t open until Rx press < 461 psig.

B: **Incorrect.** No CS Pump trips for these conditions, injection valve (F005) doesn’t open until Rx press < 461 psig.

C: **Correct.** Test return (F015) closes but nothing else happens until Rx pressure < 461 psig.

D: **Incorrect.** No CS Pump trips for these conditions.

Technical Reference(s): HC.OP-SO.BE-0001 (Attach if not previously provided)

Core Spray System

Proposed References to be provided to applicants during examination: none.

Learning Objective:	Given a set of conditions and a drawing of the controls, instrumentation and/or alarms located in the Control Room, assess the status of the Core Spray System or its components by evaluation of the controls/instrumentation/alarms.	(As available)

Question Source:	Bank #36007	
	Modified Bank #	(Note changes or attach parent)
	New	

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41 (7)

Comments:

3.3 **Interlocks**

3.3.1. Core Spray System auto starts upon the receipt of the following signals:

A. Low-Low Reactor water level (Level 1, -129" wide range) ____

B. High Drywell pressure (1.68 psig) ____

3.3.2. Core Spray Loop A(B) Inboard Isolation Valve HV-F005A(B) is interlocked to open at Reactor pressure < 461 psig with an initiation signal present. ____

3.3.3. Full Flow Test Valve HV-F015A(B) will AUTO Close on Core Spray logic initiation signal, Manual OR Automatic. ____

3.3.4. Core Spray Loop A(B) Inboard Isolation Valve HV-F005A(B) is interlocked with Outboard Isolation valve HV-F004A(B) to prevent manually opening of the HV-F005A(B) UNLESS the HV-F004A(B) is Fully Closed. (No interlock exists to prevent manual opening of HV-F004A(B) IF HV-F005A(B) is open) ____

3.3.5. Core Spray Minimum Flow Valve HV-F031A(B) will automatically open IF a respective loop pump is running
AND discharge flow is low (< 775 gpm).
WHEN flow is > 775 gpm the valve automatically closes. ____

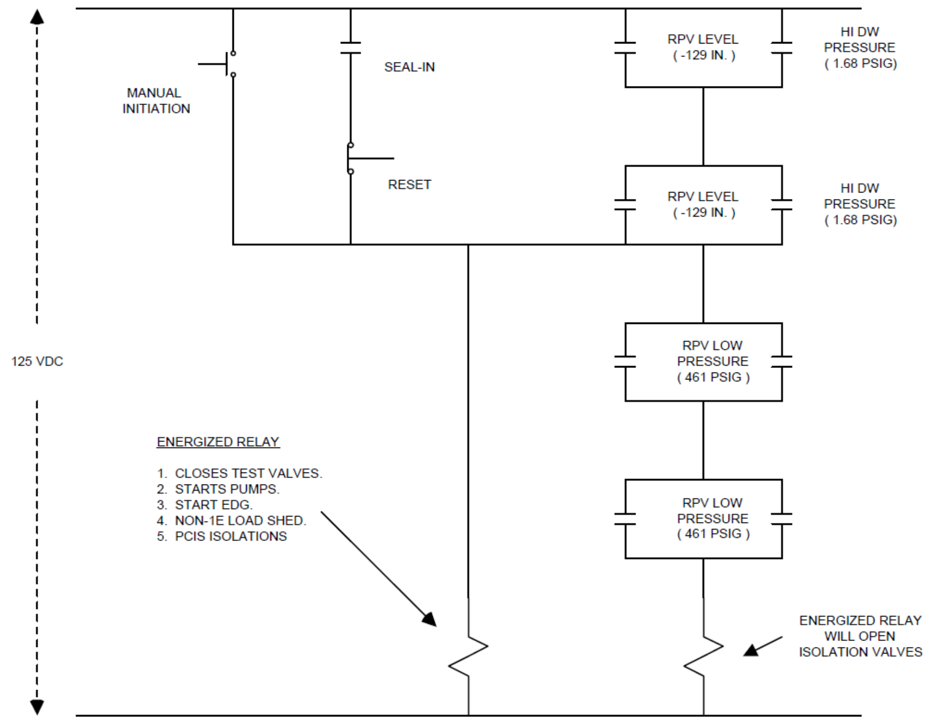
3.3.6. Core Spray logic initiation, Manual
OR Automatic will actuate Primary Containment Isolation System (PCIS)
AND Non-1E Vital Bus LOCA load shed. ____

3.3.7. The following overrides are provided during a Core Spray Initiation
AND the associated Logic Initiation Reset must be pressed to restore to AUTO position:

A. CORE SPRAY PUMP A(B,C,D)P206 (Press STOP PB). ____

B. HV-F005 A(B), CORE SPRAY INBOARD ISOLATION MOV (Depress AUTO OP OVRD PB). ____

ATTACHMENT 3
SIMPLIFIED CORE SPRAY LOGIC DIAGRAM



Facility: Hope Creek

Vendor: GE

Exam Date: 2019

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	211000 K4.03	
	Importance Rating	3.8	

K/A Statement: Knowledge of STANDBY LIQUID CONTROL SYSTEM design feature(s) and/or interlocks which provide for the following: Keeping sodium pentaborate in solution

Question: RO #34

SELECT the answer below that correctly describes what portions of the Standby Liquid Control System (SLC) that are heat traced and why this is done.

A. The pump suction lines, including the pump heads are heat traced to ensure the sodium pentaborate remains in solution.

B. All system piping, including the pump heads are heat traced to ensure the sodium pentaborate remains in solution.

C. The pump suction lines, including the pump heads are heat traced since sodium pentaborate at higher temperatures makes a better poison (neutron absorber).

D. All system piping, including the pump heads are heat traced since sodium pentaborate at higher temperatures makes a better poison (neutron absorber).

Proposed Answer: A

Explanation (Optional): The SLC pump suction lines, including the pump heads are provided with backup thermostatically controlled heat tracing, (setpt. 85˚F), which prevents the sodium.pentaborate from coming out of solution. **(Refer to the Attached print M-48-1 for the SLC system)**

A: **Correct-**.Suction line and pump heads.

B: **Incorrect-**.Just the suction line.

C: **Incorrect-** To keep the sodium pentaborate from coming out of solution.

D: **Incorrect-** Just the suction line and to keep the sodium pentaborate from coming out of solution.

Technical Reference(s): M-48-1 (Note 4) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: From memory, summarize/identify the purpose of heat tracing on the Standby Liquid Control System Storage Tank (0T204).
Given plant conditions, summarize/identify the interrelationship between the following Systems and the Standby Liquid Control System.- Heat Trace (As available)

Question Source: Bank #32593

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41 (7)

Comments:

NOTES:

1. THIS IS A SEISMIC CATEGORY I SYSTEM (EXCEPT AS NOTED: DRIP PLATE DRAINS, DEMIN. WATER AND INSTRUMENT AIR LINES, SLC TANK OVERFLOW LINE AND TEST TANK DRAIN LINE ARE NON-SEISMIC I)

2. ORIENT PRESSURE INDICATOR PI R003 SO THAT IT CAN BE READ FROM VALVE V019 LOCATION.

3. THE GE MPL NUMBER FOR THIS SYSTEM IS C41

4. STANDBY LIQUID CONTROL SYSTEM SHALL HAVE THERMOSTATICALLY CONTROLLED HEAT TRACING ON THE PUMP SUCTION LINE FROM TANK (0T204) UP TO AND INCLUDING PUMP HEADS AND UP TO VALVES V016, V053 V046, V017, V044, PSV-F029 A&B.

5. GE MPL NUMBERS ARE SHOWN IN PARENTHESIS NEXT TO BECHTEL EQUIPMENT NUMBERS WHEREVER APPLICABLE.

2)

6. VALVE V030 IS GE VALVE F036. VALVE V019 IS GE VALVE F016. VALVE V053 IS GE VALVE F031.

7. HEATERS 0E276 & 0E277 SHALL EACH BE
2) CONNECTED TO A DIFFERENT DIESEL GENERATOR BACKED MCC.

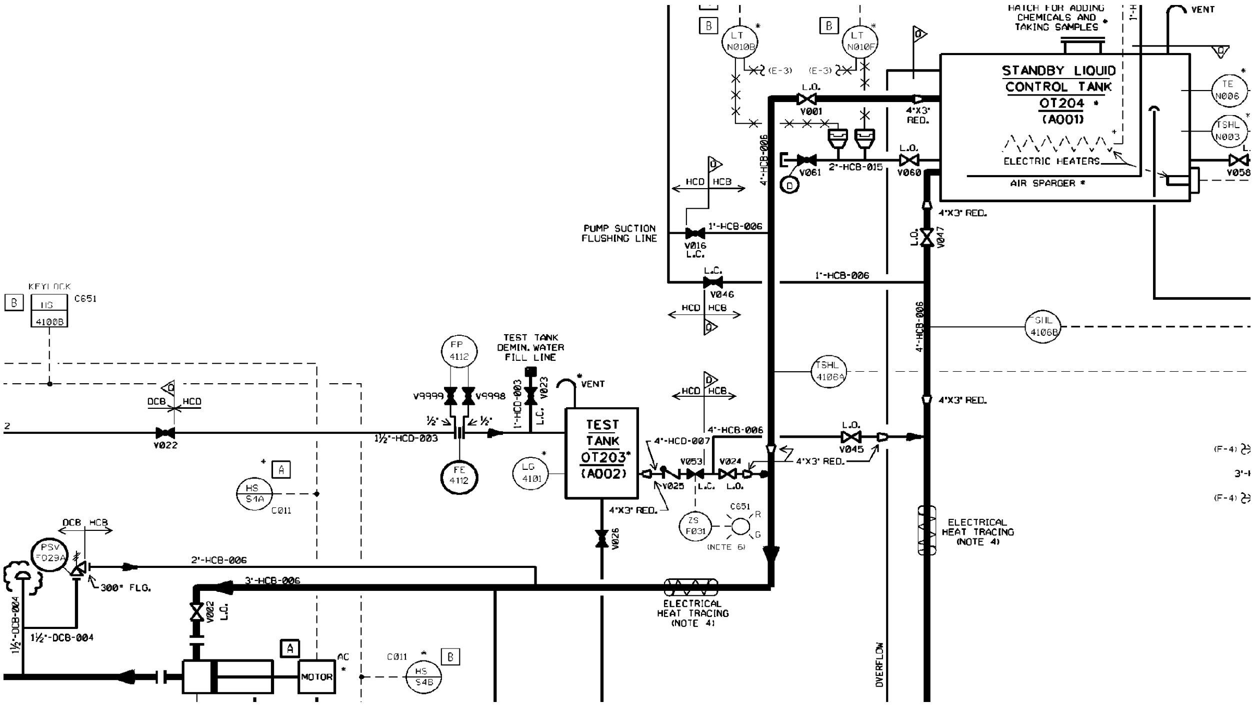
8. DELETED

9. THE ELEVATION OF THE DEMINERALIZED WATER & COMPRESSED AIR SUPPLY LINES SHALL BE ABOVE THE TOP OF THE STORAGE TANK.

10. THIS P&ID CONTAINS PORTIONS OF SYSTEMS:
    AN - DEMINERALIZED WATER MAKEUP STORAGE AND TRANSFER
    KA - SERVICE COMPRESSED AIR

11. PRESSURE GAGE ROOT VALVES ON ASME SECTION III, CLASS 2 OR 3 LINES ON THIS P&ID SHALL BE IN THE OPEN POSITION ONLY WHILE BEING READ BY AN OPERATOR. OTHERWISE THESE ROOT VALVES SHALL REMAIN IN THE CLOSE POSITION.

12. LOW PRESSURE CAPILLARY IS MOUNTED LOCALLY WITH THE DIAPHRAGM EXPOSED TO ATMOSPHERIC PRESSURE.


STANDBY LIQUID CONTROL TANK
OT204
(A001)
ELECTRIC HEATERS
AIR SPARGER
HATCH FOR ADDING CHEMICALS AND TAKING SAMPLES
VENT
TEST TANK
OT203
(A002)
TEST TANK DEMIN. WATER FILL LINE
PUMP SUCTION FLUSHING LINE
ELECTRICAL HEAT TRACING (NOTE 4)
OVERFLOW
MOTOR
KEYLOCK
300° FLG.
4'X3' RED.
4"-HCB-006
3"-HCB-006
2"-HCB-006
1"-HCB-006
1½"-HCD-003
4"-HCD-007
2"-HCB-015
V001
V002
V016
V022
V023
V024
V025
V026
V045
V046
V047
V053
V058
V060
V061
V9998
V9999
PSV F029A
TSHL 4106A
TSHL 4106B
TSHL N003
TE N006
LT N010B
LT N010F
LG 4101
FE 4112
FP 4112
ZS F031
HS S4A
HS S4B
HS 4100B
L.O.
L.C.

Facility: Hope Creek

Vendor: GE

Exam Date: 2019

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	212000 K1.10	
	Importance Rating	3.2	

K/A Statement: Knowledge of the physical connections and/or cause/effect relationships between REACTOR PROTECTION SYSTEM and the following: Main turbine.

Question: RO #35

T=0:

- A plant start up is in progress IAW HC.OP-IO.ZZ-0003, Startup from Cold Shutdown to Rated Power.
- The Reactor is at 15% rated power.
- The Main Turbine is synchronized to the grid and loaded.
- The RX RECIRC PUMPS RPS TRIP BYP alarm (C1-E3) is NOT illuminated.

T= 60 seconds:

- RPS Bus 'B' is de-energized due to a ground fault.
- Crew has entered HC.OP-AB.IC-0003, Reactor Protection System.

T= 70 seconds:

- A spurious Main Turbine Trip occurs.

What is the initial immediate response, if any, of the plant to the Main Turbine trip?

A. Reactor Scram will occur.

B. Both Reactor Recirculation Pumps will trip.

C. ONLY the 'B' Reactor Recirculation Pump will trip.

D. Neither a Reactor Recirculation Pump trip nor Reactor Scram will occur.

Proposed Answer: B

Explanation (Optional): In HC.OP-SO.SB-0001 (attached) when the turbine is shutdown or **the Turbine Stop and/or Control valves are closed (Turbine Trip)** with any RPS channel de-energized and the Recirc Pump Trip System Disable switches in NORMAL then the **EOC RPT breakers will trip for BOTH Reactor Recirculation Pumps.** IAW OP-IO.ZZ-0003 (attached), the Recirc Pump Trip System Disable switches are placed in NORMAL immediately after synchronizing and loading the Main Turbine. Due to the fact that the main turbine first stage pressure ≤ 98.1 psig (18% Reactor Power is the nominal value for Turbine first stage pressure of 98.1 psig above which a Turbine trip would result in a Reactor SCRAM), the Turbine Stop and/or Control valves RPS Scram set points are bypassed.

A: **Incorrect**- With main turbine first stage pressure ≤ 98.1 psig, the Turbine Stop and/or Control valves RPS Scram set points (see attached table) are bypassed, therefore NO reactor scram.

B: **Correct** – See above explanation.

C: **Incorrect** – Both Recirc pumps will trip. One channel of RPS will trip one of the EOC-RPT breakers associated with each of the Recirc pumps.

D: **Incorrect** – See above explanation.

Technical Reference(s):	HC.OP-SO.SB-0001 HC.OP-IO.ZZ-0003	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective:	Given labeled diagrams/drawings of the RPS trip logics, explain the coincidence requirements necessary to generate a reactor scram.	(As available)

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

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41 (7)

Comments:

1.0 PURPOSE AND SCOPE

1.1 Purpose

1.1.1. This procedure outlines the steps necessary for the operation of the Reactor Protection System (RPS).

1.2 Scope

1.2.1. Technical Specifications: 3/4.3.1 - Reactor Protection System Instrumentation; 3.6.1.1 - Primary Containment Integrity; 3.6.3 - Primary Containment Isolation Valve and Reactor Instrumentation Line Excess Flow Check Valve Operability; 3.3.7.5 - Accident Monitoring Instrumentation Channels Operability.

2.0 PRECAUTIONS AND LIMITATIONS

2.1 Precautions

2.1.1. A Full Scram should be reset as soon as possible to prevent CRD mechanism internal seal damage from excessive drive water flows AND to minimize thermal stratification of vessel with the Recirc Pumps out of service. _______

2.1.2. The transfer of RPS Bus Power supply results in a Half Scram of RPS and a Half Trip of NSSSS for MSIV isolation. Prior to transfer, verification of **NO** Scram **OR** Trip signals present in the opposite Channel should occur, as well as and when transfer is completed, **THEN**, Half Scram and Half Trip should be reset. _______

2.1.3. Transfer of RPS Bus Power Supply results in valve isolations from the associated NSSSS Logic.
(RPS Bus A - Logic A & C, RPS Bus B - Logic B & D)
HC.OP-SO.SM-0001(Q), Isolation Systems Operation, presents additional information. _______

2.1.4. Transfer of the RPS power supply will result in an EOC-RPT actuation and a Recirc Pump Trip when the Turbine Stop and/or Control Valves are closed. _______

CAUTION

A Transfer to the RPS power supply with the Main Turbine Stop and/or Control Valves closed will cause an EOC-RPT operation and a Recirc Pump Trip. EOC RPT Bypass Switches are required to be in BYPASS prior to swapping power supplies.

HC.OP-SO.SB-0001(Q)
REACTOR PROTECTION SYSTEM OPERATION

TABLE SB-001(Continued)

Variable	Setpoint	Auto Bypass
SRM Hi-Hi	≤ 2.0 X 10^5 CPS	When Shorting Links are installed (**SEE NOTE 1** and **NOTE 2**)
Mode switch SHUT-DOWN	N/A	10 seconds after Mode Switch is in SHUT-DOWN
Scram Discharge Volume Level - Hi	80.5" above Instrument zero (South Header) 83.25" above Instrument zero (North Header)	Mode Switch in SHUT-DOWN <u>OR</u> REFUEL <u>AND</u> Keylock Switch in BY-PASS
MSIV Closure	≤ 8% closed	When <u>NOT</u> in RUN
Turbine Stop Valve Closure	≤ 5% closed	When Turbine First Stage Pressure ≤ 98.1 psig
Turbine Control Valve Fast Closure, EHC Trip Oil Pressure - Lo	≥ 530 psig EHC Trip Oil Pressure	When Turbine First Stage Pressure ≤ 98.1 psig
OPRM Trip	Detection of Thermal Hydraulic Instability	Rx Pwr < 24%, <u>OR</u> Core Flow > 76% (nominal) <u>OR</u> APRM in Bypass (**SEE NOTE 3**)

4.4.8. **SYNCHRONIZE AND LOAD** the Main Generator IAW HC.OP SO.MA-0001(Z), Main Generator Exciter Operation and Switching. ________

4.4.9. **CLOSE** the following:

- HV-1026 STM LEAD S/U. ________
- HV-1013 A, B, C, D MN STM VLV BFR SEAT. ________
- HV-1015 CONT VLV BFR SEAT. ________
- HV-1017A/B STEAM LEAD 1&2 (1 PB). ________
- HV-1018B STEAM LEAD 3 ________

4.4.10. **ENSURE** #4 STEAM LEAD DRAIN HV-1018A is in AUTO. ________

4.4.11. **PLACE** EOC Recirc Pump Trip System into service as follows: [T.S. 3.3.4.2]

1. **PLACE** RECIRC PUMP TRIP SYSTEM A DISABLE Switch to NORMAL. (10C609) ________
2. **PLACE** RECIRC PUMP TRIP SYSTEM B DISABLE Switch to NORMAL. (10C611) ________

Facility: Hope Creek

Vendor: GE

Exam Date: 2019

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	212000 K5.02	
	Importance Rating	3.3	

K/A Statement: Knowledge of the operational implications of the following concepts as they apply to REACTOR PROTECTION SYSTEM: Specific logic arrangements.

Question: RO #36

Given:

- With the plant at 50% rated power.

What is the:

- Largest number of MSIVs that can be closed AND NOT directly produce a Reactor Protection System (RPS) FULL scram signal?
- Smallest number of MSIVs that can be closed AND directly produce a Reactor Protection System (RPS) FULL scram signal?

A. 5
3

B. 4
2

C. 5
2

D. 4
3

Proposed Answer: **D**

Explanation (Optional): Largest number of MSIVs that can be closed and not directly produce a full scram signal? – **[4]** Anything more than **4 MSIVs** means at least **3 lines isolated**. 3 or more lines give a full scram.
Smallest number of MSIVs that can be closed and directly produce a full scram signal? - **[3]** Even if only **one MSIV on each of 3** lines is closed, we get a full scram.

See attached logic drawing for MSIV trips.

A: **Incorrect –** Three main steam lines would be isolated and therefore a FULL scram.

B: **Incorrect –** Need three main steam lines isolated.

C: **Incorrect -**. Three main steam lines would be isolated and therefore a FULL scram.

D: **Correct** -.See above explanation

Technical Reference(s): HC.OP-SO.SB-0001 (Attach if not previously provided)

RPS

Proposed References to be provided to applicants during examination: none

Learning Objective: Given labeled diagrams/drawings of the RPS trip logics, explain the coincidence requirements necessary to generate a reactor scram. (As available)

Question Source: Bank #33105

Modified Bank # (Note changes or attach parent)

New

Question History:

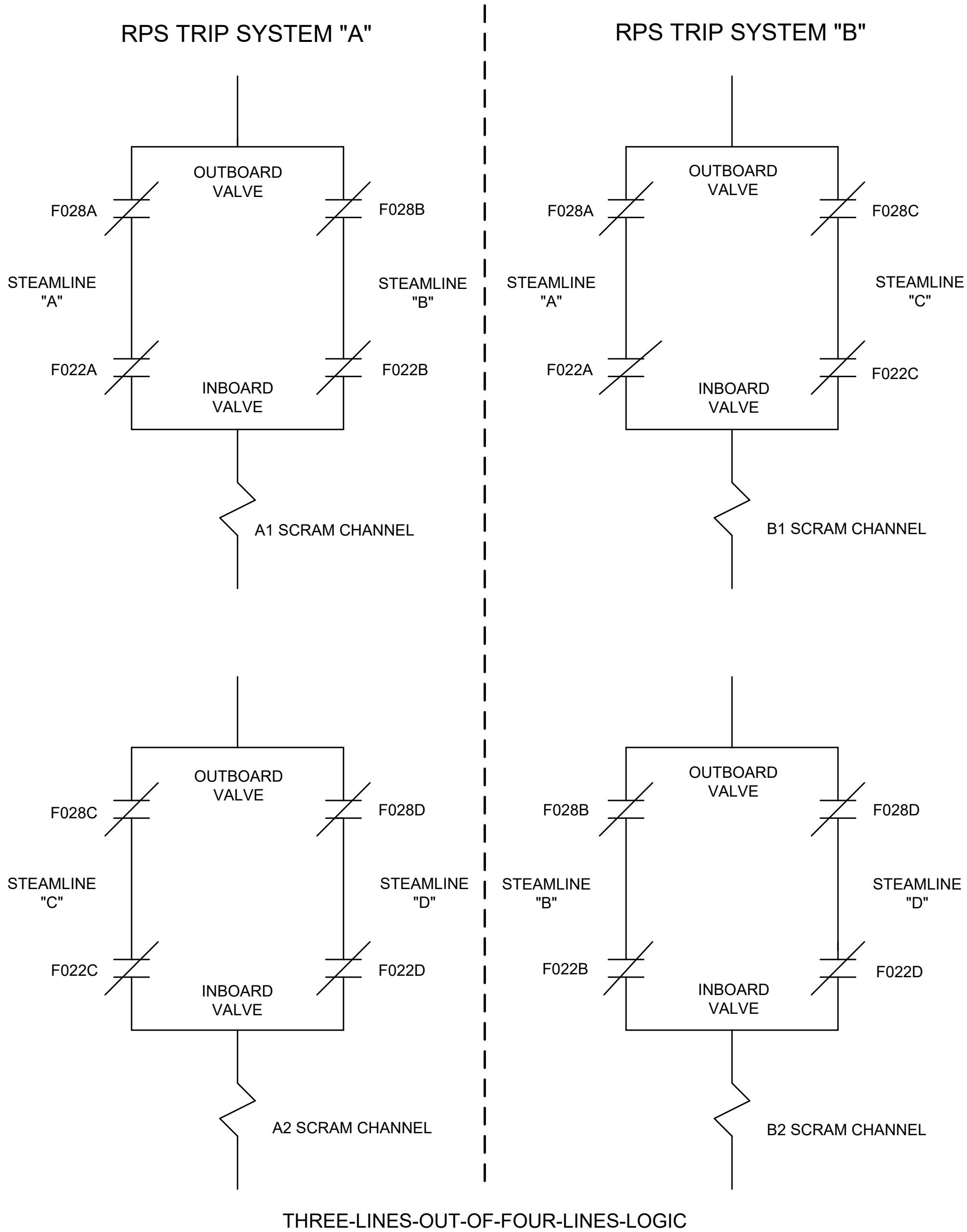
Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41 (7)

Comments:

TABLE SB-001(Continued)

Variable	Setpoint	Auto Bypass
SRM Hi-Hi	$\leq 2.0 \times 10^5$ CPS	When Shorting Links are installed (**SEE NOTE 1** and **NOTE 2**)
Mode switch SHUT-DOWN	N/A	10 seconds after Mode Switch is in SHUT-DOWN
Scram Discharge Volume Level - Hi	80.5" above Instrument zero (South Header) 83.25" above Instrument zero (North Header)	Mode Switch in SHUT-DOWN <u>OR</u> REFUEL <u>AND</u> Keylock Switch in BY-PASS
MSIV Closure	≤ 8% closed	When <u>NOT</u> in RUN
Turbine Stop Valve Closure	≤ 5% closed	When Turbine First Stage Pressure ≤ 98.1 psig
Turbine Control Valve Fast Closure, EHC Trip Oil Pressure - Lo	≥ 530 psig EHC Trip Oil Pressure	When Turbine First Stage Pressure ≤ 98.1 psig
OPRM Trip	Detection of Thermal Hydraulic Instability	Rx Pwr **<** 24%, <u>OR</u> Core Flow **>** 76% (nominal) <u>OR</u> APRM in Bypass (**SEE NOTE 3**)



**NOTE: Contacts are shown in their normal operating condition.
Any valve drifting 8% closed will open its contact.**

MSIV RPS TRIPS

Vdrive\Drawings\0001\AV365F.vsd

Rev. 2

Facility: Hope Creek

Vendor: GE

Exam Date: 2019

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	215003 K6.03	
	Importance Rating	2.8	

K/A Statement: Knowledge of the effect that a loss or malfunction of the following will have on the INTERMEDIATE RANGE MONITOR (IRM) SYSTEM: Detector drive motor.

Question: RO #37

Given:

- The reactor is being shutdown for a refueling outage.
- Reactor power is at 10% rated power with the Mode Switch in RUN.
- Seven IRM's have been inserted with their range switch positioned to establish 25 to 75% scale.
- "G" IRM failed to drive in and its range switch is placed on range 3 to establish 25 to 75% scale.

Which of the following describes the plant response, if any, if the Reactor Mode Switch is placed in Startup/Hot Standby?

A. Only a rod block will be automatically generated.

B. Only a half scram and rod block will be automatically generated.

C. A full scram and rod block will be automatically generated.

D. NO rod block, half or full scram will be generated.

Proposed Answer: A

Explanation (Optional): See attached Tables for Rod Blocks and Scram setpoints IAW HC.OP-SO.SE-0001.

A: Correct- IRM's detector in Wrong Position (Not fully inserted with mode switch not in RUN), is bypassed when the mode switch is in RUN.

B: Incorrect- all IRMs are 25 to 75% which precludes any scram signals.

C: Incorrect- all IRMs are 25 to 75% which precludes any scram signals.

D: Incorrect- IRM's detector in Wrong Position (Not fully inserted with mode switch not in RUN)

Technical Reference(s): HC.OP-SO.SE-0001 (Attach if not previously provided)

Tables SE-001 and SE-002

Proposed References to be provided to applicants during examination: none

Learning Objective: (As available)

Given a labeled diagram of, or access to, the IRM controls/indication bezel:
a. Explain the function of each indicator
b. Assess the plant conditions that will cause the indicator to light or extinguish
c. Predict the effect of each control on the IRM System
d. Select the condition or permissives required for the control switches to perform their intended function.

Question Source: Bank #34507

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41 (7)

Comments:

2.2.2. The signals listed in Table SE-001 will initiate a **CONTROL ROD BLOCK**. ________

TABLE SE-001

PARAMETER	SETPOINT	BYPASSED
SRM Downscale	≥ 3 cps	IRMs ≥ range 3 or Reactor Mode Switch in RUN
SRM Upscale	≤ 1 X 10^5 cps	IRMs ≥ range 8 or Reactor Mode Switch in RUN.
SRM Inoperative	a) Module unplugged b) Low Voltage c) Mode Switch not in Operate	IRMs ≥ range 8 or Reactor Mode Switch in RUN.
SRM Detector Wrong Position	≤ 100 cps and Detectors not fully inserted	IRMs ≥ range 3 or Reactor Mode Switch in RUN.
IRM Downscale	≥ 5/125 of full scale	IRMs in range 1 position or Reactor Mode Switch in RUN.
IRM Upscale	≤ 108/125 of full scale	Reactor Mode Switch in RUN
IRM Inoperative	a) Module unplugged b) Low Voltage c) Mode Switch not in Operate	Reactor Mode Switch in RUN.
IRM Detector Wrong Position	Detector not fully inserted	Reactor Mode Switch in RUN.
APRM Simulated Thermal Power Upscale (Flow Biased)	≤ 0.56W + 53.1% ≤ 0.56(W-10.8%) + 53.1% for SLO ≤ 108% maximum with high flow clamped	Reactor Mode Switch not in RUN.
APRM Downscale	≥ 4% of Rated Thermal Power	Reactor Mode Switch not in RUN.
APRM Upscale	≤ 11% of Rated Thermal Power	Reactor Mode Switch in RUN.
APRM Inoperative	a) Mode Switch NOT in Operate b) The Firmware/Software Watchdog timer has timed out c) A critical Self-Test fault is detected	None

2.2.3. The signals listed in Table SE-002 will initiate a **REACTOR SCRAM**. ________

TABLE SE-002

PARAMETER	SETPOINT	BYPASSED
SRM Upscale	≤ 2 X 10^5 cps	Shorting Links installed
IRM Upscale	≤ 120/125 of full scale	Reactor Mode Switch in RUN
IRM Inoperative	a) Module unplugged b) Low Voltage c) Mode Switch not in Operate	Reactor Mode Switch in RUN
APRM Upscale	≤ 116.3% of Rated Thermal Power	None
APRM Upscale	≤ 17% of Rated Thermal Power	Reactor Mode Switch in RUN.
APRM Simulated Thermal Power Upscale (Flow Biased)	≤ 0.56W + 58% ≤ 0.56(W-10.8%) + 58% for SLO maximum of 113.5% of Rated Thermal Power	None
APRM Inoperative	a) Mode Switch NOT in Operate b) The Firmware/Software Watchdog timer has timed out c) A critical Self-Test fault is detected	None
OPRM Trip	Detection of Thermal Hydraulic Instability	Rx Pwr < 24%, OR Core Flow > 76% (nominal) OR APRM in BYPASS

Facility: Hope Creek

Vendor: GE

Exam Date: 2019

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	215004 A1.06	
	Importance Rating	3.1	

K/A Statement: Ability to predict and/or monitor changes in parameters associated with operating the SOURCE RANGE MONITOR (SRM) SYSTEM controls including: Lights and alarms.

Question: RO #38

Given:

- During a reactor startup all SRMs are fully retracted (withdrawn from the core boundary).
- The Reactor Mode Switch is in Startup/Hot Standby.
- All IRMs range switches are on range 5 or 6.
- All SRM 'OUT' lamps are illuminated on the SRM Detector Select panel.

Then:

- The SRM UPSCALE OR INOPERATIVE (C3-C1) alarm is received due to a problem with SRM Channel 'A'.
- The crew enters HC.OP-AB.IC-0004, Neutron Monitoring.

Which of the following conditions would cause the above alarm?

(1) SRM drive mechanism electrical failure.
(2) SRM high voltage input is low.
(3) SRM module unplugged.
(4) SRM Mode Switch in PERIOD.

A. (1), (2) and (4)

B. (1), (2) and (3)

C. (2), (3) and (4)

D. (1), (3) and (4)

Proposed Answer: C

Explanation (Optional): See attached HC.OP-SO.SE-0001 for setpoints of a SRM INOP. Due to the fact that the SRM Detector panel is indicating 'OUT' for the SRMs and the SRMs withdrew from the core, power is available to the SRMs (10Y202) and therefore would not be the input for any alarm or setpoint of UPSCALE/INOP.

A: **Incorrect –** Power is available to the SRM detectors.

B: **Incorrect –** Power is available to the SRM detectors..

C: **Correct -**.SRM INOP setpoints (see attached SE Table).

D: **Incorrect –** Power is available to the SRM detectors..

Technical Reference(s): HC.OP-SO.SE-0001 (Attach if not previously provided)

Table SE-001

HC.OP-AB.IC-0004
Condition A

Proposed References to be provided to applicants during examination: none

Learning Objective: Given a set of conditions and a drawing of the controls, instrumentation and/or alarms located in the main control room, evaluate the status of the SRM system by observation of the controls/instrumentation alarms. (As available)

Question Source: Bank #36105

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41 (7)

Comments:

2.2.2. The signals listed in Table SE-001 will initiate a **CONTROL ROD BLOCK**. ______

TABLE SE-001

PARAMETER	SETPOINT	BYPASSED
SRM Downscale	≥ 3 cps	IRMs ≥ range 3 <u>or</u> Reactor Mode Switch in RUN
SRM Upscale	≤ 1 X 10^5 cps	IRMs ≥ range 8 <u>or</u> Reactor Mode Switch in RUN.
SRM Inoperative	a) Module unplugged b) Low Voltage c) Mode Switch not in Operate	IRMs ≥ range 8 <u>or</u> Reactor Mode Switch in RUN.
SRM Detector Wrong Position	≤ 100 cps <u>and</u> Detectors not fully inserted	IRMs ≥ range 3 <u>or</u> Reactor Mode Switch in RUN.
IRM Downscale	≥ 5/125 of full scale	IRMs in range 1 position <u>or</u> Reactor Mode Switch in RUN.
IRM Upscale	≤ 108/125 of full scale	Reactor Mode Switch in RUN
IRM Inoperative	a) Module unplugged b) Low Voltage c) Mode Switch not in Operate	Reactor Mode Switch in RUN.
IRM Detector Wrong Position	Detector not fully inserted	Reactor Mode Switch in RUN.
APRM Simulated Thermal Power Upscale (Flow Biased)	≤ 0.56W + 53.1% ≤ 0.56(W-10.8%) + 53.1% for SLO ≤ 108% maximum with high flow clamped	Reactor Mode Switch not in RUN.
APRM Downscale	≥ 4% of Rated Thermal Power	Reactor Mode Switch not in RUN.
APRM Upscale	≤ 11% of Rated Thermal Power	Reactor Mode Switch in RUN.
APRM Inoperative	a) Mode Switch NOT in Operate b) The Firmware/Software Watchdog timer has timed out c) A critical Self-Test fault is detected	None

Facility: Hope Creek

Vendor: GE

Exam Date: 2019

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	215005 A2.01	
	Importance Rating	2.7	

K/A Statement: 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: Power supply degraded.

Question: RO #39

Given:

- The plant is at 100% rated power.

When:

- The 1AN414, BD483 Neutron Monitoring Electrical Protection Assembly (EPA) breaker trips.
- The cause of the trip is unknown at this time.

What it is the status of the plant and what additional action is required?

A. Half scram from ‘B’ and ‘D’ APRM. Bypass ‘B’ and ‘D’ APRM.

B. Half scram from the ‘B’ and ‘D’ PRNM 2 out of 4 Voters. Reset 1AN414 once the cause is known.

C. Half scram from ‘B’ and ‘D’ APRM. Reset 1AN414 once the cause is known.

D. Half scram from the ‘B’ and ‘D’ PRNM 2 out of 4 Voters. Bypass ‘B’ and ‘D’ APRM.

Proposed Answer: **B**

Explanation (Optional): With the trip of the BD483 1AN414 EPA this causes a loss of power to the ‘B’ and ‘D’ PRNM Voters (two of the 2 out 4 Voters) resulting in 1/2 scram on "B" RPS (see attached). The APRMS are powered from the Quad LVPS (Low Voltage Power Supply) which are auctioneered and will fail over to the available power supply. **The 2 out of 4 Voter Modules are powered from their associated EPA breakers and NOT the QLVPS.** The remaining voters will look for the normal 2/4 logic from the APRMs. ‘B’ and ‘D’ APRMs will still be energized and indicating current power. So, the half scram has to come from the PRNM Voters NOT the APRM on the trip of the EPA breaker. .A loss of one 120 VAC bus (AJ483 or BJ483) will NOT result in a loss of power to the APRMs.

A: **Incorrect-** Half scram is from the Voters losing power due to the tripped EPA. Power is still available to the APRMs. The APRMs are still indicating current power no need to bypass the APRMs.

B: **Correct-** See above explanation

C: **Incorrect-** Half scram is from the Voters losing power due to the tripped EPA. Power is still available to the APRMs.

D: **Incorrect-**. The APRMs are still indicating current power no need to bypass the APRMs.

Technical Reference(s): HC.OP-AB.ZZ-0136 (Attach if not previously provided)

Loss of 120 VAC Inverter

Proposed References to be provided to applicants during examination: none

Learning Objective: From memory, determine the conditions under which a 2/4 Voter will generate a Reactor Protection System (RPS) trip. (As available)
Given a system which connects to or is required for the support of the APRMS, explain the function of the system interrelationship
From memory, explain the purpose of the electrical protection assemblies (EPAs).

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New **X**

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41 (7,10)

Comments:

ATTACHMENT 10
1BD483 INVERTER

Automatic Plant Response

1. Loss of power to "B" RFPT Woodward Speed Controller resulting in "B" RFPT trip.
2. "B" Feedwater Heater Train trips. 3B, 4B, and 5B Extraction Steam isolates.
3. Loss of Chill Water to "B" Recirc Motor Cooler through HV-9515B1/B2 (fail closed) (Note: Chill Water is still available through HV-9515A1/A2, powered from 1YF405 via AJ483).
4. Loss of power to 1YF406. (E-0012-1 Sht 3, E-1417-0 Sht 7A)
5. Chilled Water to DW coolers HV-9510B1/F1/D2/H2/B2/F2 which fail open.
6. Loss of power to ‘B’ and ‘D’ PRNM Voters resulting in a Half Scram. (‘B’ RPS).

Control and Indication Failures

1. Loss of all overhead annunciators.
2. Loss of control and indication from Digital Feed PDSs.
3. Loss of Reactor Narrow Range Level instrumentation.
4. Loss of various Fire Protection Monitoring Equipment and Panels.
5. Loss of control and indications for 1BX501, 1BX502 and 1BX503.
   Loss of Relay Controls for various Non-1E Switchgear.
6. Loss of various Radiation Monitoring equipment.
7. Loss of indication power to various Non-1E and BOP components.
8. Loss of Control of GS-PSV-4946C, GS-PSV-4946D and GS-PSV-5032. (indicates dual)
9. Loss of Control and Indication to various Chilled Water valves: HV-9510B1/F1/D2/H2/B2/F2.
10. Loss of ‘B’ and ‘D’ APRM Downscale signal inputs to RRCS. (Possible SLC Injection IF Reactor Scram occurs with RRCS Initiation signals present)
11. Loss of the ability to Bypass APRMs ‘B’ and ‘D’.

NOTE

The restoration of logic power may result in a change of state in equipment controls. (e.g., transfer control from AUTO to MANUAL). ___

4.15 **REQUEST** I&C to place the 1E Bailey Logic Panels 1(A-D)C652 AND 1(A-D)C655 back in service, IF de-energized in Step 4.13.1.A AND reset the Optical Isolator Cards. ___

4.16 **REQUEST** I&C to place the RRCS Panels 10C601 AND 10C602 back in service IF de-energized in Step 4.13.1.B. ___

4.17 **REQUEST** I&C to restore the ECCS Trip Units disabled in Step 4.13.2.B. ___

4.18 **RE-ENERGIZE** the load sequencer, IF it was de-energized in Step 4.13.2 AND **CLOSE** 1AD417 CB-10 OR 1BD417 CB-24 (if opened in Step 4.13.2). ___

4.19 **REQUEST** I&C to place the Non-1E Bailey Logic Panels 1(A-D)C653 back in service, IF de-energized in Step 4.13.3. ___

NOTE

Step 4.20 need only be performed for the 1(A-D)D481 Inverters. ___

4.20 IF required, **RESET** RSP TAKEOVER for the affected channel by taking the channel RSP Transfer Switch to NORMAL at 10C399. ___

4.21 **RESTORE** affected systems and equipment to a normal standby lineup IAW the applicable system operation procedure. ___

NOTE

Step 4.22 need only be performed for the 1A(B)D483 Inverters. ___

4.22 IF required, **RESET** the Neutron Monitoring EPAs. ___

4.23 **VERIFY** normal system/plant indications are restored. ___

Facility: Hope Creek

Vendor: GE

Exam Date: 2019

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	217000 A3.02	
	Importance Rating	3.6	

K/A Statement: Ability to monitor automatic operations of the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) including: Turbine startup.

Question: RO #40

Given:

- RCIC received a valid initiation signal.

The plant operator (PO) finds the following indications:

- The RCIC turbine trip throttle valve (FC-HV-4282) indicates closed.
- The RCIC Turbine Trip Status light is illuminated.
- The RCIC Inboard and Outboard Steam line Isolation valves (FC-HV-F007) and (FC-HV-F008) are open.
- The steam supply valve (FC-HV-F045) is open.
- RCIC speed is zero.

SELECT the condition that caused the current RCIC response.

A. High reactor water level at +54”.

B. Turbine Exhaust diaphragm ruptured.

C. RCIC suction pressure at 15 inHg.of vacuum.

D. Mechanical Overspeed.

Proposed Answer: **D**

Explanation (Optional): An **overspeed** condition on RCIC causes the trip throttle valve (FC-HV-4282) to close along with the RCIC Turbine Trip Status light being illuminated. The steam supply valve (FC-HV-F045) would remain open with the initiation signal present.. See attached HC.OP-SO.BD-0001 for RCIC valve interlocks.

A: INCORRECT - The steam supply valve (FC-HV-F045) would close on High reactor water level +54”.

B: INCORRECT – Turbine exhaust diaphragm rupture (High Turbine Exhaust Diaphragm pressure >10 psig) would be a system isolation that would cause the F007 and F008 to isolate.

C: INCORRECT -.RCIC Pump suction pressure trip setpoint is at more than 20inHg of vacuum. Current suction pressure (15inHg) would not trip RCIC.

D: CORRECT-.See above explanation.

Technical Reference(s): HC.OP-SO.BD-0001 (Attach if not previously provided)

RCIC Operation

Proposed References to be provided (none)

Learning Objective: Given plant conditions, evaluate and determine if any of the following should occur: (As available)
RCIC initiation
RCIC isolation
RCIC turbine trip

Question Source: Bank # 54053

Modified Bank #

New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41 (7)

Comments:

3.3 **Interlocks**

3.3.1. During operation of the RCIC System, a solenoid operated trip mechanism will energize and trip the turbine as a result of any of the following:

- RCIC Pump suction at more than 20" Hg of vacuum (PI-R604), after a 2 second time delay. ___
- Turbine exhaust pressure ≥ 50 psig (PI- R603). ___
- Any RCIC isolation signal ___
- Manual ___

3.3.2. A Turbine trip signal will cause the following to occur: ___

Trip close the turbine trip throttle valve, which closes BD-SV-F019 and BD-SV-4405 minimum flow valves. ___

3.3.4. AUTO isolation occurs upon receipt of any of the following signals: ___

- Low Reactor pressure (< 64.5 psig w/4 second time delay). ___
- High Steam Pipe Area temperature (≥160°F w/30 min time delay). ___
- High steam line flow :
  ≥ 598” H_2O w/4 sec. time delay
  - 50” H_2O w/4 sec. time delay. ___
- High Turbine Exhaust Diaphragm pressure (> 10 psig). ___
- RCIC Pump Room High temperature (≥160°F w/1 sec time delay). ___
- RCIC Pump Room Ventilation Duct High diff temperature (≥ 70°F w/1 sec time delay). ___
- RCIC Torus Compartment High temperature (≥ 128°F w/30 min time delay). ___

3.3.8. The mechanical overspeed trip of 121% rated speed will close the trip throttle valve and must be reset locally, and the limitorque must be manually run to the full closed position to relatch the valve. ___

HC.OP-SO.BD-0001(Q)

3.3.10. RCIC valve interlocks are as follows:

- HV-F007 RCIC Inboard Steam Supply Isolation Valve
  Automatically closes on "D" Channel isolation signal ___

- HV-F076 RCIC Inboard Steam Supply Isolation Valve
  Automatically closes on "D" Channel isolation signal ___

- HV-F008 RCIC Inboard Steam Supply Isolation Valve
  Automatically closes on "B" Channel isolation signal ___

- HV-F054 RCIC Steam Supply Line Drain Pot Steam Trap Bypass Valve
  Opens when drain pot level switch LSH-N010 reaches high level setpoint ___
  Closes when high level condition clears ___

- HV-F025 RCIC Inboard Steam Drain Isolation Valve
  Automatically closes when HV-045 is not full closed ___
  OR when "D" channel transfer switch on 10C399 is placed in emergency ___
  Automatically opens when HV-F045 is fully closed, provided its control switch is in OPEN ___

- HV-F026 RCIC Outboard Steam Drain Isolation Valve
  Automatically closes when HV-045 is not full closed ___
  OR when "D" channel transfer switch on 10C399 is placed in emergency ___
  Automatically opens when HV-F045 is fully closed, provided its control switch is in OPEN ___

- HV-F045 Turbine Steam Supply Isolation
  Automatically opens upon RCIC initiation signal (manual or automatic) provided
  RCIC turbine exhaust valve HV-F059 is fully open. ___
  Automatically closes on high reactor water level (level 8) signal. ___

- HV-4282 RCIC Turbine Trip and Throttle Valve Closes on a RCIC turbine trip signal ___

- HV-F084 RCIC Turbine Exhaust Line Inboard Vacuum Breaker Isolation Valve Automatically opens when "D" channel transfer switch on 10C399 is placed in emergency ___

Facility: Hope Creek

Vendor: GE

Exam Date: 2019

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	218000 A4.07	
	Importance Rating	3.5	

K/A Statement: Ability to manually operate and/or monitor in the control room: ADS valve acoustical monitor noise: Plant-Specific.

Question: RO #41

Given:

- An MSIV isolation and scram from 100% rated power occurred.
- The Lo-Lo Set SRV's cycled several times.
- CRIDS is unavailable.

How does the operator know how many times each SRV opened?

A. SRV Acoustic Monitor function switch is placed in "T" mode.

B. SRV Tailpipe Temp recorder is placed in "RECORD" mode.

C. SRV Tailpipe Temp recorder is placed in "DATA" mode.

D. SRV Acoustic Monitor function switch is placed in "C" mode.

Proposed Answer: **D**

Explanation (Optional): When the FUNCTION switch is placed in the **C(OUNT)** position, the LED window will display the number of times that the acoustic monitor in the tailpipe for the SRV chosen by the CHANNEL SELECT switch has been above the threshold value since the last time the counter was reset.

A: **Incorrect.** "T" for Threshold value is established during valve testing and represents the acoustic level at which the valve is considered open.
.

B: **Incorrect.** The temperature recorder is only a temperature recorder. It will not **count** the openings of the SRV's. Used primarily for leakage.

C: **Incorrect.** The temperature recorder is only a temperature recorder. It will not **count** the openings of the SRV's. Used primarily for leakage.

D: **Correct- see above explanation.**

Technical Reference(s): HC.OP-SO.SN-0001 (Attach if not previously provided)

HC.OP-AR.ZZ-0008

HC.OP-AB.RPV-0006

Proposed References to be provided to applicants during examination: none

Learning Objective: Given a set of conditions and a drawing of the controls, instrumentation and/or alarms located in the main control room, determine the status of the Main Steam System or its components by observation of the control, instrumentation and alarms (As available)

Question Source: Bank #32908

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41 (7)

Comments:

HC.OP-SO.SN-0001(Q)

3.0 PRECAUTIONS AND LIMITATIONS

3.1 Precautions

3.1.1. **OPEN** a different SRV each time to evenly distribute the heat being added to the Suppression Chamber. ___

3.1.2. IF an opened SRV does not indicate stable OPEN indication on its acoustic monitor, THEN it should be closed and a different SRV selected. ___

SRV Monitoring by Acoustic Monitor (XISH-4508)

There are no changes to monitoring by the Acoustic Monitor (XISH-4508). The Acoustic Monitor continues to provide an input to OHA C1-A3, ADS/SAFETY RELIEF VLV NOT CLOSED. The Acoustic Monitors only provide indication of gross leakage and that does not change with the replacement of 2-Stage SRVs with 3-Stage models.

SRV Monitoring by SRV Tailpipe Temperature Elements

SRV Tailpipe Temperature Elements continue to provide indication of SRV leakage and continue to provide an input to OHA C1-A3, ADS/SAFETY RELIEF VLV NOT CLOSED. The temperature versus leakage chart (Attachment 2 of this procedure) is revised to reflect the new 'L' SRV criteria which reflect a significantly lower threshold for action. The replacement 3-Stage SRVs may spuriously lift at a Pilot Stage leakage rate of 23 lbm/hr. Attachment 2 is revised to identify tailpipe temperatures associated with a 10 lbm/hr leak (226°F). The C1-AE alarm for the 'L' SRV has been decreased to 200 °F to provide significant early warning that SRV leakage may be approaching 10 lbm/hr.

HC.OP-AR.ZZ-0008(Q)

ATTACHMENT A3

ADS/SAFETY
RELIEF VLV
NOT CLOSED

Window Location C1-A3

OPERATOR ACTION:

1. IF ADS
   OR SRV valve(s) are open,
   REFER to HC.OP-AB.RPV-0006(Q).

CATEGORY II

SAFETY/RELIEF VALVE

ALARMS

• ADS/SAFETY RELIEF VLV NOT CLOSED	C1-A3
• SUPPR POOL TEMP HIGH	C8-F1
• COMPUTER PT IN ALARM	C1-F5

INDICATIONS

- OPEN indication on the Safety Relief Valve and/or its associated Acoustic Monitor

ADDITIONAL INFORMATION:

The following indications may be utilized to verify the SRV(s) are closed:

- CLOSED light indication on the SRV position and/or Acoustic Monitor Panels.

Facility: Hope Creek

Vendor: GE

Exam Date: 2019

Exam Type: RO

Examination Outline Cross-reference:

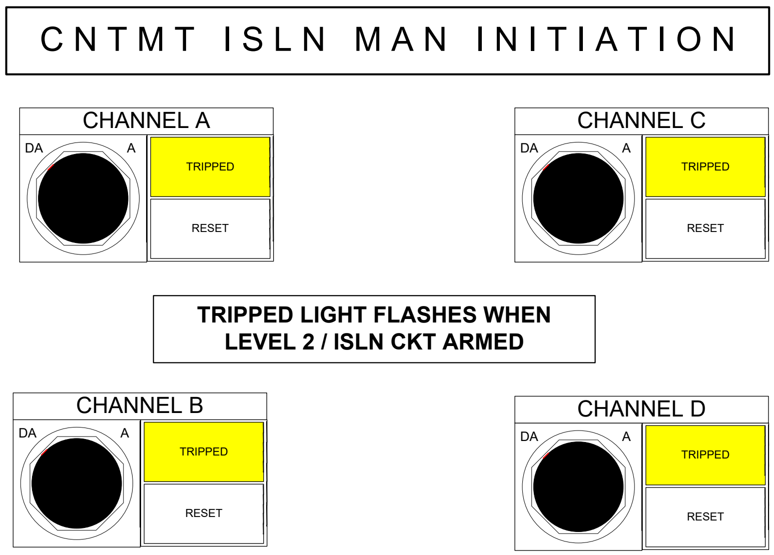
Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	223002 G2.4.50	
Importance Rating	4.2	

K/A Statement: Ability to verify system alarm setpoints and operate controls identified in the alarm response manual.- Primary Containment Isolation/Nuclear Steam Supply Shutoff

Question: RO #42

Given:

- Reactor Building ventilation has isolated due to a valid -38” RPV level condition.
- RPV level has been restored to +35”.
- CNTMT ISLN MAN INITIATION TRIPPED lights are flashing.



In order to reset the isolation IAW plant procedures AND allow the Reactor Building Isolation Dampers to be reopened, the operator must reset ______________________.

A. PCIS isolation ONLY.

B. NS^4 isolation AND PCIS isolation.

C. NS^4 isolation ONLY.

D. Core Spray initiation logic ONLY.

Proposed Answer: B

Explanation (Optional): With the valid containment isolation, the containment lights flashing indicate a half trip from NSSSS or Core Spray. RV level 2 (-38”) comes from NSSSS not Core Spray. Since both PCIS and NSSSS input to the isolation then both need to be reset IAW plant procedures. (See attached alarm response and procedure for resetting isolation).

A: **Incorrect.** the second isolation signal seals in from NS^4 which must be reset

B: **Correct.** Both inputs need to be reset.

C: **Incorrect.** the second isolation signal comes from PCIS

Technical Reference(s): HC.OP-AR.ZZ-0012 (Attach if not previously provided)

HC.OP-SO.SM-0001

Proposed References to be provided to applicants during examination: NS4 controls in the stem

Learning Objective: Describe the precautions required when resetting or overriding any Primary Containment Isolation. (As available)

Question Source: Bank #36093

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41 (7)

Comments:

HC.OP-AR.ZZ-0012(Q)

ATTACHMENT C3

NSSSS ISLN
SIG - RPV
LEVEL LO

Window Location C8-C3

OPERATOR ACTION:

1. **CHECK** NSSSS MSIV LOGIC INITIATED overhead windows AND NUCLEAR STEAM SUPPLY SHUTOFF SYSTEM MSIV TRIP LOGIC TRIPPED status lights for an isolation signal in both channels.
2. **CHECK** CNTMT ISLN MAN INITIATION Channel A - D Tripped lights.
   IF Solid, an isolation signal from both NS[4] AND Core Spray logic exists.
   IF flashing, a half isolation signal exists from NS[4]
   OR Core Spray logic.
3. IF a full isolation signal exists,
   PROCEED in accordance with HC.OP-AB.CONT-0002(Q), Primary Containment.
4. IF a full isolation signal does not exist,
   PROCEED in accordance with associated digital points.

INPUTS

Digital Point/ Indication	Nomenclature/Condition	Automatic Action
D2623	REACTOR LO WATER LVL 2 CH A	1A. Level 2 Primary Containment isolation.
D2624	REACTOR LO WATER LVL 2 CH C	1B. Filtration, Recirculation, and Ventilation System (GU) initiates.
D2627	REACTOR LO WATER LVL 2 CH B	1C. TIP withdrawal signal
D2628	REACTOR LOW WATER LVL 2 CH D	

HC.OP-SO.SM-0001(Q)

OBSERVE the Group 19 Dampers listed in Table **SM-019** have closed under the Manual or Automatic Isolation Signals and other Actions have occurred for Equipment listed as specified.

TABLE SM-019 (1 of 3)							
EQUIPMENT NUMBER	NOMENCLATURE	ACTION	MANUAL ISOLATION	AUTO ISOLATIONS			
				A	B	C	D
#GR-HD9414A	REACTOR BLDG SPLY/EXH ISLN INBD EXH	CLOSE	*C CNTMT	X	X	X	X
#GR-HD9414B	REACTOR BLDG SPLY/EXH ISLN OUTBD EXH	CLOSE	*D CNTMT	X	X	X	X
#GU-HD9370A	REACTOR BLDG SPLY/EXH ISLN OUTBD SPLY	CLOSE	*C CNTMT	X	X	X	X
#GU-HD9370B	REACTOR BLDG SPLY/EXH ISLN INBD SPLY	CLOSE	*D CNTMT	X	X	X	X
GT-HD9372A	DRYWELL PURGE EXH DRYWELL VENT	CLOSE	*A CNTMT	X	X	X	X
GT-HD9372C	DRYWELL PURGE SPLY	CLOSE	*A CNTMT	X	X	X	X
GU-HD9395A	FRVS RECIRC BYPASS	CLOSE	*A CNTMT	X	X	X	X
GU-HD9395B	FRVS RECIRC BYPASS	CLOSE	*B CNTMT	X	X	X	X
52-44024	RBVS SUPPLY FAN AVH300	TRIP	*D CNTMT	X	X	X	X
52-41024	RBVS SUPPLY FAN BVH300	TRIP	*A CNTMT	X	X	X	X
52-43024	RBVS SUPPLY FAN CVH300	TRIP	*C CNTMT	X	X	X	X
52-48024	RBVS EXHAUST FAN AV301	TRIP	*D CNTMT	X	X	X	X
52-42024	RBVS EXHAUST FAN BV301	TRIP	*B CNTMT	X	X	X	X
52-45034	RBVS EXHAUST FAN CV301	TRIP	*A CNTMT	X	X	X	X

<u>**ISOLATION**</u>	<u>**SETPOINT**</u>
A - REACTOR VESSEL WATER LEVEL 2	-38"
B - DRYWELL PRESSURE - HIGH	1.68 psig
C - REFUEL FLOOR EXHAUST RADIATION - HIGH	2 X 10^{-3} uCi/cc
D - REACTOR BUILDING EXHAUST RADIATION - HIGH	1 X 10^{-3} uCi/cc

- Group 19 Dampers

* - Can receive a Half Isolation Signal from the corresponding NSSSS Manual Isolation

5.3.2 (continued)

D. **PRESS** the following PBs:

1. NUCLEAR STEAM SUPPLY SHUTOFF SYSTEM TRIP LOGIC A RESET. MSIV TRIP LOGIC TRIPPED goes off. ____
2. NUCLEAR STEAM SUPPLY SHUTOFF SYSTEM TRIP LOGIC B RESET. MSIV TRIP LOGIC TRIPPED goes off. ____
3. NUCLEAR STEAM SUPPLY SHUTOFF SYSTEM TRIP LOGIC C RESET. MSIV TRIP LOGIC TRIPPED goes off. ____
4. NUCLEAR STEAM SUPPLY SHUTOFF SYSTEM TRIP LOGIC D RESET. MSIV TRIP LOGIC TRIPPED goes off. ____

5.3.3. To reset the Containment Isolation System, **PERFORM** the following:

A. **ENSURE** conditions causing the Isolation have been corrected,
AND NS[4] has been reset. ____

B. **PRESS** the following PBs:

1. CNTMT ISLN MAN INITIATION CHANNEL A RESET. TRIPPED goes off. ____
2. CNTMT ISLN MAN INITIATION CHANNEL B RESET. TRIPPED goes off. ____
3. CNTMT ISLN MAN INITIATION CHANNEL C RESET. TRIPPED goes off. ____
4. CNTMT ISLN MAN INITIATION CHANNEL D RESET. TRIPPED goes off. ____

Facility:	Hope Creek
Vendor:	GE
Exam Date:	2019
Exam Type:	RO

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	239002	K1.04	
	Importance Rating	3.6		

K/A Statement: Knowledge of the physical connections and/or cause/effect relationships between RELIEF/SAFETY VALVES and the following: Main steam

Question: RO #43

Given:

- The Plant is at 100% rated power.

Then, 'A' SRV fails full open:

- The operators are monitoring the main steam line flow indicators (FI-R603A-D).
- OHA C1-A3 ADS/SAFETY RELIEF VLV NOT CLOSED is in alarm.
- HC.OP-AB.RPV-0006, "Safety Relief Valve" abnormal has been entered, but NO operator actions have been taken.

Which of the following describes the indicated steam flow response (FI-R603A-D) with the open Safety Relief Valve (SRV) and the reason for that response?

A. Indicated steam flow goes down, because the SRV steam flow is not monitored by the main steam system flow detectors.

B. Indicated steam flow will remain steady, because SRV steam flow is seen as additional steam flow over what is going to the main turbine.

C. Indicated steam flow goes up, because the Turbine Control Valves and Intercept

Valves throttle open to maintain a steady MWe output.

D. Indicated steam flow will remain steady, because the Turbine Control Valves throttle closed to maintain a steady reactor pressure.

Proposed Answer: **A**

Explanation (Optional):

A: **Correct.** When the SRV opens, steam which would have gone down the steam lines through the flow venturies, to the turbine is routed to the torus. Since it does not flow past the flow venturies**, it is not monitored.**

B: **Incorrect.** When the SRV opens, steam which would have gone down the steam lines through the flow venturies, to the turbine is routed to the torus. Since it does not flow past the flow venturies, it is not monitored. Steam flow indication would change.

C: **Incorrect.** The intercept valves will not throttle for pressure control and the TCV's do not control reactor power by throttling, but rather reactor pressure.

D: **Incorrect.** TCV will throttle to maintain reactor pressure which would change steam flow indication.

Technical Reference(s): HC.OP-AB.RPV-0006 (Attach if not previously provided)

SAFETY RELIEF VALVE

Proposed References to be provided to applicants during examination: none

Learning Objective: Concerning the safety relief valves; summarize, list or identify the following: **The indications in the control room that determine if an SRV is open.** (As available)

Question Source: Bank # #2 on 2018 NRC Exam

Modified Bank # (Note changes or attach parent)

New

Question History: **2018 NRC Exam**

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41 (10)

Comments:

When a safety relief valve lifts, steam flows from the respective main steam line to the suppression chamber (torus) via the SRV, tailpipe and T-quencher and discharges below the water level normally maintained in the torus.

Steam Line Flow Restrictors (FE-NO51-N054)

Purpose

Provides for steam flow measurement and sends this flow measurement signal to NS4, Rod Worth Minimizer, Reactor Water Level Control and main control room for steam line flow indication.

Instrumentation and Control

“A” Steam Line: FE-N051 supplies PDT-N003A and PDT-N086A-D

“B” Steam Line: FE-N052 supplies PDT-N003B and PDT-N087A-D

“C” Steam Line: FE-N053 supplies PDT-N003C and PDT-N088A-D

“D” Steam Line: FE-N054 supplies PDT-N003D and PDT-N089A-D

PDT-N003A-D feeds the 10C650C panel in the main control room with individual steam line flow on FI-R603A-D.

CATEGORY II

SAFETY/RELIEF VALVE

ALARMS

• ADS/SAFETY RELIEF VLV NOT CLOSED	C1-A3
• SUPPR POOL TEMP HIGH	C8-F1
• COMPUTER PT IN ALARM	C1-F5

INDICATIONS

- OPEN indication on the Safety Relief Valve and/or its associated Acoustic Monitor
- Rising SRV Tail Pipe Temperature
- Feedwater Flow Higher than Steam Flow
- Lowering Generator Output
- Lowering Steam Flow
- Rising Suppression Pool Temperature
- >5°F Lowering 1st Stage (Pilot) Temperature for L SRV
- Rising Bellows Temperature for L SRV
- Rising Unidentified Drywell Floor Drain Leakage

Facility: Hope Creek

Vendor: GE

Exam Date: 2019

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	259002 K2.02	
	Importance Rating	3.5	

K/A Statement: Knowledge of electrical power supplies to the following: Feedwater coolant injection (FWCI) initiation logic: FWCI/HPCI.

Question: RO #44

HPCI is unavailable for manual or automatic initiation with __________ unavailable?

A. 125 VDC 10D420

B. 120 VAC 1CD482

C. 125 VDC 10D410

D. 120 VAC 1AD482

Proposed Answer: **C**

Explanation (Optional): HPCI would NOT initiate. The Initiation Logic will NOT seal-in, and HPCI is unavailable for manual or automatic initiation until 10D410 is restored. (see attached print E-0009 sh.1)

A: **Incorrect-** 125 VDC ‘B’ channel RCIC logic circuits

B: **Incorrect-** 120 VAC inverter 1CD482 Division 3 components of HPCI, RHR and Core Spray will lose status indication only. All Manual and **Automatic Signals remain functional.**

C: **Correct-** 125 VDC is required for the logic circuits and control power to HPCI valves, therefore HPCI will not initiate.

D: **Incorrect-** 120 VAC inverter 1AD482 Division 1 components of HPCI, RHR and Core Spray will lose status indication only. All Manual and **Automatic Signals remain functional**.

Technical Reference(s): E-0009 sheet 1 (Attach if not previously provided)

1E 125 VDC Channel A

HC.OP-SO.PK-0001 Power Distribution Attachment 2

HC.OP-AB.ZZ-0136 Loss of 120 VAC Inverter

Proposed References to be provided to applicants during examination: none

Learning Objective: Given plant conditions, determine the HPCI System response to any of the following: (As available)
Low CST level (HPCI in operation)
High Suppression Pool level (HPCI in operation)
Loss of 250 VDC
Loss of 480 VAC
Loss of 125 VDC

Question Source: Bank # #3 on 2018 NRC Exam

Modified Bank # (Note changes or attach parent)

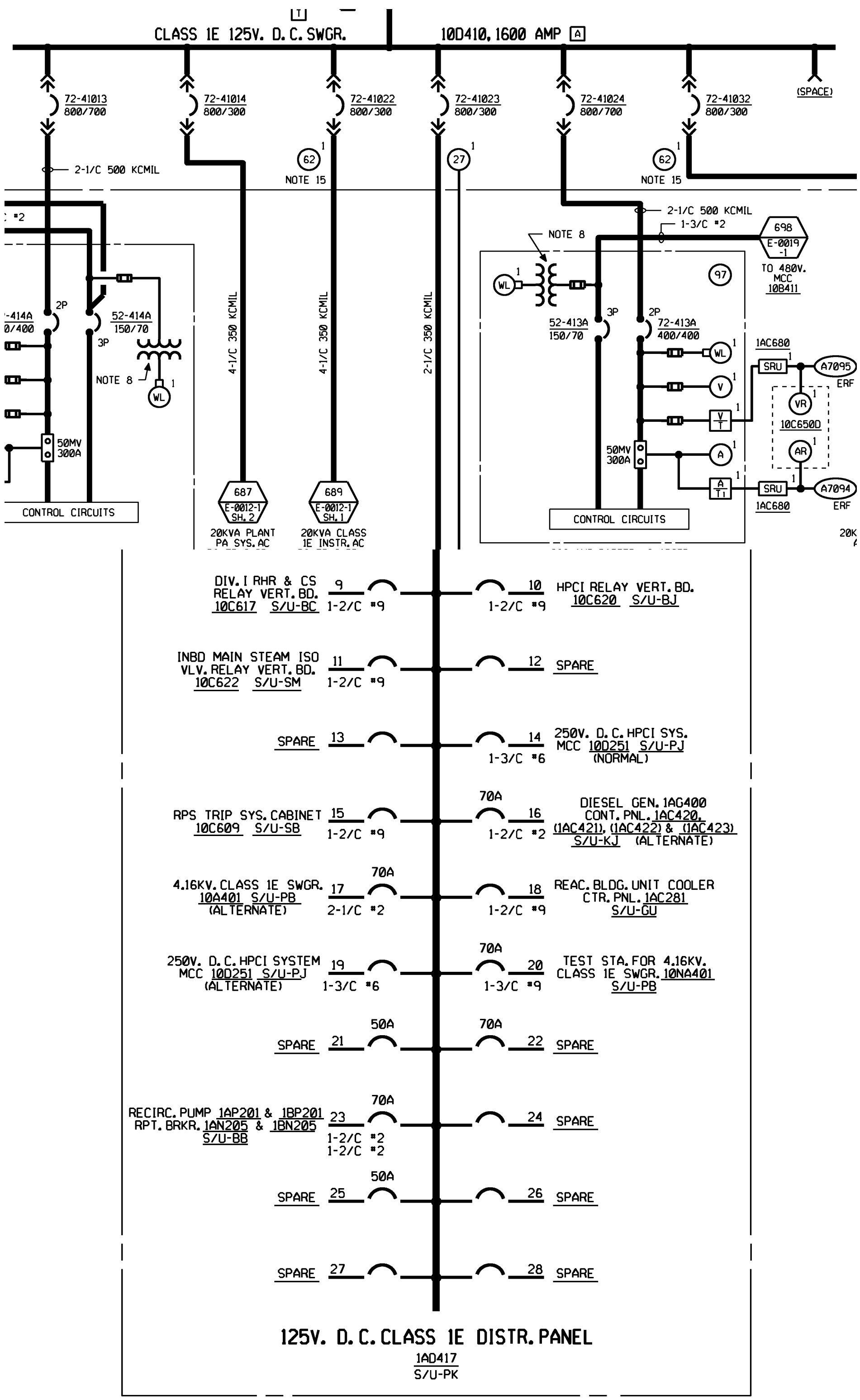
New

Question History: **2018 NRC Exam**

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41 (7)

Comments:



10D410 bus supplies the **1AD417(below)**. **Breaker 10 on 1AD417** supplies the **10C620** HPCI vertical board with all the HPCI initation logic. With no power to the **10C620** vertical board, HPCI will NOT initiate automatically or manually. **Breaker 14(19 alternate) on 1AD417** supplies the control power to the 250VDC MCC 10D251 for HPCI system. No components can be aligned without the control power to the

250VDC power supply.

ATTACHMENT 2
BATTERY COMPONENTS

Switchgear	Battery	Monitor	Transfer Switch	Charger	Distribution Panel	Loads
10D410	1AD411	1AD415	1AD412	1AD413 1AD414	1AD417	1AD481 1AD482 10D496
10D420	1BD411	1BD415	1BD412	1BD413 1BD414	1BD417	1BD481 1BD482
10D430	1CD411	1CD415	1CD412	1CD413 1CD414	1CD417	1CD481 0AD495
10D440	1DD411	1DD415	1DD412	1DD413 1DD414	1DD417	1DD481 10D485

ATTACHMENT 5
1AD482 INVERTER

Automatic Plant Response

1. Closure of TACS Supply Valve HV-2522A. Bailey Output Relay fails in de-energized (Close) state.
   IF TACS was on the A SACS Loop, the Standby SACS pump starts and the B and D TACS Supply and Return valves open.
   Water sluices from B SACS Loop to A SACS Loop due to HV-2496A failing as is (open).
   B and D TACS Supply and Return valves isolate when 'B' SACS Loop Expansion Tank reaches Lo-Lo-Lo Setpoint.
2. RWCU isolation. HV-F001 closes due to false "A" SLC Pump operating signal.
   Both RWCU Pumps trip.
3. EC-4676A closes. (IF the demin is in service, THEN the B FPCC will trip if it was in service. the 'A' FPCC pump will not trip to to loss of power to trip logic, the pump will remain inservice with no flow path.)

Control and Indication Failures

1. "A" EDG will not respond to LOCA, LOP, or Control Room Manual Start Signals.
2. Loss of 1E Analog Logic Cabinet 1AC655 will result in the loss of DIV 1/ Channel "A" analog instrumentation. Analog indicators will fail in the mid-scale position. Digital Status Indicators (valve positions, pump status, etc.) will be lost.
3. Loss of 1E Digital Logic Cabinet 1AC652 will result in the loss of control and status indication for Non-ECCS Division 1 / Channel "A" components, including alarm and computer input.
4. Division 1 components of HPCI, RHR and Core Spray will lose status indication only.
   All Manual and Automatic Signals remain functional.
5. The "A" SLC Pump will not respond to a Control Room manual start command.
   The "A" SLC Pump will start on an RRCS Initiation Signal.
6. LR-3682A-1 / PR-3684A-1, HPCI pressure and level recorder.
7. IF Demin is In Service, A Fuel Pool Cooling Pump will continue to run but will lose running indication. EC-HV-4689B, FILTER DEMIN BYPASS VALVE must be opened to establish a flow path.

ATTACHMENT 7
1CD482 INVERTER

Automatic Plant Response

1. Closure of TACS Supply Valve HV-2522C. Bailey Output Relay fails in de-energized (Close) state.
   IF TACS was on the A SACS Loop, the Standby SACS pump starts and the B and D TACS Supply and Return valves open.
   Water sluices from B SACS Loop to A SACS Loop due to HV-2496C failing as is (open).
   B and D TACS Supply and Return valves isolate when 'B' SACS Loop Expansion Tank reaches Lo-Lo-Lo Setpoint.
2. HD-9414A fails closed, causing trips of RBVS.

Control and Indication Failures

1. EDG "C" response to LOP, LOCA or Control Room Start Signals will be inhibited.
2. Loss of 1E Analog Logic Cabinet 1CC655 will result in the loss of DIV 3/ Channel "C" analog instrumentation. Analog indicators will fail in the mid-scale position. Digital status indicators (valve positions, pump status, etc.) will be lost.
3. Loss of 1E Digital Logic Cabinet 1CC652 will result in the loss of control and status indication for Non-ECCS Division 3 / Channel "C" components, including alarm and computer input.
4. Division 3 components of HPCI, RHR and Core Spray will lose status indication only.
   All Manual and Automatic Signals remain functional.

Facility: Hope Creek

Vendor: GE

Exam Date: 2019

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	261000 A4.07	
	Importance Rating	3.1	

K/A Statement: Ability to manually operate and/or monitor in the control room: System flow- Standby Gas Treatment

Question: RO #45

Given:

- The plant is at 50 percent rated power.
- The FRVS Recirculation Fans are in AUTO.

T=0:

- A Loss of Offsite Power occurs.
- HPCI and RCIC are manually initiated.
- Minimum RPV water level reached was -25 inches.

T=3 minutes:

SELECT the approximate total FRVS recirculation flow. (Assume NO other operator actions).

A. 0 cfm

B. 60,000 cfm

C. 120,000 cfm

D. 180,000 cfm

Proposed Answer: **A**

A: **Correct** - There are no automatic initiation signals present (-38” RPV LVL); no FRVS Recirc Fans are running.

B: **Incorrect** - There are no automatic initiation signals present (-38” RPV LVL); no FRVS Recirc Fans are running.

C: **Incorrect** - There are no automatic initiation signals present (-38” RPV LVL); no FRVS Recirc Fans are running.

D: **Incorrect** – If the FRVS system did receive an auto initiation. Also, FRVS does not come off the LOP sequencer (RBVS).

Technical Reference(s): HC.OP-SO.GU-0001 (Attach if not previously provided)

FRVS Operation

Proposed References to be provided to applicants during examination: none

Learning Objective: Concerning the Filtration Recirculation Ventilation System (FRVS): Distinguish between the automatic starts and stops associated with the FRVS Vent and Recirc Fans. (As available)

Question Source: Bank #35800

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41 (7)

Comments:

3.2.4. All fan operations should be performed IAW OP-HC-108-106-1001, Equipment Operational Control. ___

3.2.5. FRVS Recirc Fan flow recorders in the main control room contain a “zero-suppression equation” which displays a zero value up to 2,999 CFM. ___

3.2.6. With respect to SACS cooling and EDG power supply, the following is required for FRVS Recirculation Fan Operability: ___

In OPERATIONAL CONDITION 1, 2, and 3, SACS and EDG backing are required support systems for all fans.

In OPERATIONAL CONDITION "*" (OPDRVs), SACS and EDG backing are required support systems for only three recirc fans (additional fans may be considered operable without SACS cooling when powered from an offsite source).

OPERATIONAL CONDITION "*" (handling recently irradiated fuel in the secondary containment) is prohibited by administrative procedures.

During fuel handling of fuel that is not recently irradiated and CORE ALTERATIONS, either the RBVS or one FRVS vent fan is required to be operating and capable of drawing air into the building and exhausting through a monitored pathway.
REFER to OP-HC-108-115-1001 for details.
[80005080, 70020722] ___

3.3 Interlocks

3.3.1. FRVS Recirculation Fans AV213 through FV213 in AUTO and FRVS Vent Fan in AUTO LEAD will automatically start under any of the following conditions: ___

- High Drywell Pressure (1.68 psig).
- Low RPV Water Level (Level 2, - 38").
- Refueling Floor Exhaust Duct High Radiation (≥ 2x10 $^{-3}$ micro Ci/cc).
- Reactor Building Exhaust Air High Radiation (≥ 1x10 $^{-3}$ micro Ci/cc).

3.3.2. With the FRVS Recirculation Fans EV213 and FV213 in AUTO, and the A, B, C, and D fans running due to receiving a LOCA OR High Rad signal, a low flow signal from a running FRVS Recirculation Fan will automatically start the EV213 and FV213 fans. ___

Facility: Hope Creek

Vendor: GE

Exam Date: 2019

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	261000 K3.04	
	Importance Rating	3.1	

K/A Statement: Knowledge of the effect that a loss or malfunction of the STANDBY GAS TREATMENT SYSTEM will have on following: High Pressure Coolant Injection System: Plant Specific.

Question: RO #46

Given:

- The plant is at 100% rated power.
- RCIC is out of service for maintenance.

Then:

- A station blackout (SBO) occurs.
- RPV level dropped to -65 inches.
- HPCI automatically injected to recover RPV level.

Current conditions:

- RPV level is stable at +35 inches.

What describes the long term operations of the HPCI system?

A. HPCI will continue to operate since the barometric condenser vacuum pump discharge flow path will automatically swap to the main condenser due to high backpressure in the HPCI barometric condenser. The swap will preserve HPCI operation and prevent potential airborne contamination in the Reactor Building and/or HPCI compartment.

B. HPCI will eventually trip on turbine high exhaust pressure as non condensible gases build up in the barometric condenser due to the loss of the barometric condenser vacuum pump discharge valve flowpath. In addition, HPCI operation is undesirable without ventilation in service since air-born contamination levels may eventually rise in the Reactor Building and/or HPCI compartment.

C. HPCI will continue to operate; however, airborne contamination levels may eventually rise in the Reactor Building and/or HPCI compartment as the barometric condenser vacuum pump continues to discharge into the RBVS/FRVS ductwork without ventilation in service.

D. HPCI will continue to operate since the barometric condenser pump discharge flow path will automatically swap to the suppression chamber airspace due to high backpressure in the HPCI barometric condenser. The swap will preserve HPCI operation and prevent potential airborne contamination in the Reactor Building and/or HPCI compartment.

Proposed Answer: C

Explanation:

A: **Incorrect –** HPCI will continue to operate; however, there is no automatical swap feature to realign the vacuum pump discharge back to the main condenser

B: **Incorrect –** This condition will not cause high turbine exhaust pressure. The barometric condenser services HPCI turbine gland seal leak off.

C: **Correct –** The DC powered HPCI barometric condenser vacuum pump will continue to discharge into the RBVS/FRVS ventilation duct.. Without ventilation in service, potentially contaminated non condensable gases may eventually leak into the reactor Building.

D: **Incorrect –** HPCI will continue to operate; however, there is no automatic swap feature to suppression chamber airspace.

Technical Reference(s): M-56-1 (Attach if not previously provided)

HPCI P&ID

Proposed References to be provided to applicants during examination: none

Learning Objective: From memory summarize/identify the interrelationship(s) between the HPCI System and any of the following (As available)

RBVS/FRVS

Question Source: Bank #84236

Modified Bank # (Significant change to distractors and actual question)

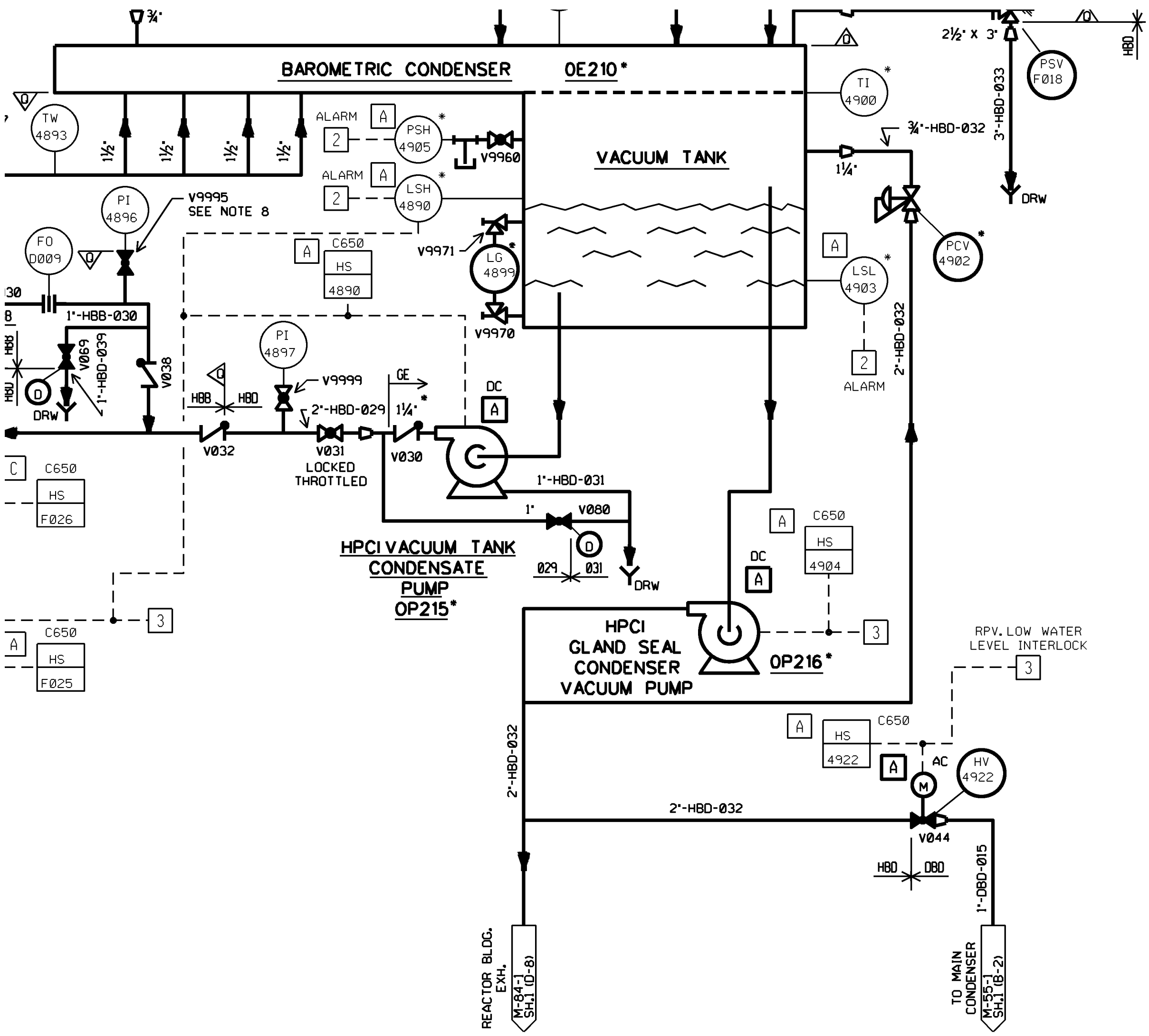
New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41 (7)

Comments:


BAROMETRIC CONDENSER
OE210*
VACUUM TANK
TW 4893
PI 4896
V9995
SEE NOTE 8
FO D009
1"-HBB-030
1"-HBD-039
V069
V038
DRW
ALARM
PSH 4905
V9960
LSH 4890
C650
HS 4890
V9971
LG 4899
V9970
TI 4900
3/4"-HBD-032
1 1/4"
PCV 4902
LSL 4903
ALARM
2"-HBD-032
PSV F018
2 1/2" X 3"
3"-HBD-033
DRW
PI 4897
V9999
2"-HBD-029
V032
V031
LOCKED
THROTTLED
V030
GE
DC
1"-HBD-031
V080
DRW
HPCI VACUUM TANK
CONDENSATE
PUMP
OP215*
C650
HS F026
HS F025
HS 4904
HPCI
GLAND SEAL
CONDENSER
VACUUM PUMP
OP216*
RPV. LOW WATER
LEVEL INTERLOCK
HS 4922
AC
HV 4922
2"-HBD-032
V044
HBD
DBD
1"-DBD-015
REACTOR BLDG.
EXH.
M-84-1
SH.1 (D-8)
TO MAIN
CONDENSER
M-55-1
SH.1 (B-2)

Facility: Hope Creek

Vendor: GE

Exam Date: 2019

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	262001 K4.06	
	Importance Rating	3.6	

K/A Statement: Knowledge of A.C. ELECTRICAL DISTRIBUTION design feature(s) and/or interlocks which provide for the following: Redundant power sources to vital buses.

Question: RO #47

Given:

- The reactor is at 55% rated power.
- All vital bus normal in-feed breakers are closed.
- The vital bus alternate in-feed breakers are open.

Then:

- Drywell pressure begins to rise and OHA “DRYWELL PRESS HI/LO” (A7-E4) alarms.

At T=0 seconds:

- The High Drywell Pressure scram setpoint is reached and drywell pressure continues to rise slowly.
- The reactor scrams and all control rods fully insert.
- Reactor level lowers to -10 inches and begins to recover.
- All expected automatic actions occur.

At T=60 seconds:

The "A" RHR pump is powered via the ____________ and the pump started via the ___________.

A. "A" Emergency Diesel Generator; RHR logic.

B. 1AX501 Station Service Transformer; LOCA Sequencer.

C. "A" Emergency Diesel Generator; LOCA Sequencer.

D. 1AX501 Station Service Transformer; RHR logic.

Proposed Answer: D

Explanation (Optional): With no LOP and a LOCA, the “A” RHR pump will start without delay on normal auxiliary power (1AX501 transformer). Even though the LOCA sequencer would be energized, the “A” RHR pump would start off of its own logic.

A: **Incorrect-** The “A” EDG is running due to the LOCA signal but the EDG output breaker remains open due to NO LOP. The “A” RHR pump is powered by the 1AX501.

B: **Incorrect-** RHR has its own pump starting logic and are NOT powered by the LOCA or LOP Sequencers. See attached Figure 3 of SOP.

C: **Incorrect-** The “A” EDG is running due to the LOCA signal but the EDG output breaker remains open due to NO LOP. The “A” RHR pump is powered by the 1AX501.

D: **Correct-** See above explanation.

Technical Reference(s): HC.OP-SO.BC-0001 (Attach if not previously provided)

E-0001

Proposed References to be provided to applicants during examination: none

Learning Objective: Given a set of conditions and a drawing of the controls, instrumentation and/or alarms located in the main control room, assess the status of the Residual Heat Removal System or its components by evaluation of the controls/instrumentation/alarms
Concerning the 1E AC distribution switchgear:
Given a list of electrical loads (motor/unit substations); choose which are powered from the 1E 4.16KV switchgear(s).
Given the procedural steps required to perform the following at the 4.16KV switchgear, explain each step; including sequencing when appropriate. (As available)

Question Source: Bank # 150868

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41 (7)

Comments:

HC.OP-SO.BC-0001(Q)

5.3 **LPCI Initiation Observation**

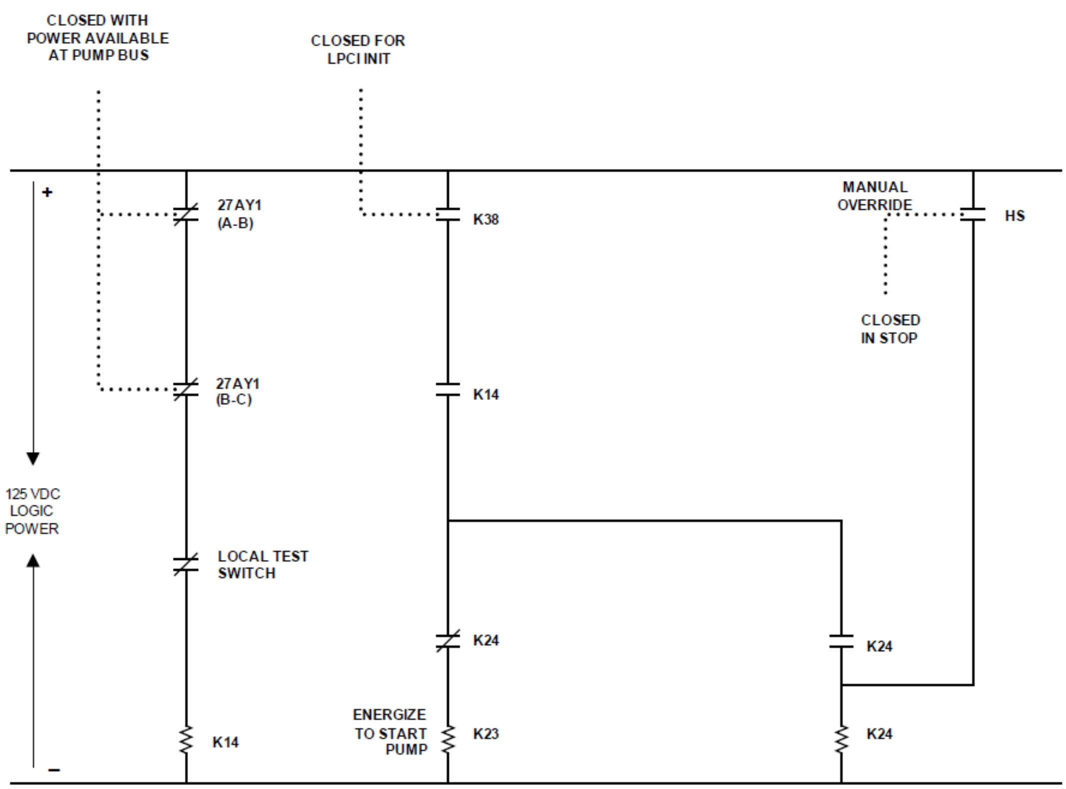
NOTE

IF the RHR System is aligned for a manual mode of operation and LPCI protection is required, the system will automatically realign and initiate LPCI operation without operator action.

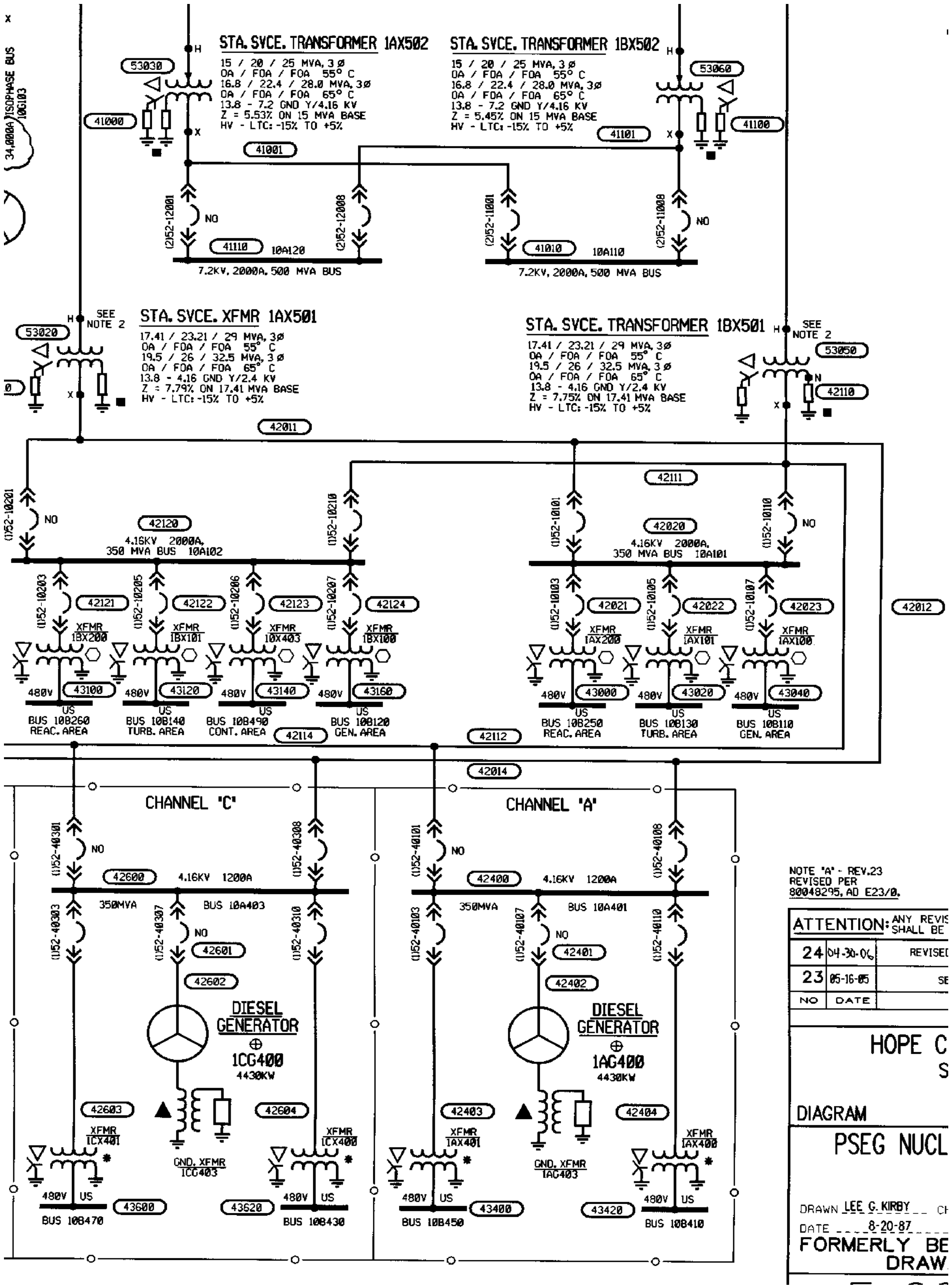
Upon receipt of LPCI initiation signal, RHR Pumps A and B start without delay on normal auxiliary power and RHR Pumps C and D start after a 5 second time delay. All four RHR Pumps start simultaneously on Standby Diesel power (LOP). ___

HC.OP-SO.BC-0001(Q)

FIGURE 3
LPCI A/B, PUMP START LOGIC



LPCI PUMP START CIRCUIT(RHR PUMPS A/B)


STA. SVCE. TRANSFORMER 1AX502
15 / 20 / 25 MVA, 3 Ø
OA / FOA / FOA 55° C
16.8 / 22.4 / 28.0 MVA, 3Ø
OA / FOA / FOA 65° C
13.8 - 7.2 GND Y/4.16 KV
Z = 5.53% ON 15 MVA BASE
HV - LTC: -15% TO +5%
STA. SVCE. TRANSFORMER 1BX502
15 / 20 / 25 MVA, 3 Ø
OA / FOA / FOA 55° C
16.8 / 22.4 / 28.0 MVA, 3Ø
OA / FOA / FOA 65° C
13.8 - 7.2 GND Y/4.16 KV
Z = 5.45% ON 15 MVA BASE
HV - LTC: -15% TO +5%
34,000A ISOPHASE BUS 10G103
7.2KV, 2000A, 500 MVA BUS
10A120
10A110
STA. SVCE. XFMR 1AX501
17.41 / 23.21 / 29 MVA, 3Ø
OA / FOA / FOA 55° C
19.5 / 26 / 32.5 MVA, 3 Ø
OA / FOA / FOA 65° C
13.8 - 4.16 GND Y/2.4 KV
Z = 7.79% ON 17.41 MVA BASE
HV - LTC: -15% TO +5%
STA. SVCE. TRANSFORMER 1BX501
17.41 / 23.21 / 29 MVA, 3Ø
OA / FOA / FOA 55° C
19.5 / 26 / 32.5 MVA, 3 Ø
OA / FOA / FOA 65° C
13.8 - 4.16 GND Y/2.4 KV
Z = 7.75% ON 17.41 MVA BASE
HV - LTC: -15% TO +5%
SEE NOTE 2
4.16KV 2000A, 350 MVA BUS 10A102
4.16KV 2000A, 350 MVA BUS 10A101
BUS 10B260 REAC. AREA
BUS 10B140 TURB. AREA
BUS 10B490 CONT. AREA
BUS 10B120 GEN. AREA
BUS 10B250 REAC. AREA
BUS 10B130 TURB. AREA
BUS 10B110 GEN. AREA
CHANNEL 'C'
CHANNEL 'A'
4.16KV 1200A 350MVA BUS 10A403
4.16KV 1200A 350MVA BUS 10A401
DIESEL GENERATOR 1CG400 4430KW
DIESEL GENERATOR 1AG400 4430KW
GND. XFMR 1CG403
GND. XFMR 1AG403
BUS 10B470
BUS 10B430
BUS 10B450
BUS 10B410
NOTE 'A' - REV.23 REVISED PER 80048295, AD E23/0.
ATTENTION: ANY REVIS SHALL BE
24 04-30-06 REVISED
23 05-16-05 SE
NO DATE
HOPE C
DIAGRAM
PSEG NUCL
DRAWN LEE G. KIRBY
DATE 8-20-87
FORMERLY BE DRAW

Facility: Hope Creek

Vendor: GE

Exam Date: 2019

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	262002 A1.02	
	Importance Rating	2.5	

K/A Statement: Ability to predict and/or monitor changes in parameters associated with operating the UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.) controls including: Motor generator outputs.

Question: RO #48

Given:

- The plant is at 100% rated power.
- Reactor Protection System (RPS) buses are aligned to their normal power supply.

Then:

- The BN410 RPS EPA (Electrical Protection Assembly) breaker trips due to a sensed 135 VAC on its Overvoltage trip device.

Which of the following is the plant response to the trip of the BN410 EPA breaker?

A. “A”' RPS bus will lose power immediately.

B. “A” RPS bus will lose power in 1-3 seconds.

C. “B” RPS bus will lose power in 1-3 seconds.

D. “B” RPS bus will lose power immediately.

Proposed Answer: **A**

Explanation (Optional): With a trip of the EPA breaker the power would immediately be loss to the bus due to the fact that the EPA breaker is downstream of the RPS MG Set and upstream of the distribution panel (see attached simplified drawing of the RPS power distribution). If the normal AC power is lost or any malfunction on the MG Set, then the flywheel on the MG Set would delay the power loss for 1-3 seconds. An EPA Breaker will trip OPEN on the following: Overvoltage (≥ 132Vac), Undervoltage (≤ 108Vac) , or Underfrequency (≤ 57 Hz)

A: **CORRECT-** Immediate loss of power to “A” RPS bus since the EPA breaker is downstream of the MG Set and the BN410 is part of the “A” RPS power distribution.

B: **INCORRECT-** Immediate loss of power. The time delay is if MG set was lost.

C: **INCORRECT-** Immediate loss of power. The time delay is if MG set was lost. The BN410 EPA breaker is an “A” RPS power distribution breaker.

D: **INCORRECT-**The BN410 EPA breaker is an “A” RPS power distribution breaker.

Technical Reference(s): HC.OP-SO.SB-0001 (Attach if not previously provided)

RPS

Proposed References to be provided to applicants during examination: none

Learning Objective: Identify the normal and alternate sources of power to RPS Bus A and RPS Bus B. Identify the power supplies to RPS MG Set A and RPS MG Set B. (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New **X**

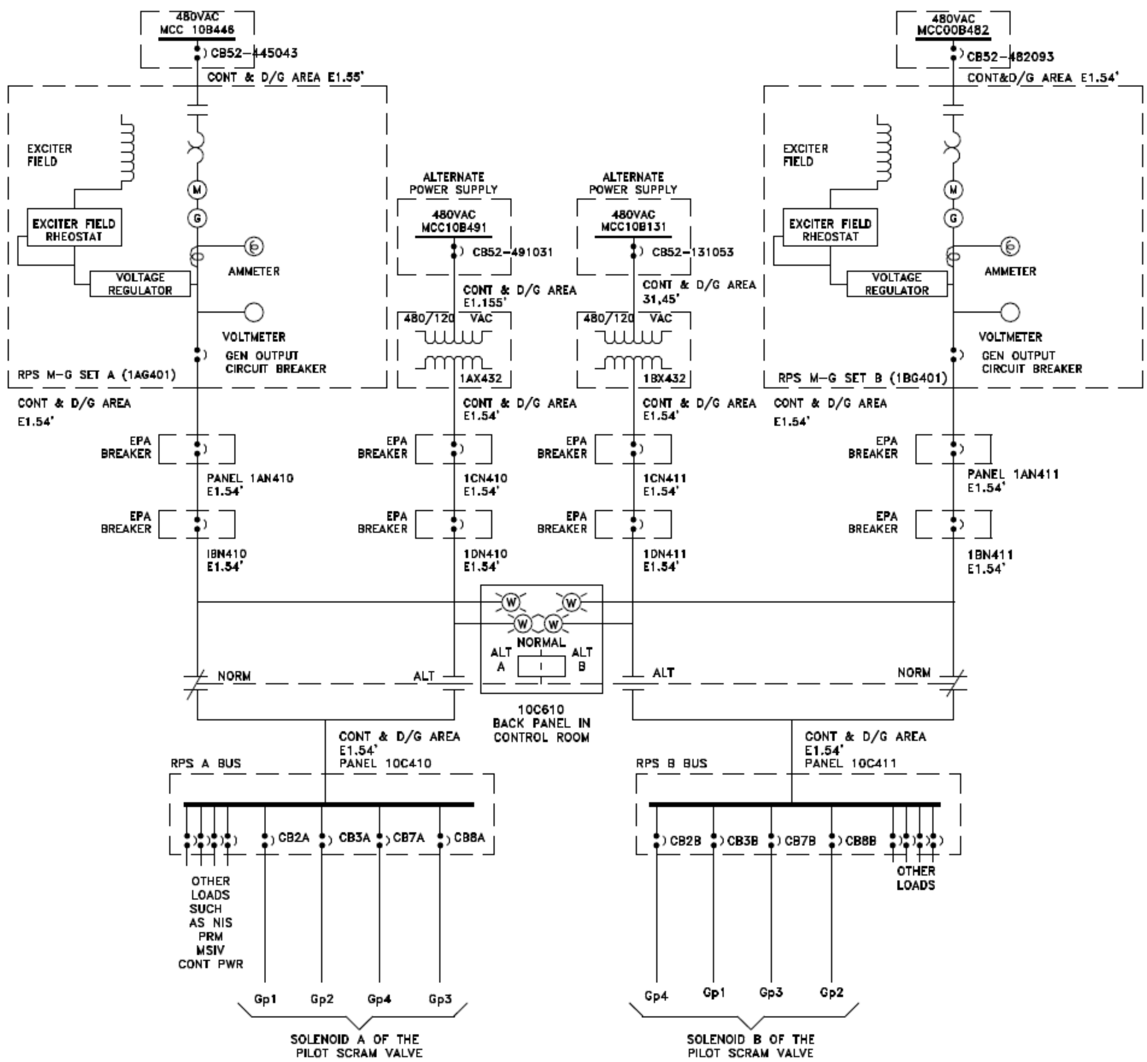
Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41 (7)

Comments:

Attachment 4
RPS POWER DISTRIBUTION SIMPLIFIED DRAWING



Facility: Hope Creek

Vendor: GE

Exam Date: 2019

Exam Type: RO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	263000 K6.01	
Importance Rating	3.2	

K/A Statement: Knowledge of the effect that a loss or malfunction of the following will have on the D.C. ELECTRICAL DISTRIBUTION: A.C. electrical distribution.

Question: RO #49

Given:

- The plant is operating at 100% rated power.

When:

- A LOCA Level 1 concurrent with a LOP occurs.
- All Emergency Diesel Generators started and responded as designed.

30 seconds after the event:

Which of the following describes the status of the 1E and Non-1E 125 VDC Battery Chargers?

A. 1E and Non-1E battery chargers are load shed.
1E battery chargers will be automatically restored by load sequencing.
Non-1E battery chargers will be restored 2 minutes after the sequencer starts.

B. 1E and Non-1E battery chargers are load shed but will be automatically restored at the same time by load sequencing.

C. 1E battery chargers are in service.
Non-1E battery chargers are load shed and can be manually restored by overriding the load shed and re-energizing the MCC's.

D. 1E battery chargers are in service.
Non-1E battery chargers are load shed and CANNOT be returned to service.

Proposed Answer: **C**

Explanation (Optional): Upon a LOCA, the MCCs that supply the non 1E battery chargers are shed from the Class 1E 480 VAC Unit Substations that normally supply their power. Shedding of the MCCs places the 125 VDC (non- 1E) power requirements on the respective batteries. The LOCA signal for the MCC feeder breakers can be overridden in the control room at 10C650 with the TRIP OVRD PB (This Pb is used to override the LOCA trip signal and restore power to the associated load(s)) and restore the non-1E chargers. For the 1E 125VDC chargers due to the fact that there is **NO load shed of the 1E breaker** once the power comes back (EDGs) the chargers will be energized.

A: **Incorrect-** The 1E chargers are not load shed. Non-1E battery chargers are not automatically restored.

B: **Incorrect-** The Non 1E chargers are not automatically restored after a LOCA.

C: **Correct-** The 1E battery chargers supply breakers are not load shed. The Non-1E battery chargers can be restored manually.

D: **Incorrect-** The Non-1E battery chargers can be restored manually.

Technical Reference(s):	HC.OP-SO.SM-0001 Isolations HC.OP-SO.PK-0001 125 VDC	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective:	Given various plant conditions/malfunctions associated with the A.C. Electrical Distribution Systems, determine the effect that the malfunction will have on the D.C. Distribution System.	(As available)

Question Source:	Bank # 33891	
	Modified Bank #	(Note changes or attach parent)
	New	

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41 (7)

Comments:

HC.OP-SO.SM-0001(Q)

OBSERVE the Equipment or Valves listed in Table **SM-020** have met their required Action under the Manual or Automatic Isolation Signals

TABLE SM-020 (1 of 8)							
EQUIPMENT NUMBER	**NOMENCLATURE**	**ACTION**	**MANUAL ISOLATION**	**AUTO ISOLATIONS**			
				A	**B**	**C**	**D**
"52-232094	INSTRUMENT GAS CPRSR AK202	TRIP	+C CNTMT	X		X	X
"52-242094	INSTRUMENT GAS CPRSR BK202	TRIP	+D CNTMT	X		X	X
"52-41011	RACS PMP AP209 BRKR	TRIP	A CS	X		X	
"52-42011	RACS PMP BP209 BRKR	TRIP	B CS	X		X	
"52-43014	CRD PMP AP207 BRKR	TRIP	C CS	X		X	
"52-44014	CRD PMP BP207 BRKR	TRIP	D CS	X		X	
"52-41014	MCC 10B313 BRKR	TRIP	A CS	X		X	
"52-42014	MCC 10B323 BRKR	TRIP	B CS	X		X	
"52-47011	MCC 10B272 BRKR	TRIP	C CS	X		X	
"52-48011	MCC 10B282 BRKR	TRIP	D CS	X		X	
"52-45011	MCC 10B252 BRKR	TRIP	A CS	X		X	
"52-46011	MCC 10B262 BRKR	TRIP	B CS	X		X	
"52-47031	MCC 00B474 BRKR	TRIP	C CS	X		X	
52-451023	INVERTER 10D496 B/U PWR (PA SYST)	TRIP	A CS	X		X	

ISOLATION	**SETPOINT**
A - REACTOR VESSEL WATER LEVEL 1	-129"
C - DRYWELL PRESSURE - HIGH	1.68 psig
D - REACTOR BUILDING EXHAUST RADIATION - HIGH	1 X 10^{-3} uCi/cc

+ - Also Isolates from the corresponding Core Spray Manual Initiation PB
* - Can receive a Half Isolation from the corresponding NSSSS Manual Isolation
" - Isolation can be bypassed by TRIP OVRD PB

ATTACHMENT 3
INVERTER POWER SUPPLIES

Inverter	Normal AC	Backup AC	DC Bus	Battery	Batt Charg	Charg Pwr
1AD481 **a**	10B451 **a**	10B411 **a**	10D410 **a**	1AD411 **a**	1AD413 **a** 1AD414 **a**	10B411 **a** 10B451 **a**
1AD482 **a**	10B411 **a**	10B451 **a**	10D410 **a**	1AD411 **a**	1AD413 **a** 1AD414 **a**	10B411 **a** 10B451 **a**
1AD483 **n**	10B252 **a**	10B313 **a**	10D470 **n**	1A1D471 **n** 1A2D471 **n**	1A1D473 **n** 1A2D473 **n**	10B313 **a** 10B252 **a**
1AD484 **n**	10B313 **a**	10B252 **a**	10D476 **n**	1A1D477 **n** 1A2D477 **n**	1A1D474 **n** 1A2D474 **n**	10B272 **c** 10B282 **d**
1BD481 **b**	10B461 **b**	10B421 **b**	10D420 **b**	1BD411 **b**	1BD413 **b** 1BD414 **b**	10B421 **b** 10B461 **b**
1BD482 **b**	10B421 **b**	10B461 **b**	10D420 **b**	1BD411 **b**	1BD413 **b** 1BD414 **b**	10B421 **b** 10B461 **b**
1BD483 **n**	10B262 **b**	10B323 **b**	10D480 **n**	1B1D471 **n** 1B2D471 **n**	1B1D473 **n** 1B2D473 **n**	10B323 **b** 10B262 **b**
1BD484 **n**	10B323 **b**	10B262 **b**	10D486 **n**	1B1D477 **n** 1B2D477 **n**	1B1D474 **n** 1B2D474 **n**	10B252 **a** 10B262 **b**
1CD481 **c**	10B471 **c**	10B431 **c**	10D430 **c**	1CD411 **c**	1CD413 **c** 1CD414 **c**	10B431 **c** 10B471 **c**
1CD482 **c**	10B431 **c**	10B471 **c**	10D436 **c**	1CD447 **c**	1CD444 **c**	10B431 **c**
1CD483 **n**	10B272 **c**	10B252 **a**	10D470 **n**	1A1D471 **n** 1A2D471 **n**	1A1D473 **n** 1A2D473 **n**	10B313 **a** 10B252 **a**
1CD484 **n**	10B272 **c**	10B313 **a**	10D476 **n**	1A1D477 **n** 1A2D477 **n**	1A1D474 **n** 1A2D474 **n**	10B272 **c** 10B282 **d**
1DD481 **d**	10B481 **d**	10B441 **d**	10D440 **d**	1DD411 **d**	1DD413 **d** 1DD414 **d**	10B441 **d** 10B481 **d**
1DD482 **d**	10B441 **d**	10B481 **d**	10D446 **d**	1DD447 **d**	1DD444 **d**	10B441 **d**
1DD483 **n**	10B282 **d**	10B262 **b**	10D480 **n**	1B1D471 **n** 1B2D471 **n**	1B1D473 **n** 1B2D473 **n**	10B323 **b** 10B262 **b**
1DD484 **n**	10B282 **d**	10B323 **b**	10D486 **n**	1B1D477 **n** 1B2D477 **n**	1B1D474 **n** 1B2D474 **n**	10B252 **a** 10B262 **b**

LOCA Load Shed

a = 'A' Div 1E **b** = 'B' Div 1E **c** = 'C' Div 1E
d = 'D' Div 1E **n** = NON-1E

ATTACHMENT 3
INVERTER POWER SUPPLIES
Page 2 of 2

Inverter		Normal AC		Backup AC		DC Bus		Battery		Batt Charg		Charg Pwr	
10D485 NS4 comp	d	10B481	d	10B441	d	10D440	d	1DD411	d	1DD413 1DD414	d d	10B441 10B481	d d
1AD492 BOP comp	n	10B323	b	10B282	d	10D476	n	1A1D477 1A2D477	n n	1A1D474 1A2D474	n n	10B272 10B282	c d
1BD492 BOP comp	n	10B313	a	10B272	c	10D486	n	1B1D477 1B2D477	n n	1B1D474 1B2D474	n n	10B252 10B262	a b
10D496 PA	n	10B411	a	10B451	a	10D410	a	1AD411	a	1AD413 1AD414	a a	10B411 10B451	a a
0AD495 SEC sys	n	10B431	c	10B471	c	10D430	c	1CD411	c	1CD413 1CD414	c c	10B431 10B471	c c
0BD495 SEC sys	n	10B532	n	10B531	n	10D510	n	10D511	n	10D514	n	10B531	n
1AD491 stby lighting	n	DC only	n	10B446 manual xfr	n	10D470	n	1A1D471 1A2D471	n n	1A1D473 1A2D473	n n	10B313 10B252	a a
1BD491 stby lighting	n	DC only	n	10B491 manual xfr	n	10D480	n	1B1D471 1B2D471	n n	1B1D473 1B2D473	n n	10B323 10B262	b b
1CD491 stby lighting	n	DC only	n	10B491 manual xfr	n	10D470	n	1A1D471 1A2D471	n n	1A1D473 1A2D473	n n	10B313 10B252	a a
0DD491 stby lighting	n	DC only	n	10B491 manual xfr	n	10D480	n	1B1D471 1B2D471	n n	1B1D473 1B2D473	n n	10B323 10B262	b b

LOCA Load Shed

a = 'A' Div 1E   **b** = 'B' Div 1E   **c** = 'C' Div 1E
d = 'D' Div 1E   **n** = NON-1E

Facility: Hope Creek

Vendor: GE

Exam Date: 2019

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	264000 A1.03	
	Importance Rating	2.8	

K/A Statement: Ability to predict and/or monitor changes in parameters associated with operating the EMERGENCY GENERATORS (DIESEL/JET) controls including: Operating voltages, currents, and temperatures.

Question: RO #50

Given:

- The “A” EDG is paralleled to the 10A401 bus for a surveillance test IAW HC.OP-ST.KJ-0001, 1AG400 Operability Test.
- Generator Terminal Voltage is at 4160 Volts.
- Generator Phase Current for all three phases is at 600 amps.
- Generator Power is at 4300 KWs.
- Generator Reactive Load is at 1000 KVARS.
- Generator Frequency is at 60Hz.
- All EDG pressures and temperatures are within specifications IAW the surveillance test.

Then:

- The Plant Operator (PO) begins to unload the “A” EDG to remove it from the 10A401 bus by depressing the DIESEL ENG GOV DECR pushbutton, when it becomes stuck in the depressed position.

If NO operator action is taken, what will be the EDG terminal voltage **and** eventual diesel response to this condition?

A. lower AND then the diesel will trip on reverse power.

B. lower AND then the diesel will trip on generator overcurrent.

C. remain constant until the diesel trips on reverse power.

D. remain constant until the diesel trips on generator overcurrent.

Proposed Answer: C

Explanation (Optional): Adjusting the **generator controls** while the EDG is paralleled to the 10A401 bus will adjust the generator load (KW). Adjusting the **voltage regulator** while the EDG is in parallel with the 10A401 bus will adjust the power (KVARS). Adjusting the **voltage regulator** while the EDG is NOT in parallel with the 10A401 bus will adjust the terminal voltage. Reducing KWs or load would eventually cause a trip on reverse power. Reducing load would not be an overcurrent condition. (See attached procedures)

A: **Incorrect.** Due to the diesel being paralleled, the voltage and frequency will remain constant only KWs would be affected with the governor controls.

B: **Incorrect.** Due to the diesel being paralleled, the voltage and frequency will remain constant only KWs would be affected with the governor controls.

C: **Correct.** Due to the diesel being paralleled, the voltage and frequency will remain constant but load will be removed (DIESEL ENG GOV DECR pushbutton) causing a reverse power trip which will trip the diesel generator lockout (86T).

D: **Incorrect -** Lowering on the governor control (DIESEL ENG GOV DECR pushbutton) will remove load not increase load. Therefore, there would not be an overcurrent condition.

Technical Reference(s): HC.OP-SO.KJ-0001 (Attach if not previously provided)

EDG Operations

HC.OP-ST.KJ-0001
"A" EDG Operability Test

HC.OP-AR.KJ-0002
Remote Generator Control Panel
1AC422

Proposed References to be provided to applicants during examination: none

Learning Objective: Given a set of conditions and a drawing of the controls, instrumentation, and/or alarms located in the main control room, determine the status of the Emergency Diesel Generators by evaluation of the controls/instrumentation/alarms in the main control room. (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New **X**

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41 (7)

Comments:

4.4.13. **USING** GEN VR RAISE **OR** LOWER PBs,
LOWER KiloVar (KvAR), loading to approx. 50 to 200 KvAR. ______

> **NOTE**
>
> Control Room Instrumentation may be used to obtain the incremental Kw values in this step.

______

4.4.14. **PRESS** DIESEL ENG GOV DECR PB
AND LOWER generator load in 500 Kw increments,
STOP for 3 to 5 minutes at each 1000 Kw increment.
STOP decreasing generator load between 50 and 200 Kw.
[CD-931D] ______

1. 4000 Kw ______
2. 3000 Kw ______
3. 2000 Kw ______
4. 1000 Kw ______
5. 50 - 200 Kw ______

4.4.15. **TRIP** EDG AG400 40107 Gen Brkr. ______

> **NOTE**
>
> Control Room Instrumentation may be used to obtain the voltage values in this step.

______

4.4.16. **USING** GEN VR RAISE **OR** LOWER PBs
CYCLE Generator terminal voltage
between 4056V and 4394V (one complete cycle)
THEN RETURN to voltage to approximately 4160V. ______

4.4.17. **PRESS** GEN VR MAN PB **AND ENSURE** the MAN light is on. ______

> **NOTE**
>
> Control Room Instrumentation may be used to obtain the voltage values in this step.

______

4.4.18. **USING** GEN VR RAISE **OR** LOWER PBs,
CYCLE Generator terminal voltage
between 4056V and 4394V (one complete cycle),
THEN RETURN to voltage to approximately 4160V. ______

4.4.19. **PRESS** GEN VR AUTO PB **AND ENSURE** the AUTO light is on. ______

2.3 Interlocks

2.3.1. Emergency Diesel Generator auto starts upon receipt of the following signals:

- Voltage at both the 4.16Kv preferred incoming feeder breakers is < 92% of normal voltage. (time delay 20 sec.)
  OR
- 4.16Kv bus voltage is < 70% of normal **AND** the voltage at both the preferred incoming feeder breakers is < 92% of normal voltage.
  OR
- Low-Low Reactor Water Level (-129" inches wide range).
  OR
- High Drywell pressure (1.68 psig).
  OR
- Manual initiation of Core Spray System
  AND
- The following permissives are satisfied:
  - The Barring Device is disconnected.
  - EDG Breaker Failure Lockout Relay (86F) is reset.
  - EDG Backup Lockout Relay (86B) is reset.
  - EDG Regular Lockout Relay (86R) is reset.
  - The Start Failure Relay (SFR) is de-energized.
  - Shutdown Relay (SDR) is de-energized.
  - Control Power Circuits are energized.

2.3.2. Manual operations performed from Local Control Panels are NOT overridden by auto-initiation signals. ______

2.3.3. An Emergency Diesel Generator trips upon receipt of any of the following signals to the Regular 86R or Backup 86B Lockout Relays: ______

- Engine Overspeed
- Generator Regular Differential Overcurrent
- Generator Backup Differential Overcurrent
- Generator Phase Overcurrent
- Emergency Stop PB
- Lube Oil Pressure Low
- Bus Differential Relay for bus to which EDG is connected.

2.3.4. While in TEST, an Emergency Diesel Generator Output Breaker trips upon receipt of any of the following signals to the 86T Test Lockout Relay: ______

- Bus Overcurrent
- EDG Reverse Power
- EDG Low Field Current
- EDG Over Excitation

HC.OP-AR.KJ-0002(Q)

ATTACHMENT 7

LOCATION A-3

GENERATOR OVERCURRENT TRIP	**SETPOINT** N/A **ORIGIN** (1)51X-A(B,C) (2 OUT OF 3)

EXPLANATION

Generator Overcurrent Trip

HC.OP-AR.KJ-0002(Q)

ATTACHMENT 7

LOCATION A-3

CAUSE/SYMPTOM	ACTION
1. Gen. overload condition	1A. **MAINTAIN** Gen. load $\leq$ 4430 Kw.
2. Gen. overvoltage condition	2A. **MAINTAIN** Gen. volts about 4160V. (3828V-4580V)
3. Electrical short circuit	3A. **REQUEST** CRS initiate corrective action.

ATTACHMENT 14

LOCATION **D-4**

GENERATOR
REVERSE
POWER

SETPOINT **N/A**

ORIGIN 32AX

EXPLANATION

Generator Reverse Power

> **NOTE**
>
> No automatic trips are expected for this condition, IF the Emergency Diesel Generator is running as the result of a valid LOCA or LOP signal.

AUTOMATIC ACTION

Trip and lockout D.G. Ckt Brkr (1)52-40107
IF D-G Ckt Brkr fails to open, Emergency Diesel Generator will trip (IF in parallel).

OPERATOR ACTION

1. IF D.G. operating in parallel,
   CHECK D.G. Ckt Brkr (1)52-40107 open
2. **CHECK** tripped (1)32 REVERSE POWER,
   (1)86T TEST LOCKOUT
   AND **RECORD** tripped relays.

CAUSE/SYMPTOM	ACTION
1. Reverse power	1A. **REQUEST** CRS initiate corrective action.

Facility: Hope Creek

Vendor: GE

Exam Date: 2019

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	300000 K3.02	
	Importance Rating	3.3	

K/A Statement: Knowledge of the effect that a loss or malfunction of the INSTRUMENT AIR SYSTEM will have on the following: Systems having pneumatic valves and controls.

Question: RO #51

Given:

- The plant is starting up IAW HC.OP-IO.ZZ-0003, Startup from Cold Shutdown to Rated Power.
- Feedwater flow is being controlled thru the Startup Level Control Valve-LV-1785 manually.
- The “B” RFPT is warmed up and operating in minimum flow recirc.

Then:

- Vibration causes the air supply line to Startup Level Control Valve-LV-1785 AND RFP Minimum Flow Recirculation Valve-FV-1783B to break loose from the valve controllers.

How does the Feedwater System respond?

A. The feedwater pump remains running and the FV-1783B fails open. The startup level control flowpath is lost.

B. The feedwater pump trips when the FV-1783B fails closed. The startup level control flowpath is lost.

C. The feedwater pump remains running and the FV-1783B fails closed. The startup level control flowpath remains in service.

D. The feedwater pump remains running and the FV-1783B fails open. The startup level control flowpath remains in service.

Proposed Answer: A

Explanation (Optional): HC.OP-AB.COMP-0001 -Attachment 2 (See attached)
1AEFV-1783A/B/C - Reactor Feed Pump Recirc Valves - <u>Valves fail open resulting in reduced feedwater flow.</u>
1AELV-1785 - Reactor Start-up Level Control Valve - <u>Valve fails closed resulting in loss of Feedwater flow.</u>

A: **CORRECT-** On loss of air the 1783B fails open and the pump would remain in service. The 1785 valve fails closed and the flow path is lost.

B: **INCORRECT-**. The 1783B fails open and the pump remains running.

C: **INCORRECT-** The 1785 fails closed and the flow path is lost.

D: **INCORRECT-**. The 1785 fails closed and the flow path is lost.

Technical Reference(s): HC.OP-AB.COMP-0001 (Attach if not previously provided)
Instrument and/or Service Air
Attachment 2

Proposed References to be provided to applicants during examination: none

Learning Objective: Explain the reasons for how plant/system parameters respond when implementing Instrument and/or Service Air abnormal. (As available)

Question Source: Bank #120298

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41 (7)

Comments:

SUBSEQUENT OPERATOR ACTIONS (continued)

CONDITION	ACTION
D. Instrument Air Header pressure is $\leq$ 70 psig. Date/Time: ___________	___ ****NOTE 1**** ___ D.1 **ENSURE** HV-7595 is CLOSED
E. Service Air Compressor Tripped OR Failed to Start AND Required to be placed in-service. Date/Time: ___________	___ E.1 **CORRECT** the cause for the Trip/Failure to Auto Start. ___ E.2 **SET** the Post Lube AND Coastdown Timers to Zero. ___ E.3 **START** the Service Air Compressor. (AB-0001)
F. Loss of instrument air to a component. Date/Time: ___________	___ F.1 **REVIEW** system response(s) per Attachment 2 for contingency and restoration actions. ___ F.2 **IMPLEMENT** actions identified above as directed by SM/CRS.

ATTACHMENT 2
Plant Response and Applicable Actions on a Loss of Instrument Air [CD-009F]
Page 1 of 8

SYSTEM	COMPONENT / DESCRIPTION		SYSTEM RESPONSE	CONTINGENCY ACTIONS	RESTORATION ACTIONS
Feedwater	1AEFV-1783A 1AEFV-1783B 1AEFV-1783C	Reactor Feed Pump Recirc Valves	Valves fail open resulting in reduced feedwater flow.	**REDUCE** reactor power as necessary to restore Reactor water level between LEVEL 4 AND LEVEL 7.	**REFER** to HC.OP-SO.AE-0001(Q) to return Reactor Feed Pumps to service.
	1AELV-1785	Reactor Start-up Level Control Valve	Valve fails closed resulting in loss of Feedwater flow.	**REFER** to HC.OP-AB.RPV-0004(Q), Reactor Level Control	

Facility: Hope Creek

Vendor: GE

Exam Date: 2019

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	300000 A2.01	
	Importance Rating	2.9	

K/A Statement: Ability to (a) predict the impacts of the following on the INSTRUMENT AIR SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation: Air dryer and filter malfunctions.

Question: RO #52

Given:

- The plant is at 100% rated power.
- Instrument Air Dryers 00-F-104 and 10-F-104 are both IN SERVICE.
- Instrument Air Dryer 1A-F-104 is in standby with hand switch HS-11416 in the STANDBY position.

When:

- Lowering of the instrument air header pressure to ≈ 80 psig occurs.

The following is the current status of the Instrument Air System:

- The Standby Service Air Compressor has auto started.
- The Emergency Instrument Air Compressor has auto started.
- RACS Demineralizers have isolated.
- Both 00-F-104 and 10-F-104 Air Dryers are INSERVICE.
- Instrument Air Header is at ≈ 78 psig and continues to slowly lower.

Has the Instrument Air System responded properly to the lowering of the Instrument Air Header pressure and what action(s) will need to be completed IAW HC.OP-AB.COMP-0001, Instrument and/or Service Air?

A. Did NOT respond properly; REDUCE Recirc to minimum and LOCK the mode switch in shutdown when instrument air header pressure lowers to 75 psig.

B. Did respond properly; ISOLATE system leaks to determine where the leak is coming from.

C. Did NOT respond properly; ENSURE HV-11416 for 1A-F-104 Instrument Air Dryer is open.

D. Did respond properly; ENSURE HV-7595 Service Air Supply Header Isolation Valve is closed when instrument air header pressure lowers to 70 psig.

Proposed Answer: C

Explanation (Optional): See attached HC.OP-AB.COMP-0001

A: **INCORRECT**, The system did NOT respond properly, however IAW AB.COMP-0001 Retainment Override has the MS in shutdown at 70# with the HV-7595 closed.

B: **INCORRECT**, The system did NOT respond properly with only two of the three dryers INSERVICE. Isolating system leaks would be a proper action for lowering air header.

C: **CORRECT**, The HV-11416 for 1A-F-104 should have opened at 85#. IAW AB.COMP-0001 subsequent action C with Inst. Air Header pressure at 85# all air dryers should be inservice.

D: **INCORRECT**,. The system did NOT respond properly due to Dryer operation. The HV-7595 will auto close at 70#.

Technical Reference(s):	HC.OP-AB.COMP-0001 Instrument and/or Service Air	(Attach if not previously provided)
Proposed References to be provided to applicants during examination:		none
Learning Objective:	Given a set of conditions and a drawing of the controls, instrumentation and/or alarms located on the local control panel, summarize/identify the status of the Instrument Air System or its components by evaluation of the controls/ instrumentation/alarms.	(As available)
Question Source:	Bank #	
	Modified Bank #	(Note changes or attach parent)
	New **X**	
Question History:		
Question Cognitive Level:	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41 (7)	
Comments:		

AUTOMATIC ACTIONS

IF	THEN
< 92 psig Service Air Pressure.	Standby Service Air Compressor Auto Start
≤ 85 psig Emergency Instrument Air Receiver Pressure	• EIAC Auto Start • RACS Demineralizers Isolate
≤ 85 psig Instrument Air Pressure OR Loss of Power to 00-F-104	1-KBHV-7620 OPENS
≤ 85 psig Instrument Air Pressure OR Loss of Power to 10-F-104	1-KBHV-7619 OPENS
≤ 85 psig Instrument Air Pressure OR Loss of Power to 1A-F-104	1-KBHV-11416 OPENS
≤ 70 psig Instrument Air Pressure	1-KAHV-7595 AUTO CLOSES

SUBSEQUENT OPERATOR ACTIONS

CONDITION	ACTION
A. Instrument Air Pressure Lowering. Date/Time: ________	___ A.1 **ANNOUNCE** the following (or equivalent) over the plant paging system: "Attention all plant personnel - Terminate all unnecessary use of Service and Instrument Air. Immediately report all significant air leaks to the Control Room." A.2 At the SM /CRS discretion, **PERFORM** the following: ___ • **START** the standby Service Air Compressor (AB-0001). ___ • **ISOLATE** system leaks **REFER** to guidance in Attachments 2, 3 and HC.OP-AR.KA-0001 for isolating various building risers. ___ **<u>NOTE 1</u>** ___ • **CONSIDER** closing HV-7595. ___ A.3 **PERFORM** the CONTINGENCY <u>AND</u> RESTORATION actions of Attachment 2.
B. Additional Air Dryer Capacity is Required. Date/Time: ________	B.1 **ENSURE** <u>BOTH</u> 00-F-104 <u>AND</u> 10-F-104 Air Dryers In Service as follows: ___ • **ENSURE** 00-F-104 in service by placing Switch HS-7620 in ON. ___ • **ENSURE** 10-F-104 in service by placing Switch HS-7619 in ON. B.2 **PERFORM** the following as directed by the SM/CRS: ___ • **PLACE** 1A-F-104 in service by placing Switch HS-11416 in ON.
C. Instrument Air Header <u>Pressure</u> is $\leq$ 85 psig. Date/Time: ________	C.1 **ENSURE** the following: ___ • EIAC is running IAW AB.ZZ-0001 ___ • HV-7620 is OPEN ___ • HV-7619 is OPEN ___ • HV-11416 is OPEN ___ • RACS Demineralizers have ISOLATED.

HC.OP-AB.COMP-0001(Q)
INSTRUMENT AND/OR SERVICE AIR

SUBSEQUENT OPERATOR ACTIONS (continued)

CONDITION	ACTION
D. Instrument Air Header pressure is ≤ 70 psig. Date/Time: ____________	___ **NOTE 1** ___ D.1 **ENSURE** HV-7595 is CLOSED

HC.OP-AB.COMP-0001(Q)
INSTRUMENT AND/OR SERVICE AIR

RETAINMENT OVERIDE	
CONDITION	**ACTION**
I. 1-KA-HV-7595 is CLOSED AND Instrument Air Header Pressure ≤ 70 psig. (PI-7603A(B) or A3016) AND Instrument Air Header Pressure continues to Lower Date/Time: ____________	___ **I.a** **REDUCE** Recirc. Pump speed to MINIMUM ___ **I.b** **LOCK** the Mode Switch in SHUTDOWN.

Facility: Hope Creek

Vendor: GE

Exam Date: 2019

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	400000 A3.01	
	Importance Rating	3.0	

K/A Statement: Ability to monitor automatic operations of the CCWS including: Setpoints on instrument signal levels for normal operations, warnings, and trips that are applicable to the CCWS

Question: RO #53

Given:

- EA-HV-2346, RACS Heat Exchanger Outlet AND EA-HV-2207 RACS Heat Exchanger Inlet Valves, have isolated.
- No surveillances are in progress.
- No overhead alarms are received due to the valves isolating.

Which of the following caused the valves to stroke closed?

A. RACS room flooded $\geq$ 1" isolation on the "C" channel (LSH-2365C).

B. Reactor Level at -140".

C. Loss of Offsite Power (LOP).

D. High Drywell Pressure at 1.98 psig.

Proposed Answer: **A**

Explanation (Optional): See attached

A: **Correct** - Channel "C" room flooded detector LSH-2365C, will cause isolation with no alarm. A and B detectors will provide an overhead alarm, but don't isolate the HV-2346 and HV-2207 valves. (See attached)

B: **Incorrect** – The valves would isolate on -129" Reactor Level, however HV-2203 and HV-2204 would have isolated too. (see attached)

C: **Incorrect** – The valves are motor operated and would not isolate on a LOP.

D: **Incorrect** - The valves would isolate on 1.68# Drywell Pressure, however HV-2203 and HV-2204 would have isolated too. (see attached)

Technical Reference(s): HC.OP-AB.COOL-0003 (Attach if not previously provided)

RACS

HC.OP-SO.EA-0001 3.3.7
SSW

Proposed References to be provided to applicants during examination: none

Learning Objective: Give a brief description of what happens to the Station Service Water System on the following plant conditions: (As available)

a. LOCA
b. LOP
c. Pipe Break
   - In RACS HX Room
   - Safety-related piping

Question Source: Bank # 72634

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41 (7)

Comments:

IMMEDIATE OPERATOR ACTIONS

NONE

AUTOMATIC ACTIONS

IF	THEN
RACS Pump Room LSH-2365A (1 inch)	• EA-HV-2203 Closes.
RACS Pump Room LSH-2365B (1 inch)	• EA-HV-2204 Closes.
RACS Pump Room LSH-2365C (1 inch)	****NOTE 1**** The following Valves CLOSE: • 1-EA-HV-2207 • 1-EA-HV-2346

NOTES:

1. An isolation signal resulting from the LSH-2365C will NOT result in an overhead alarm.

ADDITIONAL INFORMATION:

Valves:

- EA-HV-2203, LOOP "A" RACS HX HDR SUP VLV.
- EA-HV-2204, LOOP "B" RACS HX HDR SUP VLV.
- EA-HV-2207, RACS HX HDR INLET VLV.
- EA-HV-2346, RACS HX HDR OUTLET VLV.

3.3.7. HV-2203, LOOP A RACS HX HDR SUP, HV-2204, LOOP B RACS HX HDR SUP, HV-2207, RACS HX HDR INLET VLV and HV-2346, RACS HX HDR OUTLET VLV, will close when any of the following conditions exist (HV-2204 will NOT auto close if control is transferred to the RSP):

- Low Reactor Vessel water level (< level 1, -129 inches)
- RACS Rm. Flooded (1 inch above floor)
- High Drywell Pressure (> 1.68 psig)

Facility:	Hope Creek
Vendor:	GE
Exam Date:	2019
Exam Type:	RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	201002 K1.03	
	Importance Rating	3.4	

K/A Statement: Knowledge of the physical connections and/or cause/effect relationships between REACTOR MANUAL CONTROL SYSTEM and the following: Control rod block interlocks/power operation refueling.

Question: RO #54

Given:

- Refueling is in progress.
- Mode switch in REFUEL.
- Fuel Grapple loaded with a fuel bundle.
- Refuel bridge is over the spent fuel pool.
- All control rods fully inserted.

Which one of the following conditions will generate a Rod Block?

A. A control rod is selected on the Rod Select Matrix.

B. The refuel bridge is moved over the core

C. The Main Hoist Loaded light is illuminated.

D. The Fuel Grapple control is placed in the RAISE position.

Proposed Answer: **B**

Explanation (Optional): See attached Refueling Platform Interlocks.

A: **INCORRECT -** Requires second rod being selected when in REFUEL. One rod out interlock.

B: **CORRECT –** See attachment

C: **INCORRECT –**With the fuel bundle loaded the light should be illuminated however, the bridge is currently over the pool, not the core, so no Rod Block until it moves over the core..

D: **INCORRECT -** Even if the grapple is not fully up, no block will occur when taken to “raise” because the bridge is over the pool, not the core.

.

Technical Reference(s): HC.OP-SO.KE-0001 (Attach if not previously provided)

Refuel Platform & Fuel Grapple

Proposed References to be provided to applicants during examination: none

Learning Objective: (As available)

Question Source: Bank #30794

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

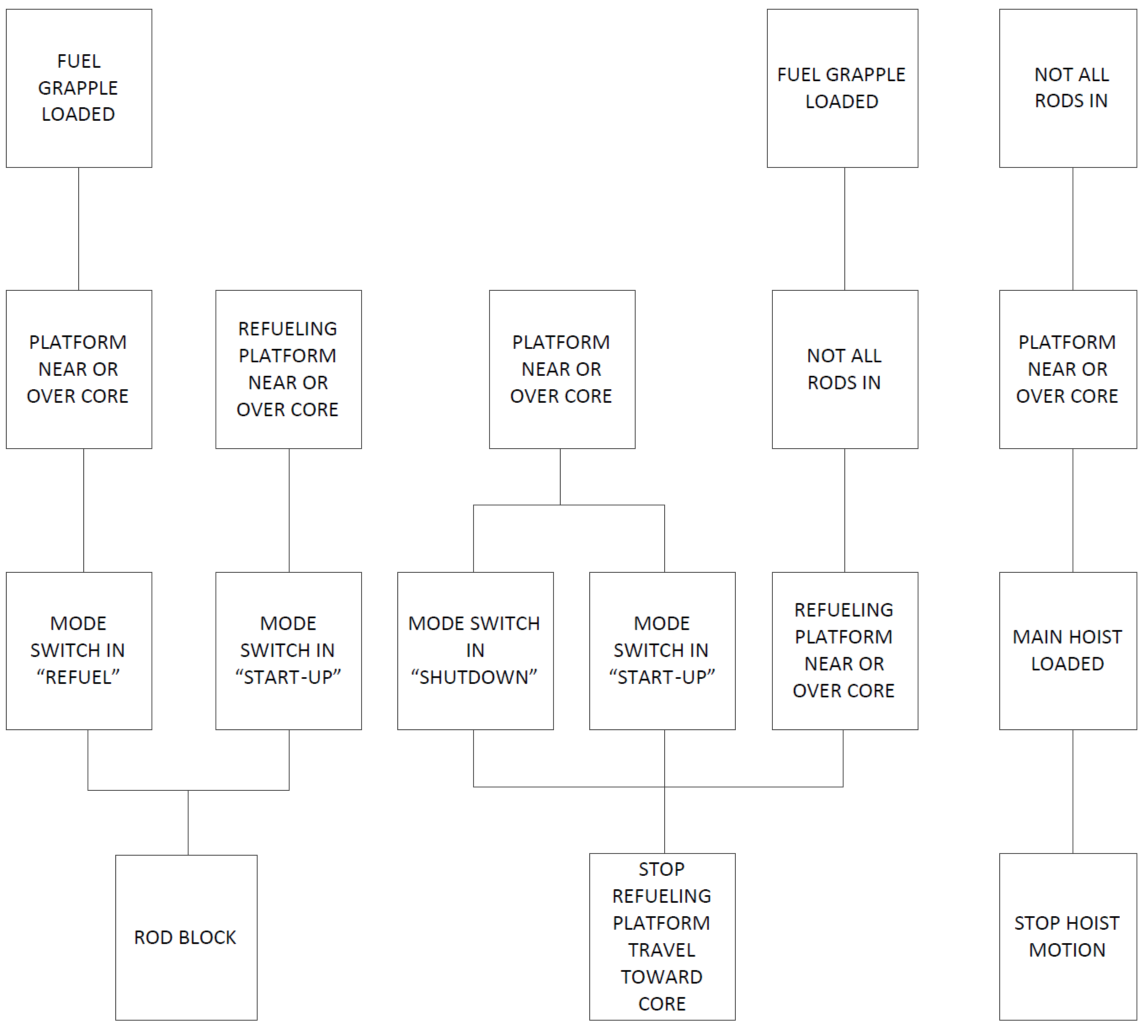
10 CFR Part 55 Content: 55.41 (7)

Comments:

HC.OP-SO.KE-0001(Q)

ATTACHMENT 3
Refueling Platform Interlocks

> **NOTE**
>
> HC.IC.GP.SF-0001 may be performed to allow bridge motion over the core.
> NO CORE ALTERATIONS will be permitted per TS 3.9.1.



Facility: Hope Creek

Vendor: GE

Exam Date: 2019

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	201003 K3.01	
	Importance Rating	3.2	

K/A Statement: Knowledge of the effect that a loss or malfunction of the CONTROL ROD AND DRIVE MECHANISM will have on following: Reactor Power

Question: RO #55

Given:

- The plant is operating at 100% rated power near the end of core life.
- All control rods are fully withdrawn.

When:

- The scram outlet valve for control rod 30-31 develops an air leak and the valve slowly opens.

T=5 minutes:

Which one of the following describes the response of the plant?

Reactor power will ______ (1) ________ and the control rod 30-31 will _______ (2) _________.

A. (1) lower, but the plant will continue to operate at power
(2) insert with NO leakage into the Scram Discharge Volume

B. (1) remain at 100 % reactor power
(2) NOT move along with NO leakage into the Scram Discharge Volume

C. (1) lower, but the plant will continue to operate at power
(2) insert with leakage into the Scram Discharge Volume

D. (1) be <4% on APRMs
(2) insert due to the Reactor Scram on Scram Discharge Volume Level-Hi

Proposed Answer: **C**

Explanation (Optional): The scram valve will leak from the CRDM, mechanism over piston area into the Scram Discharge Volume. The rod will slowly insert. Reactor power will lower but no scram will occur because the SDV vent and drain valves are still open to Radwaste. The SDV will not fill up (no RPS scram setpoint is reached).

A: **Incorrect.** There will be leakage to the SDV due to the outlet valve opening to the SDV.

B: **Incorrect.** Power will lower due to the control rod inserting from the scram outlet valve opening.

C: **Correct.** See above explanation.

D: **Incorrect**. No scram setpoint on SDV level will be reached due to the SDV vents and drain remaining open to Radwaste.

Technical Reference(s): M-47-1 (Attach if not previously provided)

CRD P&ID

Proposed References to be provided to applicants during examination: none

Learning Objective: Given P&ID M-47-1, select the various riser isolation valves, scram pilot valve assemblies, scram valves, directional flow control valves, accumulators and instrumentation assemblies. (As available)

Question Source: Bank #30630

Modified Bank #

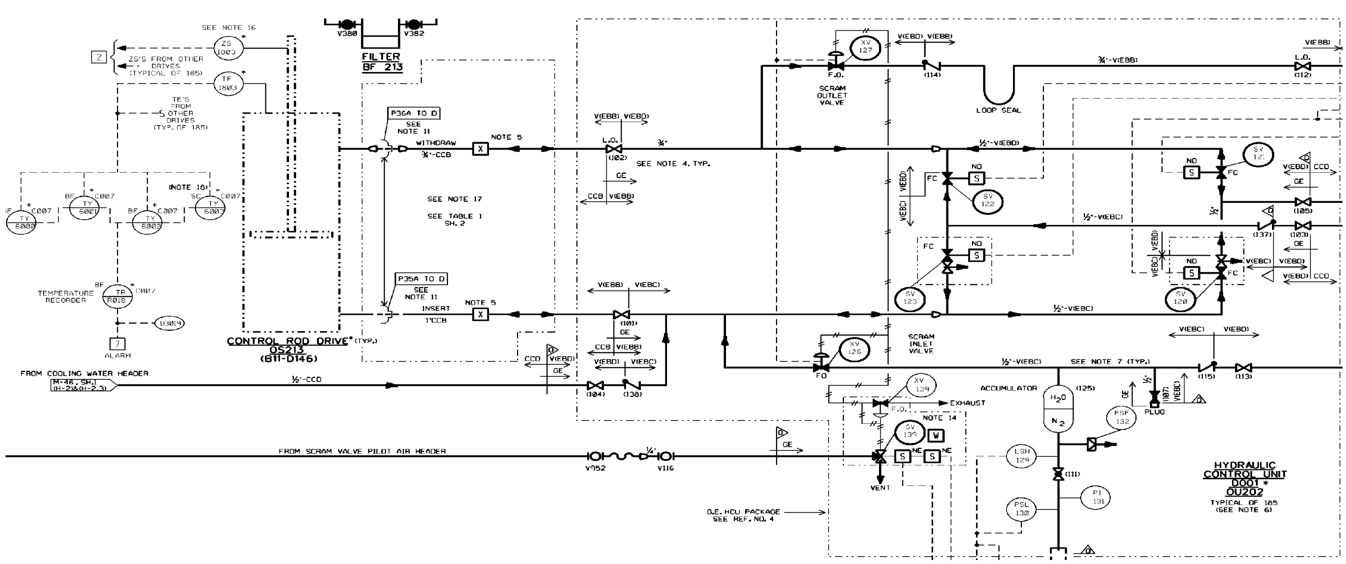
New

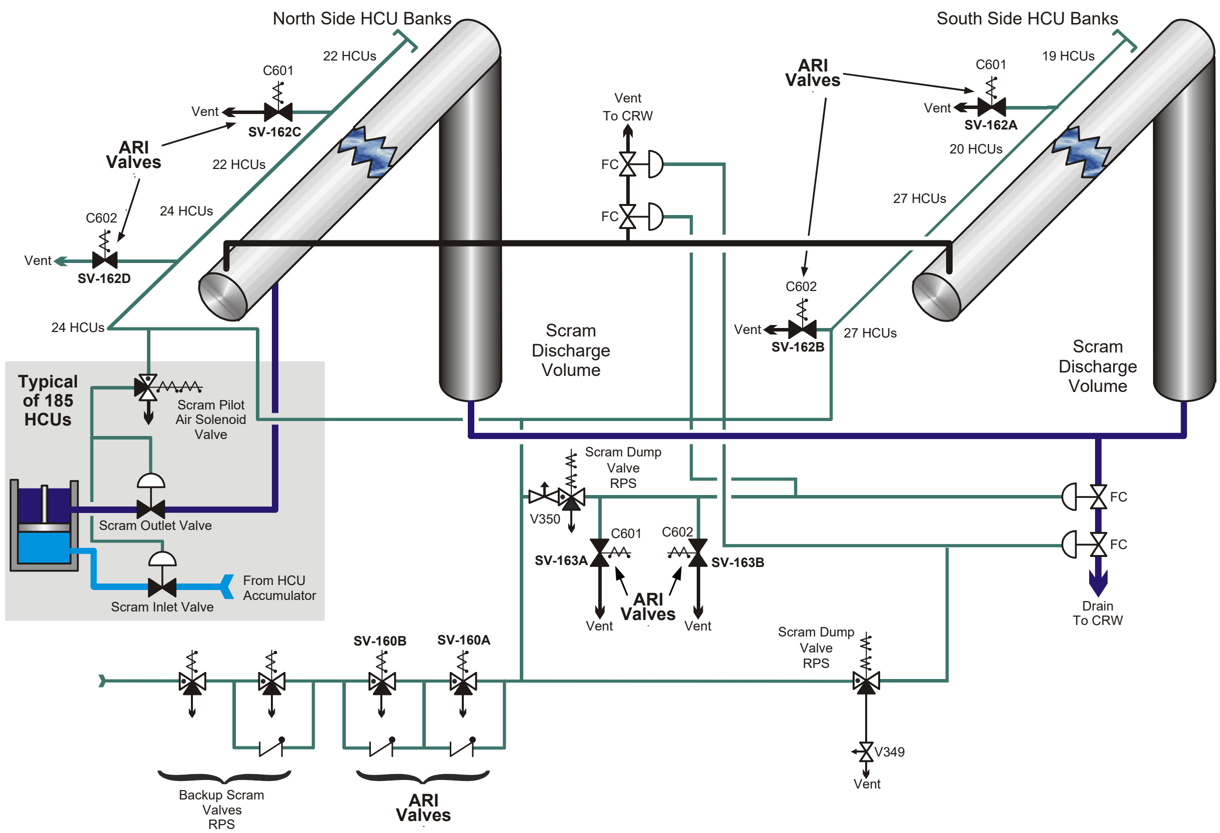
Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41 (6)

Comments:





North Side HCU Banks
South Side HCU Banks
22 HCUs
C601
Vent
SV-162C
ARI
Valves
22 HCUs
24 HCUs
C602
Vent
SV-162D
24 HCUs
Vent
To CRW
FC
FC
ARI
Valves
C601
19 HCUs
Vent
SV-162A
20 HCUs
27 HCUs
C602
Vent
SV-162B
27 HCUs
Scram
Discharge
Volume
Scram
Discharge
Volume
Typical
of 185
HCUs
Scram Pilot
Air Solenoid
Valve
Scram Outlet Valve
Scram Inlet Valve
From HCU
Accumulator
Scram Dump
Valve
RPS
V350
C601
C602
SV-163A
SV-163B
ARI
Valves
Vent
Vent
FC
FC
Drain
To CRW
SV-160B
SV-160A
Scram Dump
Valve
RPS
V349
Vent
Backup Scram
Valves
RPS
ARI
Valves

Facility: Hope Creek

Vendor: GE

Exam Date: 2019

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	202001 K4.01	
	Importance Rating	3.9	

K/A Statement: Knowledge of RECIRCULATION System design feature(s) and/or interlocks which provide for the following: 2/3 core coverage: Plant-Specific.

Question: RO #56

Given:

- The plant is operating at 100% rated power.

When:

- A hole develops in the diffuser section of a jet pump.

Which one of the following is a potential consequence of continued operations with this condition?

A. The vibration related "noise" will cause power oscillations.

B. Less than 2/3 of the core height will be re-flooded following a DBA LOCA.

C. Total reactor power will be higher than expected.

D. The associated reactor coolant loop reactor recirculation pump will lose NPSH.

Proposed Answer: B

Explanation (Optional): See attached system design criteria.

A: **Incorrect-** Vibration related noise will mask the presence of power oscillations, but will not cause power oscillations.

B: **Correct-**. A failure of piping integrity does compromise the ability of the reactor vessel internals to provide a re-floodable volume (2/3 core height).

C: **Incorrect-** The loss of flow through the failed jet pump will lower total core flow and therefore would not raise total core power.

D: **Incorrect-** The recirculation pump suction will be more subcooled as the steam separator/dryer reject flow will be further diluted by the flow bypassing the core through the failed jet pump. NPSH will rise.

Technical Reference(s): 10855-D3.30 (Attach if not previously provided)

Engineering Design Criteria for the Reactor Recirculation System

Proposed References to be provided to applicants during examination: none

Learning Objective: Explain how the Recirculation System is designed to ensure a 2/3 core height re-floodable volume is maintained. (As available)

Question Source: Bank #30807

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41 (3)

Comments:

1 SYSTEM DESIGN CRITERIA

1.1 GENERAL

The Reactor Recirculation system (RRS) circulates reactor water through the reactor core. The circulation rate of the coolant will vary in response to the main turbine steam demand.

1.2 DESIGN REQUIREMENTS BY COMMITMENT

a. The reactor recirculation system shall provide sufficient subcooled water to the core during normal operation to maintain normal operating temperatures. (PSAR Section 4.3.2.1)

b. The reactor recirculation system shall be designed to minimize maintenance situations that require core assembly and fuel removal. (PSAR Section 4.3.2.3)

c. The reactor recirculation system shall be designed so that an adequate fuel barrier thermal margin is assured following recirculation pump malfunctions. (PSAR Section 4.3.3.1)

d. The reactor recirculation system shall be designed so that failure of piping integrity does not compromise the ability of the reactor vessel internals to provide a refloodable system. The break in the recirculation system will not prevent reflooding the core to the level of the jet pump suction inlet. (PSAR Section 4.3.3.2)

e. The reactor recirculation system shall be designed to withstand adverse combination of loadings and forces resulting from operation during abnormal, accident and special event conditions. (PSAR Section 4.3.3.3)

f. The recirculation pumps are to be designed in accordance with the requirements of the ASME Standard Code for Pump and Valves for Nuclear Power. (PSAR Section 4.3.4)

g. The recirculation loops are provided with a system of supports designed to limit pipe motion so that reaction forces associated with any split or circumferential break does not jeopardize containment integrity. This support system provides adequate clearance for normal thermal expansion due to movement of the loop. (PSAR Section 4.3.4)

Facility: Hope Creek

Vendor: GE

Exam Date: 2019

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	204000 K5.05	
	Importance Rating	2.6	

K/A Statement: Knowledge of the operational implications of the following concepts as they apply to REACTOR WATER CLEANUP SYSTEM: Flow controllers.

Question: RO #57

Given:

- Core Alterations are in progress.
- RHR Loop "A" is in Shutdown Cooling.
- RPV Coolant Temperature is at 100°F.
- RHR Loop "B" is in Fuel Pool Cooling Assist.
- Both Fuel Pool Cooling Pumps are Cleared & Tagged for repairs.
- RWCU is in service with Blowdown flow at 25 gpm for level control.
- A flow controller malfunction causes Blowdown Flow Control Valve (HV-F033) to open fully.
- No operator action is taken.

Which of the following will be the effect of the conditions above?

A. Skimmer Surge Tank level will lower until the Fuel Pool weirs are uncovered, and then will remain at the level of the weirs.

B. Cavity level will continue to lower until level reaches +12.5”.

C. Cavity level will lower until the Cavity weirs are uncovered, and then will remain at the level of the weirs.

D. Cavity level will continue to lower until level reaches -38".

Proposed Answer: D

Explanation (Optional): See attached NOTE in the Blowdown Operations (Level control) of the RWCU SOP.

A: **Incorrect:** The Skimmer Surge Tank receives water from the weirs. When level drops below the weirs, tank level will continue to lower and be maintained by makeup from CST.

B: **Incorrect:** RWCU takes suction from the bottom head and receives an isolation signal at -38". Level will not reach -129 inches.

C: **Incorrect:** The mass loss is from the RPV. Cavity weirs only direct water to the Skimmer Surge Tank.

D: **Correct:** RWCU takes suction from the bottom head and receives an isolation signal at -38". The isolation signals for RWCU are NOT bypassed (see attached note).

Technical Reference(s): HC.OP-SO.BG-0001 (Attach if not previously provided)

HC.OP-SO.SM-0001

Proposed References to be provided to applicants during examination: none

Learning Objective: Given a set of conditions and a drawing of the controls, instrumentation and/or alarms located in the Main Control Room, identify the status of the RWCU System or its components by evaluation of the controls/instrumentation/alarms. (As available)

Question Source: Bank #33635

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41 (7)

Comments:

HC.OP-SO.BG-0001(Q)

NOTE
This mode of operation is used to maintain Reactor water level when operating at low power levels due to CRD input AND Reactor coolant expansion during heatup.
Use of RWCU for inventory control in Steps 5.4.4 through 5.4.10 is not considered an OPDRV provided instrumentation and valves that provide automatic isolation are maintained. DO NOT defeat the L2 automatic isolation of both BG-F001 and BG-F004 while performing blow-down operation in OpCon 4 or 5 UNLESS the OPDVR requirements from OP-HC-108-102 are implemented. **[SOER 87-2]** ___

5.4.5. **ENSURE** RWCU return to feedwater flow (CRIDS A2856) remains greater than 10 gpm AND:

A. **THROTTLE** HV-F033, HIC-R606 DRAIN FLOW CONTROL, by pressing INCREASE/DECREASE push-button until Computer Point A2947 RWCU DISCH to COND AND EQUIP DRAIN FLOW indicates desired flow. ___

HC.OP-SO.SM-0001(Q)

OBSERVE the Group 7 Valves listed in Table **SM-007** have closed under the Manual or Automatic Isolation Signals specified.

TABLE SM-007

VALVE NO.	NOMENCLATURE	MANUAL ISOLATION	AUTO ISOLATIONS A	B	C	D	E	F	G	H	I
BG-HVF001	RWCU PMP SUCT CONT INBD ISOLATION VALVE	A NSSSS	X	X	X	X	X	X	X		
BG-HVF004	RWCU PMP SUCT CONT OUTBD ISOLATION VALVE	D NSSSS	X	X	X	X	X	X	X		

ISOLATION	SETPOINT
A - REACTOR VESSEL WATER LEVEL 2	-38"
B - RWCU DIFFERENTIAL FLOW - HIGH	≥ 56 gpm (45 sec time delay)
C - RWCU AREA TEMPERATURE - HIGH (PIPE CHASE, ROOM 4402)	160°F
D - RWCU AREA TEMPERATURE - HIGH (PUMP ROOM & HEAT EXCHANGER ROOMS)	140°F
E - RWCU AREA TEMPERATURE - HIGH (PIPE CHASE, ROOM 4505)	135°F
F - RWCU AREA VENTILATION DIFFERENTIAL TEMPERATURE - HIGH	60°F
G - STANDBY LIQUID CONTROL SYSTEM INITIATED	N/A

Facility: Hope Creek

Vendor: GE

Exam Date: 2019

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	216000 K6.01	
	Importance Rating	3.1	

K/A Statement: Knowledge of the effect that a loss or malfunction of the following will have on the NUCLEAR BOILER INSTRUMENTATION: A.C. electrical distribution.

Question: RO #58

Given:

- 120 VAC UPS TROUBLE Annunciator D3-E3 alarms.
- 1BD481 inverter 120 VAC output is lost.

What affect will the loss of the 120 VAC have on the Nuclear Boiler Instrumentation System?

A. “B” and “D” Channel ECCS Rosemount Trip Units will lose power and the analog RPV level indications will fail upscale.

B. “B” and “F” Channel ECCS Rosemount Trip Units will lose power and the analog RPV level indications will fail upscale.

C. “B” and “D” Channel ECCS Rosemount Trip Units will lose power and the analog RPV level indications will fail downscale.

D. “B” and “F” Channel ECCS Rosemount Trip Units will lose power and the analog RPV level indications will fail downscale.

Proposed Answer: D

Explanation (Optional): Loss of DIV II ECCS/RCIC Auto Trip Units and Start Relays - in general, Process Signal Transmitter failures affecting initiation signals, Min. Flow Valves, pressure permissives, etc. Channels "B" and "F" affected. Loss of instrument power affecting various systems. Instrument **indications fail low**. The student could interpret AD481 for “A” and “C” channels and therefore BD481 for “B” and “D” channels. (AD-DD481 power the four channels respectively).

A: **INCORRECT-** “D” channel is powered from DD481 and the RPV level fails downscale.

B: **INCORRECT** RPV level fails downscale.

C: **INCORRECT-** “D” channel is powered from DD481.

D: **CORRECT-** “B” and “F” channels are powered from BD481 inverter and the RPV level indications fail downscale (see attached).

Technical Reference(s): HC.OP-AB.ZZ-0136 (Attach if not previously provided)

Loss of 120VAC

Proposed References to be provided to applicants during examination: none

Learning Objective: Summarize/identify the systems/components supplied by the Uninterruptible Power Supplies System.
(R) Analyze the power flowpath for a given 1E UPS inverter and determine the affect on the loads if malfunctions occur.

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New **X**

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41 (7)

Comments:

ATTACHMENT 2
1BD481 INVERTER

Automatic Plant Response

1. HV-2522B fails closed (Bezel for HV-2522B/HV-2496B initially indicates OPEN)
   IF TACS is on the "B" Loop, it will automatically swap to "A" Loop.
2. Trip of running FPCC Pumps due to low flow caused by closing of FPC F/D Valves.
3. "B" Channel PCIS initiation/actuation signals as a result of RFE/RBE radiation.
4. RCIC suction swap from the CST to the Suppression Pool due to loss of power to level switches
   H1BD -1BDLSL-N035A & E CNDS STOR TANK LEVEL LO.

Control and Indication Failures

General

1. Channel "B" LOCA/LOP Sequencer is inoperative.
2. Loss of DIV II ECCS/RCIC Auto Trip Units and Start Relays
   - in general, Process Signal Transmitter failures affecting initiation signals, Min. Flow Valves, pressure permissives, etc. Channels "B" and "F" affected.
3. Loss of Optical Isolator power for RHR, Core Spray, RCIC and ADS inhibiting output to annunciators and computer points.
4. BV213 and FV213 FRVS Recirc
   AND BV206 Vent Fans receive trip signals due to false Deluge Activation signals.
5. Loss of instrument power affecting various systems. Instrument indications fail low.
6. Loss of Channel "B" RSP controls.
7. Loss of Channel "B" RBE and RFE Radiation Monitors.
8. Loss of power to 1BC200 H_2/O_2 Analyzer.
9. Loss of power to solenoid/pneumatically actuated valves of various systems.
   ECCS Solenoid Valves lose indication.
10. Loss of 1BJ481 120 VAC power to 10C602 RRCS Panel affects indications and Self-Test Circuits only.
11. Loss of voltage and ampere transducers for 1BD413 and 1BD414 Battery Chargers.
12. Loss of satellite phones in the Control Room, OCC and TSC. Loss of DC remote radio console in Control Room. Hand-held satellite phones are unaffected.
13. Loss of Control Room indication for Spent Fuel Pool level instrument LR-4670B. Display in Lower Relay Room continues to function on independent battery for up to 7 days.

Facility: Hope Creek

Vendor: GE

Exam Date: 2019

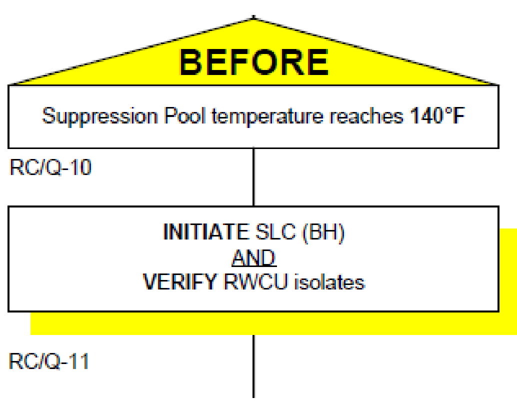
Exam Type: RO

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	2		
	K/A #	223001	A1.09	
	Importance Rating	3.5		

K/A Statement: Ability to predict and/or monitor changes in parameters associated with operating the PRIMARY CONTAINMENT SYSTEM AND AUXILIARIES controls including: Suppression pool temperature.

Question: RO #59

In EOP 101A, “ATWS – RPV Control,” step RC/Q-10, what is the primary basis for initiating Standby Liquid Control (SLC) before suppression pool temperature reaches 140°F?



A. To initiate a reactor shutdown, which will help prevent further suppression pool temperature rise and avoid challenging or exceeding the maximum design temperature limit of the suppression pool (torus) structure.

B. To permit injection of the Cold Shutdown Boron Weight of boron before suppression pool temperature exceeds the Heat Capacity Temperature Limit.

C. To permit injection of the Hot Shutdown Boron Weight of boron before suppression pool temperature exceeds the Heat Capacity Temperature Limit.

D. To initiate a reactor shutdown and ensure cooldown rates of the reactor do not exceed Technical Specification or design requirements.

Proposed Answer: **C**

Explanation (Optional): To avoid depressurizing the RPV with the reactor at power, it is desirable to shut down the reactor prior to reaching the Heat Capacity Temperature Limit. The Boron Injection Initiation Temperature is defined so as to achieve this goal when practicable. The Boron Injection Initiation Temperature (BIIT) is the greater of:

The highest suppression pool temperature at which initiation of boron injection will permit injection of the **Hot Shutdown Boron Weight** of boron before suppression pool temperature exceeds the **Heat Capacity Temperature Limit**

The suppression pool temperature at which a reactor scram is required by plant Technical Specifications.

The BIIT for power levels at or below 4% is approximately **144°F**. The value has been conservatively rounded to **140°F**. (See attached 101A BASES)

A: **INCORRECT-** The maximum design temperature limit of the suppression pool (torus) is **310°F**, and is not close to being exceeded.

B: **INCORRECT** - Knowing differences between Cold versus Hot Shutdown Boron Weight.

C: **CORRECT-** see above explanation.

D: **INCORRECT –** A reactor shutdown is desired, however cooldown rates are not a primary concern.

Technical Reference(s): HC.OP-EO.ZZ-0101A-BASES (Attach if not previously provided)

Proposed References to be provided to applicants during examination: Stem of question EOP-101A RC/Q-10,11

Learning Objective: Given any step of the procedure, explain the reason for performance of that step and/or evaluate the expected system response to control manipulations prescribed by that step. (As available)

Question Source: Bank # #26 on 2016 NRC Exam

Modified Bank # (Note changes or attach parent)

New

Question History: 2016 NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41 (10)

Comments:

7.7 RC/Q-6 Is reactor power above 4% or unknown

RC/Q-10 Before Suppression Pool temperature reaches 140°F

RC/Q-11 Initiate SLC and verify RWCU isolates

If reactor power is below the APRM downscale trip setpoint, tripping the recirculation pumps results in little, if any, reduction in reactor power since power is already near the decay heat level. In this case, forced recirculation flow is continued, if possible, to enhance boron mixing if boron injection is later required.

If suppression pool temperature and RPV pressure cannot be maintained below the Heat Capacity Temperature Limit, emergency RPV depressurization will be required. To avoid depressurizing the RPV with the reactor at power, it is desirable to shut down the reactor prior to reaching the Heat Capacity Temperature Limit. The Boron Injection Initiation Temperature is defined so as to achieve this goal when practicable.

HC.OP-EO.ZZ-0101A-BASES

The Boron Injection Initiation Temperature (BIIT) is the greater of:

- The highest suppression pool temperature at which initiation of boron injection will permit injection of the Hot Shutdown Boron Weight of boron before suppression pool temperature exceeds the Heat Capacity Temperature Limit.
- The suppression pool temperature at which a reactor scram is required by plant Technical Specifications.

The BIIT for power levels at or below 4% is approximately 144°F (See Appendix C, WS-1). The value has been conservatively rounded to 140°F.

The BIIT is a function of reactor power. If boron injection is initiated before suppression pool temperature reaches the BIIT, emergency RPV depressurization may be precluded at lower reactor power levels. At higher reactor power levels, however, the suppression pool heatup rate may become so high that the Hot Shutdown Boron Weight of boron cannot be injected before suppression pool temperature reaches the Heat Capacity Temperature Limit even if boron injection is initiated early in the event.

Since failure-to-scram conditions may present severe plant safety consequences, the requirement to initiate boron injection is independent of any anticipated success of control rod insertion. When attempts to insert control rods satisfactorily achieve reactor shutdown under all conditions without boron, the requirement for boron injection no longer exists, and the operator is directed by the first override in Step LP-1 to terminate boron injection, exit EOP-101A, and enter EOP-101.

Until efforts to insert the control rods are successful and the reactor is shutdown under all conditions without boron, SLC injection continues until the entire usable contents of the SLC tank have been injected. The tank is monitored for this condition in subsequent steps and the SLC pumps are verified to be tripped when this condition is achieved. One other SLC tank level–dependent action exists; within the pressure control leg RC/P, when the Cold Shutdown Boron Weight has been injected, RPV cooldown may be initiated.

Facility: Hope Creek

Vendor: GE

Exam Date: 2019

Exam Type: RO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	2	
K/A #	256000 A2.01	
Importance Rating	3.3	

K/A Statement: Ability to (a) predict the impacts of the following on the REACTOR CONDENSATE SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Pump trips.

Question: RO #60

Given:

- The plant is operating at 60% rated power during a plant startup.
- “A”, “B”, and “C” Primary Condensate Pumps (PCP) are running.
- “B” and “C” Secondary Condensate Pumps (SCP) are running.
- “A” and “C” Reactor Feed Pumps (RFP) are running.

When:

- The “B” and “C” Primary Condensate Pumps (PCP) trip.

Which of the following actions, in regards to the Condensate or Feedwater System pumps, is required IAW HC.OP-SO.AD-0001, Condensate System Operations?
[Reference attached]

A. Manually trip “A” or “C” Reactor Feed Pump.

B. Manually trip “B” and “C” Secondary Condensate Pumps.

C. Manually trip “A” and “C” Reactor Feed Pumps.

D. Manually trip “B” or “C” Secondary Condensate Pump.

Proposed Answer: D

Explanation (Optional): In order to remove Single Point Vulnerabilities for Secondary Condensate Pump (SCP) AUTO TRIP control logic trips, the SCPs are NOT automatically tripped. When two Primary Condensate Pumps (PCPs) are stopped with two or more SCPs running, all but one SCP must be tripped manually. In order to remove Single Point Vulnerabilities for Reactor Feedwater Pump (RFP) AUTO TRIP control logic trips, the RFPs are NOT automatically tripped. When two Secondary Condensate Pumps (SCPs) are stopped with two or more RFPs running, all but one RFP must be tripped manually. **(See attached Pump Trip Logic Attachments)**

A: **INCORRECT**: No feed pumps need to be tripped.

B: **INCORRECT** Need to have one SCP left.

C: **INCORRECT** No feed pumps need to be tripped.

D: **CORRECT**: IAW attachment 3 Condensate Pump Trip Logic of HC.OP-SO.AD-0001.

Technical Reference(s): HC.OP-SO.AD-0001 Attachment 3 (Attach if not previously provided)

HC.OP-AB.RPV-0004

Proposed References to be provided to applicants during examination: Attachment 3 of HC.OP-SO.AD-0001

Learning Objective: Given initial conditions and the loss of one or more condensate pumps, explain the interlocks and automatic actuations associated with the runback and/or trip logic of the condensate, feedwater and reactor recirculation systems. (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New **X**

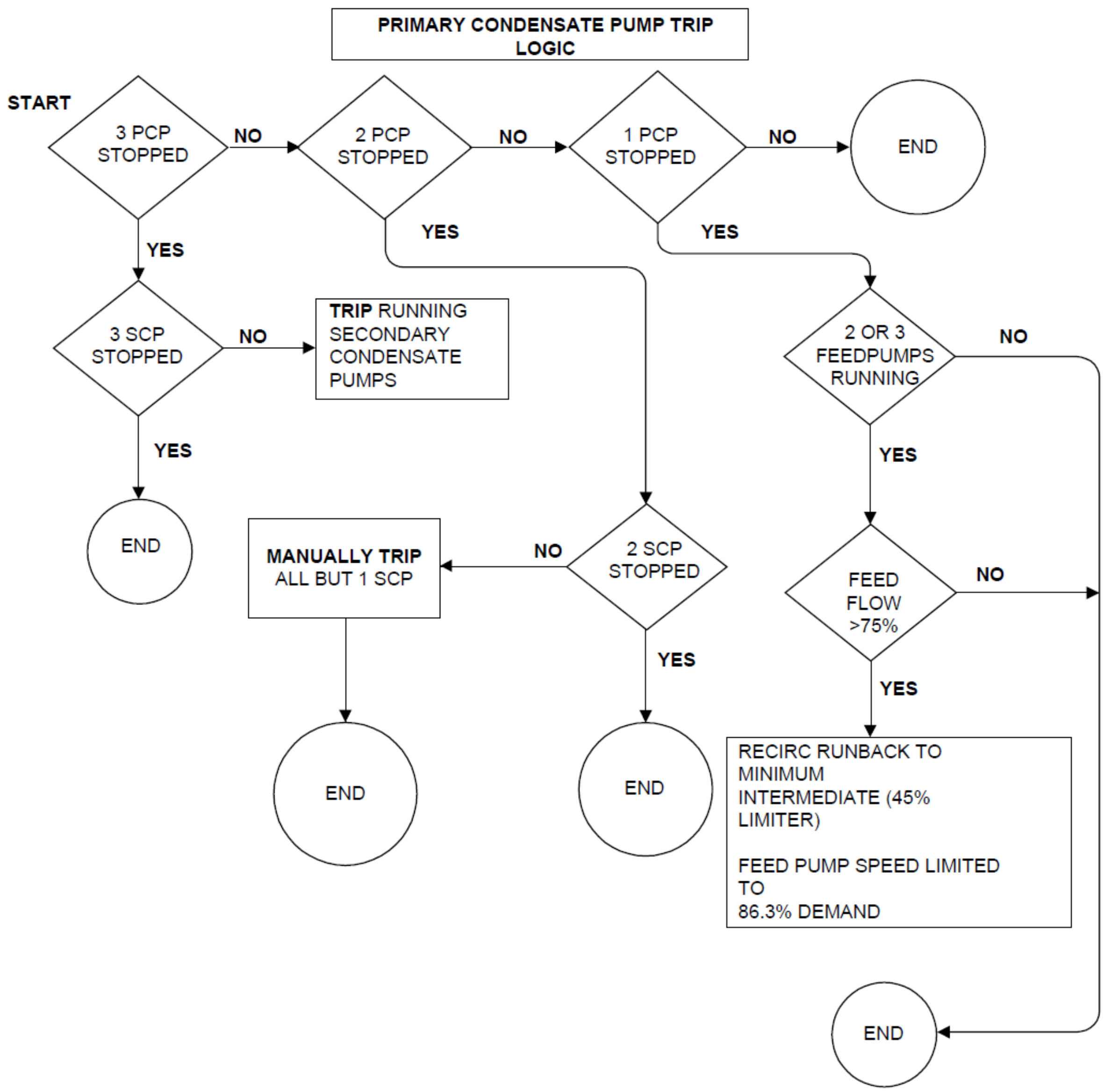
Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41 (10)

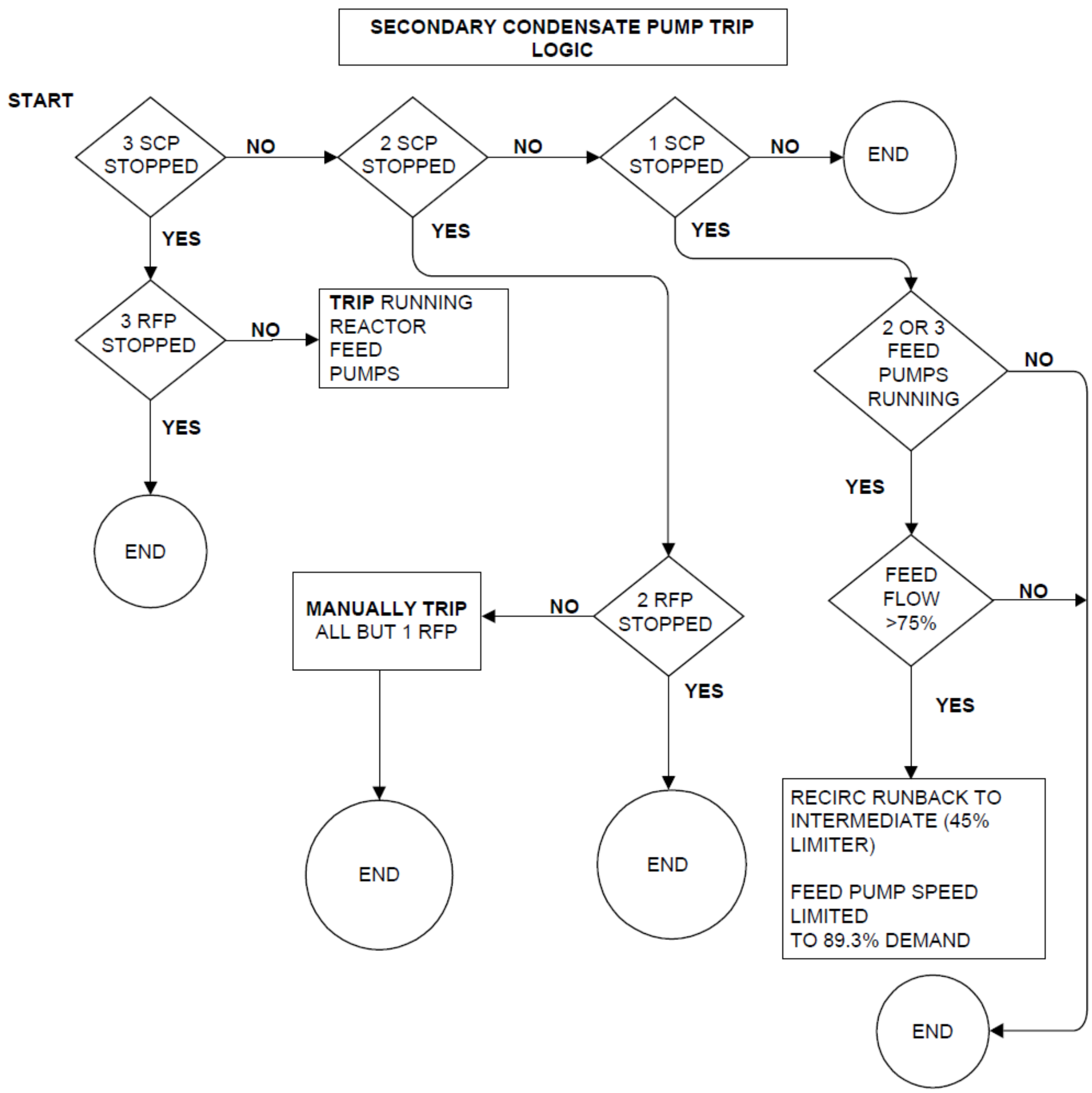
Comments:

ATTACHMENT 3
CONDENSATE PUMP TRIP LOGIC
Page 1 of 2



HC.OP-SO.AD-0001(Q)

ATTACHMENT 3
CONDENSATE PUMP TRIP LOGIC
Page 2 of 2



<u>SUBSEQUENT OPERATOR ACTIONS</u> (continued)

CONDITION		ACTION
D. 1 PCP Tripped. Date/Time: ____________	___ D.1	**ENSURE** Feedwater and Recirc. Pump Runbacks have occurred. (If Armed)
	___ D.2	**ENSURE** Reactor Power is within Feedwater delivery capability of Table 1.
	___ D.3	**CLOSE** HV-1680A(B,C) for tripped pump.
	D.4	**BYPASS** the Reactor Feedwater Runback when conditions permit as follows:
	___	• **OBSERVE** the FLOW >75% ARMED light on RFPT RUNBACK bezel extinguished.
	___	• **PRESS** the PUSH TO BYPASS PCP LIMIT PB.
	___ D.5	<u>WHEN</u> the condition causing the Reactor Recirculation Runback has been corrected, <u>AND</u> <u>WHEN</u> directed by the CRS, **RESET** the Reactor Recirculation Runback.(BB)
E. 2 PCP's Tripped/Stopped. Date/Time: ____________	___	**<u>NOTE 2</u>**
	___ E.1	<u>IF</u> 2 SCP's are <u>NOT</u> TRIPPED, <u>THEN</u> at the discretion of the SM/CRS **TRIP** SCP's to 1 remaining SCP.
	___ E.2	**CLOSE** HV-1680A(B,C) for tripped pumps.
F. 3 PCP's Tripped. Date/Time: ____________	___ F.1	**ENSURE** all SCP's have TRIPPED.
	___ F.2	**CLOSE** HV-1680A, B and C.
	___ F.3	**VERIFY** adequate hotwell levels.

Facility: Hope Creek

Vendor: GE

Exam Date: 2019

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	259001 A3.09	
	Importance Rating	3.0	

K/A Statement: Ability to monitor automatic operations of the REACTOR FEEDWATER SYSTEM including: Lights and alarms.

Question: RO #61

Given:

- The plant is operating at 100% rated power.
- The “A” Reactor Feedwater Pump is in service and in AUTO.

T= 0:

- The reactor failed to scram when required.
- Reactor power is at 14%.
- Reactor pressure is at 1105 psig.
- Reactor water level is at 15”.

T= 1 minute:

- Reactor power is at 10%.
- Reactor pressure is at 1080 psig.
- Reactor water level is at 25”.
- The "A" Reactor Feed Pump speed is at approximately 2500 rpm and remains steady.
- The “RFP TURBINE AUTO XFR TO MANUAL” (B3-F3) annunciator is in alarm.
- The ‘MAN CONTROL AVAIL’ indicating light on 10C651B is illuminated.

(The lowest Reactor water level reached was 15”.)

The "A" Reactor Feed Pump is responding to _________________.

A. the Setpoint Setdown feature of Digital Feedwater Control.

B. a Redundant Reactivity Control System runback.

C. a Control Signal Failure.

D. a gross failure of a Main Steam Flow transmitter.

Proposed Answer: B

Explanation If Rx Pressure exceeded 1071 psig (sealed in) and APRMs are not DOWNSCALE (or are INOP) after a **25 second** time delay, RRCS initiates a runback of any operating RFPT (in AUTO or MAN) to 2500 rpm and transfers the RFPT to MANUAL control. Annunciator RFP TURBINE AUTO XFR TO MANUAL (B3-F3) informs the operator of this condition. After **30 seconds**, control is returned to the operator (MAN CONTROL AVAIL indicating light illuminated). See attached.

A: **Incorrect-** Setpoint setdown does not occur unless Level 3 (12.5”) is reached.

B: **Correct-** With RPV pressure at 1105 psig, RRCS will swap the RFP to manual, initiate a RFP Runback signal and lower speed to 2500 rpm(RRCS Setpoint: >1071psig w/ > 4% pwr for 25 sec.)

C: **Incorrect-** Control Signal Failure will cause the auto transfer to manual, however the speed will remain essentially constant and there is no runback to 2500 rpms associated with this failure.

D: **Incorrect-** A single steam flow transmitter failure will change the Feedwater control from three element to single element. This is not a runback condition for the RFP’s.

Technical Reference(s):	HC.OP-SO.SA-0001 RRCS HC.OP-SO.AE-0001 Feedwater HC.OP-AR.ZZ-0007	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective:	Given a set of conditions and a drawing of the controls, instrumentation and/or alarms located in the Main Control Room, identify the status of the Feedwater Control System or its components by evaluation of the controls/instrumentation/alarms. From memory, describe the three possible RFP runback signals including conditions, setpoints and time delays if applicable.	(As available)

Question Source:	Bank #	36257/30862	
	Modified Bank #		(Note changes or attach parent)
	New		

Question History:

Question Cognitive Level: Analysis and Comprehension

10 CFR Part 55 Content: 55.41 (7)

Comments:

3.0 PRECAUTIONS AND LIMITATIONS

3.1 Precautions

3.1.1. IF resetting ARI will cause the SCRAM to reset, **ENSURE** the Scram Discharge Volume has been surveyed when required. ___

3.2 Limitations

3.2.1. All steps of this procedure shall be performed in sequence unless otherwise noted. ___

3.2.2. Ensure compliance with the requirements of Technical Specifications 3.3.4.1, ATWS Recirculation Pump Trip System Instrumentation. ___

3.3 Interlocks

3.3.1. RRCS-ALTERNATE ROD INSERTION (ARI) will initiate upon receipt of the following signal:

A. Reactor Steam Dome Pressure High (1071 PSIG) ___

OR

B. Reactor Low Low Water Level (Level 2, -38") ___

OR

C. Manual Initiation ARI (Local Test Switches) ___

OR

D. Manual Initiation RRCS ___

3.3.2. RRCS - Feedwater Pump Runback will initiate upon receipt of the following signals:

A. High Reactor Steam Dome Pressure (1071 PSIG) sealed in after 25 second time delay ___

AND

B. APRMs NOT downscale (> 4%) or INOP ___

ATTACHMENT F3

RFP TURBINE **AUTO XFR** **TO MANUAL**

Window Location **B3-F3**

OPERATOR ACTION:

1. **CHECK** REACTOR LEVEL TURB A(B,C) CONT SIG FAIL for indication of which RFP's are affected.
2. **STABILIZE** Reactor water level.
3. **REFER TO** HC.OP-AB.RPV-0004(Q), Reactor Level Control.
4. IF required, affected RFP may be controlled by REACTOR FEED PUMP A(B,C) SPEED CONTROLLER DMND INC
   OR DEC pushbuttons OR if illuminated, TURB CONTROL INC SPEED DEC SPEED pushbuttons.

INPUTS

Digital Point/ Indication	Nomenclature/Condition	Automatic Action
D5363	RFPT FEEDWATER CONTROL SIGNAL.	Affected RFP speed is locked at last correct speed. Speed will remain fixed at this value until it is adjusted manually or returned to automatic control.
	RFPT A(B,C) Speed Control transferred to "M" (Manual) due to one of the following:	.
	• Control Signal Failure	
	• Failure of the RFP common disch press or reactor pressure	
	• RRCS Runback	For the RRCS Runback, control will be returned in the manual mode after a 30 second time delay.

3.3.19. FV-1783A(B,C), Reactor Feed Pump A(B,C) Recirc Vlv, will NOT operate in automatic unless:

- Reactor Feed Pump speed is > 200 rpm ___

  AND

- Reactor Feed Pump disch flow is < 8000 gpm. ___

3.3.20. The following interlocks are associated with the operation of FV-1783, Recirc Flow Control Valve:

- RFPT trip signal (Control Oil pressure) causes the RFPT Recirc Flow Control Valve to close. ___
- When the RFPT speed increases to > 200 rpm, the Recirc Valve receives a 10% open signal if it is in Automatic ___
- Total pump flow exceeding 8000 gpm provides a full closed signal (fast close) to the Recirc Valve. ___
- In AUTO the RFPT A(B,C) Recirc Valve Controller maintains the flow through the Recirc Valve at the auto setpoint unless pump discharge flow exceeds 1000 gpm. When pump discharge flow exceeds 1000 gpm the Recirc Valve Controller output will decrease linearly to 0% demand at 5000 gpm pump discharge flow. ___

3.3.21. The Master Level Controller will automatically switch from single element to three element level control at > 31.4% total steam flow after a 1 minute time delay (assuming all feed flow and steam flow inputs are good) AND will switch from three element to single element level control instantaneously at < 27.8% total steam flow. ___

3.3.22. The Startup Level Controller cannot be placed in automatic mode if total steam flow is > 50%. ___

3.3.23. When the Startup Level Controller is in the AUTO Mode, RFPT(s) placed in AUTO will operate in the startup valve differential pressure mode.
When the Startup Level Controller is in the MANUAL Mode, RFPT's placed in AUTO will operate on the Master Level Controller. ___

Facility: Hope Creek

Vendor: GE

Exam Date: 2019

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	268000 A4.01	
	Importance Rating	3.4	

K/A Statement: Ability to manually operate and/or monitor in the control room: Sump integrators.

Question: RO #62

Given:

- The plant is operating at 100% rated power.
- The first reactor building equipment drain sump pump (AP-266) starts.
- Two minutes later the reactor building equipment sump (AT266) reaches the hi-hi level setpoint.
- OHA D3-C2 “REACTOR BLDG SUMP LVL HI/LO” is in alarm.
- No other overhead alarms are in.

Based on this:

A. The high leak rate detection timer (AP/BP266 TIMER RESET) must be reset.

B. HC.OP-EO.ZZ-0103/4, Reactor Building & Rad Release Control, must be entered.

C. HC.OP-AB.CONT-0006, Drywell Leakage, must be entered.

D. The second reactor building equipment drain sump pump (BP-266) will be running.

Proposed Answer: **D**

Explanation (Optional): Sump High Level - First pump starts. Pumps will switch as to which one starts first. Sump High-High Level - Second pump starts. Sump Low Level - Pumps stop.Sump Low-Low Level - Pumps receive a redundant stop signal. Sump timer reset pbs, which resets a timer used for Hi leak rate detection.

A: **Incorrect:** This timer is for a high leak detection which would bring in the REACTOR BLDG SUMP LEAK HI OHA. See attached HC.OP-AR.ZZ-0014.

B: **Incorrect:** The entry condition is 1" of water on the floor in certain areas IAW the Table 2 of EOP-103. There are no indications of any rooms in the reactor building that are flooding. RX BLDG sump leak hi alarm would be indication of excessive inputs. No entry into EOP-103 is needed.

C: **Incorrect:** There is no indication of Drywell leakage and the Reactor Building sumps would not receive inputs from the drywell if there was a leak. OHA “DRYWELL SUMP LEVEL HI/LO”. No entry into Drywell Leakage AB.

D: **Correct:** the second reactor building equipment drain sump pump should start. - The 2nd sump pump starts on high-high level. See attached HC.OP-AR.ZZ-0014.

Technical Reference(s): HC.OP-AR.ZZ-0014 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: Given a labeled diagram/drawing of the Radwaste System controls/indications identify/state:(NCO and Above)
Function of each indicator
The conditions that will cause the indicator to light or extinguish
Effect of each control switch on the Radwaste System
Conditions or permissives required for the control to perform their intended function. (As available)

Question Source: Bank #111085

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Analysis and Comprehension

10 CFR Part 55 Content: 55.41 (7)

Comments:

HC.OP-AR.ZZ-0014(Q)

ATTACHMENT C2

REACTOR **BLDG SUMP** **LVL HI/LO**

Window Location D3-C2

OPERATOR ACTION:

1. IF level is hi hi
   THEN **ENSURE** both sump pumps in affected sump are running.

2. IF level is lo lo,
   AND pump is running,
   THEN **STOP** associated sump pump.

3. **CONTACT** Radwaste Control Room to verify alarm
   AND to verify PLC Logic Controller lineup.

INPUTS

Digital Point/ Indication	Nomenclature/Condition	Automatic Action
D2265	RB EQUIP DRAIN SUMP LEVEL A	1. Standby pump starts IF level is hi hi. 2. Alarm only IF level is lo lo.
D2266	RB EQUIP DRAIN SUMP LEVEL B	1. Standby pump starts IF level is hi hi. 2. Alarm only IF level is lo lo.
D2267 AT265	RB FLOOR DRAIN SUMP LEVEL N005 North Floor Drain Sump	1. Standby pump starts IF level is hi hi. 2. Alarm only IF level is lo lo.
D2268 BT265	RB FLOOR DRAIN SUMP LEVEL N006 South Floor Drain Sump	1. Standby pump starts IF level is hi hi. 2. Alarm only IF level is lo lo.

HC.OP-AR.ZZ-0014(Q)

ATTACHMENT C1

REACTOR
BLDG SUMP
LEAK HI

Window Location D3-C1

OPERATOR ACTION:

1. **CONFIRM** proper sump pump operation.
2. **CONTACT** RadWaste to verify PLC Logic Controller lineup **AND**, IF sump run is excessive, determine if run times have increased.
3. **INVESTIGATE** possible excess input to sumps.

INPUTS

Digital Point/ Indication	Nomenclature/Condition	Automatic Action
D5874	REAC BLDG EQUIP DRN SUMP A LVL	Alarm only
D5875	REAC BLDG EQUIP DRN SUMP B LVL	Alarm only
D5876 AT265	REAC BLDG FLOOR DRN SUMP A LVL	Alarm only
D5877 BT265	REAC BLDG FLOOR DRN SUMP B LVL	Alarm only

NOTE

Alarm occurs when:

1. Sump pump is unable to reduce level to pump shutoff point before a timer times out, OR
2. Sump refills AND restarts pump before a second timer times out.

Facility: Hope Creek

Vendor: GE

Exam Date: 2019

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	271000 G 2.4.4	
	Importance Rating	4.5	

K/A Statement: Ability to recognize abnormal indications for system operating |parameters that are entry-level conditions for emergency and abnormal operating procedures.-Offgas.

Question: RO #63

Given:

- A plant startup is in progress following a forced outage.
- The plant has been operating with a known fuel leak.
- 'A' Mechanical Vacuum Pump (MVP) is in service with the suction valve throttled.
- The Main Condenser Vacuum Breakers are closed.

What action is required if the South Plant Vent (SPV) RMS Effluent reaches the HIGH (RED) alarm setpoint?

A. Enter HC.OP-AB.CONT-0004, Radioactive Gaseous Release, and stop the MVP.

B. Enter HC.OP-AB.CONT-0004, Radioactive Gaseous Release, and throttle MVP Suction valve further closed.

C. Enter HC.OP-EO.ZZ-0103/4, Reactor Building & Radioactive Release Control, and stop the MVP.

D. Enter HC.OP-EO.ZZ-0103/4, Reactor Building & Radioactive Release Control, and throttle MVP Suction valve further closed.

Proposed Answer: **A**

Explanation (Optional):

A: **Correct.** Required because the HIGH Setpoint is reached. See attached HC.OP-AB.CONT-0004 Step C.1.

B: **Incorrect.** the MVP does not need to be stopped if the MVP suction is throttled until the HIGH alarm setpoint is reached. However, the HIGH setpoint has been reached and the MVP needs to be secured.

C: **Incorrect**. HC.OP-AB.CONT-0004 Step C.1.

D: **Incorrect.** HC.OP-AB.CONT-0004 Step C.1. the MVP does not need to be stopped if the MVP suction is throttled until the HIGH alarm setpoint is reached. However, the HIGH setpoint has been reached and the MVP needs to be secured.

Technical Reference(s): HC.OP-AB.CONT-0004 (Attach if not previously provided)

Radioactive Gaseous Release

Proposed References to be provided to applicants during examination: none

Learning Objective: Recognize abnormal indications/alarms and/or procedural requirements for implementing Radioactive Gaseous Release. (As available)

Question Source: Bank #35504

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41 (10)

Comments:

PSEG Internal Use Only **HC.OP-AB.CONT-0004(Q)**
RADIOACTIVE GASEOUS RELEASE

SUBSEQUENT OPERATOR ACTIONS (continued)

CONDITION	ACTION
C. South Plant Vent Activity Rising. Date/Time: ________	___ ** **NOTE 1**** ___ C.1 IF the South Plant Vent RMS Effluent (9RX580) is at the HIGH alarm setpoint, THEN **STOP** the Mechanical Vacuum Pumps.

PSEG Internal Use Only **HC.OP-AB.CONT-0004(Q)**
RADIOACTIVE GASEOUS RELEASE

NOTES:

1. While operating the Mechanical Vacuum Pump(s) at reduced flow with known or suspected fuel damage, it is not necessary to stop the Mechanical Vacuum Pump(s) until the SPV RMS Effluent (9RX580) alarms at the HIGH alarm setpoint.

Facility: Hope Creek

Vendor: GE

Exam Date: 2019

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	288000 K1.04	
	Importance Rating	2.6	

K/A Statement Knowledge of the physical connections and/or cause/effect relationships between PLANT VENTILATION SYSTEMS and the following: Applicable component cooling water system: Plant Specific.

Question: RO #64

Given:

- The plant is at 100% rated power.
- The ‘A’ Control Room Chiller (AK400) and the circulating water pump (1AP-400) is in-service.
- The ‘A’ TSC Chiller (AK403) and the circulating water pump (1AP-414) is in-service.
- The Reactor Building Ventilation System is in service with Reactor Building Unit Cooler Control Panel, 1A(B,C,D)C-281 hand switches positioned as follows:
  - AVH214 SACS room cooler handswitch in 'AULD'
  - BVH214 SACS room cooler handswitch in 'AULD'
  - CVH214 SACS room cooler handswitch in 'AUTO'
  - DVH214 SACS room cooler handswitch in 'AUTO'

Then:

- The AVH214 SACS room cooler trips.

Which of the following describes how the SACS Room ventilation is affected?

A. Standby SACS Pumps Room Fan CVH214 auto starts. No other auto actions occur.

B. The Standby Control Area Chiller and associated fans auto-start. The Standby SACS Pumps Room Fan CVH214 auto starts.

C. The associated SACS pumps room is without ventilation unless the standby fan is manually started. No other plant equipment is affected.

D. The associated SACS pumps room is without ventilation unless the standby fan is manually started. The Standby Control Area Chiller and associated fans auto-start.

Proposed Answer: B

Explanation (Optional): See attached interlocks 3.3.5 of HC.OP-SO.GK-0001

A: **Incorrect-** The Standby Control Area Chiller and associated fans auto-start.

B: **Correct-** a trip of the fan will cause the associated chill water pump to trip which in turn cause the standby fan (CVH-214) and control area ventilation train to swap automatically. See attached interlocks.

C: **Incorrect-** Standby SACS Pumps Room Fan CVH214 auto starts. The Standby Control Area Chiller and associated fans auto-start.

D: **Incorrect-** Standby SACS Pumps Room Fan CVH214 auto starts.

2019 NRC Written Examination

Technical Reference(s): HC.OP-SO.GK-0001 (Attach if not previously provided)

Control Area HVAC

Proposed References to be provided to applicants during examination: none

Learning Objective: Given plant conditions, summarize/identify the interrelationship between: The Safety Auxiliary Cooling System (SACS) Room Cooler and the Control Area Chilled Water System. (As available)

Question Source: Bank #120293

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41 (7)

Comments:

HC.OP-SO.GK-0001(Q)

3.3.4. Pressing the START PB of any Chiller starts the Program Timer to perform the following: ___

- Starts the Lube Oil Pump
- Prevents restart for fifteen minutes to limit Compressor Motor starts

3.3.5. The tripping of any of the following fans, (which are all being supplied by Control Room Chill Water), will cause a trip of the associated Chilled Water Pump, Chiller, and the remaining fans on the list. This will in turn result in an auto start of the standby Control Room Ventilation System and Chiller: ___

- 1A(B)VH403, Cont Rm Supply Fan
- 1A(B)VH407, Cont Eq Rm Sply Fan
- 1A(C)VH214, A SACS Rm Unit Cooler
- 1B(D)VH214, B SACS Rm Unit Cooler

Facility: Hope Creek

Vendor: GE

Exam Date: 2019

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	290001 K6.01	
	Importance Rating	3.5	

K/A Statement: Knowledge of the effect that a loss or malfunction of the following will have on the SECONDARY CONTAINMENT: Reactor Building Ventilation: Plant Specific.

Question: RO #65

Given:

- The plant has experienced a Loss of Coolant Accident (LOCA) from a high drywell pressure signal of 1.68 psig.
- The LOCA signal has cleared and has been reset.

WHAT actions must be taken to restore the Reactor Building Ventilation Supply (RBVS) and Reactor Building Ventilation Exhaust (RBVE) fans?

The ____(1)____ breakers for both Rx Building Supply and Exhaust fans must be manually closed, then the fans will be restarted from the ____(2)_____.

A. (1) 1-E
(2) local controls (10C382)

B. (1) 1-E
(2) main control room (MCR)

C. (1) Non 1-E
(2) main control room (MCR)

D. (1) Non 1-E
(2) local controls (10C382)

Proposed Answer: **A**

Explanation (Optional): The Class 1-E breakers are located in the respective channel 1-E Unit Sub Station switchgears and **automatically open upon a LOCA signal 1.68 psig drywell pressure (see attached).** The non 1-E breaker is the breaker actuated for routine equipment operation via normal STOP/START control switches at the local panel 10C382. To restore reactor building (secondary containment) ventilation requires the **LOCA signal be reset** and the Unit Sub-Station **1-E breakers to be reclosed.** The Non 1-E breakers, which are in series with the 1-E breakers, will be cycled **when the local start/stop switch is cycled at the local panel 10C382.**

A: **Correct-** See above explanation

B: **Incorrect-** The fans are exclusively operated at the local panel 10C382.

C: **Incorrect-** The non-1E breakers cycle off of the STOP/START local switches. The 1E breakers are upstream and needs to be closed for the non-1E to provide power to the fans. The RBVS and RBVE system is operated from the local panel 10C382.

D: **Incorrect-** The non-1E breakers cycle off of the STOP/START local switches. The 1E breakers are upstream and needs to be closed for the non-1E to provide power to the fans

Technical Reference(s):	HC.OP-SO.GR-0001 RBVS Operations HC.OP-SO.SM-0001 Isolation System Operations	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective:	Given plant conditions associated with the Reactor Building Ventilation Exhaust (RBVE) and Supply (RBVS) system: Summarize/identify the normal line-up Summarize/identify the electrical power supply lineup. Summarize/identify the automatic trips of the electric supply. Summarize/identify the Nuclear Equipment Operator actions required to restore power to the fans following a LOCA/LOP.	(As available)

Question Source:	Bank #115987	
	Modified Bank #	(Note changes or attach parent)
	New	

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41 (7)

Comments:

4.0 INSTRUCTIONS

NOTE

- All operations are performed from the Reactor and Auxiliary Bldg HVAC Panel 10C382 unless otherwise noted.
- Local Panel locations are as follows:
  - 10C382 Reactor and Auxiliary Building HVAC Panel, Serv & Radwaste Bldg 153'
  - 1A(B)(C)C281 Reactor Building Unit Cooler Control Panel, Reac Bldg 102'
  - 1EC281 Cooler Switch Status Alarm Panel, Reac Bldg 102'
  - 00C391 Tech Support Center HVAC Panel, Serv & Radwaste Bldg 153'
  - 1DC281 Reactor Building Unit Cooler Control Panel Reac. Bldg.77'
- During normal operation, the Reactor Building Ventilation System will operate at a dp of approximately - 0.55" WG. **[80035787]**

HC.OP-SO.SM-0001(Q)

OBSERVE the Group 19 Dampers listed in Table **SM-019** have closed under the Manual or Automatic Isolation Signals and other Actions have occurred for Equipment listed as specified.

TABLE SM-019 (1 of 3)

EQUIPMENT NUMBER	NOMENCLATURE	ACTION	MANUAL ISOLATION	AUTO ISOLATIONS			
				A	B	C	D
#GR-HD9414A	REACTOR BLDG SPLY/EXH ISLN INBD EXH	CLOSE	*C CNTMT	X	X	X	X
#GR-HD9414B	REACTOR BLDG SPLY/EXH ISLN OUTBD EXH	CLOSE	*D CNTMT	X	X	X	X
#GU-HD9370A	REACTOR BLDG SPLY/EXH ISLN OUTBD SPLY	CLOSE	*C CNTMT	X	X	X	X
#GU-HD9370B	REACTOR BLDG SPLY/EXH ISLN INBD SPLY	CLOSE	*D CNTMT	X	X	X	X
GT-HD9372A	DRYWELL PURGE EXH DRYWELL VENT	CLOSE	*A CNTMT	X	X	X	X
GT-HD9372C	DRYWELL PURGE SPLY	CLOSE	*A CNTMT	X	X	X	X
GU-HD9395A	FRVS RECIRC BYPASS	CLOSE	*A CNTMT	X	X	X	X
GU-HD9395B	FRVS RECIRC BYPASS	CLOSE	*B CNTMT	X	X	X	X
52-44024	RBVS SUPPLY FAN AVH300	TRIP	*D CNTMT	X	X	X	X
52-41024	RBVS SUPPLY FAN BVH300	TRIP	*A CNTMT	X	X	X	X
52-43024	RBVS SUPPLY FAN CVH300	TRIP	*C CNTMT	X	X	X	X
52-48024	RBVS EXHAUST FAN AV301	TRIP	*D CNTMT	X	X	X	X
52-42024	RBVS EXHAUST FAN BV301	TRIP	*B CNTMT	X	X	X	X
52-45034	RBVS EXHAUST FAN CV301	TRIP	*A CNTMT	X	X	X	X

ISOLATION	**SETPOINT**
A - REACTOR VESSEL WATER LEVEL 2	-38"
B - DRYWELL PRESSURE - HIGH	1.68 psig
C - REFUEL FLOOR EXHAUST RADIATION - HIGH	2×10^{-3} uCi/cc
D - REACTOR BUILDING EXHAUST RADIATION - HIGH	1×10^{-3} uCi/cc

- Group 19 Dampers
* - Can receive a Half Isolation Signal from the corresponding NSSSS Manual Isolation

Facility: Hope Creek

Vendor: GE

Exam Date: 2019

Exam Type: RO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	3	
Group #	1	
K/A #	2.1.4	
Importance Rating	3.3	

K/A Statement: Knowledge of individual licensed operator responsibilities related to shift staffing, such as medical requirements, “no-solo” operation, maintenance of active license status, 10CFR55, etc.

Question: RO #66

You are a licensed Reactor Operator. Due to illness, you have worked the following schedule over the past quarter of 2018 (July thru September).

- July 1 - Off
- July 2 - Off
- July 3 - 12 hour day shift as RO
- July 4 -12 hour day shift as RO
- July 8 -12 hour night shift as RO
- July 9 - 12 hour night shift as RO
- July 10 Through September 30 – Off Shift due to illness.

All licensed operator training is up to date.
You're received medical clearance to stand watch.

Which one of the following describes the status of your license and an additional requirement, if any, to stand watch during the 4th quarter of 2018 IAW OP-AA-105-102 "NRC ACTIVE LICENSE MAINTENANCE?

A. Your license is Inactive. You must perform shift functions under the sole direct supervision of an active licensed RO for at least seven (7) eight hour shifts OR five (5) 12 hour shifts as part of the reactivation and perform a complete plant walkdown.

B. Your license is Active because you stood watch for at least 40 hours the previous quarter. NO additional requirements are needed to stand watch in the 4th quarter of 2018.

C. Your license is Inactive. You must perform shift functions under the sole direct supervision of ONLY an active licensed SRO for at least 40 hours as part of the reactivation and perform a complete plant walkdown.

D. Your license is Inactive. You must perform shift functions under the sole direct supervision of an active licensed RO for at least 40 hours as part of the reactivation and perform a complete plant walkdown.

Proposed Answer: D

Explanation (Optional): See attached OP-AA-105-102 Section 4.1.1 and 4.2.1

A: **Incorrect.** These are the requirements for maintaining an active license not reactivation

B: **Incorrect.**. Previous quarter requirements are not met. See attached section 4.1.1.

C: **Incorrect.** Under the direct supervision of a licensed RO for the RO position. Section 4.2.1

D: **Correct** see attached section 4.2.1 for reactivation which requires 40 hours and under the appropriate position RO.

Technical Reference(s): OP-AA-105-102 (Attach if not previously provided)
NRC Active License Maintenance

Proposed References to be provided to applicants during examination: none

Learning Objective: Provided access to control room references Determine the requirements for maintaining an operator license active. IAW OP-AA-105-102 (As available)

Question Source: Bank #119575

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41 (10)

Comments:

4.0 INSTRUCTIONS

> **NOTE**
>
> The quarterly shift watch requirement may be completed with a combination of complete 8- and 12-hour shifts (in a position appropriately credited for watch-standing proficiency as discussed below) at sites having a mixed shift schedule, and watches shall **NOT** be truncated when the minimum quarterly requirement (56 hours) is satisfied. (NUREG 1021, Revision 9).

4.1 Active License Maintenance

> **NOTE**
>
> As specified in 10 CFR 55.4, "Definitions," "Actively performing the functions of an operator or senior operator," means that "the individual carries out and is responsible for the duties covered by that position".
> For RO and SRO watches being credited for license maintenance, administrative tasks not related to the licensed position should be minimized. Non-position related administrative tasks performed outside the control room, e.g., NRC physicals, all hands meetings, etc., shall not be scheduled during a credited shift. Non-position related administrative tasks performed inside the control room are allowed during a credited shift provided the activities do not impact the individual's ability to perform assigned licensed responsibilities; i.e., the individual is in a position to provide prompt assistance to or oversight of the RO at the controls.

4.1.1. **MAINTAIN** an active license by actively performing the functions of RO, SRO or LSRO.

1. RO licenses by performing the duties of the Unit RO and/or Unit PO for a minimum of seven 8-hour or five 12-hour shifts per calendar quarter, including turnover to the next shift.

2. SRO licenses by performing the duties of Shift Manager or Unit Supervisor for a minimum of seven 8-hour or five 12-hour shifts per calendar quarter, including turnover to the next shift.

3. RO/SRO licenses by performing the duties of Unit RO or Supervisor.

4.2 License Reactivation

NOTE

- If more than 3 months have passed between the time of NRC examination results being issued and issuance of the NRC license, then the license must be activated in accordance with the requirements of this procedure and 10CFR55.53(f).
- The complete plant tour must include all readily accessible major areas of the plant (both units at Salem) routinely toured by in-plant operators that contain safety related equipment. For LSRO's, the plant tour shall consist of all levels of containment as well as the Fuel Handling Building refueling areas.

4.2.1. **REACTIVATE** an RO or SRO license to an "active status" by performing 40 hours of shift functions in the presence and under the sole direct supervision of an active RO or SRO, as appropriate and in the position to which the individual will be assigned. The 40 hours will include completion of a position-specific activation guide containing specific and detailed activation requirements (if required), a plant tour in the presence and under the sole direct supervision of an active RO or SRO, as appropriate, and a review of the position-specific turnover procedures. LSRO's (or SRO licenses that will be activated for fuel handling only) need to complete only one 8-hour shift under the sole direction of an active SRO or LSRO. All parts of the reactivation for LSRO, or SRO licenses activated for fuel handling only, will be performed with the accompaniment of an active SRO or LSRO.

Facility:	Hope Creek
Vendor:	GE
Exam Date:	2019
Exam Type:	RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	1	
	K/A #	2.1.31	
	Importance Rating	4.6	

K/A Statement: Ability to locate control switches, controls, and indications, and to determine that they correctly reflect the desired plant lineup.

Question: RO #67

With the plant at 100% rated power, which of the following set of keys are required to be inserted into their key lock switches IAW OP-HC-112-101-1001, Shift Turnover Responsibilities?

A. Scram Discharge Volume Hi Level Scram Bypass Switch and 4 RPS Channel Switches.

B. Scram Discharge Volume Hi Level Scram Bypass Switch and Rod Worth Minimizer Switch.

C. HPCI and RCIC Steam Line Isolation Valve Switches and 4 RPS Channel Switches.

D. HPCI and RCIC Steam Line Isolation Valve Switches and Rod Worth Minimizer Switch.

Proposed Answer: **A**

Explanation (Optional): At the end of each shift, using OP-HC-108-101-1002, the RO/PO should audit the placement of the keys for keylock control switches on Panels 10C650, 10C651, 1AC633, 1BC633, 10C609, 10C611, and 10C607. These keys are individually labeled according to their own unique control switch function and are normally removed from their respective keylock switches during operation. **The Rx Mode Switch and the Scram Discharge Volume Hi Level Scram Bypass Switch must be inserted into their keylock switches. The 4 RPS Channel Switches must be inserted into their keylock switches.** The remaining keys must be affixed to the control room panels with magnetic strips.

A: **Correct** – See above explanation.

B: **Incorrect** – Rod Worth is off to the side of the rod worth minimizer.

C: **Incorrect** – HPCI and RCIC are attached to magnets and are attached to the panel by the associated valves.

D: **Incorrect** -.Not required to be inserted.

Technical Reference(s): OP-HC-112-101-1001 (Attach if not previously provided)
Shift Turnover Responsibilities
OP-HC-108-101-1002
Key Control

Proposed References to be provided to applicants during examination: none

Learning Objective: From memory State the responsibilities of the following personnel with regard to Key Control: (As available)
WCC Supervisor
NCO-Work Control
Control Room Operators

Question Source: Bank #110671
Modified Bank # (Note changes or attach parent)
New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41 (10)

Comments:

4.2 Shift/Individual Relief and Turnover

4.2.1. Refer to the appropriate section from OP-AA-112-101 for guidance on preparing the shift turnover.

4.2.2. Off-going and On-coming watches will conduct face to face discussion of Protected Equipment.

4.2.3. Utilize appropriate form from this T&RM to document the turnover.

4.3 Off-Going Shift Personnel

4.3.1. At the end of each shift, using OP-HC-108-101-1002, the RO/PO should audit the placement of the keys for keylock control switches on Panels 10C650, 10C651, 1AC633, 1BC633, 10C609, 10C611, and 10C607. These keys are individually labeled according to their own unique control switch function and are normally removed from their respective keylock switches during operation. The Rx Mode Switch and the Scram Discharge Volume Hi Level Scram Bypass Switch must be inserted into their keylock switches. The 4 RPS Channel Switches must be inserted into their keylock switches. The remaining keys must be affixed to the control room panels with magnetic strips.

4.3.2. The appropriate form(s) of this T&RM should be implemented by the Off-Going Shift Personnel.

END of Instructions

ATTACHMENT 3
CONTROL CONSOLE KEY INVENTORY
Page 1 of 3

RPS-A1	RPS TRIP LOGIC A1	H661
RPS-A2	RPS TRIP LOGIC A2	H661
RPS-B1	RPS TRIP LOGIC B1	H661
RPS-B2	RPS TRIP LOGIC B2	H661
	NO ROD SLCT	137
	SCRAM DISCH VOL	402
	RPS MODE SWT	NO ID
	ROD WORTH MINIMIZER	SHH591

Facility: Hope Creek

Vendor: GE

Exam Date: 2019

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	1	
	K/A #	2.1.1	
	Importance Rating	3.8	

K/A Statement: Knowledge of conduct of operations requirements.

Question: RO #68

Given:

- The on-duty RO was selected for random fitness-for-duty testing.
- The CRS provided relief for the RO.
- The time away from the Control Room was 45 minutes.

What is the minimum requirement(s) that must be completed prior to the on-duty RO resuming the watch IAW OP-AA-112-101, Shift Turnover And Relief?

A. Review the Shift Turnover Checklist and a brief of any changes that occurred during the relief period.

B. A brief on plant status to include all safety-related equipment.

C. A complete board walkdown, and Shift Turnover brief is required.

D. Complete a turnover that is comprehensive enough to ensure the relief person is fully cognitive of plant status and aware of ongoing activities.

Proposed Answer: D

Explanation (Optional): (see attached) **CONDUCT** mid-shift turnover and relief between individuals. The turnover and relief should be **comprehensive enough to ensure the relief person is fully cognitive** of plant status and aware of ongoing activities.

- Control room watch station turnover and relief for less than one (1) hour with the individual remaining in the control room area only requires a review of the current shift turnover checklist and update of any deviations of plant status or activities from this checklist.

A: **Incorrect-** Individual who remains in the control room area. FFD is outside the protected area.

B: **Incorrect-** not comprehensive enough. Need to be aware of ongoing activities.
.

C: **Incorrect-** A complete board Walkdown would be required when the RO first takes the watch.
.

D: **Correct-** see above explanation.

Technical Reference(s): OP-AA-112-101 (Attach if not previously provided)

Shift Turnover and Relief

4.1.8 Mid-Shift Turnover

Proposed References to be provided to applicants during examination: none

Learning Objective: From Memory Summarize the definition for the following terms: (As available)
a. Shift Relief
b. Short Term Relief As used in OP-AA-112-101 and OP-HC-112-101-1001.

Question Source: Bank #36016

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41 (10)

Comments:

NOTE
Any turnover in progress may be temporarily suspended to allow participation in the shift turnover meeting. The turnover should be completed immediately after the shift turnover meeting.

4.1.6. **PERFORM** turnovers at the normal workstation unless plant conditions dictate a change (e.g. during an EP condition, or field work in progress).

1. For non-position specific turnover of a job in progress in the field, utilize a general turnover that addresses the following attributes:
   - A. Oncoming operator's name
   - B. Date and Shift of oncoming shift
   - C. Major activities in progress related to the job
   - D. Activities to be accomplished by the oncoming shift
   - E. Special notes or conditions
   - F. Signatures of off-going and oncoming operators acknowledging turnover
2. **If** a turnover occurs in the field during performance of a task, the Shift Manager will ensure the individual(s) receive a follow-up shift turnover briefing.

4.1.7. **DOCUMENT** shift turnovers on a written hard copy or computer generated facsimile or on the operator logs. The turnover documents must contain all pertinent status, items in progress or that have occurred, abnormal conditions and actions being taken and other issues or concerns.

4.1.8. **CONDUCT** mid-shift turnover and relief between individuals. The turnover and relief should be comprehensive enough to ensure the relief person is fully cognitive of plant status and aware of ongoing activities.

- Control room watch station turnover and relief for less than one (1) hour with the individual remaining in the control room area only requires a review of the current shift turnover checklist and update of any deviations of plant status or activities from this checklist.

Facility: Hope Creek

Vendor: GE

Exam Date: 2019

Exam Type: RO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	3	
Group #	2	
K/A #	2.2.6	
Importance Rating	3.0	

K/A Statement: Knowledge of the process for making changes to procedures.

Question: RO #69

Given:

- In the process of reviewing the procedure for a system to be started during your shift, it is determined that an initial condition must be deleted.
- This deletion will change the operation of the system.

Which of the following correctly describes how the change to this procedure should be handled IAW AD-AA-101-101, Implementing and Technical Procedure Process?

A. A partial use.

B. A non-permanent change.

C. An on-the-spot change.

D. A full revision.

Proposed Answer: D

Explanation (Optional): See attached AD-AA-101-101 section 1.3 for “Change of Intent Criteria” and the use of OTSC (on the spot changes) Criteria #12.

A: **Incorrect**. If a procedure cannot be used as written then utilize either OTSC (implementing procedures) or a non-permanent change (O.E.M technical procedures) or full revision. (see attached HU-AA-104-101)

B: **Incorrect.** Used for technical procedures not implementing procedures, full revision is needed due to the change of intent of the procedure (see attached).

C: **Incorrect**. – changes to initial conditions not allowed as OTSC (see Attachment 1)

D: **Correct-** because an OTSC cannot be used for this change a full revision will be required.

Technical Reference(s): AD-AA-101-101 (Attach if not previously provided)

OTSC Process

HU-AA-104-101
Procedure Use and Adherence.

Proposed References to be provided to applicants during examination: none

Learning Objective: From Memory Describe what requirements must be satisfied to make an On-the-Spot change, and the required approval signatures. (As available)

Question Source: Bank #33438

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41 (10)

Comments:

1. **PURPOSE**

1.1. This procedure provides instructions for performing On-The-Spot-Changes (OTSCs) to PSEG Nuclear's Department Implementing Procedures (DIPs) and Operational Excellence Model (O.E.M.) Technical Procedures.

1.2. **Permitted Uses**

1.2.1. On-The-Spot-Change (OTSC) process may be used to revise a procedure **when** the change is of an urgent nature **and** routine procedure revision process will not support resolution of condition in a timely manner.

1.2.2. Changes of a minor nature are **not** considered to be such that their urgency would require processing an OTSC. Normal revision request process (writing a notification) should be used to initiate a change or correction. Job supervisor is expected to provide ample clarification or guidance so that work can proceed without having to change procedure. **[80028543]**

1.2.3. Obvious typographical errors, as determined from proper listing in other steps or other sections of the procedure.

1.3. **Prohibited Uses [80073759]**

- OTSC shall **not** be used for Administrative Procedures.
- OTSC shall **not** be used for T&RMs.
- OTSC shall **not** change intent of procedure as described in Attachment 1, Change of Intent Criteria.
- OTSC change shall **not** be used to change procedures:
  - For rewording or clarification of statements based on personnel preferences, except when failure to do so will prevent task completion
  - For reordering of steps, except when failure to do so will prevent task completion
- In lieu of using a normal revision request (writing a notification) for a procedure in accordance with AD-AA-101-1004, Requesting Changes to PSEG Procedures and T&RMs.

ATTACHMENT 1

CHANGE OF INTENT CRITERIA

“Changes of Intent” are text changes that meet ANY ONE (or more) of the following conditions:

1. A change that alters purpose or scope statement of procedure.
2. A change that affects operating parameter(s) being monitored by Technical Specification Implementing Procedures.
3. A change that affects testing methodology for Technical Specification–required implementing, troubleshooting or special test procedures.
4. A change that affects technical content of Technical Specification acceptance criteria, a setpoint, or a required operating parameter range **unless** change is to bring procedure into alignment with Technical Specifications.
5. A change that affects a condition or activity that could result in exceeding limits allowed by Technical Specifications.
6. A change that affects facility as described in UFSAR or which causes a deviation from Technical Specifications or UFSAR.
7. A change that affects a step or section of a procedure described in the UFSAR. Note that a procedure being “named by title” in UFSAR is not the same as a procedure “described in the UFSAR.” See LS-AA-104-1000 for further definition.
8. A change to a proposed test or experiment not described in UFSAR.
9. A change to technical content of a Non-Technical Specification acceptance criteria, setpoint, or required operating parameter range, **without documented technical justification** in the form of an approved calculation, engineering procedure or engineering evaluation, etc.
10. A change that causes entry into a Limiting Condition for Operation (LCO) Action Statement if not required by or resulting from original procedure.
11. A change in the manner in which equipment or a component is operated that is not consistent with design or function of equipment or component, or causes uncertainty as to how change may affect or impact operation or function of equipment or component.
12. A change to initial conditions or prerequisites of procedure such that change is not consistent with purpose of procedure.
13. A change that alters or deletes an Inspection Hold Point (IHP).
14. A change that affects a current commitment or commitment step(s) such that original intent of commitment is no longer satisfied.
15. A change that affects safe condition of personnel or equipment, or which has potential to introduce ALARA concerns not previously evaluated.
16. A change in amount or level of management oversight or level of approval authority, **unless** change results in an **increased** amount or level of oversight or level of approval.

C. **If** the procedure cannot be performed as written, **then**:

1. **UTILIZE** other appropriate action such as an On-The-Spot Change (implementing procedure) or a Non-Permanent Change (O.E.M. technical procedures) **or**

2. **HALT** the activity and abandon continuation of the procedure, **then WRITE** a revision request notification documenting the reason procedure cannot be performed as written.

Facility: Hope Creek

Vendor: GE

Exam Date: 2019

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	2	
	K/A #	2.2.13	
	Importance Rating	4.1	

K/A Statement: Knowledge of tagging and clearance procedures.

Question: RO #70

IAW the Safety Tagging Program procedure, OP-AA-109-115, Worker's Blocking Tags (WBTs) ________________________.

A. May also contain Yellow Permissive Tags (YPTs) on the Work Clearance Document (WCD).

B. May have more than one simultaneously installed on the same blocking point.

C. The label designating the Worker and Job Technician is placed on the tag by the Clearing Agent.

D. May be used to isolate a high voltage energy source (>600 volts).

Proposed Answer: **A**

Explanation (Optional):

A: **Correct-** WCDs containing WBTs may contain other tag types such as RBTs and YPTs. See Attachment 3, Tagging Rules.

B: **Incorrect-** The WBT shall not be installed on any blocking point that is already tagged with any safety tag except for a White Caution Tag.

C: **Incorrect-** A label designating the Worker and the Clearing Agent shall be placed on the WBT by the Worker.

D: **Incorrect-** WBTs shall not be used to isolate equipment from a High Voltage (>600 volts) energy source. Low voltage.

Technical Reference(s): OP-AA-109-115 (Attach if not previously provided)

Safety Tagging Operations

Proposed References to be provided to applicants during examination: none

Learning Objective: Identify tagging rules and conditions IAW the Safety Tagging Procedure. (As available)

Question Source: Bank #163323

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41 (10)

Comments:

Attachment 3, Tagging Rules (continued)

4. WORKER'S BLOCKING TAG (WBT)

- WBTs are used to place equipment in a safe condition for activities, which require components to be operated as part of testing, inspections or troubleshooting. Approved uses for WBTs are below
- WBT stickers vice tags may be used when the blocking point is a switch/bezel so as to not obstruct the view of other remote switches in the area. WBT stickers do not require the name of the Clearing Agent/Worker to be placed on the tag, so control room operators must validate the worker requesting a control room switch/bezel manipulation is currently on the tagging request.
- WBTs shall not be used if work scope changes from that which the WBT use was initially intended.
- Work shall not be performed on the component to which the WBT is affixed.
- WBTs shall not be used to isolate equipment from a High Voltage (>600 volts) energy source.
- WBTs shall not be placed on any blocking point that is tagged with any other Safety Tag except a White Caution Tag.
- Tagouts containing WBTs will only allow activation of one Job Supervisor and one Worker at a time; only one worker shall be designated to operate the equipment
(or hold responsibility for the tag) at a time.
  - The individual worker specified on Tagout shall be the lead worker for the job (the lead worker may designate another worker directly involved with the job to operate the equipment).
  - After the Clearing Agent has signed onto the request, the Worker shall place a label designating the Worker and the Clearing Agent on the WBT.
  - When work is completed, components tagged with WBTs shall be returned to the tagged position as specified on the Tagout.
If conditions warrant, then the SM/CRS/WCCS may grant permission to leave components in an alternate position. This permission SHALL be documented in the long text or the Tag Removal Notes of the Tagout.
- A Tagout containing WBTs can be turned over from one Clearing Agent/worker to another. This allows jobs to continue from one shift to another without the necessity of swapping tags.
- Tagouts containing WBT(s) may contain other tag types such as red blocking tags and yellow permissive tags.

Facility: Hope Creek

Vendor: GE

Exam Date: 2019

Exam Type: RO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	3	
Group #		
K/A #	2.3.12	
Importance Rating	3.2	

K/A Statement: Knowledge of radiological safety principles pertaining to licensed operator duties, such as containment entry requirements, fuel handling responsibilities, access to locked high-radiation area, aligning filters, etc.

Question: RO #71

The following conditions exist for a job to be performed on a system.

- The general area radiation levels are 20 mrem/hr in the room.
- The hot spot in the room is a pipe elbow that has a radiation level of 50 mrem/hr.
- The job will be performed near the hot spot area.

(Assumptions: ALL 4 cases below have the same transition time to and from destinations. All shielding placement and removal is at 50 mrem/hr. The hot spot with shielding in place is 25 mrem/hr)

Which one of the following methods of job performance would result in the lowest total mrem received (ALARA)?

A. The job is performed by 3 operators for 1.5 hours each on the job at the hot spot and a fourth operator reading instructions in the general room area for an hour.

B. The job is performed by using 2 operators for 3 hours each on the job at the hot spot.

C. 2 Radiation Protection personnel hang/remove lead shielding at the hot spot in 1.0 hours. The job is performed after the lead shielding is in place by using 2 operators for 3 hours each on the job.

D. The job is performed by 2 operators for 2 hours each on the job at the hot spot and a third operator reading instructions in the general room area for 2 hours.

Proposed Answer: **D**

Explanation (Optional):

A: **Incorrect-** The job is performed by 3 operators for 1.5 hrs each on the job at the hot spot and a fourth operator reading instructions in the general room area for 1 hr.(3 operators X 50 mrem/hr x 1.5 hrs) + (1 operators X 20 mrem/hr x 1hr) = **245 mrem/hr.**

B: **Incorrect-** The job is performed by using 2 operators for 3 hrs each on the job at the hot spot. (2 operators X 50 mrem/hr x 3hrs) = **300 mrem/hr.**

C: **Incorrect-** Two Radiation Protection personnel hang and remove 1 tenth thickness of lead shielding on the hot spot in 1.0 hours for the job. The job is performed after the lead shielding is in place by using 2 operators for 3 hrs each on the job. (2 rad techs X 50 mrem/hr x 1.0 hr) + (2 operators X 25 mrem/hr x 3hr) = **250 mrem/hr.**

D: **Correct-** The job is performed by 2 operators for 2 hrs each on the job at the hot spot and a third Operator reading instructions in the general room area for 2 hrs. (2 operators X 50 mrem/hr x 2hr) + (1 operators X 20 mrem/hr x 2hrs) = **240 mrem/hr.**

Technical Reference(s): RP-AA-401 (Attach if not previously provided)

OPERATIONAL ALARA PLANNING AND CONTROLS

Proposed References to be provided to applicants during examination: none

Learning Objective: From Memory Describe what the worker is acknowledging when signing a RWP prior to use. (As available)

Question Source: Bank #120394

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41 (12)

Comments:

ATTACHMENT 4
ALARA Plan
Page 1 of 4

RWP/Task Number: ____________ ALARA Plan: ________ Work Order Number: ____________

Job Description:

Estimated Exposure: __________ Person-rem Estimated Time:________ Person-hours

Exposure Analysis: (e.g., site historical data, challenge exposure goal, etc.)

RWP #/AR #	Description	Person-rem

1. **Expected Radiological Conditions:**
   A. Dose Rates (mrem/hr)

Whole Body	**Skin**	**Extremity**	**Neutron**

Maximum Expected Whole Body Dose Rate: __________ mrem/hr (Gamma)

β/γ	α

β/γ	α

Reference survey data:

B. Contamination (dpm/100cm^2)

C. Airborne (DAC)

SRD Alarm Set Points:	**Low:** 00 mrem	**High:** 00 mrem	**Dose Rate:** 00 mrem/hr

ATTACHMENT 4
ALARA Plan
Page 2 of 4

2. ALARA Plan Estimates

TASK	DETAILED TASK DESCRIPTION	PERSON-HOURS	PERSON-REM
1.			
2.			
3.			
4.			
5.			
6.			
7.			
8.			
9.			
10.			
TOTAL TIME / EXPOSURE ESTIMATES:			

Facility: Hope Creek

Vendor: GE

Exam Date: 2019

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	3	
	K/A #	2.3.7	
	Importance Rating	3.5	

K/A Statement: Ability to comply with radiation work permit requirements during normal or abnormal conditions.

Question: RO #72

Given:

- A RWCU system valve independent verification is being completed in the field.

When:

- The on-duty NCO discovers that two valves on the verification list are in the "A" RWCU pump room.
- The shift radiation protection technician and the independent verifier reviewed the Radiation Work Permit (RWP) survey for the "A" RWCU pump room.
- The general area dose rate at the valves is 115 mRem/hr.
- The job is estimated to take six minutes.

What is the estimated dose the verifier will receive and will the "Hands On" independent verification be required IAW OP-AA-108-101-1002, Component Configuration Control?

A. 19 mRem; independent verification is NOT required.

B. 12 mRem; independent verification is required.

C. 12 mRem; independent verification is NOT required.

D. 19 mRem; independent verification is required.

Proposed Answer: C

Explanation (Optional): Dose rate calculation- **[6 minutes/ 60, then 0.1 x 115= 11.5 mRem, estimate 12 mRem.]** Calculation of 115/ 6 = 19.2 mRem, estimate 19 mRem, if the candidate did not carry the units correctly. IAW OP-AA-108-101-1002 Attachment 5 General Rules for Verification,
If significant cumulative radiation exposure **(10 mRem)** would be received by the person performing the Independent Verification or by persons assisting the performance of the Independent Verification.
"Hands On" Independent Verification is NOT required.

A: **Incorrect:** Dose calculation would be 12 mRem. Independent verification would NOT be required due to the dose >10 mRem.

B: **Incorrect:** Independent verification would NOT be required due to the dose >10 mRem.

C: **Correct: See Explanation.**

D: **Incorrect:** Dose calculation would be 12 mRem. Independent verification would NOT be required due to the dose >10 mRem.

Technical Reference(s): OP-AA-108-101-1002 (Attach if not previously provided)

Component Configuration Control

Attachment 5 (1.5 and 1.5.1)

Proposed References to be provided to applicants during examination: none

Learning Objective: Describe what the worker is acknowledging when signing a RWP prior to use. (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New **X**

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41 (10)

Comments:

ATTACHMENT 5
INDEPENDENT/CONCURRENT VERIFICATION
Page 1 of 4

1. **GENERAL RULES FOR VERIFICATION [CD-787D, CD-828D, CD-927E, CD-037F, CD-421Y, CD-424Y]**

1.1 Equipment operation shall be performed only by qualified personnel in the department responsible for the equipment/system. Equipment lineups shall be performed utilizing approved procedures or Work Clearance Module (WCM) lineups. Station equipment is labeled using unique identifiers that are also incorporated in WCM and/or applicable procedures to reduce the possibility of misoperation as determined by individual Departments. This procedure identifies methods to be used to verify the position of locked/sealed valves.
[CD-408A, CD-435A, CD-828D, CD-147X, CD-390X, CD-424Y]

1.2 A system shall not be considered operable until the Instrument and Controls Department / Qualified Operations Department Personnel have verified the correct position of instrument valves at the discretion of the SM/CRS. **[CD-408A, CD-957D]**

1.3 When a safety related component (Q, Qs, F, R at HCGS) in a system listed in Attachment 6 or 7 is being aligned to its final position to support future operations, an independent verification shall be performed. (Attachment 6 system list is not all-inclusive. For instance, valves listed in HC.OP-ST.ZZ-0002(Q), Primary Containment Integrity Verification - Monthly/Cold Shutdown, require independent verification regardless of whether their systems appear on Attachment 6 or 7.) Non-safety related components in systems listed in Attachment 6 or 7 do not require independent verification unless independent verification of them is required by an implementing procedure (SO, IS, ST, etc.) **[CD-583D] [70021376]**

1.4 Personnel that perform Independent Verifications shall be trained by Operations Training to meet or exceed the requirements of Equipment Operator safety tagging training as described in approved Nuclear Training procedures.

1.5 If significant cumulative radiation exposure (10 mRem) would be received by the person performing the Independent Verification or by persons assisting the performance of the Independent Verification (for example, scaffold builders):

1.5.1. "Hands On" Independent Verification is not required.

1.5.2. Independent Verification shall be accomplished by observing appropriate process parameters such as flow, current, or voltage, annunciator, or status/position.
[CD-828D]

Facility: Hope Creek

Vendor: GE

Exam Date: 2019

Exam Type: RO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	3	
Group #	4	
K/A #	2.4.2	
Importance Rating	4.5	

K/A Statement: Knowledge of system set points, interlocks and automatic actions associated with EOP entry conditions.

Question: RO #73

Given:

- The plant is operating at 100% rated power.

Then:

- A Feedwater Level Control malfunction results in lowering reactor water level.
- Reactor water level reaches +10".
- A full Reactor Recirculation Pump Runback occurs.
- NO other automatic system response occurs.

What are the required Reactor Operator actions for these conditions?

A. Do NOT insert a manual reactor scram until the Feedwater failure has been verified by two separate individuals with the crew entering HC.OP-AB.ZZ-0000, Reactor Scram.

B. Inform the Control Room Supervisor of the condition and insert a manual reactor scram with the crew entering HC.OP-EO.ZZ-0101, RPV Control.

C. Inform the Control Room Supervisor of the condition and when directed, insert a manual reactor scram with the crew entering HC.OP-EO.ZZ-0101, RPV Control.

D. Perform an immediate pressure reduction to raise reactor water level back to normal as soon as possible with the crew entering HC.OP-AB.RPV-0004, Reactor Level Control.

Proposed Answer: B

Explanation (Optional):

A: **Incorrect** – Each operator is required to verify indications using redundant means and then respond accordingly. The RO has the authority and responsibility to scram when operating parameters exceed any RPS setpoint.

B: **Correct –** The RO has the authority and responsibility to scram when operating parameters exceed any RPS setpoint. EOP-101 would be entered for the <+12.5” reactor water level condition.

C: **Incorrect** – The RO is not expected to wait for direction to respond to an RPS failure to scram. EOP-101 would be entered for the <+12.5” reactor water level condition.

D: **Incorrect** - While RPV level restoration is important, the first response should be to the RPS failure. The I.O.A for RPV-0004 level control is to reduce power/pressure to restore level. EOP-101 would be entered for the <+12.5” reactor water level condition.

Technical Reference(s): OP-AA-101-111 (Attach if not previously provided)

ROLES AND RESPONSIBILITIES OF ON-SHIFT PERSONNEL

Section 4.6.2.4

Proposed References to be provided to applicants during examination: none

Learning Objective: From memory. Recognize the conditions under which the Licensed Operator is responsible to shutdown the reactor. (As available)

Question Source: Bank #33060

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41 (10)

Comments:

4.6 Reactor Operators (ROs)

4.6.1. **REPORT** to the CRS.

4.6.2. **OPERATE** the plant in accordance with approved procedures and within the Limiting Conditions for Operation of the Technical Specifications to ensure the reactor is operated in a safe, conservative, and efficient manner at all times.

NOTE
The RO's immediate actions to stabilize the plant during transient conditions take priority over verbalization to the CRS. If possible, verbalization should be accomplished to inform the CRS of actions being taken.

1. During transient conditions, the RO may perform immediate operator actions of abnormal procedures from memory, while verbalizing actions being taken to the CRS.

2. Subsequent actions taken during transient conditions will be based on direction of the CRS per the applicable procedure(s).

3. **COORDINATE** **AND**/**OR** **PERFORM** necessary reactivity changes on the unit during the shift.

4. Shutdown the reactor when the RO determines the safety of the reactor is in jeopardy or when operating parameters exceed any of the reactor protection circuit setpoints and automatic shutdown does **NOT** occur.

5. Manually initiate safety systems' automatic actions when operating parameters exceed the systems' automatic initiation setpoints and automatic initiation does **NOT** occur.

Facility:	Hope Creek
Vendor:	GE
Exam Date:	2019
Exam Type:	RO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	3	
Group #		
K/A #	2.4.39	
Importance Rating	3.2	

K/A Statement: Knowledge of RO responsibilities in emergency plan implementation.

Question: RO #74

You are the Primary Communicator (CM1) when an Unusual Event (UE) is declared.

Which one of the following are the communication time limits required IAW the ECG ATT.6, Primary Communicators Log (CM1) for New Jersey (NJ), Delaware (DE), and the Nuclear Regulatory Commission (NRC)?

A. Notify NJ and DE within 15 minutes, and the NRC within 60 minutes of the event CLASSIFICATION.

B. Notify NJ and DE within 15 minutes, and the NRC within 90 minutes of the event OCCURRENCE.

C. Notify NJ and DE within 15 minutes, and the NRC within 60 minutes of the event OCCURRENCE.

D. Notify NJ and DE within 15 minutes, and the NRC within 90 minutes of the event CLASSIFICATION.

Proposed Answer: **A**

Explanation (Optional): ECG Att. 6 states that the 15 minute clock for NJ and DE starts at the time of event classification. (NRC 60 minutes, ANI Insurer is 90 minutes). (See attached CMI callout)

A: **Correct-** The two states within 15 minutes and the NRC within 60 minutes at the time of the classification NOT occurrence.

B: **Incorrect-** From Classification. NRC is at 60 minutes.

C: **Incorrect**- From Classification.

D: **Incorrect-** NRC is at 60 minutes.

Technical Reference(s): ECG. Att. 6 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: ECG/E-Plan/Fire & Medical Questions (As available)

Question Source: Bank #119576

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41 (10)

Comments:

NOTE

IF **Step 15.a** is NOT SUCCESSFULLY COMPLETED, then **PERFORM Steps 15. b-d**, below.
IF **Step 15.a** is SUCCESSFULLY COMPLETED, then **PROCEED** to **Step 15.e** - Delaware.

______ CM1/TSC1/EOF1

TIME LIMIT - 15 minutes		**INITIAL NOTIFICATIONS**				**TERMINATION OF EVENT**
Step	ORGANIZATION / INDIVIDUALS	NAME OF CONTACT	TIME/ DATE	CALLER		NAME OF CONTACT/ TIME
Step 15.a.	**NEW JERSEY STATE POLICE/OEM** Primary: NETS 5400 Secondary: 609-963-6900, **PRESS 1** BACKUP: EMRAD (not in TSC)	 Call Back:	____ TIME ____ DATE ____ TIME ____ DATE			

Step 15.b	SALEM COUNTY Primary: NETS 5402 Secondary: 856-769-2959		____ TIME ____ DATE			
Step 15.c	CUMBERLAND COUNTY Primary: NETS 5403 Secondary: 856-455-8770		____ TIME ____ DATE			
Step 15.d	U.S. COAST GUARD (Speak Only to Duty Desk) Primary: 215-271-4807 Secondary: 215-271-4940		____ TIME ____ DATE			

PROCEED immediately to next page for Delaware Communications.

NOTE

- IF DEMA has called to report acceptance of emergency responsibilities from Delaware State Police THEN CONTACT DEMA for subsequent Delaware notifications IAW **Step 15.h**. ________ CM1/TSC1/EOF1
- IF **Step 15.e** is NOT SUCCESSFULLY COMPLETED, then **PERFORM Steps 15. f-g**, below. IF **Step 15.e** is SUCCESSFULLY COMPLETED, then **PROCEED** to **Step 30.a** - LAC. ________ CM1/TSC1/EOF1

TIME LIMIT - 15 minutes		**INITIAL NOTIFICATIONS**				**TERMINATION OF EVENT**
Step	ORGANIZATION / INDIVIDUALS	NAME OF CONTACT	TIME/ DATE	CALLER		NAME OF CONTACT/ TIME
Step 15.e	**DELAWARE STATE POLICE / DEMA** Initial contact: : **(DE STATE POLICE)** Primary: NETS 5406 Secondary: 302-659-2341 Backup: NAWAS	 Call Back:	TIME DATE TIME DATE			
Step 15.f	NEW CASTLE COUNTY Primary: NETS 5408 Secondary: 302-571-7331		TIME DATE			
Step 15.g	KENT COUNTY Primary: NETS 5409 Secondary: 302-678-9111		TIME DATE			

For initial notifications, PROCEED immediately to next page for LAC Communications.

NOTE

Step 15.h is to be used when DEMA calls to report acceptance of emergency responsibilities from Delaware State Police.

Turnover from Delaware State Police to DEMA is complete. ☐ Yes ☐ No ________ CM1/TSC1/EOF1

TIME LIMIT - 15 minutes		**INITIAL NOTIFICATIONS**				**TERMINATION OF EVENT**
Step	ORGANIZATION / INDIVIDUALS	NAME OF CONTACT	TIME/ DATE	CALLER		NAME OF CONTACT/ TIME
Step 15.h	Subsequent contact: **(DEMA)** Primary: NETS 5407 Secondary: 302-659-2251, -2256 BACKUP: NAWAS	 Call Back:	TIME DATE TIME DATE			

TIME LIMIT - 60 minutes		INITIAL NOTIFICATIONS				TERMINATION OF EVENT
Step	ORGANIZATION / INDIVIDUALS	NAME OF CONTACT	TIME/ DATE	CALLER		NAME OF CONTACT/ TIME
Step 60.a	**NRC OPERATIONS CENTER ICMF** Primary:(ENS) 301-816-5100 First back-up: 301-951-0550 Second back-up: 301-415-0550 Third back-up: 301-415-0553 FAX 301-816-5151		____ TIME ____ DATE			
No time limit for Data Sheet. As soon as practical.	**NRC Data Sheet** (per above **NOTE**, *not required for initial call to NRC*)		____ TIME ____ DATE			
Step 60.b	**INPO** Emergency Director (ED) 404-290-3977 Assistant Emergency Director (AED) 404-290-3980		____ TIME ____ DATE			

Facility: Hope Creek

Vendor: GE

Exam Date: 2019

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	4	
	K/A #	2.4.29	
	Importance Rating	3.1	

K/A Statement: Knowledge of the emergency plan.

Question: RO #75

Given:

- An ATWS with fuel damage has occurred.
- The Emergency Duty Officer (EDO) decides that it is necessary to send someone into the Reactor Building (with Radiation Protection) to individually scram rods.

What is the maximum allowable dose limit that the EDO may authorize for this evolution?

A. 5 REM

B. 10 REM

C. 25 REM

D. 75 REM

Proposed Answer: C

Explanation (Optional): Planned Emergency Exposure Limit (PEEL) for accident mitigation is **25 REM** EDE. Planned Emergency Exposure Limit (PEEL) for life saving is **75 REM** EDE.

A: **Incorrect-** For accident mitigation it is **25 REM**

B: **Incorrect-** For accident mitigation it is **25 REM**

C: **Correct-** IAW Accident Mitigation Emergency Exposure Criteria (NC.EP-EP.ZZ-0304).

D: **Incorrect-**.For life saving it is **75 REM** EDE.

Technical Reference(s): NC.EP-EP.ZZ-0304 (Attach if not previously provided)

Attachment 3

Proposed References to be provided to applicants during examination: none

Learning Objective: Given an Emergency Event, APPLY principles of the OSC duties, responsibilities, and requirements to determine the actions of an EP OSC Team member in accordance with approved EP Procedures: (As available)

- Describe the Emergency Dose procedure.

Question Source: Bank #36210

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41 (10)

Comments:

ATTACHMENT 3
SELECTION AND AUTHORIZATION FOR EMERGENCY EXPOSURES

- Emergency Exposure should only be authorized by the Emergency Duty Officer (EDO) and cannot be delegated. The SM has this responsibility until the EDO assumes his responsibilities.
- EDO's Emergency Exposure authorization may be done via telephone.
- Emergency Exposure should be voluntary.
- Individuals who do volunteer should:
  - ♦ Have attended and passed Radiation Worker Training
  - ♦ Be above age 45, if available, physically and technically qualified to perform the task.
  - ♦ Not have previously received Emergency Exposure.
- Emergency Exposure received should be added to the individual's current occupational radiation exposure history.
- An individual's exposure is not considered to be an Emergency Exposure if his or her total exposure for the year is $\leq$ 4.5 rem upon finishing an accident mitigation or life saving task and may still volunteer to receive Emergency Exposure.
- Declared pregnant women **SHALL NOT** be allowed to volunteer for Emergency Exposure.

3.0 **EXPOSURE CRITERIA LIFE SAVING EMERGENCY**

- Any and all actions necessary to preserve life, including, but not limited to:
  - ♦ Removal of injured personnel.
  - ♦ Providing medical treatment/first aid.
  - ♦ Providing ambulance service to injured personnel.
- Planned Emergency Exposure Limit (PEEL) for life saving is 75 rem EDE.

4.0 **ACCIDENT MITIGATION EMERGENCY EXPOSURE CRITERIA**

- Any and all actions necessary to mitigate an accident, including, but not limited to:
  - ♦ Performance of actions to prevent immediate deterioration of the plant status.

ATTACHMENT 3
SELECTION AND AUTHORIZATION FOR EMERGENCY EXPOSURES

- ♦ Performance of actions to cause significant reduction of onsite or offsite radiological hazards.

- Planned Emergency Exposure Limit (PEEL) for accident mitigation is 25 rem EDE.
- Emergency Exposure should be voluntary.

2019 NRC Written Examination

Facility: Hope Creek

Vendor: GE

Exam Date: 2019

Exam Type: SRO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		1
Group #		1
K/A #		295016/2.4.35
Importance Rating		4.0

K/A Statement: Knowledge of local auxiliary operator tasks during an emergency and the resultant operational effects. Control Room Abandonment.

Question: SRO #76

Given:

- The reactor was operating at 100% rated power.

When:

- At 1120, dense smoke starts billowing from the control room ventilation ducts.
- At 1125, due to the dense smoke the SM determines that the plant can no longer be operated from the Main Control Room.
- The CRS orders the evacuation of the Main Control Room.
- At 1130, the reactor is scrammed remotely from the RPS distribution panels.
- At 1150, the SM determines that emergency takeover is successful and control has been established at the Remote Shutdown Panel (RSP).

Which of the following describes how the scram can be verified IAW HC.OP-IO.ZZ-0008, Shutdown from Outside Control Room and how should this event be classified?

A. RPS Backup Scram Air Solenoids verified de-energized; Site Area Emergency.

B. RMCS Activity Control Cards; Site Area Emergency.

C. RPS Backup Scram Air Solenoids verified de-energized; Alert.

D. RMCS Activity Control Cards; Alert.

Proposed Answer: B

Explanation (Optional): IAW IO-0008 step 5.1.3 "If the Rx scram was not verified prior to evacuating the Control Room, then verify Rods Full In (SPDS/CRIDS or RMCS Activity Control Cards OR Other). IAW EP-HC-325-112 (see attached). The time to establish control of the plant from outside the control room exceeded **15 minutes**. With control of the RSP being established after **15 minutes**, RPV water level and/or RCS heat removal is not under control of the crew, therefore the event is classified as a site area emergency.

A: **INCORRECT-**. BU scram valves are de-energized with the scram reset. If they were energized, that would indicate a reactor scram.

B: **CORRECT** – see above explanation.

C: **INCORRECT-** Alert is the classification for transferring control to the RSP, but due to the transfer taking **>15 minutes** the classification is upgrade to site area emergency.

D: **INCORRECT-** Alert is the classification for transferring control to the RSP, but due to the transfer taking **>15 minutes** the classification is upgrade to site area emergency.

Technical Reference(s):	HC.OP-IO.ZZ-0008 Shutdown from Outside Control Room EP-HC-325-112 Control Room Evacuation	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: EALs and RALs (No EAL attachments)

Learning Objective:	Given plant conditions and plant procedures, determine required actions of the retainment override(s) and subsequent operator actions in accordance with Control Room Environment. ECG/E-Plan/Fire & Medical Questions	(As available)

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

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content:

55.43 (5)

Comments

Section H - Hazards
H6 – Control Room Evacuation

Initiating Condition	Control Room evacuation resulting in transfer of plant control to alternate locations	Inability to control a key safety function from outside the Control Room
OPCON	All	All
EAL #	**HA6.1**	**HS6.1**
EMERGENCY ACTION LEVELS	**IF** An event has resulted in plant control being transferred from the Control Room to the Remote Shutdown Panel (RSP)	**IF** An event has resulted in plant control being transferred from the Control Room to the Remote Shutdown Panel (RSP) **AND** Control of **ANY** of the following key safety functions is **NOT** reestablished within **15 min.** of the last licensed operator leaving the Control Room: • Reactivity (OPCON 1 and 2 **only**) • RPV water level • RCS heat removal (Note 1)
	THEN	**THEN**
Action Required	Refer to Attachment 1 **ALERT**	Refer to Attachment 2 **SITE AREA EMERGENCY**

Note 1: The Emergency Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded

H6

5.0 PROCEDURE

NOTE

Each step should be initialed upon completion of the step.

Attachment 3 should be referred to for RSP redundant instrumentation/ controls.

Attachment 4 should be referred to for placing 'A' Loop RHR in Suppression Pool Cooling.

Attachments 7 and 8 should be referred to for plant communications information. When dispatching an operator to a remote shutdown control station, the operator should be provided with an electro-sound-powered phone OR radio to assist with communication. ___

5.1 Establish Control from Outside the Control Room

5.1.1. **ENSURE** that all prerequisites have been satisfied IAW Section 2.0 of this procedure. ___

NOTE

IF the Reactor was NOT scrammed AND the MSIVs are still open, then the Feedwater System AND the Main Turbine Bypass Valves may be regulating Rx level AND Rx pressure at this time.

Opening the circuit breakers in Attachment 13 will deenergize the RPS busses, scramming the plant, AND deenergize the NSSSS busses, closing the MSIVs.

10C410(10C411) RPS PWR Dist. Panels A(B) are located in Control/DG Bldg. El. 54'. ___

5.1.2. IF required to Scram the reactor or close the MSIVs, **PERFORM** the Attachment 13. ___

5.1.3. IF the Rx scram was NOT verified prior to evacuating the Control Room, THEN **VERIFY** Rods Full In on CRIDS/SPDS, OR RMCS Activity Control Cards; ___

A. IF unable to verify Rods Full In on CRIDS/SPDS OR RMCS Activity Control Cards THEN CRS may **DIRECT** another method of verifying the scram. ___

Vendor: GE

Exam Date: 2019

Exam Type: SRO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #		295019/AA2.01
	Importance Rating		3.6

K/A Statement: Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR: Instrument air system pressure.

Question: SRO #77

Given:

- Hope Creek is operating at 100% rated power.

When:

- An instrument air line in the Turbine Building ruptures.
- The air compressors are unable to keep up with the loss of air.
- Instrument air header pressure is lowering quickly.

What will be the strategy for Reactor Pressure Vessel (RPV) level control and pressure control IAW OP-HC-103-102-1013, Transient Mitigation Strategies?

A. SRVs for pressure control, HPCI/RCIC for level control.

B. Main Steam Line Drains for pressure control, HPCI/RCIC for level control.

C. SRVs for pressure control, Maximize CRD for level control.

D. Main Steam Line Drains for pressure control, Maximize CRD for level control.

Proposed Answer: **A**

Explanation (Optional): Outboard MSIVs will go closed on a loss of air, therefore NO steam for feedpumps or use of the main condenser for decay heat (heat sink). Condensate will be unavailable due to NO feedpath on a loss of air. CRD flow control valves will fail closed with the loss of level control through the cooling water path.

A: **CORRECT** – SRVs (pressure) with HPCI/RCIC (level). EOP-101 and IAW OP-HC-103-102-1013, Transient Mitigation Strategies.

B: **INCORRECT** - Condenser is NOT available for pressure control and not IAW Transient Mitigation Strategies.

C: **INCORRECT** - CRD flow control valves fail closed on a loss of air and NO condensate line up is possible due to level control valves fail closed on a loss of air. (AB-COMP-0001)

D: **CORRECT** - Condenser is NOT available and NO condensate line up is possible due to level control valves fail closed on a loss of air and not IAW Transient Mitigation Strategies.

Technical Reference(s): EOP-101 (Attach if not previously provided)
RPV Control

HC.OP-AB.COMP-0001

Instrument/Service Air
OP-HC-103-102-1013 Transient Mitigation Strategies

Proposed References to be provided to applicants during examination: none

Learning Objective: Given a system or component that is either physically connected to or required for support of the Instrument Air System or emergency instrument air compressor, assess the interrelationship (As available)
Given any step of the procedure, explain the reason for performance of that step and/or evaluate the expected system response to control manipulations prescribed by that step

Question Source: Bank # 116024

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

MONITOR AND CONTROL RPV PRESSURE

RC/P-1

IF while executing the following steps:	THEN:
DRYWELL pressure is above **1.68 psig**	**PREVENT** injection from Core Spray AND LPCI pumps **NOT** required to assure adequate core cooling prior to depressurizing the RPV below their maximum injection pressures
Emergency RPV Depressurization is anticipated per EOP 102 OR EOP 103/4	**OPEN** the Main Turbine Bypass Valves, **EXCEED** cooldown rate, if necessary
Rapid RPV Depressurization has been required	**EXIT** RC/P AND **ENTER** 202
Emergency RPV Depressurization is OR has been required	**EXIT** RC/P AND **ENTER** 202
RPV level **CANNOT** be determined	**EXIT** RC/P AND **ENTER** 206

RC/P-2

IF while executing the following steps:	THEN:
Supp pool temperature **CANNOT** be maintained below Curve SPT-P	**MAINTAIN** RPV press below the limit, **EXCEED** cooldown rate, if necessary
Steam Cooling is required	**EXIT** RC/P AND **GO** to SC-1 SC
SRVs are being used to stabilize pressure AND PCIG is OR becomes unavailable	**PLACE** the control switch for each SRV in the CLOSE or AUTO position

RC/P-3

STABILIZE RPV press below **1037 psig** with the following

- Main Turbine Bypass Valves (AB.ZZ-0001)

Pressure control may be augmented by:

- SRVs (AB.ZZ-0001) ONLY when supp pool level is above **0 in** — 3
- Main Steam Line Drains (AB.ZZ-0001)
- HPCI (AB.ZZ-0001) with suction from the CST — 2, 3 4, 7, 9, 10
- RCIC (AB.ZZ-0001) with suction from the CST
- SJAEs (CG)
- RFPTs (AE)
- Remote operation of SRVs, ONLY when supp pool level is above **0 in** **USE** HC.OP-AM-TSC-0024

RC/P-4

Table 1	**PREFERRED INJECTION SYSTEMS**	
SYSTEMS	PRESSURE	NOTES
Feedwater (AB.ZZ-0001)	720-1250	
HPCI (AB.ZZ-0001) (2, 3 4, 7, 9, 10)	100-1250	Suction from CST, if available IF Necessary: • Bypass high torus level suction transfer, **USE** OP-AB.ZZ-0001 • Bypass high area temperature isolation, **USE** OP-AB.ZZ-0135 Attachment 2 • Bypass high RPV water level trip **USE** OP-EO.ZZ-0325
RCIC (AB.ZZ-0001) (8)	65-1250	Suction from CST, if available IF Necessary: • Bypass low RPV pressure isolation, **USE** OP-EO.ZZ-0321 • Bypass high area temperature isolation, **USE** OP-AB.ZZ-0135 Attachment 2 • Bypass high RPV water level shutdown **USE** OP-EO.ZZ-0324
CRD (BF)	0-1500	IF Necessary: • System Re-Start **USE** OP-AB.ZZ-0001 • 2-Pump Operation (BF)
Condensate (AD)	0-720	
Core Spray (AB.ZZ-0001)	0-380	IF Necessary, throttle CS injection valves **USE** OP-EO.ZZ-0326
LPCI (AB.ZZ-0001)	0-340	Inject through HXs as soon as possible **USE** OP-AB.ZZ-0001 IF Necessary: • Throttle RPV injection flow- 'A' or 'B' loops **USE** OP-EO.ZZ-0323 • Throttle RPV injection flow- 'C' or 'D' loops **USE** OP-EO.ZZ-0326

ATTACHMENT 2
Plant Response and Applicable Actions on a Loss of Instrument Air [CD-009F]

SYSTEM	COMPONENT / DESCRIPTION		SYSTEM RESPONSE	CONTINGENCY ACTIONS	RESTORATION ACTIONS
Feedwater	1AEFV-1783A 1AEFV-1783B 1AEFV-1783C 1AELV-1785	Reactor Feed Pump Recirc Valves Reactor Start-up Level Control Valve	Valves fail open resulting in reduced feedwater flow. Valve fails closed resulting in loss of Feedwater flow.	**REDUCE** reactor power as necessary to restore Reactor water level between LEVEL 4 AND LEVEL 7. **REFER** to HC.OP-AB.RPV-0004(Q), Reactor Level Control	**REFER** to HC.OP-SO.AE-0001(Q) to return Reactor Feed Pumps to service.
Condensate	1ADFV-1650A 1ADFV-1650B 1ADFV-1650C 1ADLV-1657-1 1ADLV-1657-2 1ADFV-1677	Secondary Condensate Pump Recirc Valves Condensate Makeup Condensate Reject SCP Suction Reject Bypass	Valves fail open, probable Feed Pump trip on Lo-Lo suction pressure. Condensate Makeup and Reject valves fail closed, SCP Suction Reject Bypass fails open - causes continuous reject to the CST with no hotwell makeup	* **REDUCE** Reactor power as necessary to restore Reactor water level between LEVEL 4 AND LEVEL 7. **REFER** to HC.OP-AB.RPV-0004(Q), Reactor Level Control * **SEND** an operator to TB 77'. * As necessary, **OPEN** 1-APV-091, Condensate Make-up Bypass Valve to restore Hotwell level.	**REFER** to HC.OP-SO.AD-0001(Q), Condensate System Operation, Section 5.1 to return the Condensate System to service.
Containment	1ABHV-F028's 1GSHV-4958 1GSHV-4980 1GSFV-4971 1GSHV-4978 1GSHV-4979 1GSHV-4956 1GSHV-4962 1GSHV-4964 1GSHV-11541 1GSHV-4963: 1GSHV-4950: 1GSHV-4951: 1GSHV-4952:	Outboard MSIV's Containment Atmosphere Control Isolation Valves	MSIV's and Containment Atmosphere Control Isolation Valves fail closed resulting in loss of Main Condenser as a heat sink and normal Containment make-up and vent paths.	* **ENSURE** valves fail closed. * **ENTER** EOP-101, RPV Control, as required to monitor and control RPV pressure, * **ENTER** EOP-102, Primary Containment Control, as required to monitor and control Drywell pressure.	* **REFER** to HC.OP-SO.AB-0001(Q) to return the Main Steam System to service. * **REFER** to HC.OP-SO.GS-0001(Q), to return the Containment Atmosphere Control System to service.
CRD	1BFFV-F002A 1BFFV-F002B	CRD Flow Control Valves	Valves fail closed - Loss of CRD Drive and Cooling water. The charging water header is still supplied.	**REFER** to HC.OP-AB.IC-0001(Q) CONTROL ROD.	**REFER** to HC.OP-SO.BF-0001 to reestablish CRD system flow and pressure.

5.3.5. Aggressive EOP Cooldowns with SRV's

A. Related Fundamentals (Plant Observed Phenomena)

- Opening an SRV caused approximately 10" or more of swell.
- Closing an SRV caused approximately 20" of shrink
- Opening an SRV for 1 minute starting at 1000 psig results in:
  - A level reduction of approximately 15"
  - A pressure reduction of approximately 150 psi

B. Strategies

- Maintain an RPV level band of +30" to -30" when depressurizing with SRVs and maintaining RPV level with HPCI/RCIC. With RPS not reset, it is not necessary to announce every 12.5" RPV level re-entry condition when maintaining this RPV level band. Note 3 on EOP-101 provides the guidance for using this band and re-entry into EOP-101 is not necessary each time 12.5" RPV level is reached since EOP-101 steps RC/L- 5 and RC/L-6 are being used to continually assess if RPV can be maintained above -129" while in the RPV level band.
- When the swell can be sustained without causing a Level 8 trip with an SRV open, maintain RPV level by establishing an injection rate of 1500 gpm to 2000 gpm (.75 to 1 Mlbs/hr)
- Close the SRVs at the high end of the level band and allow injection to turn level following the shrink
- Reduce injection flow to that necessary to maintain pressure in the new band and if necessary slowly raise water level back into preferred RPV level band. This minimizes the reactor pressure drop due to "overfeeding."

Facility: Hope Creek

Vendor: GE

Exam Date: 2019

Exam Type: RO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		1
Group #		1
K/A #		295021 G2.4.47
Importance Rating		4.2

K/A Statement: Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material.- Loss of Shutdown Cooling.

Question: SRO #78

Given:

- Shutdown cooling is in service.

T= 12:00

- A complete loss of shutdown cooling occurs.
- RPV temperature is at 148°F.

T= 12:20

- RPV temperature is at 182°F.

(Assume the heat up rate is constant)

Which one of the following describes how the current heat up rate will affect the plant Operational Condition and Technical Specification (TS) heat up limits?

A. Before T=12:30, a mode change would occur. At T=13:00, the TS heatup rate limit will be exceeded.

B. After T=12:30, a mode change will occur. At T=13:00, the TS heatup rate limit will be exceeded.

C. After T=12:30, a mode change will occur. At T=13:00, the TS heatup rate limit will NOT be exceeded.

D. Before T=12:30, a mode change would occur. At T=13:00, the TS heatup rate limit will NOT be exceeded.

.

Proposed Answer: B

Explanation (Optional): TS 3.4.6.1.a. & TS definitions Mode change occurs at >200°F. per TS definitions. The TS heatup limit is < or =100°F in a one hour period.

A: **Incorrect-** 199°F at 12:30.

B: **Correct-** Mode change has NOT occurred before 12:30 (199°F). TS heat up Limit will be exceeded form 148°F to 250°F by 1300 (102°F/hr).

C: **Incorrect-** 102°F/hr from 1200 to 1300.

D: **Incorrect-** 199°F at 12:30. 102°F/hr from 1200 to 1300.

Technical Reference(s):	TS 3.4.6.1.a	(Attach if not previously provided)
	Reactor Coolant System	
	PTLR	
	TS Definitions	

Proposed References to be provided to applicants during examination: none

Learning Objective:	Given access to Technical Specifications, select those sections which are applicable to the RPV & Internals IAW HCGS Technicals Specifications.	(As available)

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

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43 (5)

Comments:

REACTOR COOLANT SYSTEM

3/4.4.6 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.6.1 The reactor coolant system temperature and pressure shall be limited in accordance with the limits specified in the PTLR with:

a. A maximum heatup rate within limits specified in the PTLR,

b. A maximum cooldown rate within limits specified in the PTLR,

c. A maximum temperature change within limits specified in the PTLR during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves, and

d. The reactor vessel flange and head flange metal temperature shall be maintained within limits specified in the PTLR when reactor vessel head bolting studs are under tension.

APPLICABILITY At all times.

- Normal Operating Heat-up/Cool-down rate limit (Figures 2, 5, and 8: Curve B – non-nuclear heating, and Figures 3, 6, and 9: Curve C – nuclear heating): ≤ **100°F/hour**[2] [7].
- RPV bottom head coolant temperature to RPV coolant temperature ΔT limit during Recirculation Pump startup: ≤ **145°F** [1].
- Recirculation loop coolant temperature to RPV coolant temperature ΔT limit during Recirculation Pump startup: ≤ **50°F** [1].
- RPV flange and adjacent shell temperature limit ≥ **79°F** [7].

To address the NRC condition regarding lowest service temperature in Reference [2, Section 4.0], the minimum temperature is set to 79°F, which is equal to $RT_{NDT,max}$ + 60°F. This value is consistent with the minimum temperature limits and minimum bolt-up temperature in the current docketed P-T curves [9] (approved for use by the NRC in Reference [10]) and bounds the minimum temperature in the first set of P-T limits approved for initial operation [11].

5.0 Discussion

The adjusted reference temperature (ART) of the limiting beltline material is used to adjust beltline P-T curves to account for irradiation effects. RG 1.99 [5] provides the methods for determining the ART. The RG 1.99 methods for determining the limiting material and adjusting the P-T curves using ART are discussed in this section.

The vessel beltline copper (Cu) and nickel (Ni) values were obtained from the evaluation of the HCGS vessel plate, weld, and forging materials [6]. This evaluation included the results of two surveillance capsules for the representative plate and weld materials. The Cu and Ni values were used with Table 1 of RG 1.99 to determine a chemistry factor (CF) per Paragraph 1.1 of RG 1.99 for welds. The Cu and Ni values were used with Table 2 of RG 1.99 to determine a CF per Paragraph 1.1 of RG 1.99 for plates and forgings. However, for materials where surveillance data

[2] Interpreted as the temperature change in any 1-hour period is less than or equal to 100°F.

TABLE 1.2

OPERATIONAL CONDITIONS

CONDITION	MODE SWITCH POSITION	AVERAGE REACTOR COOLANT TEMPERATURE
1. POWER OPERATION	Run	Any temperature
2. STARTUP	Startup/Hot Standby	Any temperature
3. HOT SHUTDOWN	Shutdown#,***	> 200°F
4. COLD SHUTDOWN	Shutdown#,##,***	≤ 200°F+
5. REFUELING*	Shutdown or Refuel**,#	≤ 140°F

#The reactor mode switch may be placed in the Run, Startup/Hot Standby, or Refuel position to test the switch interlock functions and related instrumentation provided that the control rods are verified to remain fully inserted by a second licensed operator or other technically qualified member of the unit technical staff. If the reactor mode switch is placed in the Refuel position, the one-rod-out interlock shall be OPERABLE.

##The reactor mode switch may be placed in the Refuel position while a single control rod drive is being removed from the reactor pressure vessel per Specification 3.9.10.1.

*Fuel in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

**See Special Test Exceptions 3.10.1 and 3.10.3.

***The reactor mode switch may be placed in the Refuel position while a single control rod is being recoupled or withdrawn provided that the one-rod-out interlock is OPERABLE.

+See Special Test Exception 3.10.8.

Facility: Hope Creek

Vendor: GE

Exam Date: 2019

Exam Type: SRO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		1
Group #		1
K/A #		295023 AA2.04
Importance Rating		4.1

K/A Statement: Ability to determine and/or interpret the following as they apply to REFUELING ACCIDENTS: Occurrence of fuel handling accident.

Question: SRO #79

Given:

- The plant is operating at 100% rated power.
- Irradiated fuel is being moved in the spent fuel pool.

When:

- A spent fuel bundle is dropped from the refueling bridge grapple, resulting in a Refuel Floor HVAC Exhaust rad level of 1.5E-03 µCi/ml and steady.

Based on this, the CRS should enter ___________ and _____________.

A. EOP 103/4, "Reactor Building & Rad Release Control"
restore rad release rate below 5.25E+6 µCi/sec.

B. EOP 103/4, "Reactor Building & Rad Release Control"
monitor and control reactor building radiation levels, reactor building temperatures, and reactor building floor levels.

C. HC.OP-AB.CONT-0005, "Irradiated Fuel Damage"
return the dropped spent fuel bundle to its original location.

D. HC.OP-AB.CONT-0005, "Irradiated Fuel Damage"
within 60 minutes, implement OP-HC-108-115-1001, Attachment 5, Operability Evaluation Log, to seal secondary containment breaches.

Proposed Answer: **B**

Explanation (Optional):Both procedures would be entered, however the procedural hierarchy would have you enter the EOP for direction of mitigation. Within each procedure the directions given in the distractors are not for the current conditions.

A: **Incorrect.** There are no indications of an offsite release and certainly not at the ALERT level needed for entry into 103/4.

B: **Correct.** EOP 103/4 and monitor and control reactor building temps, reactor building rad levels, reactor building floor levels. Concurrently IAW the EOP. (see attached)

C: **Incorrect**. There is no direction to do this at this time IAW the AB. This would be a post event evolution. If the fuel was still attached to the grapple then IAW AB.CONT-0005 the fuel could be put back in its original location. The I.O.A for the AB.CONT-0005 would have the crew suspend all handling of irradiated fuel/components. (see attached)

D: **Incorrect.** OP-HC-108-115-1001, - Procedure to be implemented within 30 minutes vice 60.

Technical Reference(s):	EOP-103/4	(Attach if not previously provided)
	Reactor Building & Rad Release Control HC.OP-AB.CONT-0005 Irradiated Fuel Damage	

Proposed References to be provided to applicants during examination: none

Learning Objective:	Given a set of plant conditions, analyze and determine if entry conditions into HC.OP-EO.ZZ-0103/4 exists.	(As available)

Question Source:	Bank # 112293	
	Modified Bank #	(Note changes or attach parent)
	New	

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(5)

Comments:

:TOR BLDG ROOM TEMP
:VE TABLE 1 COLUMN 1

REACTOR BLDG LOCAL RAD
MONITORING ALARM

REFUEL FLOOR HVAC EXHAUST RAD
LEVELS ABOVE 1E-3 UCI/ML

REACTOR BLDG HVAC EXHAUST RAD
MONITOR LEVELS ABOVE 5E-4 UCI/ML

TABLE 1

Area Description & Room Number	Column 1 Max Normal Op Temp	Column 2 Max Safe Op Temp
CRD Pump Room (4202)	115°F	140°F
HPCI (4111)	115°F	250°F
Core Spray Pump Rooms A(4118) & C(4116)	115°F	140°F
RHR Pump Rooms A(4113) & C(4114)	115°F	140°F
SACS A & C (4309)	115°F	140°F
RCIC Pump Room (4110)	115°F	250°F
Core Spray Pump Rooms B(4104) & D(4105)	115°F	140°F
RHR Pump Rooms B(4109) & D(4107)	115°F	140°F
SACS B & D (4307)	115°F	140°F
RWCU Pipe Chase (4402)	160°F	350°F

9

IF while executing the following steps:	THEN:
All entry conditions have cleared	**EXIT** this procedure
Reactor Bldg HVAC Exhaust rad level exceeds **1 x10-3 uci/ml** OR Refuel Floor HVAC Exhaust rad level exceeds **2x10-3 uci/ml**	**VERIFY** isolation of RBVS AND initiation of FRVS
SAG entry is required	**EXIT** this procedure AND **ENTER** SAG

RB-1

EXECUTE the following steps concurrently

MONITOR AND CONTROL Reactor Bldg Temps

MONITOR AND CONTROL Reactor Bldg Rad Levels

MONITOR AND CONTROL Reactor Bldg Floor Levels

GASEOUS RADIOACTIVE RELEASE RATE
ABOVE ALERT LEVEL EP-HC-325-103

IF while executing the following steps:	THEN:
SAG entry is required	**EXIT** this procedure AND **ENTER** SAG

RR-1

RESTORE Rad Release Rate below ALERT level

RR-2

PSEG Internal Use Only | **HC.OP-AB.CONT-0005(Q)**
IRRADIATED FUEL DAMAGE

SUBSEQUENT OPERATOR ACTIONS

CONDITION	ACTION
A. Irradiated Fuel Damage. Date/Time: ____________	A.1 **PERFORM** the following: ___ • **ENSURE** either RBVS OR FRVS is in service • **PERFORM** ONE of the following: ___ 1. **ENSURE** Secondary Containment Integrity is in effect. ___ 2. WITHIN 30 minutes, **IMPLEMENT** OP-HC-108-115-1001, Attachment 5 to seal secondary containment breaches. ___ • **DIRECT** Radiation Protection to take air samples AND control access to the Reactor Bldg and Refuel floor. ___ • **EVACUATE** all UNNECESSARY personnel from the affected area.
B. Damaged irradiated fuel is attached to the fuel handling grapple. Date/Time: ____________	___ B.1 IF Radiation levels permit, **PLACE** the damaged fuel in the Spent Fuel Pool.

PSEG Internal Use Only | **HC.OP-AB.CONT-0005(Q)**
IRRADIATED FUEL DAMAGE

IMMEDIATE OPERATOR ACTIONS

CONDITION	ACTION
Irradiated fuel damage Date/Time: ____________	___ **SUSPEND** the handling of Irradiated Fuel/Components.

Facility:	Hope Creek
Vendor:	GE
Exam Date:	2019
Exam Type:	SRO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #		295024 G2.2.40
	Importance Rating		4.7

K/A Statement: Ability to apply Technical Specifications for a system. High Drywell Pressure.

Question: SRO #80

Given:

- The plant is operating at 100% rated power.
- Pressure transmitter BB-PT-N050A has failed its surveillance calibration check.
- The failure is non-conservative.

Which one of the following describes the action(s) required by Technical Specification?
[Reference attached]

A. Commence a normal shutdown within one hour and be in at least Startup within 6 hours, Hot Shutdown with 6 hours and Cold Shutdown within the following 24 hours.

B. Place the inoperable channel(s) and/or that trip system in the tripped condition within 24 hours.

C. Place at least one trip system in the tripped condition within one hour and be in at least HOT SHUTDOWN within 12 hours.

D. Place the inoperable channel(s) and/or that trip system in the tripped condition within 12 hours.

Proposed Answer: **D**

Explanation (Optional):

A: **Incorrect**- These are actions IAW 3.0.3 when there is no applicable condition. This is not the case.

B: **Incorrect-** This is the action for instruments NOT common to RPS.

C: **Incorrect-** This is an action for OPERABLE channels less than required for BOTH trip systems.

D: **Correct-** Actions for number of OPERABLE channels less than required for ONE trip system. **N050A is common to NSSSS and RPS (The student determines this from M-42-1 sheet 2).** Actions are more limiting for instruments common to RPS. Tech Specs **3.3.1** and **3.3.2.b.1.b**.

Technical Reference(s): M-42-1 Sheet 2 (Attach if not previously provided)

Boiler Instrumentation

T.S. 3.3.1 and 3.3.2
RPS and NSSSS

Proposed References to be provided to applicants during examination: M-42-1 Sheet 2
T.S. 3.3.1, 3.3.2
Including Tables ¾ 3-3 through 3-4 and ¾ 3-11 and 3-16
(As available)

Learning Objective: Given a scenario of applicable operating conditions and access to Technical Specifications:
Select those sections that are applicable to Nuclear Boiler Instrumentation System
Evaluate Nuclear Boiler Instrumentation operability and determine required actions based upon system inoperability

Question Source: Bank #

Modified Bank # 84112 (**Note changes**: Modified the failed instrument to fit the K/A for Drywell pressure.
Original instrument was BB-PT-N078A for reactor pressure. Rest of question is the same.)

New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43 (2)

Comments
:

BB-PT-NØ5ØA-C71	C-7	LOCAL	PIS-N65ØA-C71	C6Ø9	HIGH DRYWELL PRESS. TO NS4 ISLN AND RPS TRIP LOGIC	D-21Ø9	W	14,27
			PS-N651-C71	C6Ø9	LOW DRYWELL PRESSURE TO ANNUNCIATOR			
			PS-N653-C71	C6Ø9	HIGH DRYWELL PRESSURE TO ANNUNCIATOR			
BB-PT-NØ5ØB-C71	C-3	LOCAL	PIS-N65ØB-C71	C611	HIGH DRYWELL PRESS. TO NS4 ISLN AND RPS TRIP LOGIC	D-211Ø	X	14,27
BB-PT-NØ5ØC-C71	B-7	LOCAL	PIS-N65ØC-C71	C6Ø9	HIGH DRYWELL PRESS. TO NS4 ISLN AND RPS TRIP LOGIC	D-2111	Y	14,27
BB-PT-NØ5ØD-C71	B-3	LOCAL	PIS-N65ØD-C71	C611	HIGH DRYWELL PRESS. TO NS4 ISLN AND RPS TRIP LOGIC	D-2112	Z	14,27
	G-3	C114	PIS-N654A-C11	C613	FIRST STAGE TURBINE PRESS. TO RSCS LOGIC		NON-1E	13
	G-3	C114	PIS-N654B-C11	C613	FIRST STAGE TURBINE PRESS. TO RSCS LOGIC		NON-1E	13

M-42-1 Sheet 2

3/4.3 INSTRUMENTATION

3/4 3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the reactor protection system instrumentation channels shown in Table 3.3.1-1 shall be OPERABLE.

APPLICABILITY: As shown in Table 3.3.1-1.

ACTION:

a. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for one trip system, place the inoperable channel(s) and/or that trip system*** in the tripped condition* within twelve hours.

b. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for both trip systems, place at least one trip system** in the tripped condition within one hour and take the ACTION required by Table 3.3.1-1.

INSTRUMENTATION

3/4.3.2 ISOLATION ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2 The isolation actuation instrumentation channels shown in Table 3.3.2-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.2-2.

APPLICABILITY: As shown in Table 3.3.2-1.

ACTION:

a. With an isolation actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.2-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

b. With the number of OPERABLE channels less than required by the minimum OPERABLE channels per trip system requirement for one trip system, either

   1) place the inoperable channel(s) in the tripped condition within

      a) 1 hour for trip functions without an OPERABLE channel,
      b) 12 hours for trip functions common to RPS instrumentation, and
      c) 24 hours for trip functions not common to RPS instrumentation,

   or

   2) take the ACTION required by Table 3.3.2-1.

c. With the number of OPERABLE channels less than required by the minimum OPERABLE channels per trip system requirement for both trip systems,

   1) place the inoperable channel(s) in one trip system in the tripped condition within one hour, and

   2) a) place the inoperable channel(s) in the remaining trip system in the tripped condition within

         1) 1 hour for trip functions without an OPERABLE channel,
         2) 12 hours for trip functions common to RPS instrumentation, and
         3) 24 hours for trip functions not common to RPS instrumentation,

      or

      b) take the ACTION required by Table 3.3.2-1.

TABLE 3.3.2-1
ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE ACTUATION GROUPS OPERATED BY SIGNAL	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)	APPLICABLE OPERATIONAL CONDITION	ACTION
1. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level				
1) Low Low, Level 2	2, 8, 9, 12, 13, 14, 15, 17, 18	2	1, 2, 3	20
2) Low low Low, Level 1	10, 11, 15, 16	2	1, 2, 3	20
b. Drywell Pressure - High	8, 9, 10, 11, 12, 13, 14, 15, 16, 17, 18	2(j)	1, 2, 3	20
c. Reactor Building Exhaust Radiation - High	8, 9, 12 13, 14, 15, 17, 18	3	1, 2, 3	28
d. Manual Initiation	8, 9, 10 11, 12, 13, 14, 15, 16, 17. 18	1	1, 2, 3	24

(j) Trip functions common to RPS instrumentation.

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	APPLICABLE OPERATIONAL CONDITIONS	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)	ACTION
6. This item intentionally blank			
7. Drywell Pressure - High	1, 2[h]	2	1
8. Scram Discharge Volume Water Level - High			
a. Float Switch	1, 2 5[i]	2 2	1 3
b. Level Transmitter/Trip Unit	1, 2 5[i]	2 2	1 3
9. Turbine Stop Valve - Closure	1[j]	4[k]	6
10. Turbine Control Valve Fast Closure, Valve Trip System Oil Pressure - Low	1[j]	2[k]	6
11. Reactor Mode Switch Shutdown Position	1, 2 3, 4 5	2 2 2	1 7 3
12. Manual Scram	1, 2 3, 4 5	2 2 2	1 8 9

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

ACTION

ACTION 1 - Be in at least HOT SHUTDOWN within 12 hours.

ACTION 2 - Verify all insertable control rods to be inserted in the core and lock the reactor mode switch in the Shutdown position within one hour.

ACTION 3 - Suspend all operations involving CORE ALTERATIONS* and insert all insertable control rods within one hour.

ACTION 4 - Be in at least STARTUP within 6 hours.

ACTION 5 - This ACTION is deleted.

ACTION 6 - Initiate a reduction in THERMAL POWER within 15 minutes and reduce turbine first stage pressure to less than the automatic bypass setpoint within 2 hours.

ACTION 7 - Verify all insertable control rods to be inserted within one hour.

ACTION 8 - Lock the reactor mode switch in the Shutdown position within one hour.

ACTION 9 - Suspend all operations involving CORE ALTERATIONS*, and insert all insertable control rods and lock the reactor mode switch in the SHUTDOWN position within one hour.

ACTION 10 - a) Initiate action to implement the Manual BSP Regions defined in the CORE OPERATING LIMITS REPORT immediately and b) implement the Automated BSP Scram Region using the modified APRM Simulated Thermal Power – Upscale scram setpoints defined in the CORE OPERATING LIMITS REPORT within 12 hours and c) initiate action in accordance with Specification 6.9.3.

ACTION 11 - If unable to complete Action 10 within required completion time: a) Initiate action to implement the Manual BSP Regions defined in the CORE OPERATING LIMITS REPORT immediately and b) restore required channel to OPERABLE with 120 days. LCO 3.0.4 is not applicable.

ACTION 12 - If unable to complete Action 11 within the required completion time: Reduce THERMAL POWER to less than 19% RATED THERMAL POWER within 4 hours.

---

*Except replacement of LPRM strings provided SRM instrumentation is OPERABLE per Specification 3.9.2.

Facility: Hope Creek

Vendor: GE

Exam Date: 2019

Exam Type: SRO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #		295025 EA2.01
	Importance Rating		4.3

K/A Statement: Ability to determine and/or interpret the following as they apply to HIGH REACTOR PRESSURE: Reactor pressure.

Question: SRO #81

Given:

- The plant was operating at 100% rated power.

When:

- A transient occurs resulting in a Reactor Scram.
- RPV pressure reached 1330 psig before turning downward.

Which one of the following states the required action(s) for RPV steam dome pressure reaching 1330 psig IAW Technical Specifications?

A. Perform an engineering evaluation on the out-of-limits condition within 24 hours.

B. Restore to within limits within 15 minutes or be in COLD SHUTDOWN within the next 6 hours.

C. Prepare and submit a Safety Limit Violation Report within 30 days.

D. Restore to within limits within 1 hour or be in COLD SHUTDOWN within the next 12 hours.

Proposed Answer: C

Explanation (Optional): Tech Specs TS 2.1.3
2.1.3 States with the RPV pressure above 1325 psig, be in at least HOT SHUTDOWN with the RPV pressure $\leq$ 1325 psig within 2 hours and comply with the requirements of 6.7.1 (see attached).

A: **Incorrect** – TS 2.1.3 states with the RPV pressure above 1325 psig, be in at least HOT SHUTDOWN with the RPV pressure $\leq$ 1325 psig within 2 hours and comply with the requirements of 6.7.1

B: **Incorrect** – Section 3.4.6.2 requires if exceeding 1020 psig reduce to < 1020 psig within 15 minutes or be in HOT SHUTDOWN within 12 hours.

C: **Correct** – Action for Safety Limit Violation is specified in TS Admin controls section 6.7.1.c

D: **Incorrect** – Section 3.4.6.2 requires if exceeding 1020 psig reduce to < 1020 psig within 15 minutes or be in HOT SHUTDOWN within 12 hours.

Technical Reference(s): T.S. 2.1.3 (Attach if not previously provided)

Safety Limits

T.S. 6.7.1
Administrative Controls
T.S. 3.4.6.2
Reactor Steam Dome

Proposed References to be provided to applicants during examination: (none)

Learning Objective: Given Technical Specifications, determine the administrative and operational actions that must be performed if a Safety Limit is violated. (As available)

Question Source: Bank #36142

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43 (2)

Comments:

REACTOR COOLANT SYSTEM PRESSURE

2.1.3 The reactor coolant system pressure, as measured in the reactor vessel steam dome, shall not exceed 1325 psig.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3 and 4.

ACTION:

With the reactor coolant system pressure, as measured in the reactor vessel steam dome, above 1325 psig, be in at least HOT SHUTDOWN with reactor coolant system pressure less than or equal to 1325 psig within 2 hours and comply with the requirements of Specification 6.7.1.

6.7. SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within 1 hour. The senior corporate nuclear officer and the senior management position with responsibility for independent nuclear safety assessment activities and quality program oversight shall be notified within 24 hours.

b. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the SORC. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon unit components, systems, or structures, and (3) corrective action taken to prevent recurrence.

c. The Safety Limit Violation Report shall be submitted to the Commission, the senior management position with responsibility for independent nuclear safety assessment activities and quality program oversight and the senior corporate nuclear officer within 30 days of the violation.

d. Critical operation of the unit shall not be resumed until authorized by the Commission.

REACTOR COOLANT SYSTEM

REACTOR STEAM DOME

LIMITING CONDITION FOR OPERATION

3.4.6.2 The pressure in the reactor steam dome shall be less than 1020 psig.

APPLICABILITY: OPERATIONAL CONDITION 1* and 2*.

ACTION:

With the reactor steam dome pressure exceeding 1020 psig, reduce the pressure to less than 1020 psig within 15 minutes or be in at least HOT SHUTDOWN within 12 hours.

Facility: Hope Creek

Vendor: GE

Exam Date: 2019

Exam Type: SRO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		1
Group #		1
K/A #		295031/ 2.1.20
Importance Rating		4.6

K/A Statement: Ability to interpret and execute procedure steps. - Reactor Low Water Level.

Question: SRO #82

Given:

- The reactor has been in COLD SHUTDOWN for two (2) days following power operation.
- Reactor vessel water level is +30 inches.
- Neither Reactor Recirculation pump is available.
- HC.OP-GP.SM-0001, Defeating NSSSS isolations for Shutdown Cooling, has NOT been completed.
- Alternate decay heat removal using RHR ‘D’ and ‘B’ cross-tie has been established IAW HC.OP-AB.RPV-0009, Shutdown Cooling Attachment 3.

Then:

- RPV level lowers to +10”.

WHICH ONE of the following describes the status of the ‘D’ RHR Pump and what are the actions that need to be directed?

The ‘D’ RHR pump is __ (1) __ and as the CRS direct __ (2) __.

A. (1) tripped
(2) isolating the HV-F015B, RHR Loop ‘B’ Return to Recirc Loop ‘B Isolation

B. (1) running
(2) immediately securing the ‘D’ RHR pump

C. (1) tripped
(2) isolating the HV-F008 and HV-F009 Shutdown Cooling Isolation valves

D. (1) running
(2) throttling the HV-F015B, RHR Loop ‘B’ Return to Recirc Loop ‘B Isolation, to establish flow

Proposed Answer: **B**

Explanation (Optional):

A: **INCORRECT** – 'D' RHR pump trips are bypassed when alternate decay heat removal using D to B cross-tie is established IAW HC.OP-AB.RPV-0009; therefore pump 'D' would remain running. The F015B would isolate due to the fact the GP.SM-0001 has not been completed, which would bypass the 12.5" NSSSS isolation.

B: **CORRECT** 'D' RHR pump trips are bypassed when alternate decay heat removal using D to B cross-tie is established IAW HC.OP-AB.RPV-0009; therefore pump 'D' would remain running, and due to the level below auto isolation setpoint 12.5", F008, F009, and F015 would all isolate. Due to no suction for the 'D' RHR pump the CRS would direct securing the pump immediately IAW AB.RPV-0009 attachment 3 (see attached).

C: **INCORRECT -** 'D' RHR pump trips are bypassed when alternate decay heat removal using D to B cross-tie is established IAW HC.OP-AB.RPV-0009; therefore pump 'D' would remain running. The F008 and F009 would isolate due to the fact the GP.SM-0001 has not been completed, which would bypass the 12.5" NSSSS isolation.

D: **INCORRECT -** 'D' RHR pump trips are bypassed when alternate decay heat removal using D to B cross-tie is established IAW HC.OP-AB.RPV-0009; therefore pump 'D' would remain running. The F015B would isolate due to the fact the GP.SM-0001 has not been completed, which would bypass the 12.5" NSSSS isolation.

Technical Reference(s): HC.OP-AB.RPV-0009 (Attach if not previously provided)

Shutdown Cooling

Attachment 3

Proposed References to be provided to applicants during examination: none

Learning Objective: Explain the reasons for how plant/system parameters respond when implementing Shutdown Cooling. (As available)

Question Source: Bank #

Modified Bank # 147151 (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43 (5)

Comments: Original Question (modified the question to an action needed for the transient. The changes provide a direction by the CRS IAW the abnormal)

Given:

The reactor has been in COLD SHUTDOWN for two (2) days following power operation

Reactor vessel water level is +30 inches

Neither Reactor Recirculation pump is available

GP.SM-0001 has NOT been completed

Due to several component failures, Shutdown Cooling is lost, and alternate decay heat removal using D to B cross-tie has been established.

Then, RPV level lowers to +10”.

WHICH ONE of the following describes the status of the "D" RHR Pump and HV-F015B, RHR LOOP B RET TO RECIRC one minute later?

The "D" RHR Pump is __(1)__ and the HV-F015B is __(2)__.

A. (1) Running
(2) CLOSED

B. (1) Running
(2) OPEN

C. (1) NOT Running
(2) CLOSED

D. (1) NOT Running
(2) OPEN

Answer: A

ATTACHMENT 3
ALTERNATE DECAY HEAT REMOVAL USING D TO B CROSS-TIE

NOTE

Interlock override will allow D RHR Pump to operate when pump is aligned to alternate suction from RPV, when and if required.

In Cross-Tie mode, D RHR Pump will not be automatically protected against loss of suction from RPV. ___

1.23 **PERFORM** the following at Panel 10C640:

1.23.1. **INSERT** key obtained in Step 1.1 in 1-BC-HS-11682, Keylock Switch. ___

1.23.2. **OVERRIDE** HV-F004D, Valve/Pump D Interlock, using 1-BC-HS-11682 Keylock Switch. ___

1.23.3. **LOG** in Control Room Narrative Log position of Keylock Switch. ___

1.24 IF during cross-tie operation mode, HV-F008 or HV-F009, close (e.g., on RPV Low Level 3 signal), THEN immediately **STOP** D RHR Pump AND **TAKE** corrective action. ___

1.25 IF HV-F015B, RHR LOOP B RET TO RECIRC LOOP B ISLN MOV, does not open immediately to establish flow, THEN **STOP** RHR Pump. ___

PSEG Internal Use Only

HC.OP-AB.RPV-0009(Q)
SHUTDOWN COOLING

IMMEDIATE OPERATOR ACTIONS

NONE

AUTOMATIC ACTIONS

IF	THEN
Reactor Pressure > 82 psig	The following valves cannot be opened from the Control Room OR their Remote Shutdown controls: • HV-F008 • HV-F009 • HV-F015A • HV-F015B
Reactor Pressure > 82 psig OR Reactor Level < 12.5" OR Loss of EITHER RPS Bus AND RSP Takeover Switches in NORMAL	The following valves will isolate: • HV-F008* • HV-F009* • HV-F015A* • HV-F015B*
RHR Pump in Shutdown Cooling AND Closure of ANY of the following: • HV-F008 • HV-F009 • Associated HV-F006	RHR pump trips.

*If GP.SM-0001 has been performed, these isolations are bypassed.

Facility: Hope Creek

Vendor: GE

Exam Date: 2019

Exam Type: SRO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #		295009 AA2.01
	Importance Rating		4.2

K/A Statement: Ability to determine and/or interpret the following as they apply to LOW REACTOR WATER LEVEL: Reactor water level.

Question: SRO #83

Given:

- Reactor power was at 100% rated power.

When:

- An ATWS and small break LOCA occurred.

Current plant conditions:

- Reactor power is at 7% and slowly lowering.
- SRVs are cycling on LO-LO Set.
- RPV level is at -45 inches and is being intentionally lowered IAW EOP-101A.
- SLC, CRD, and RCIC are injecting.
- Drywell pressure is at 4.5 psig and slowly rising.
- Suppression Pool temperature is at 112°F.
- 'A' and 'B' RHR loops are in Suppression Pool Cooling mode.

Which one of the following would permit RPV water level to be stabilized between -185 inches and the RPV level when that condition is achieved?

A. All SRVs remain closed.

B. Suppression Pool temperature lowers to 108°F.

C. All MSIVs are closed.

D. Reactor power reaches 3%.

Proposed Answer: **D**

Explanation (Optional): This item test SRO ability to resolve the question by correctly applying the stem conditions to the flowchart of EOP-101A level control.

A: **INCORRECT -** Would be correct if Drywell Pressure was below 1.68 psig.(see attached LP-13).

B: **INCORRECT –** With Stem conditions and at 7 percent power, Step LP-10 and 11 are answered YES. This requires lowering level until power is less than 4 percent. You cannot back up and change the answer to LP-10 using the retainment step LP-6. Plausible misconception.

C: **INCORRECT -**Based on stem conditions, Steps LP-10 and LP-11 must be answered YES. Level must be lowered until the conditions of step LP-13 are met.

D: **CORRECT -** The conditions provided in the stem require terminate and prevent injection and RPV level reduction to lower power. The conditions require EOP-101A Step LP-13 implemented for level reduction. If reactor power reaches 3 percent from 7percent, EOP-101A Step LP-13 allows level reduction to be stopped. Step LP-14 then allows RPV level to be maintained between that level and -185 inches. (see attached).

Technical Reference(s): HC.OP-EO.ZZ-0101A ATWS-RPV Control (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: Given any step of the procedure, explain the basis for performance of that step and/or evaluate the expected system response to control manipulations prescribed by that step. (As available)

Question Source: Bank #36215

Modified Bank # (note changes or attach parent)

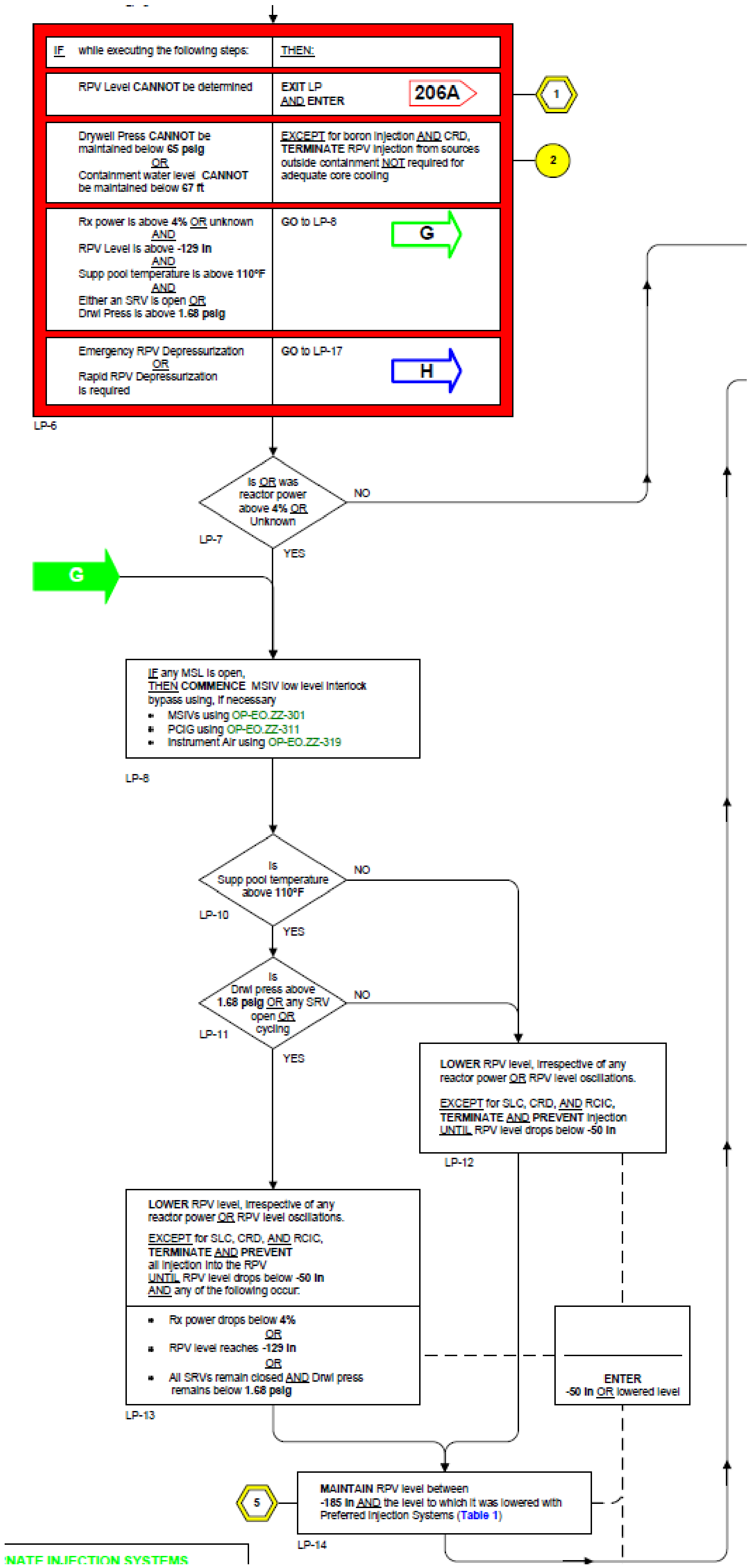
New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43 (5)

Comments:


IF while executing the following steps:
THEN:
RPV Level CANNOT be determined
EXIT LP AND ENTER
206A
1
Drywell Press CANNOT be maintained below 65 psig OR Containment water level CANNOT be maintained below 67 ft
EXCEPT for boron injection AND CRD, TERMINATE RPV injection from sources outside containment NOT required for adequate core cooling
2
Rx power is above 4% OR unknown AND RPV Level is above -129 in AND Supp pool temperature is above 110°F AND Either an SRV is open OR Drwl Press is above 1.68 psig
GO to LP-8
G
Emergency RPV Depressurization OR Rapid RPV Depressurization is required
GO to LP-17
H
LP-6
Is OR was reactor power above 4% OR Unknown
NO
YES
LP-7
G
IF any MSL is open, THEN COMMENCE MSIV low level interlock bypass using, if necessary
MSIVs using OP-EO.ZZ-301
PCIG using OP-EO.ZZ-311
Instrument Air using OP-EO.ZZ-319
LP-8
Is Supp pool temperature above 110°F
NO
YES
LP-10
Is Drwl press above 1.68 psig OR any SRV open OR cycling
NO
YES
LP-11
LOWER RPV level, irrespective of any reactor power OR RPV level oscillations.
EXCEPT for SLC, CRD, AND RCIC, TERMINATE AND PREVENT injection UNTIL RPV level drops below -50 in
LP-12
LOWER RPV level, irrespective of any reactor power OR RPV level oscillations.
EXCEPT for SLC, CRD, AND RCIC, TERMINATE AND PREVENT all injection into the RPV UNTIL RPV level drops below -50 in AND any of the following occur:
Rx power drops below 4% OR
RPV level reaches -129 in OR
All SRVs remain closed AND Drwl press remains below 1.68 psig
LP-13
ENTER -50 in OR lowered level
MAINTAIN RPV level between -185 in AND the level to which it was lowered with Preferred Injection Systems (Table 1)
5
LP-14
NATE INJECTION SYSTEMS

Facility: Hope Creek
Vendor: GE
Exam Date: 2019
Exam Type: SRO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		1
Group #		2
K/A #		295009/2.1.20
Importance Rating		4.6

K/A Statement: Ability to interpret and execute procedure steps. - Low Reactor Water Level.

Question: SRO #84

Given:

- The reactor is shutdown (all rods full-in) due to a scram on low RPV level.
- All sources of injection to the RPV have been lost.
- Water level is slowly lowering due to decay heat boil off.
- Reactor pressure is stable and being controlled by SRVs.
- All efforts to restore a source of injection have been unsuccessful.

As RPV water level goes below level 1 and continues to lower, describe the next operator action(s).

A. When RPV water level reaches the Minimum Zero-Injection Water Level (MZIRWL), open 5 ADS valves.

B. When RPV water level reaches Minimum Steam Cooling Reactor Water Level (MSCRWL), open 5 ADS valves.

C. When RPV water level reaches Top of Active Fuel (TAF), open 5 ADS valves.

D. Immediately open 5 ADS valves, and enter Severe Accident Guidelines (SAG).

Proposed Answer: **A**

Explanation (Optional): Steam cooling is effected by allowing RPV water level to decrease through boil-off until it drops to the Minimum Zero-Injection RPV Water Level (MZIRWL). During this period, the fuel temperatures in the uncovered portion of the core increase and heat is transferred from the fuel rods to the steam. The MZIRWL is the lowest RPV water level at which the covered portion of the reactor core will generate sufficient steam to preclude any clad temperature in the uncovered portion of the core from exceeding 1800°F. Emergency RPV depressurization is then performed in accordance with EOP-202. Unless the RPV is already depressurized, it is expected that the resulting swell will be sufficient to quench the uncovered portion of the fuel and reduce PCT almost to the value that would exist if the core were submerged. As the swell subsides and steam flow through the open SRVs decreases, however, PCT turns and again rises. Opening the SRVs before RPV water level reaches the MZIRWL would reduce the time over which the core remains adequately cooled with no injection. Waiting to emergency depressurize the RPV much after RPV water level reaches the MZIRWL could result in significant core damage due to excessive fuel temperatures. See attached bases for Minimum Zero-Injection Water Level (MZIRWL).

A: **Correct** – See above explanation

B: **Incorrect** – Emergency RPV Depressurization before RPV water level reaches -185 in. (MSCRWL) is based on having injection sources available. (see attached 101)

C: **Incorrect** - Based upon current conditions, you do not meet the requirements to anticipate ED per step RC/P-3.

D: **Incorrect** – Based upon current conditions, you do not yet meet the entry conditions for SAG.

Technical Reference(s): HC.OP-EO.ZZ-0101BASES (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none.

Learning Objective: Given any step of the procedure, describe the reason for performance of that step and/or expected system response to control manipulation prescribed by that step. (As available)

Question Source: Bank #152791

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Comprehension or Analysis

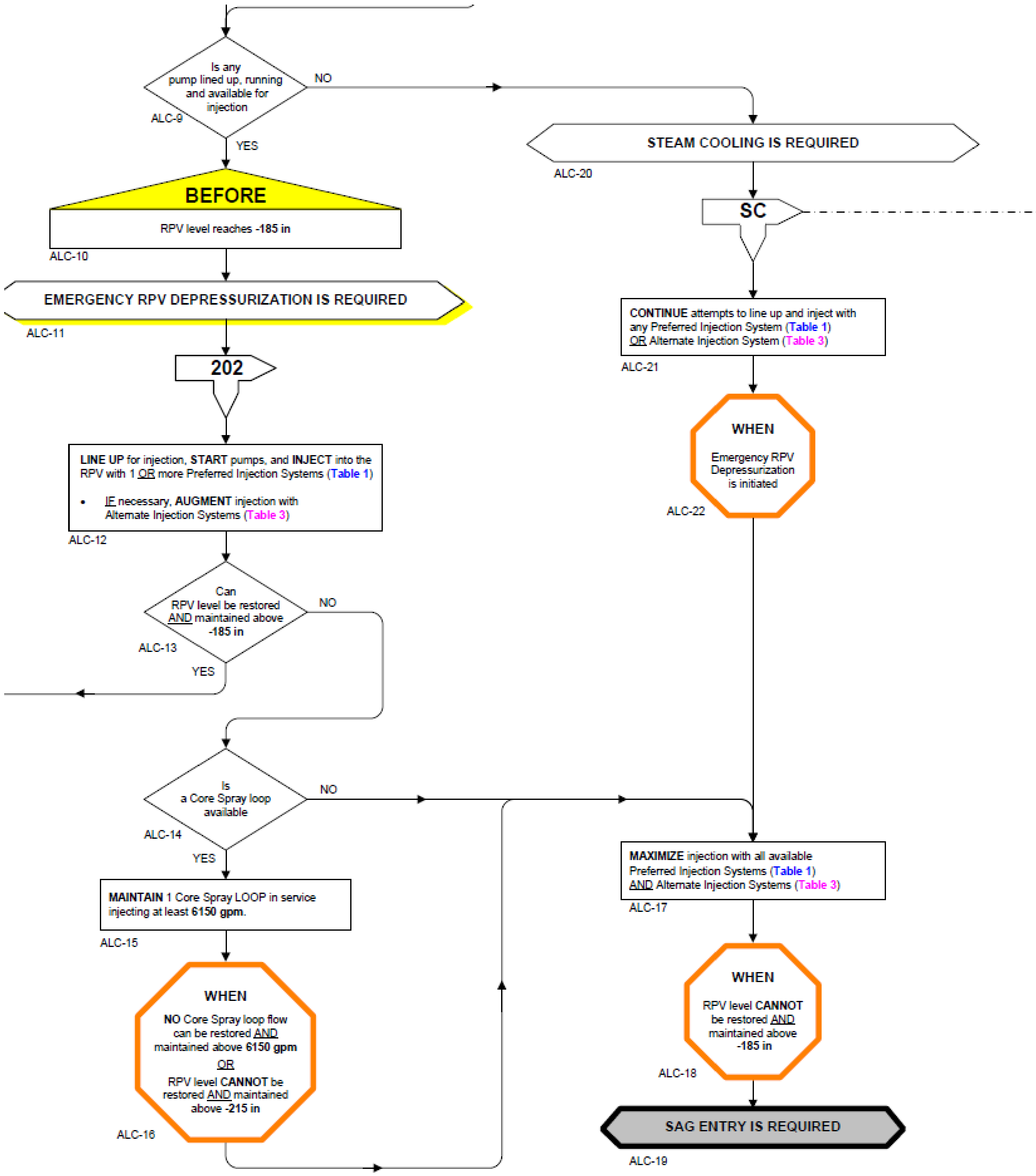
10 CFR Part 55 Content: 55.43 (5)

Comments:

9.3 SC-3 When RPV level drops to -198 in.
SC-4 Emergency RPV Depressurization is required

Steam cooling is effected by allowing RPV water level to decrease through boil-off until it drops to the Minimum Zero-Injection RPV Water Level (MZIRWL). During this period, the fuel temperatures in the uncovered portion of the core increase and heat is transferred from the fuel rods to the steam. The MZIRWL is the lowest RPV water level at which the covered portion of the reactor core will generate sufficient steam to preclude any clad temperature in the uncovered portion of the core from exceeding 1800°F.

When RPV water level drops below the MZIRWL, steam cooling may no longer be sufficient to preclude the peak clad temperature from exceeding 1800°F. Emergency RPV depressurization is then performed in accordance with EOP-202. Unless the RPV is already depressurized, it is expected that the resulting swell will be sufficient to quench the uncovered portion of the fuel and reduce PCT almost to the value that would exist if the core were submerged. As the swell subsides and steam flow through the open SRVs decreases, however, PCT turns and again rises. Opening the SRVs before RPV water level reaches the MZIRWL would reduce the time over which the core remains adequately cooled with no injection. Waiting to emergency depressurize the RPV much after RPV water level reaches the MZIRWL could result in significant core damage due to excessive fuel temperatures.


Is any pump lined up, running and available for injection
ALC-9
NO
YES
BEFORE
RPV level reaches -185 in
ALC-10
EMERGENCY RPV DEPRESSURIZATION IS REQUIRED
ALC-11
202
LINE UP for injection, START pumps, and INJECT into the RPV with 1 OR more Preferred Injection Systems (Table 1)
IF necessary, AUGMENT injection with Alternate Injection Systems (Table 3)
ALC-12
Can RPV level be restored AND maintained above -185 in
ALC-13
NO
YES
Is a Core Spray loop available
ALC-14
NO
YES
MAINTAIN 1 Core Spray LOOP in service injecting at least 6150 gpm.
ALC-15
WHEN
NO Core Spray loop flow can be restored AND maintained above 6150 gpm
OR
RPV level CANNOT be restored AND maintained above -215 in
ALC-16
STEAM COOLING IS REQUIRED
ALC-20
SC
CONTINUE attempts to line up and inject with any Preferred Injection System (Table 1) OR Alternate Injection System (Table 3)
ALC-21
WHEN
Emergency RPV Depressurization is initiated
ALC-22
MAXIMIZE injection with all available Preferred Injection Systems (Table 1) AND Alternate Injection Systems (Table 3)
ALC-17
WHEN
RPV level CANNOT be restored AND maintained above -185 in
ALC-18
SAG ENTRY IS REQUIRED
ALC-19

SC

IF while executing the following steps:	THEN:
RPV level is rising	**EXIT** SC AND **GO** to **B**
RPV level **CANNOT** be determined	**EXIT** SC AND **ENTER** **206**
Emergency RPV Depressurization is required	**EXIT** SC AND **ENTER** **202**
Any pump is lined up for injection AND running AND RPV level **CANNOT** be restored AND maintained above **-185 in**	Emergency RPV Depressurization is required
SRVs are being used to stabilize pressure AND the continuous pneumatic supply is unavailable	Emergency RPV Depressurization is required

SC-1

STABILIZE RPV pressure using 1 OR more of the following systems:

- SRVs (AB.ZZ-0001) only if torus level is above **0 in**
- HPCI (AB.ZZ-0001)
- RCIC (AB.ZZ-0001)

2, 3, 4, 7, 9, 10

SC-2

WHEN

RPV level drops to **-198 in**

SC-3

EMERGENCY RPV DEPRESSURIZATION IS REQUIRED

SC-4

202

2019 NRC Written Examination

Facility: Hope Creek

Vendor: GE

Exam Date: 2019

Exam Type: SRO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	295034	EA2.01
	Importance Rating		4.2

K/A Statement: Ability to determine and/or interpret the following as they apply to SECONDARY CONTAINMENT VENTILATION HIGH RADIATION: Ventilation radiation levels

Question: SRO #85

Given:

- The plant is operating at 100% rated power.

Then:

- The crew reports a MWe drop with no change in recirculation system or rod position.
- SPDS shows rising HPCI room temperatures and a drop in HPCI steam inlet pressure on the 10C650 panel.

The following OHA's are in alarm:

- RADIATION MONITORING ALARM/TRBL C6-C1
- RBVS & WING AREA HVAC PNL 10C382 E6-C5

Secondary Containment d/p is -0.25 inches WG and rising toward 0 inches WG.

The Radiation monitors in alarm are from the ____(1)____, as the CRS direct __(2)___.

A. (1) North Plant Vent and Reactor Building Exhaust,
(2) isolating HPCI, placing FRVS in service, and enter HC.OP-AB.CONT-0003 "Reactor Building" and HC.OP-AB.CONT-0004 – "Radioactive Gaseous Release".

B. (1) South Plant Vent and Reactor Building Exhaust,
(2) isolating HPCI and enter HC.OP-AB.CONT-0004 – "Radioactive Gaseous Release".

C. (1) South Plant Vent and Reactor Building Exhaust,
(2) isolating HPCI, placing FRVS in service, and enter HC.OP-AB.CONT-0003 "Reactor Building" and HC.OP-AB.CONT-0004 – "Radioactive Gaseous Release".

D. (1) North Plant Vent and Reactor Building Exhaust,
(2) placing FRVS in service and enter HC.OP-AB.CONT-0003 – "Reactor Building".

Proposed Answer: C

Explanation (Optional):

A: **Incorrect-** RBVS vents through the South Plant Vent not North Plant Vent. With South Plant Vent Activity Rising HC.OP-AB.CONT-0004 Rad Gas Release will be entered to isolate HPCI. (see attached) With RBVS unable to maintain at least -.30 inches WG then HC.OP-AB.CONT-0003 “Reactor Building” will be entered to place FRVS in service.

B: **Incorrect-** RB Exhaust radiation monitors would be in alarm with the South Plant Vent radiation monitors. With RBVS unable to maintain at least -.30 inches WG then HC.OP-AB.CONT-0003 “Reactor Building” will be entered to place FRVS in service. With South Plant Vent Activity Rising HC.OP-AB.CONT-0004 Rad Gas Release will be entered to isolate HPCI.

C: **Correct-** RBVS vents through the South Plant Vent. The RBVS exhaust radiation monitoring would be in alarm with the South Plant Vent radiation monitoring. With RBVS unable to maintain at least -.30 inches WG then HC.OP-AB.CONT-0003 “Reactor Building” will be entered to place FRVS in service. With South Plant Vent Activity Rising HC.OP-AB.CONT-0004 Rad Gas Release will be entered to isolate HPCI. (see attached)

D: **Incorrect-** RBVS vents through the South Plant Vent. With RBVS unable to maintain at least -.30 inches WG then HC.OP-AB.CONT-0003 “Reactor Building” will be entered to place FRVS in service.

Technical Reference(s): HC.OP-AB.CONT-0003 (Attach if not previously provided)

Reactor Building

Proposed References to be provided to applicants during examination: none

Learning Objective: Explain the reasons for how plant/system parameters respond when implementing Reactor Building Integrity. (As available)

Question Source: Bank #116252

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content:

55.43 (5)

Comments:

CATEGORY II

REACTOR BUILDING

ALARMS

- RB PRESSURE HI/LO **E6 - B5**
- RBVS & WING AREA HVAC PNL 10C382 **E6 - C5**
- FRVS VENT FAN MALFUNCTION **E6 - D5**
- FRVS RECIRC FAN TROUBLE **E6 - E5**

INDICATIONS

- Reactor Building vacuum is < 0.30 inches water gauge (When either RBVS or FRVS is in service).

SUBSEQUENT OPERATOR ACTIONS

CONDITION	ACTION
A. RBVS CANNOT maintain a Reactor Building vacuum of at least 0.30 inches WG. **[T/S 3.6.5.1]** Date/Time: ____________	___ ****NOTE 1** ** A.1 **PERFORM** the following: ___ • **VERIFY** HD-9414A/B and HD-9370A/B are open. ___ • **ENSURE** proper operation of RBVS. (GR) ___ • **TERMINATE** Fuel handling operations ___ • **TERMINATE** Core Alterations ___ • **TERMINATE** Operations with the potential for draining the Reactor Vessel. ___ A.2 IF RBVS cannot maintain Rx Bldg vacuum > 0.30 inches WG, THEN **PLACE** FRVS in service. ___ A.3 **ENSURE** the transient inoperability of secondary containment duration has been entered in the narrative log OR Active LCO paperwork has been completed as appropriate.

Effective Date 5/29/12

CATEGORY II

RADIOACTIVE GASEOUS RELEASE

ALARMS

- RB EXH RADIATION ALARM/TRBL **E6 - A5**
- REFUEL FLR EXH RAD ALARM/TRBL **E6 - A3**
- RADIATION MONITORING ALARM/TRBL **C6 - C1**

SUBSEQUENT OPERATOR ACTIONS (continued)

CONDITION	ACTION
C. South Plant Vent Activity Rising. Date/Time: ______________ *** Continued on Page 15 ***	___ ** **NOTE 1**** ___ C.1 IF the South Plant Vent RMS Effluent (9RX580) is at the HIGH alarm setpoint, THEN **STOP** the Mechanical Vacuum Pumps. ___ C.2 **TERMINATE** Drywell Venting OR Purging. (GS) C.3 **MONITOR** the following for indications of a steam leak: ___ • Main Gen MWe. ___ • Fire alarms. ___ • Steam Tunnel temperature. ___ • HPCI/RCIC room temperatures. ___ • HPCI/RCIC Steam Supply Pressure ___ • RWCU room/pipechase temperatures.

SUBSEQUENT OPERATOR ACTIONS (continued)

CONDITION	ACTION
*** Continued from Page 13 *** C. South Plant Vent Activity Rising. *** Continued on Page 17 ***	*** Continued from Page 13 *** C.7 IF HPCI isolation is required, THEN **CLOSE** the following: ___ • FD-HV-F002 ___ • FD-HV-F003 ___ • FD-HV-F100

Facility:	Hope Creek
Vendor:	GE
Exam Date:	2019
Exam Type:	SRO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #		203000 A2.04
	Importance Rating		3.6

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

Question: SRO #86

Given:

- RHR 'A' pump is operating at 10,450 gpm in "full flow test" mode.

During the test, a LOCA occurs and a valid LPCI initiation signal is generated concurrent with a loss of offsite power.

- The PO reports HPCI and RCIC both fail to start.
- The RO reports the "A" and “B” Emergency Diesel Generators are not able to energize their respective buses due to generator lockouts.
- RPV water level is -129” and lowering.
- Reactor Pressure is at 400 psig and lowering.
- .

All applicable Emergency Operating Procedure actions are taken.

LPCI injection valves (F017’s) ___(1)___will be open. Once Emergency Depressurization occurs, OP-HC-103-102-1013, Transient Mitigation Strategies requires RPV water level band to be restored and maintained between____(2)_____.

A. (1) ‘C’ and ‘D’
(2) -38” to +54”

B. (1) ‘A’, ‘B’, ‘C’, and ‘D’
(2) -38” to +54”

C. (1) ‘C’ and ‘D’
(2) -185” to +54”

D. (1) ‘A’, ‘C’, and ‘D’
(2) +12.5” to +54”

Proposed Answer: **A**

Explanation (Optional):

A: **Correct-** 'A' and 'B' LPCI will not be available due to no 4.16kv power ('A' and 'B' EDGs). Core Spray is also not available for injection due to the injections valves (F005A/B) are powered from the 'A' and 'B' 1E channels ('A' and 'B' EDGs). 'C' and 'D' RHR LPCI injection valves (F017) will open when reactor pressure is less than 450 psig. For non-ATWS emergency depressurizations, the Control Room Supervisor should establish and maintain an RPV level band between -38" to +54" on Wide Range indication. (See attached SOP and mitigation strategies).

B: **Incorrect –** 'A' and 'B' LPCI will not be available due to no 4.16kv power ('A' and 'B' EDGs).

C: **Incorrect -** For non-ATWS emergency depressurizations, the Control Room Supervisor should establish and maintain an RPV level band between -38" to +54" on Wide Range indication.

D: **Incorrect –** 'A' and 'B' LPCI will not be available due to no 4.16kv power ('A' and 'B' EDGs). For non-ATWS emergency depressurizations, the Control Room Supervisor should establish and maintain an RPV level band between -38" to +54" on Wide Range indication.

Technical Reference(s): HC.OP-SO.BC-0001 (Attach if not previously provided)

RHR (Figures 2,3,&4)

OP-HC-103-102-1013
Transient Mitigation Strategies

Proposed References to be provided to applicants during examination: none

Learning Objective: Given a system which physically connects to or is required to support the operation of the RHR System or components therein, explain the function of the supporting system (As available)

Question Source: Bank #32630

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43 (5)

Comments:

TABLE 1
VALVE TABLE

Valve	Power	Auto Actuations	Interlocks
F017A B C D (LPCI Injection)	10B212 10B222 10B232 10B242	Auto open on LPCI init. if following condition exist: (1) LPCI init. present in respective RHR loop logic (2) Power is avail. on respective pump bus (3) Reactor press. less than 450 psig (may be overridden by "AUTO OPEN OVRD")	Rx press. must be less than 450 psig to open valve either MAN or AUTO. Placing Ch. B RSP to EMERG will close F017B & inhibit all associated automatic & OVLD protection Features. The LPCI injection valve must be 100% closed (in the respective loop with a LPCI initiation signal present) to OPEN: F027A(B), F024A(B), F016, F021

HC.OP-SO.BC-0001(Q)

FIGURE 2
LPCI INITIATION LOGIC DIV 1 / RHR A

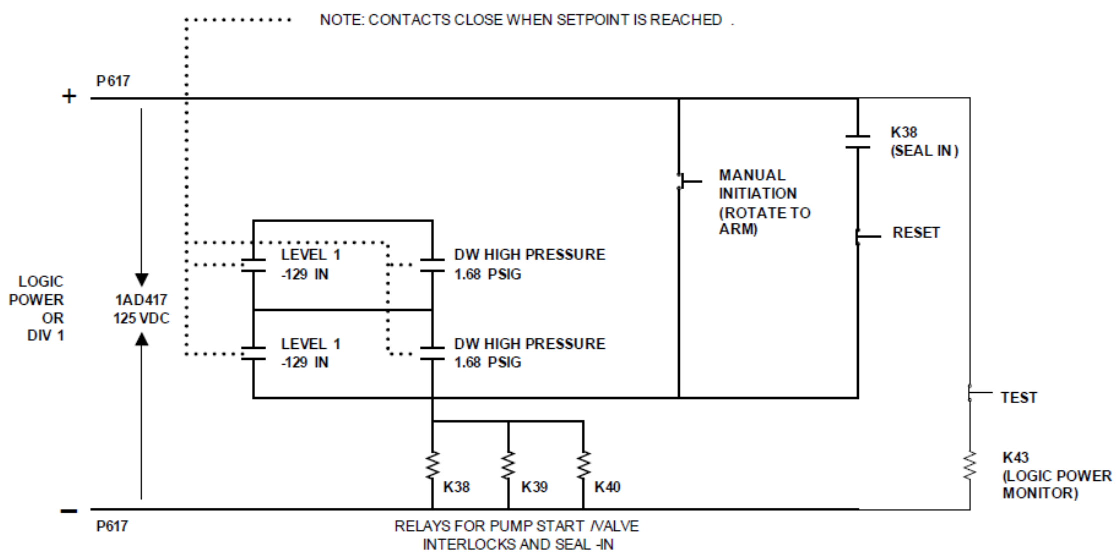


FIGURE 3
LPCI A/B, PUMP START LOGIC

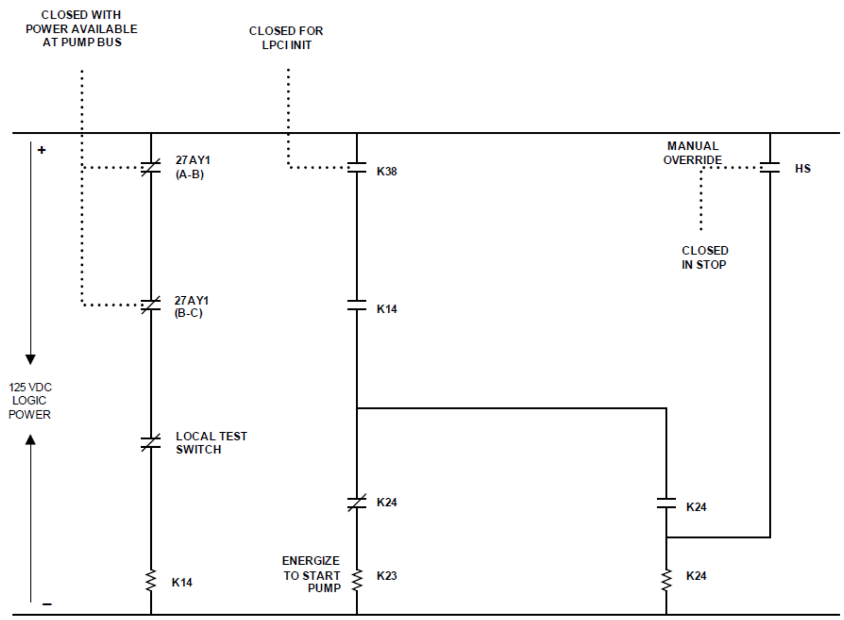
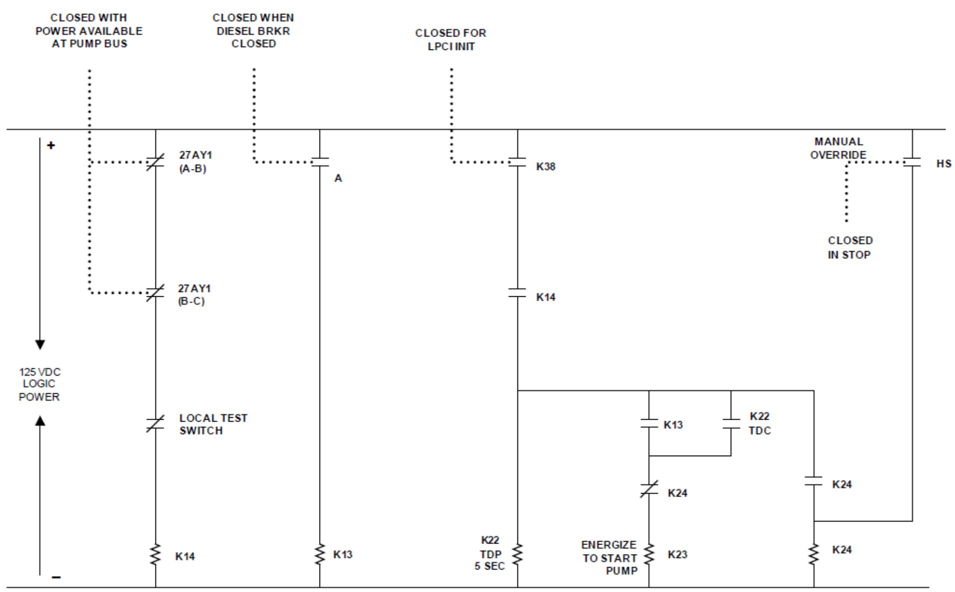


FIGURE 4
LPCI C/D PUMP START LOGIC



5.13.4. For non-ATWS emergency depressurizations, the Control Room Supervisor should establish and maintain an RPV level band between -38” to +54” on Wide Range indication. This transient level band will prevent adverse hydraulic effects caused by high outside shroud reactor level while maintaining adequate core submergence based on Fuel Zone indication. Following the emergency depressurization, level will be restored and maintained to the preferred EOP reactor level band by using a suggested Wide Range compensated level band of +12.5” to +54” if possible using Condensate and Startup Level Control or the Control Room Supervisor can maintain an RPV level band of -38” to +54” if still batch feeding with low pressure ECCS to maintain RPV level.

Facility: Hope Creek

Vendor: GE

Exam Date: 2019

Exam Type: SRO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		2
Group #		1
K/A #		215004 G2.2.40
Importance Rating		4.7

K/A Statement: Ability to apply Technical Specifications for a system. Source-Range Monitor

Question: SRO #87

Given:

- A complete core offload was completed at the beginning of the refueling outage.
- Fuel reload is ready to commence.
- All SRM's are fully inserted with the following count rates:
    - 'A' – 5 cps
    - 'B' – 2 cps
    - 'C' – 6 cps
    - 'D' – 1 cps

Based on these conditions, which of the following actions is required IAW plant procedure?

A. Spiral fuel reload may commence in 'A' and 'C' quadrants only, until either 'B' or 'D' quadrant SRM is reading > 3 cps at which time complete reload may be commenced.

B. A movable SRM detector must be hooked up to the normal SRM channel instrumentation and be placed in either 'B' or 'D' quadrant, indicating > 3 CPS prior to Spiral fuel reload commencement.

C. Spiral fuel reload may commence up to the first 16 bundles, at which time all four SRM's must read > 3 cps to perform a complete reload.

D. Spiral Reload may commence with no restrictions as long as any two SRM's are reading > 3 cps.

Proposed Answer: **C**

Explanation (Optional):

A: **Incorrect.** up to four fuel assemblies may be loaded in the four bundle locations immediately surrounding each of the four SRMs prior to obtaining 3 cps.

B: **Incorrect**. up to four fuel assemblies may be loaded in the four bundle locations immediately surrounding each of the four SRMs prior to obtaining 3 cps.

C: **Correct**. IAW T.S. 3.9.2e, During a SPIRAL RELOAD, up to four fuel assemblies may be loaded in the four bundle locations immediately surrounding each of the four SRMs prior to obtaining 3 cps. Until these assemblies have been loaded, the 3 cps count rate is not required. IAW IO-0009 step 5.2.14, WHEN performing a SPIRAL RELOAD, THEN IMMEDIATELY following the loading of the first 16 bundles, **VERIFY** the SRM channel count rate is 3 cps.

D: **Incorrect**. Up to four fuel assemblies may be loaded in the four bundle locations immediately surrounding each of the four SRMs prior to obtaining 3 cps.

Technical Reference(s): T.S. 3.9.2 Instrumentation (Attach if not previously provided)

Refueling Operations

HC.OP-IO.ZZ-0009
Refueling Operations

Proposed References to be provided to applicants during examination: none

Learning Objective: Analyze plant conditions and parameters to determine if plant operation is in accordance with the REFUELING OPERATIONS Integrated Operating Procedure, supporting System Operating Procedures and Technical Specifications (As available)

Question Source: Bank #34062

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43 (2)

Comments:

REFUELING OPERATIONS

3/4.9.2 INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.9.2 At least 2 source range monitor* (SRM) channels shall be OPERABLE and inserted to the normal operating level with:##

a. Annunciation and continuous visual indication in the control room,

b. One of the required SRM detectors located in the quadrant where CORE ALTERATIONS are being performed and the other required SRM detector located in an adjacent quadrant, and

c. Unless adequate shutdown margin has been demonstrated per Specification 3.1.1, the "shorting links" removed from the RPS circuitry prior to and during the time any control rod is withdrawn.#

d. During a SPIRAL UNLOAD, the count rate may drop below 3 cps when the number of assemblies remaining in the core drops to sixteen or less.

e. During a SPIRAL RELOAD, up to four fuel assemblies may be loaded in the four bundle locations immediately surrounding each of the four SRMs prior to obtaining 3 cps. Until these assemblies have been loaded, the 3 cps count rate is not required.

APPLICABILITY: OPERATIONAL CONDITION 5.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS and fully insert all insertable control rods.

SURVEILLANCE REQUIREMENTS

4.9.2 Each of the above required SRM channels shall be demonstrated OPERABLE by:

a. In accordance with the Surveillance Frequency Control Program:

1. Performance of a CHANNEL CHECK,

---

* The use of special movable detectors during CORE ALTERATIONS in place of the normal SRM nuclear detectors is permissible as long as these special detectors are connected to the normal SRM circuits.

\# Not required for control rods removed per Specification 3.9.10.1 and 3.9.10.2.

\## Three SRM channels shall be OPERABLE for critical shutdown margin demonstrations. An SRM detector may be retracted provided a channel indication of at least 100 cps is maintained.

5.2.12. IF the "One-Rod-Out" Refuel Position interlock was BYPASSED during the removal of control rod(s) AND/OR control rod drive mechanism(s) IAW T/S 3.9.10.2
THEN, **PERFORM** the applicable portions of HC.OP-ST.KE-0001(Q) Refuel Interlock Operability Functional Test, as a retest, to ensure the operability of the "One Rod Out" interlock. **[TS 4.9.10.2.2]**

___ SM/CRS

5.2.13. **COMPLETE** the fuel transfer sequence prepared IAW HC.RE-FR.ZZ-0001(Q), Hope Creek Special Nuclear Material and Core Component Movement, to reload the Reactor for the next fuel cycle.

___ Reactor Engineering

5.2.14. WHEN performing a SPIRAL RELOAD,
THEN IMMEDIATELY following the loading of the first 16 bundles, **VERIFY** the SRM channel count rate is 3 cps, IAW HC.OP-ST.SE-0005(Q), SRM Channel Count Rate Surveillance. **[T/S 3.9.2.e, 4.9.2.C.2, 4.9.2.C.3]**

___ SM/CRS

5.2.15. **VERIFY** the location AND orientation of the Fuel Assemblies in the Reactor Vessel for the new core loading IAW HC.RE-FR.ZZ-0008(Q), Verification of Fuel Location.

___ Reactor Engineering

Facility: Hope Creek

Vendor: GE

Exam Date: 2019

Exam Type: SRO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #		215005 A2.08
	Importance Rating		3.4

K/A Statement: Ability to (a) predict the impacts of the following on the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGEMONITOR SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Faulty or erratic operation of detectors/systems.

Question: SRO #88

Given:

- The plant is operating at 100% rated power.
- With no rods selected on the rod display, the following alarm is received:
    - LPRM UPSCALE (C3-D5)
- The operator confirms that one LPRM is upscale as shown on the PPC OD-8 (Plant Process Computer).

What subsequent action will have to be taken IAW HC.OP-AB.IC-0004, Neutron Monitoring, after bypassing the failed LPRM and what is the requirement for APRM operability?

A. Direct the reactor engineer to evaluate the failed LPRM.
APRM operability requires a minimum of 4 LPRMs per level.

B. Reset the tripped RPS channel.
APRM operability requires a minimum of 3 LPRMs per level.

C. Reset the tripped RPS channel.
APRM operability requires a minimum of 4 LPRMs per level.

D. Direct the reactor engineer to evaluate the failed LPRM.
APRM operability requires a minimum of 3 LPRMs per level.

Proposed Answer: D

Explanation (Optional)

A: **Incorrect-** An APRM channel is inoperable if there are less than **3 LPRM** inputs per level or less than 20 LPRM inputs to an APRM channel.

B: **Incorrect-** There is no RPS trip for LPRM inputs (administrative operability concern).

C: **Incorrect-** An APRM channel is inoperable if there are less than **3 LPRM** inputs per level or less than 20 LPRM inputs to an APRM channel. There is no RPS trip for LPRM inputs (administrative operability concern).

D: **Correct-**IAW AB.IC-0004 subsequent action C.3. There is no RPS trip for LPRM inputs (administrative operability concern). An APRM channel is inoperable if there are less than **3 LPRM** inputs per level or less than 20 LPRM inputs to an APRM channel.

Technical Reference(s): HC.OP-AB.IC-0004 (Attach if not previously provided)
Neutron Monitoring

HCGS Tech Specs- 3.3.1 and Table 3.3.1-1

HC.OP-SO.SE-0001
NI Operations

Proposed References to be provided to applicants during examination: none

Learning Objective: Given a scenario of applicable conditions and access to Technical Specifications: (As available)
a. Select those sections which are applicable to the LPRMS/APRMS
b. Evaluate LPRM/APRM operability and determine required actions applicable to the APRMS (SRO Only)
c. Explain the bases for those Technical Specifications associated with the APRMS IAW HCGS Technical Specifications.
Explain the reasons for how plant/system parameters respond when implementing Neutron Monitoring.

Question Source: Bank #

Modified Bank # 84451 (attach parent)

New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43 (2)(5)

Comments: Original Question (Complete modification to fit K/A. First part of question and answer is based on the abnormal not thermal limits)

The Unit is operating at 100% power with no rods selected and the following alarm is received:

LPRM UPSCALE (C3-D5)

The operator confirms that one LPRM is upscale as shown on the PPC OD-8 (Plant Process Computer).

Which ONE of the following describes how this will affect the core thermal limit calculations and the requirement for APRM operability?

A. MFLPD and MAPRAT values will rise.
APRM operability requires a minimum of 3 LPRMs per level.

B. MFLPD and MAPRAT values will rise.
APRM operability requires a minimum of 4 LPRMs per level.

C. MFLPD and MAPRAT values will lower.
APRM operability requires a minimum of 3 LPRMs per level.

D. MFLPD and MAPRAT values will lower.
APRM operability requires a minimum of 4 LPRMs per level.

Answer: A

SUBSEQUENT OPERATOR ACTIONS (continued)

CONDITION	ACTION
B. APRM Malfunction. [T/S 3/4.3.1, 3/4.3.6], Date/Time: ________________	___ B.1 **BYPASS** the Malfunctioning APRM.
C. LPRM Malfunction. [T/S 3/4.3.1, 3/4.3.6, 3/4.1.4.3] Date/Time: _________	C.1 **DETERMINE** if an LPRM is Malfunctioning as follows: ___ a. **SELECT** a control rod that will display the LPRM on RBM ODA display (10C651). ___ b. **COMPARE** the LPRM output with other LPRMs at the same elevation AND/OR directly adjacent. ___ C.2 **BYPASS** the Malfunctioning LPRM (SE-0001). ___ C.3 **DIRECT** the Reactor Engineer to evaluate the failed LPRM.
D. TIP detector retracts past the in-shield limit. Date/Time: ___________	___ D.1 **NOTIFY** plant personnel via the PAGE of potential high radiation in the vicinity of the TIP Drive Mechanisms. ___ D.2 **DIRECT** Radiation Protection to perform a survey of the TIP Drive Mech. area.

2.2.3. The signals listed in Table SE-002 will initiate a **REACTOR SCRAM**. ________

TABLE SE-002

PARAMETER	SETPOINT	BYPASSED
SRM Upscale	≤ 2 X 10^5 cps	Shorting Links installed
IRM Upscale	≤ 120/125 of full scale	Reactor Mode Switch in RUN
IRM Inoperative	a) Module unplugged b) Low Voltage c) Mode Switch not in Operate	Reactor Mode Switch in RUN
APRM Upscale	≤ 116.3% of Rated Thermal Power	None
APRM Upscale	≤ 17% of Rated Thermal Power	Reactor Mode Switch in RUN.
APRM Simulated Thermal Power Upscale (Flow Biased)	≤ 0.56W + 58% ≤ 0.56(W-10.8%) + 58% for SLO maximum of 113.5% of Rated Thermal Power	None
APRM Inoperative	a) Mode Switch NOT in Operate b) The Firmware/Software Watchdog timer has timed out c) A critical Self-Test fault is detected	None
OPRM Trip	Detection of Thermal Hydraulic Instability	Rx Pwr < 24%, OR Core Flow > 76% (nominal) OR APRM in BYPASS

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

TABLE NOTATIONS

(a) A channel may be placed in an inoperable status for up to 6 hours for required surveillance without placing the trip system in the tripped condition provided at least one OPERABLE channel in the same trip system is monitoring that parameter.

(b) This function shall be automatically bypassed when the reactor mode switch is in the Run position.

(c) Unless adequate shutdown margin has been demonstrated per Specification 3.1.1, the "shorting links" shall be removed from the RPS circuitry prior to and during the time any control rod is withdrawn*.

(d) The non-coincident NMS reactor trip function logic is such that all channels go to both trip systems. Therefore, when the "shorting links" are removed, the Minimum OPERABLE Channels Per the Trip System are 6 IRMS and 2 SRMS.

(e) An APRM channel is inoperable if there are less than 3 LPRM inputs per level or less than 20 LPRM inputs to an APRM channel.

Facility: Hope Creek

Vendor: GE

Exam Date: 2019

Exam Type: SRO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #		262002 G2.1.20
	Importance Rating		4.6

K/A Statement: Ability to interpret and execute procedure steps.- Uninterruptable Power Supply (AC/DC)

Question: SRO #89

Given:

- The plant is at 85% rated power.

Then:

- The DC supply breaker to the AD481 Inverter (72-41022) trips open due to undervoltage.

(1) Which procedure provides the required actions to mitigate this plant condition?
(2) What Technical Specification action(s), if any, is (are) required?
[Reference attached]

A. (1) HC.OP-SO.PK-0001(Q), 125 VDC Electrical Distribution System Operation.
(2) Enter a tracking LCO. No Technical Specification LCO entry is required because the backup power source is still available to supply power to the distribution panel.

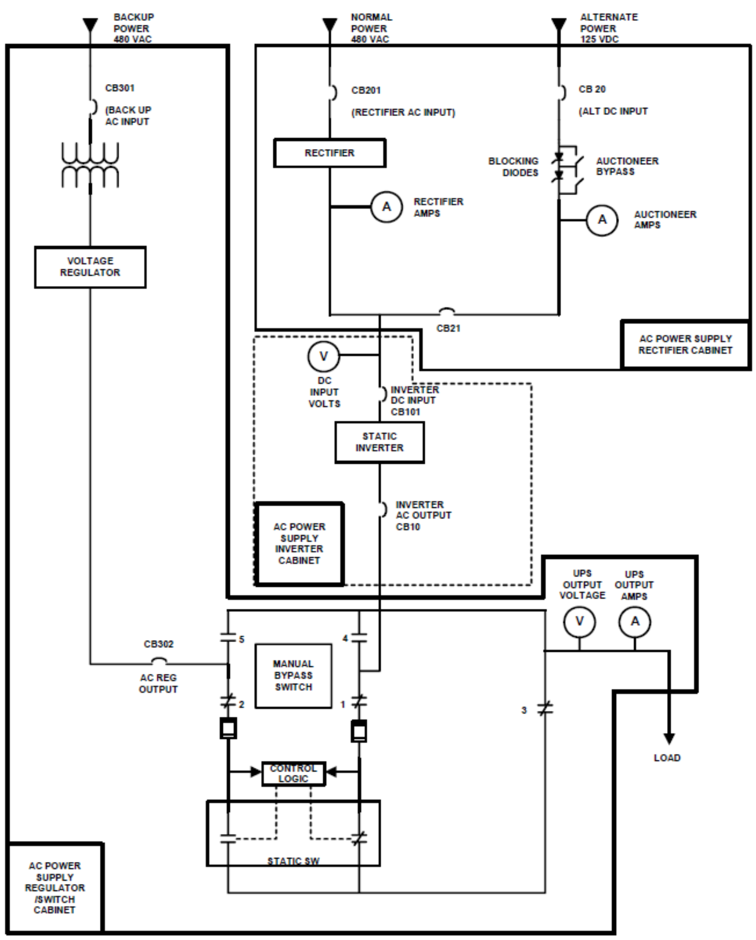
B. (1) HC.OP-SO.PK-0001(Q), 125 VDC Electrical Distribution System Operation.
(2) Be in Cold Shutdown within 44 hours of the time breaker 72-41022 was opened, if the associated inverter cannot be made Operable.

C. (1) HC.OP-AB.ZZ-0136(Q), Loss of 120 VAC Inverter.
(2) The associated inverter must be made Operable within 7 days to prevent additional required actions.

D. (1) HC.OP-AB.ZZ-0136(Q), Loss of 120 VAC Inverter.
(2) The associated 120 VAC distribution panel must be made Operable within 8 hours to prevent additional required actions.

Proposed Answer: **C**

Explanation (Optional): The power supply arrangement to loads having an Uninterruptable Power Supply (UPS) is as follows:
480 VAC (Normal) is rectified and auctioneered with 125 VDC to supply the static inverters. The static switch selects either the inverter output or the regulated 120 VAC backup supply to provide power to the loads.



TS 3.8.3.1 Action d:
With one or both inverters in one channel inoperable, energize the associated 120 volt A.C.

distribution panel(s) within 8 hours, and restore the inverter(s) to OPERABLE status within 7 days; or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.
From TS Bases 3.8.3.1:
"Energized" 120 VAC distribution panels [A-D]J48[1/2] require the panels to be energized to their proper voltage from the associated inverter via inverted **DC voltage,** inverter using the normal AC source, or Class 1E backup AC source via voltage regulator. OPERABLE inverters require the associated 120 VAC distribution panels ([A-D]J48[1/2]) to be powered by the inverter with output voltage within tolerances, and power input to the inverter from the associated station battery. Alternatively, the power supply may be from an internal AC source via rectifier as long as the OPERABLE station battery is available as the uninterruptible power supply.
HC.OP-SO.PN-0001- P&L 2.1.2:
For the purpose of defining the OPERABILITY requirements of the 120 VAC distribution panels specified in T/S 3.8.3.1 & 3.8.3.2, the phrase "energized" is defined in the Technical Specification Basis

A: **Incorrect** – HC.OP-SO.PK-0001(Q), "125 VDC Electrical Distribution System Operation," would be used to remove or place a 125 VDC electrical distribution system in service. Troubleshooting and repair should be completed before this procedure is implemented. HC.OP-AB.ZZ-0136(Q), "Loss of 120 VAC Inverter" is more appropriate for this condition and the expected alarms. Additionally, a TS entry is required. (see attached)

B: **Incorrect-** The 120 VAC distribution panel is energized, but the inverter is inoperable based upon not having the DC input. The requirement for COLD SHUTDOWN in this event would be in 60 hours.

C: **Correct** – HC.OP-AB.ZZ-0136(Q), "Loss of 120 VAC Inverter," and appropriate Attachment should be entered to determine verify automatic plant response and. TS 3.8.3.1.d - The 120 VAC distribution panel must be energized within 8 hours (it is). The remaining action is to restore the inverter to OPERABLE status within 7 days.

D: **Incorrect -** This describes the correct TS (3.8.3.1.d), but the 120 VAC distribution panel is energized, so the 8 hour requirement in the first part of this TS is not applicable. The DC supply breaker is open, which requires the inverter to be declared inoperable and restored in 7days.

Technical Reference(s):	T.S. 3.8.3.1 and BASES	(Attach if not previously provided)
	A.C. Sources	
	HC.OP-SO.PN-0001 120VAC Distribution	

HC.OP-AB.ZZ-0136
Loss of 120VAC Inverter

Proposed References to be provided to applicants during examination: T.S. 3.8.3.1

Learning Objective: (As available)

Given a scenario of applicable conditions and access to Technical Specifications:

a. Identify those sections that are applicable to Inverters IAW Technical Specifications.
b. Evaluate Inverter operability and determine required actions associated with the Inverters IAW Technical Specifications
c. Explain the bases of those technical specification items associated with the Inverters IAW Technical Specifications.

Question Source: Bank # 2018 Audit Exam

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43 (5)

Comments:

ELECTRICAL POWER SYSTEMS

3/4.8.3 ONSITE POWER DISTRIBUTION SYSTEMS

DISTRIBUTION - OPERATING

LIMITING CONDITION FOR OPERATION

3.8.3.1 The following power distribution system channels shall be energized:

a. A.C. power distribution:

1. Channel A, consisting of:

a) 4160 volt A.C. switchgear bus	10A401
b) 480 volt A.C. load centers	10B410
	10B450
c) 480 volt A.C. MCCs	10B212
	10B411
	10B451
	10B553
d) 208/120 volt A.C. distribution panels	10Y401(source:10B411)
	10Y411(source:10B451)
	10Y501(source:10B553)
e) 120 volt A.C. distribution panels	1AJ481 and inverter AD481
	1YF401(source: 1AJ481)
	1AJ482 and inverter AD482

2. Channel B, consisting of:

a) 4160 volt A.C. switchgear bus	10A402
b) 480 volt A.C. load centers	10B420
	10B460
c) 480 volt A.C. MCCs	10B222
	10B421
	10B461
	10B563
d) 208/120 volt A.C. distribution panels	10Y402(source:10B421)
	10Y412(source:10B461)
	10Y502(source:10B563)
e) 120 volt A.C. distribution panels	1BJ481 and inverter BD481
	1YF402(source:1BJ481)
	1BJ482 and inverter BD482

3. Channel C, consisting of:

a) 4160 volt A.C. switchgear bus	10A403
b) 480 volt A.C. load centers	10B430
	10B470
c) 480 volt A.C. MCCs	10B232
	10B431
	10B471
	10B573
d) 208/120 volt A.C. distribution panels	10Y403(source:10B431)
	10Y413(source:10B471)
	10Y503(source:10B573)

ELECTRICAL POWER SYSTEMS

LIMITING CONDITION FOR OPERATION (Continued)

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With one of the above required A.C. distribution system channels not energized, re-energize the channel within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With one of the above required 125 volt D.C. distribution system channels not energized, re-energize the division within 2 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. With any one of the above required 250 volt D.C. distribution systems not energized, declare the associated HPCI or RCIC system inoperable and apply the appropriate ACTION required by the applicable Specifications.

d. With one or both inverters in one channel inoperable, energize the associated 120 volt A.C. distribution panel(s) within 8 hours, and restore the inverter(s) to OPERABLE status within 7 days; or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.8.3.1 Each of the above required power distribution system channels shall be determined energized in accordance with the Surveillance Frequency Control Program by verifying correct breaker/switch alignment and voltage on the busses/MCCs/panels.

1.0 PURPOSE AND SCOPE

1.1 Purpose

1.1.1. This procedure describes energizing and de-energizing the 120VAC Electrical Distribution System that utilize Cyberex UPSs, including the 120VAC Uninterruptible Power Supplies (UPS), and all 120VAC Distribution Panels (Class 1E and Non-Class 1E) except as noted in Scope section 1.2 **[CD 267B]**

1.2 Scope

1.2.1. The portions of the 120VAC Electrical Distribution System that utilize Cyberex UPSs, including the 120VAC Uninterruptible Power Supplies (UPS), and all 120VAC Distribution Panels (Class 1E and Non- Class 1E) except as noted in 1.2.2 and 1.2.3. **[CD 267B]**

1.2.2. This procedure does not describe 120VAC Standby Lighting Electrical Distribution. **REFER** to HC.OP-SO.QB-0001(Z), Lighting System Operation.

1.2.3. This procedure does NOT apply to (AMETEK) Inverters 1AD484, 1AD492, 1BD492, 1BD483, 1BD484, 0BD595, 1CD483, 1CD484, or 0AD495.
REFER to HC.OP-SO.PN-0002(Q), 120 VAC Electrical Distribution Utilizing AMETEK Solid State Controls.

2.0 PRECAUTIONS AND LIMITATIONS

2.1 Precautions

2.1.1. **WHEN** de-energizing a Class 1E 120VAC power supply and/or associated distribution panels, **THEN** the Class 1E Electrical System requirements of T/S 3.8.3.1 **AND** 3.8.3.2 shall be observed. _______

2.1.2. For the purpose of defining the OPERABILITY requirements of the 120 VAC distribution panels specified in T/S 3.8.3.1 & 3.8.3.2, the phrase “energized” is defined in the Technical Specification Basis. _______

2.1.3. UPS Inverters may be damaged by operations NOT in sequence with this procedure. _______

2.1.4. Mis-operation of the MANUAL BYPASS switch can cause a loss of control power to vital instruments. Prior to moving the switch to any position, the operator shall ensure that power is available **AND** that the proper lamp indications are present IAW this procedure. **[CD-043F]** _______

2.1.5. Manual restoration of loads on Distribution Panel(s) may cause spurious overhead alarms due to Inverter Static Switch Transfer operation. _______

CATEGORY II

LOSS OF 120 VAC INVERTER

1.0 SYMPTOMS

1.1 Alarms

1.1.1. 120 VAC UPS TROUBLE* **D3-E3**

* Except for loss of 1BD483 which supplies power to the overhead annunciators. A loss of 1BD483 results in a loss of all overhead annunciators.

* Loss of A/B/C/D 482 Inverters does not result in 120VAC Trouble due to loss of Bailey logic to drive alarm.

1.1.2. Various multiple system alarms.

1.2 Simultaneous failures of indication and/or control of multiple systems.

1.3 Loss of instrumentation and/or control of 1E Systems of the same channel/division.

2.0 AUTOMATIC ACTIONS

2.1 Possible auto transfer of the affected inverter to either its alternate power (DC) **OR** back-up power supply.

2.2 **REFER** to the specific Attachment for the Inverter failure description for the automatic plant responses.

3.0 IMMEDIATE OPERATOR ACTIONS

None

ELECTRICAL POWER SYSTEMS

BASES

3/4.8.1, 3/4.8.2 and 3/4.8.3 A.C. SOURCES, D.C. SOURCES and ONSITE POWER DISTRIBUTION SYSTEMS (Continued)

Operation with a battery cell's parameter outside the normal limit but within the allowable value specified in Table 4.8.2.1-1 is permitted for up to 31 days. During this 31 day period: (1) the allowable values for electrolyte level ensures no physical damage to the plates with an adequate electron transfer capability; (2) the allowable value for the average specific gravity of all the cells, not more than .020 below the manufacturer's recommended full charge specific gravity ensures that the decrease in rating will be less than the safety margin provided in sizing; (3) the allowable value for an individual cell's specific gravity, ensures that an individual cell's specific gravity will not be more than .040 below the manufacturer's full charge specific gravity and that the overall capability of the battery will be maintained within an acceptable limit; (4) the allowable value for an individual cell's float voltage, greater than 2.07 volts, ensures the battery's capability to perform its design function; (5) the TABLE 4.8.2.1-1 NOTATION 31 day ACTION time was derived taking into consideration that while battery capacity is degraded, sufficient capacity exists to perform the intended function while providing a time period adequate to permit full restoration of the battery cell parameters to normal limits.

"Energized" 120 VAC distribution panels [A-D]J48[1/2] require the panels to be energized to their proper voltage from the associated inverter via inverted DC voltage, inverter using the normal AC source, or Class 1E backup AC source via voltage regulator. OPERABLE inverters require the associated 120 VAC distribution panels ([A-D]J48[1/2]) to be powered by the inverter with output voltage within tolerances, and power input to the inverter from the associated station battery. Alternatively, the power supply may be from an internal AC source via rectifier as long as the OPERABLE station battery is available as the uninterruptible power supply.

Facility:	Hope Creek
Vendor:	GE
Exam Date:	2019
Exam Type:	SRO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #		206000 A2.01
	Importance Rating		4.0

K/A Statement: Ability to (a) predict the impacts of the following on the HIGH PRESSURE COOLANT INJECTION SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Turbine trips: BWR-2,3,4

Question: SRO #90

Given:

- The plant was at 100% rated power when a Station Blackout occurred.
- Reactor scrammed, all control rods at 00.
- HPCI initiated and immediately tripped.
- RCIC is injecting.
- 1 SRV has not reseated and is partially open.

10 minutes later, conditions are as follows:

• Reactor pressure	650 psig and stable
• Reactor level	-170 inches and lowering 1 inch/min
• Drywell pressure	1.5 psig rising slowly
• Drywell temperature	190°F rising slowly
• Suppression pool level	80 inches
• Suppression chamber temp	140°F
• Suppression pool temp	190°F
• All SRVs are capable of being opened	

Based on the above conditions, what is the next required action?

A. Perform Emergency RPV Depressurization immediately.

B. When RPV level drops to -198 inches, perform Emergency RPV Depressurization.

C. Perform Rapid RPV Depressurization immediately.

D. When RPV level drops to -185 inches, perform Rapid RPV Depressurization.

Proposed Answer: **C**

Explanation (Optional): See attached 105 BASES

A: **Incorrect-** This would be correct if HCTL was exceeded, but it is not being exceeded.

B: **Incorrect-** This would be the correct step if there was no injection source. In this case, RCIC is injecting.

C: **Correct-** If an injection source is available, rapid depressurization should be delayed at least until RPV water level reaches RPV Level 1 (-129" RPV level), but may be performed anytime RPV water level is between the top of the active fuel and the Minimum Steam Cooling RPV Water Level (-185" RPV level).

D: **Incorrect-** This is not a true statement. ALC-38 states "**BEFORE** RPV level reached -185 in."

Technical Reference(s): HC.OP-EO.ZZ.0105-BASES (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: Given any step of the procedure, describe the reason for performance of that step and/or expected system response to control manipulation prescribed by that step. (As available)

Question Source: Bank #154775

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43 (5)

Comments:

8.4 ALC-37 Is any pump lined up, running and available for injection
ALC-38 Before RPV level reaches -185 in.
ALC-39 Rapid RPV depressurization is required
ALC-40 Line up for injection, start pumps, and inject into the RPV with 1 or more preferred injection systems (Table 1)

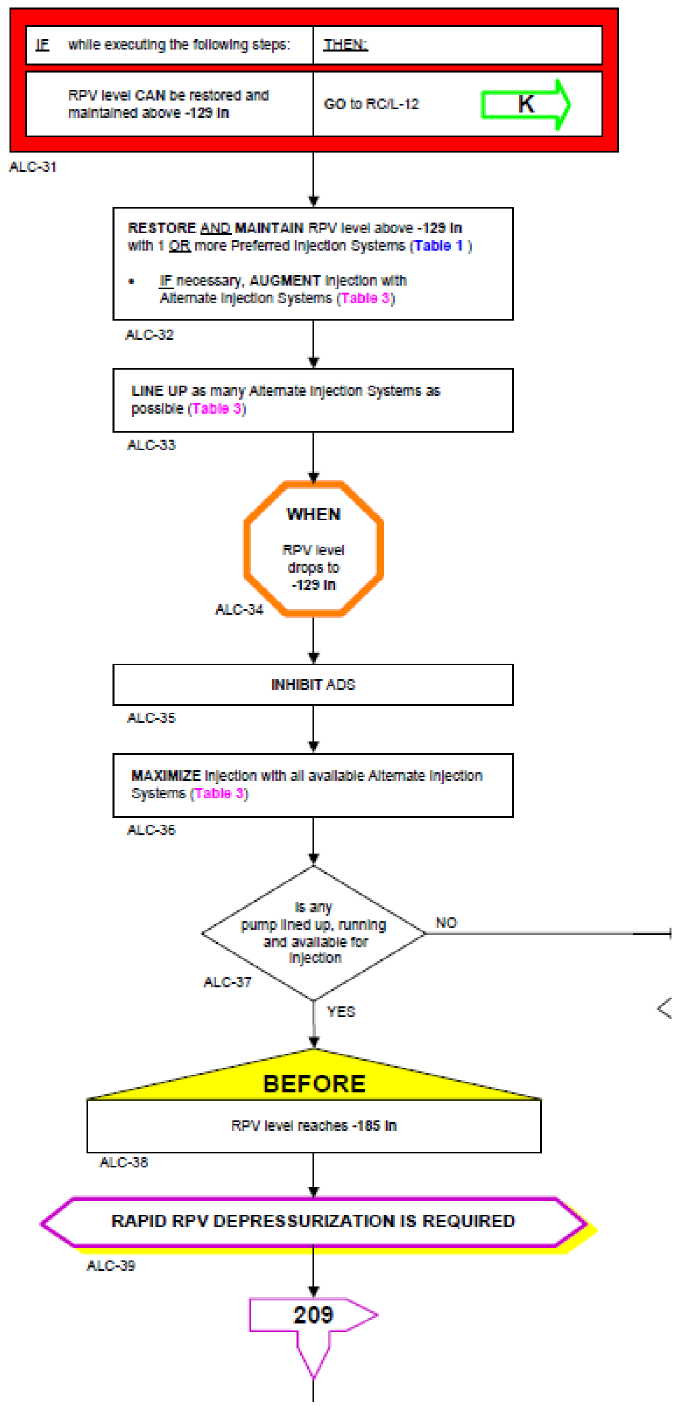
- **If necessary, augment injection with Alternate Injection Systems (Table3)**

If EOP-105 has been entered, HPCI and/or RCIC are the only source of injection to the RPV. As a result, the “concurrently perform” step after ALC-39 directs the operator to EOP-209, where the use of rapid RPV depressurization reduces RPV pressure while maintaining reactor pressure high enough to ensure the continued viability of these systems. Although it will not fully depressurize the reactor, rapid RPV depressurization permits injection from many low head systems, maximizes the total injection flow of operating systems, and minimizes the flow through any primary system break.

If an injection source is available, rapid depressurization should be delayed at least until RPV water level reaches RPV Level 1 (-129” RPV level), but may be performed anytime RPV water level is between the top of the active fuel and the Minimum Steam Cooling RPV Water Level (-185” RPV level). The MSCRWL is the lowest RPV water level at which the covered portion of the reactor core will generate sufficient steam to preclude any clad temperature in the uncovered portion of the core from exceeding 1500°F assuming the most limiting top peaked power shape prior to reactor shutdown.

8.0 ALC LEG OF EOP-105

Alternate Level Control –E



Facility: Hope Creek

Vendor: GE

Exam Date: 2019

Exam Type: SRO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		2
	K/A #		201001 G2.4.45
	Importance Rating		4.3

K/A Statement: Ability to prioritize and interpret the significance of each annunciator or alarm.- CRD Hydraulic.

Question: SRO #91

Given:

- A reactor startup is in progress.
- Reactor pressure is at 750 psig.
- Control rod withdrawal is in progress.
- CRD Hydraulic system parameters are all normal.

When:

- A withdrawn control rod receives an accumulator alarm due to nitrogen pressure.
- The NEO reports from the field that the accumulator nitrogen pressure is at 750 psig.

Which of the following describes when a reactor scram would be required?

A. When two accumulators are determined to be inoperable.

B. When the accumulator alarm is verified to be valid.

C. If the charging water header pressure was to lower below 940 psig.

D. If the control rod is NOT inserted within one hour.

Proposed Answer: C

Explanation (Optional): See attached T.S. 3.1.3.5

A: **Incorrect** – Can continue to operate at power. Would need for the charging header pressure to lower below 940 psig (CRD pump trip).

B: **Incorrect** -.Can continue to operate at power. Once the CRS determines the accumulator and then the charging water header would have to lower below 940 psig (CRD pump trip).

C: **Correct** -. TS requires an immediate scram if charging water header pressure is < 940 and any control rod with an inoperable accumulator is withdrawn.

D: **Incorrect** – This would be required after the plant has scrammed due to charging header low pressure of < 940 psig.

Technical Reference(s): T.S. 3.1.3.5 CONTROL ROD SCRAM ACCUMULATORS (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: Given a scenario of applicable operating conditions and access to Technical Specifications complete each of the following (NCO and Above) (As available)
Select those sections applicable to the CRDH System.
Evaluate CRDH System operability and determine required actions and time limits associated with inoperable components. (SRO Only)
Explain the bases for those Technical Specification sections associated with the CRDH System.

Question Source: Bank #30874

Modified Bank # (Note changes or add parent)

New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43 (2)

Comments:

REACTIVITY CONTROL SYSTEMS

CONTROL ROD SCRAM ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.1.3.5 Each control rod scram accumulator shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 5*.

ACTION:

---NOTE---
Separate condition entry is allowed for each control rod

---

a. In OPERATIONAL CONDITIONS 1 or 2:

1. With one control rod scram accumulator inoperable and reactor pressure ≥ 900 psig, within 8 hours,

    a) Restore the inoperable accumulator to OPERABLE status, or

    b) Declare the associated control rod scram time "slow"***, or

    c) Insert the associated control rod, declare the associated control rod inoperable and disarm the associated control valves by closing the drive water and exhaust water isolation valves.

    Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

2. With two or more control rod scram accumulators inoperable and reactor pressure ≥ 900 psig,

    a) Within 20 minutes of discovery of this condition concurrent with charging water pressure < 940 psig, restore charging water header pressure to ≥ 940 psig otherwise place the mode switch in the shutdown position**, and

    b) Within one hour, declare the associated control rod scram time "slow"***, or

    c) Within one hour insert the associated control rods, declare the associated control rods inoperable and disarm the associated control valves by closing the drive water and exhaust water isolation valves.

    Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

---

* At least the accumulator associated with each withdrawn control rod. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

** Not applicable if all inoperable control rod scram accumulators are associated with fully inserted control rods.

*** Only applicable if the associated control rod scram time was within the limits of Table 3.1.3.3-1 during the last scram time Surveillance. Rods that are already considered "slow" should be declared inoperable and fully inserted.

REACTIVITY CONTROL SYSTEMS

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

3. With one or more control rod scram accumulators inoperable and reactor pressure < 900 psig,

a) Immediately upon discovery of charging water header pressure < 940 psig, verify all control rods associated with inoperable accumulators are fully inserted otherwise place the mode switch in the shutdown position**, and

b) Within one hour insert the associated control rod(s), declare the associated control rod(s) inoperable and disarm the associated control valves either electrically or hydraulically by closing the drive water and exhaust water isolation valves.

Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

b. In OPERATIONAL CONDITION 5*:

1. With one or more withdrawn control rods inoperable, upon discovery immediately initiate action to fully insert inoperable withdrawn control rods.

SURVEILLANCE REQUIREMENTS

4.1.3.5 Each control rod scram accumulator shall be determined OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by verifying that the indicated pressure is greater than or equal to 940 psig unless the control rod is inserted and disarmed or scrammed.

* At least the accumulator associated with each withdrawn control rod. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

** Not applicable if all inoperable control rod scram accumulators are associated with fully inserted control rods.

Facility: Hope Creek

Vendor: GE

Exam Date: 2019

Exam Type: SRO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		2
Group #		2
K/A #		214000 A2.02
Importance Rating		3.7

K/A Statement: Ability to (a) predict the impacts of the following on the ROD POSITION INFORMATION SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Reactor SCRAM.

Question: SRO #92

Given:

- The plant is at 10% rated power:
- ‘B’ RHR is in Suppression Pool Cooling
- HPCI in Full Flow test IAW HC.OP-IS.BJ-0001, HPCI Main and Booster Pump Set.

When:

- A complete loss of the AD483 (AJ483) inverter output occurs causing a loss of all RPIS (Rod Position Information System) indications.

Then:

- A Loss of Offsite Power occurs.
- NO operator actions have been taken.

Current plant conditions:

- The MSIVs are closed.
- Reactor Power: <1%.
- Reactor Pressure: 1040 psig and rising 1 psig every 2 minutes.
- SRV Lo-Lo Set is NOT armed.
- Reactor Level: 15” and lowering 1" every 2 minutes.
- NO systems are injecting to the RPV.
- Suppression Pool temperature: 96ºF and rising 1ºF every 2 minutes.
- Suppression Pool level: 79” and rising 1" every 5 minutes.

Which of the following actions are required to be directed for these conditions IAW plant procedures?

A. Commence a Cooldown at less than 100ºF per hour and restore Suppression Pool Level to between 74.5" and 78.5" with ‘B’ RHR letdown to Radwaste.

B. Stabilize RPV pressure below 1037psig and maintain RPV level between +54” and –185”.

C. Depressurize the reactor at a cooldown rate of 100ºF per hour and initiate SLC before Suppression Pool temperature reaches 140ºF.

D. Lower RPV level to between -50” and –185” and initiate SLC before Suppression Pool temperature reaches 110ºF.

Proposed Answer: **B**

Explanation (Optional): See attached 101A

A: **Incorrect** – The Loss of AJ483 causes a loss of RPIS. It will not be immediately possible to verify the Reactor Shutdown under all conditions without boron. This will require entry into EOP-101A. Due to the ATWS condition, cooldown is not allowed. Additionally due to the LOP, NSSSS is tripped and letdown from ‘B’ RHR to radwaste is isolated.

B: **Correct-** The Loss of AJ483 causes a loss of RPIS. It will not be immediately possible to verify the Reactor Shutdown under all conditions without boron. This will require entry into EOP-101A. With no SRV cycling, RC/P-4 directs stabilizing RPV press below 1037 psig. With power <4%, LP-10 directs maintaining RPV level between +54” and –185”.

C: **Incorrect** – Due to the ATWS condition, cooldown is not allowed. Additionally, AB.ZZ-0135 Loss of Offsite Power/SBO, directs a 100 degF per hour cooldown rate only for a Station Blackout. Since all EDGs started and loaded properly, this guidance is not applicable.

D: **Incorrect -** Step LP-13 of EOP-101A only directs lowering RPV level if power is >4% or unknown.

Technical Reference(s): HC.OP-EO.ZZ-0101A (Attach if not previously provided)

ATWS-RPV Control

HC.OP-AB.ZZ-0136
Loss of 120 VAC Inverter

Proposed References to be provided to applicants during examination: none

Learning Objective: Recognize abnormal indications/alarms and/or procedural requirements for implementing, Loss of 120 VAC Inverter, Abnormal Operating Procedure. State the conditions for the ATWS - RPV Control Emergency Operating Procedure Entry. (As available)

Question Source: Bank #59764

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43 (5)

Comments:

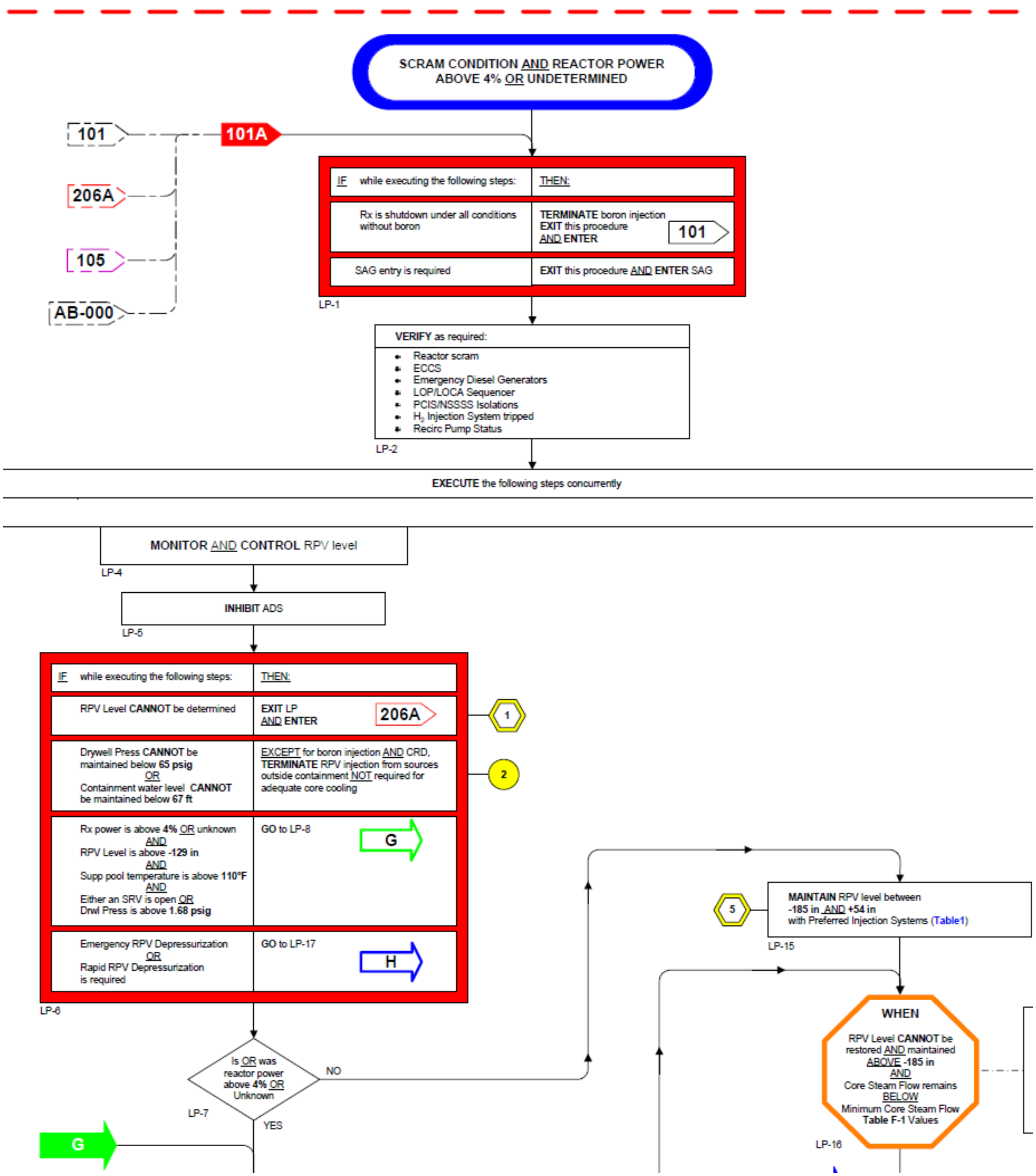
ATTACHMENT 9
1AD483 INVERTER

Automatic Plant Response

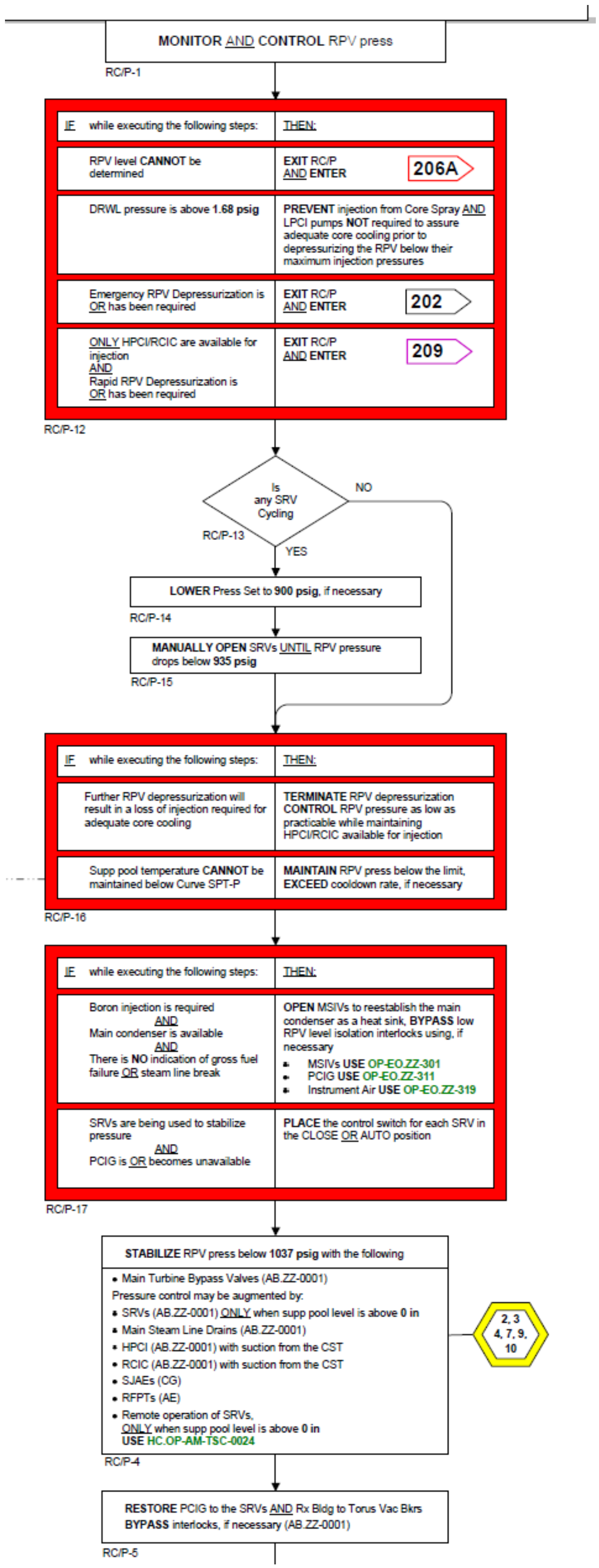
1. Loss of "A" RFPT Woodward Speed Controller resulting in "A" RFPT trip.
2. "A" Feedwater Heater Train trips. 3A, 4A, and 5A Extraction Steam isolates.
3. Reactor Feed Pump Startup Level Control Valve AE-HV-1785 fails closed.
4. Loss of Main Turbine EHC Normal power results in system transfer to Backup Supply (10Y107).
5. Chilled Water to "B" Recirc Motor Cooler HV-9515A1/A2 fail open.
6. Chilled Water to DW coolers HV-9510A1/E1/D1/H1/A2/E2 fail open.
7. Loss of power to 1YF405. (E-0012-1 Sht 3, E-1417-0 Sht 6A)
8. Loss of power to 'A' and 'C' PRNM Voters resulting in a Half Scram. ('A' RPS)

Control and Indication Failures

1. Loss of power to Rod Position Information Cabinet (RPIS).
   No Control Rod Position Data to RMCS, PPC, CRIDS, SPDS.
2. Loss of Main Turbine EHC Normal power, system transfers to the Backup Supply (10Y107).
   IF Backup Supply is NOT available, all EHC operation is lost.
   Main Turbine Bypass Valves will be inoperable in the event the transient results in a unit scram.
3. Loss of various Fire Protection Monitors and Suppression Systems.
4. Loss of TIP Controls and Monitors.
5. Loss of 1AX501, 1AX502 and 1AX503 Controls and Monitors.
6. Loss of various Radiation Monitoring instrumentation.
7. Loss of indication power to various BOP and Non-1E components.
8. Reactor Feed Pump Startup Level Control Valve AE-HV-1785.
9. Loss of LEFM (Leading Edge Flow Meter) Computer.
10. Loss of Control and Indication to various Chilled Water valves:
    HV-9515A1/A2, 9510A1/E1/D1/H1/A2/E2.
11. Loss of Control and Indication to GS-PSV-4946A, B, G, and H. (They will function normally on DP)
12. Loss of 'A' and 'C' APRM Downscale signal inputs to RRCS. (Possible SLC Injection IF Reactor Scram occurs with RRCS Initiation signals present)
13. Loss of the ability to Bypass APRMs 'A' and 'C'.



SCRAM CONDITION AND REACTOR POWER ABOVE 4% OR UNDETERMINED
101
101A
206A
105
AB-000
IF while executing the following steps:
THEN:
Rx is shutdown under all conditions without boron
TERMINATE boron injection
EXIT this procedure
AND ENTER
101
SAG entry is required
EXIT this procedure AND ENTER SAG
LP-1
VERIFY as required:
Reactor scram
ECCS
Emergency Diesel Generators
LOP/LOCA Sequencer
PCIS/NSSSS Isolations
H_2 Injection System tripped
Recirc Pump Status
LP-2
EXECUTE the following steps concurrently
MONITOR AND CONTROL RPV level
LP-4
INHIBIT ADS
LP-5
IF while executing the following steps:
THEN:
RPV Level CANNOT be determined
EXIT LP
AND ENTER
206A
1
Drywell Press CANNOT be maintained below 65 psig
OR
Containment water level CANNOT be maintained below 67 ft
EXCEPT for boron injection AND CRD, TERMINATE RPV injection from sources outside containment NOT required for adequate core cooling
2
Rx power is above 4% OR unknown
AND
RPV Level is above -129 in
AND
Supp pool temperature is above 110°F
AND
Either an SRV is open OR
Drwl Press is above 1.68 psig
GO to LP-8
G
Emergency RPV Depressurization
OR
Rapid RPV Depressurization is required
GO to LP-17
H
LP-6
Is OR was reactor power above 4% OR Unknown
NO
YES
LP-7
G
5
MAINTAIN RPV level between -185 in AND +54 in with Preferred Injection Systems (Table1)
LP-15
WHEN
RPV Level CANNOT be restored AND maintained ABOVE -185 in
AND
Core Steam Flow remains BELOW
Minimum Core Steam Flow
Table F-1 Values
LP-16


MONITOR AND CONTROL RPV press
RC/P-1
IF while executing the following steps:
THEN:
RPV level CANNOT be determined
EXIT RC/P AND ENTER
206A
DRWL pressure is above 1.68 psig
PREVENT injection from Core Spray AND LPCI pumps NOT required to assure adequate core cooling prior to depressurizing the RPV below their maximum injection pressures
Emergency RPV Depressurization is OR has been required
EXIT RC/P AND ENTER
202
ONLY HPCI/RCIC are available for injection AND Rapid RPV Depressurization is OR has been required
EXIT RC/P AND ENTER
209
RC/P-12
Is any SRV Cycling
NO
RC/P-13
YES
LOWER Press Set to 900 psig, if necessary
RC/P-14
MANUALLY OPEN SRVs UNTIL RPV pressure drops below 935 psig
RC/P-15
IF while executing the following steps:
THEN:
Further RPV depressurization will result in a loss of injection required for adequate core cooling
TERMINATE RPV depressurization CONTROL RPV pressure as low as practicable while maintaining HPCI/RCIC available for injection
Supp pool temperature CANNOT be maintained below Curve SPT-P
MAINTAIN RPV press below the limit, EXCEED cooldown rate, if necessary
RC/P-16
IF while executing the following steps:
THEN:
Boron injection is required AND Main condenser is available AND There is NO indication of gross fuel failure OR steam line break
OPEN MSIVs to reestablish the main condenser as a heat sink, BYPASS low RPV level isolation interlocks using, if necessary
• MSIVs USE OP-EO.ZZ-301
• PCIG USE OP-EO.ZZ-311
• Instrument Air USE OP-EO.ZZ-319
SRVs are being used to stabilize pressure AND PCIG is OR becomes unavailable
PLACE the control switch for each SRV in the CLOSE OR AUTO position
RC/P-17
STABILIZE RPV press below 1037 psig with the following
• Main Turbine Bypass Valves (AB.ZZ-0001)
Pressure control may be augmented by:
• SRVs (AB.ZZ-0001) ONLY when supp pool level is above 0 in
• Main Steam Line Drains (AB.ZZ-0001)
• HPCI (AB.ZZ-0001) with suction from the CST
• RCIC (AB.ZZ-0001) with suction from the CST
• SJAEs (CG)
• RFPTs (AE)
• Remote operation of SRVs, ONLY when supp pool level is above 0 in USE HC.OP-AM-TSC-0024
2, 3 4, 7, 9, 10
RC/P-4
RESTORE PCIG to the SRVs AND Rx Bldg to Torus Vac Bkrs BYPASS interlocks, if necessary (AB.ZZ-0001)
RC/P-5

Facility: Hope Creek

Vendor: GE

Exam Date: 2019

Exam Type: SRO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		2
Group #		2
K/A #		272000 G2.1.31
Importance Rating		4.3

K/A Statement: Ability to locate control room switches, controls, and indications, and to determine that they correctly reflect the desired plant lineup- Radiation Monitoring.

Question: SRO #93

Given:

- The plant is at 100% rated power.

When :

- OHA C6-C1, “RADIATION MONITORING ALARM/TRBL”, is received.

The NCO Observes the following condition:

- SPV channel 9RX581 (High Range Monitor) has a check source test failure alarm on the RM-11.

Then:

- The NCO initiates a second check source test for SPV channel 9RX581 (High Range Noble Gas Monitor).

Resulting in:

- Failure of the second source check as indicated on the RM-11.
- All other channels have normal indication.

Assuming releases through the South Plant Vent are to continue, then ________________.
[Reference attached]

A. obtain noble gas samples at least once per 12 hours.

B. estimate SPV flow rate at least once per 4 hours, obtain noble gas samples at least once per 12 hours, and establish an alternate method for sampling.

C. either restore to OPERABLE status within 72 hours or establish a preplanned alternate method for monitoring AND prepare and submit a Special Report to the NRC within 14 days.

D. estimate SPV flow rate at least once per 4 hours, obtain noble gas samples at least once per 12 hours, establish an alternate method for sampling, and either restore to OPERABLE status within 72 hours or prepare and submit a Special Report to the NRC within 14 days.

Proposed Answer: **C**

Explanation (Optional):

A: **Incorrect-** Low range is functioning. Also omits T.S. 3.3.7.5 actions.

B: **Incorrect-** Low range is functioning. Also omits T.S. 3.3.7.5 actions.

C: **Correct-** Tech Spec Table 3.3.7.5-1 Action 81. See attached.

D: **Incorrect-** Low range is functioning. Only T.S. 3.3.7.5 actions are required.

Technical Reference(s): T.S 3.3.7.5 (Attach if not previously provided)

Accident Monitoring Instrumentation

Proposed References to be provided to applicants during examination: T.S. 3.3.7.5 and Table 3.3.7.5-1

Learning Objective: Given specific plant operating conditions and a copy of the Hope Creek Offsite Dose Calculation Manual (ODCM), evaluate plant/system conditions and determine required actions (if any) to be taken IAW the ODCM. (As available)

Question Source: Bank #65482

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43 (2)

Comments:

INSTRUMENTATION

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.7.5 The accident monitoring instrumentation channels shown in Table 3.3.7.5-1 shall be OPERABLE.

APPLICABILITY: As shown in Table 3.3.7.5-1.

ACTION:

With one or more accident monitoring instrumentation channels inoperable, take the ACTION required by Table 3.3.7.5-1.

SURVEILLANCE REQUIREMENTS

4.3.7.5 Each of the above required accident monitoring instrumentation channels shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3.7.5-1.

TABLE 3.3.7.5-1

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	REQUIRED NUMBER OF CHANNELS	MINIMUM CHANNELS OPERABLE	APPLICABLE OPERATIONAL CONDITIONS	ACTION
1. Reactor Vessel Pressure	2	1	1,2,3	80
2. Reactor Vessel Water level	2	1	1,2,3	80
3. Suppression Chamber Water level	2	1	1,2,3	80
4. Suppression Chamber Water Temperature*	2	1	1,2,3	80
5. Suppression Chamber Pressure	2	1	1,2,3	80
6. Drywell Pressure	2	1	1,2,3	80
7. Drywell Air Temperature	2	1	1,2,3	80
8. Deleted				
9. Deleted				
10. Drywell Atmosphere Post-Accident Radiation Monitor	2	1	1,2,3	80
11. North Plant Vent Radiation Monitor #	1	1	1,2,3	81
12. South Plant Vent Radiation Monitor #	1	1	1,2,3	81
13. FRVS Vent Radiation Monitor #	1	1	1,2,3	81
14. Primary Containment Isolation Valve Position Indication ##	2/valve	1/valve	1,2,3	82

* Average bulk pool temperature.

\# High range noble gas monitors.

\## One channel consists of the open limit switch, and the other channel consists of the closed limit switch.

Table 3.3.7.5-1 (Continued)

ACCIDENT MONITORING INSTRUMENTATION
ACTION STATEMENTS

ACTION 80 -

a. With the number of OPERABLE channels less than the Required Number of Channels shown in Table 3.3.7.5-1, restore the inoperable channel to OPERABLE status within 30 days, or immediately initiate actions in accordance with 6.9.2.

b. With the number of OPERABLE channels less than the Minimum Number of Channels shown in Table 3.3.7.5-1, (except for the Drywell Atmosphere Post Accident Radiation Monitor) restore at least one inoperable channel to OPERABLE status within 7 days, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. Deleted

d. With the number of OPERABLE Drywell Atmosphere Post Accident Radiation Monitor channels less than the Minimum Number of Channels requirement shown in Table 3.3.7.5-1, initiate action in accordance with ACTION 81, below.

ACTION 81 - With the number of OPERABLE accident monitoring instrumentation channels less than required by the Minimum Channels OPERABLE requirement, either restore the inoperable channel(s) to OPERABLE status within 72 hours, or:

a. Initiate the preplanned alternate method of monitoring the appropriate parameter(s), and

b. Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

ACTION 82 -

a. With the number of OPERABLE accident monitoring instrumentation channels less than the Required Number of Channels shown in Table 3.3.7.5-1, verify the valve(s) position by use of alternate indication methods. If the affected penetration is not isolated by either (i) a closed manual valve, (ii) a blind flange, or (iii) a deactivated automatic valve located outside primary containment, restore the inoperable channel(s) to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

Table 3.3.7.5-1 (Continued)

ACCIDENT MONITORING INSTRUMENTATION
ACTION STATEMENTS

ACTION 82 -

b. With the number of OPERABLE accident monitoring instrumentation channels less than the Minimum Channels OPERABLE requirements of Table 3.3.7.5-1, verify the valves) position by use of alternate indication methods. If the affected penetration is not isolated by either (i) a closed manual valve, (ii) a blind flange, or (iii) a deactivated automatic valve located outside primary containment, restore the inoperable channel(s) to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

Facility: Hope Creek

Vendor: GE

Exam Date: 2019

Exam Type: SRO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		
	K/A #		2.1.2
	Importance Rating		4.4

K/A Statement: Knowledge of operator responsibilities during all modes of plant operation.

Question: SRO #94

IAW OP-AA-108-115, 'OPERABILITY DETERMINATIONS & FUNCTIONALITY ASSESSMENTS', the determination of whether systems, structures, or components (SSCs) are operable is the responsibility of _______________.

A. ONLY a senior licensed operator on the operating shift crew

B. ANY senior licensed operator (does not need to be assigned to the operating shift crew)

C. the Site Engineering Director

D. the Operations Director

Proposed Answer: **A**

Explanation (Optional): (see attached procedure) **Operability Declaration**: An operability declaration is a decision by senior licensed operator/SRO on the operating shift crew that there is a reasonable expectation that an SSC can perform its specified safety function / specified function. **Senior Licensed Operator/SRO/Operations Shift Management** . A senior licensed operator/SRO on the operating shift crew with the responsibility for plant operations makes the declaration of operability or functionality; i.e., makes the call on whether an SSC described in TSs is operable or inoperable, or an SSC that is not described in the TSs is functional or not functional. (See attached)

A: **Correct** – See above explanation.

B: **Incorrect-** On Shift SRO.

C: **Incorrect** – SRO qualified..

D: **Incorrect** – On Shift SRO.

Technical Reference(s): OP-AA-108-115 (Attach if not previously provided)

Operability Determination & Functionality Assessments

Proposed References to be provided to applicants during examination: none

Learning Objective: State the responsibilities of the following personnel:
SM/CRS (SRO ONLY)
SRO Screener (SRO ONLY)
IAW OP-AA-108-115 and OP-HC-108-115-1001. (As available)

Question Source: Bank #109371

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43 (5)

Comments:

operable, an SSC must be capable of performing the specified safety function / specified function of its design, within the required range of physical conditions, initiation times, and mission times in the CLB. In addition, TS operability considerations require that an SSC meet all surveillance requirements (as specified in Surveillance Requirement (SR) Applicability SR 3.0.1). An SSC that does not meet an SR must be declared inoperable because the LCO operability requirements are not met. For operability determination purposes, the mission time is the duration of SSC operation that is credited in the design basis for the SSC to perform its specified safety function / specified function. A system is expected to be tested and maintained to perform as designed. When an SSC capability is degraded to a point where it cannot perform with reasonable expectation or reliability, the SSC should be judged inoperable, even if at this instantaneous point in time the system could provide the specified safety function / specified function.

2.14 **Operability Determination (OD)**: The process is used to assess operability of SSCs and their support functions for compliance with TSs when a degraded or nonconforming condition is identified for a specific SSC required to be operable by TSs, or when a degraded or nonconforming condition is identified for a necessary and related support function.

2.15 **Operability Declaration**: An operability declaration is a decision by senior licensed operator/SRO on the operating shift crew that there is a reasonable expectation that an SSC can perform its specified safety function / specified function.

3. RESPONSIBILITIES

3.1 All Plant Personnel

3.1.1. All personnel IAW LS-AA-120 have the responsibility to document identified conditions adverse to safe and reliable operation and report these conditions to a management staff member in accordance with the corrective action program. This includes any conditions that could potentially impact the operability of an SSC.

3.2 Operating Shift Crew

3.2.1. The operating shift crew is responsible for overall control of facility operation. As part of that responsibility, the operating shift crew must be aware of the operability and functionality of plant SSCs, and the status of degraded or nonconforming conditions that may affect plant operation.

3.3 Senior Licensed Operator/SRO/Operations Shift Management

3.3.1. A senior licensed operator/SRO on the operating shift crew with the responsibility for plant operations makes the declaration of operability or functionality; i.e., makes the call on whether an SSC described in TSs is operable or inoperable, or an SSC that is not described in the TSs is functional or not functional.

Facility: Hope Creek

Vendor: GE

Exam Date: 2019

Exam Type: SRO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		3
Group #		
K/A #		2.1.29
Importance Rating		4.0

K/A Statement: Knowledge of how to conduct system lineups, such as valves, breakers, switches, etc.

Question: SRO #95

Given:

- While performing a HPCI system lineup check the equipment operator finds that a local Limitorque position indicator for FD-HV-F003, HPCI Outboard Steam Supply Isolation Valve, does not indicate fully open.
- To verify position, he disengages the motor actuator and attempts to open the valve further.
- After manually back seating the valve to ensure that it is fully open, he notes the problem on the system lineup to document the actions taken.
- The valve had not been subsequently operated from the Control room prior to the manual operation.

As the CRS on-shift, what is your evaluation of the Equipment Operator's actions and current valve status?

The actions taken by the Equipment Operator are _______ (1) _________.

The valve ________ (2) __________.

A. (1) in accordance with plant procedures.
(2) should be considered OPERABLE.

B. (1) in accordance with plant procedures.
(2) must be declared INOPERABLE.

C. (1) NOT in accordance with plant procedures.
(2) should be considered OPERABLE.

D. (1) NOT in accordance with plant procedures.
(2) must be declared INOPERABLE.

Proposed Answer: **D**

Explanation (Optional): OP-AA-108-101-1002 Attachment 4 (see attached) Valve Operations- MOVS **When a MOV has been manually seated or backseated to a position that would require it to change position in order to fulfill a safety function, the MOV shall be declared inoperable**. OP-AA-101-111-1003, USE OF PROCEDURES, Attachment 1, Skill of the Craft (Common), the actions taken by the EO are NOT in accordance with plant procedures. The list includes engaging a MOV, **NOT manually back-seating the valve.**

A: **Incorrect -**. the valve must be declared Inoperable

B: **Incorrect –** backseating of MOVs valves requires CRS permission. NOT skill of the craft.

C: **Incorrect** – the valve must be declared Inoperable

D: **Correct** –.See above explanation.

Technical Reference(s):	OP-AA-108-101-1002 Attachment 4 Valve Operations MOVs OP-AA-101-111-1003, USE OF PROCEDURES	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective:	Given a set of conditions, determine when a Motor Operated Valve must be declared Inoperable due to manual operation. IAW OP-AA-108-101-1002.	(As available)

Question Source:	Bank #35451	
	Modified Bank #	(Note changes or attach parent)
	New	

Question History: 2018 NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43 (5)

Comments:

ATTACHMENT 4
VALVE OPERATIONS
Page 4 of 12

4. **MOVS**

> **NOTE**
>
> MOVs are not normally operated by hand. **[CD-128A, CD-716D]**
>
> Manual torquing of MOVs is not allowed when performing Integrated Leak Rate Test (ILRT) lineups. **[CD-747A]**
>
> Excessive force exerted on a MOV handwheel during manual operation may cause serious valve damage. **[CD-716D, CD-040F, CD-193C]**

4.1 When a MOV has been manually seated or backseated to a position that would require it to change position in order to fulfill a safety function, the MOV shall be declared inoperable. **[CD-189B, CD-716D]**

4.2 If excessive torque is used to manually seat or backseat a MOV a NOTF shall be written and a Deficiency Report issued to Engineering. **[CD-716D]**

4.3 After a MOV has been manually seated or backseated, the valve shall be electrically stroked in accordance with the appropriate Tech Spec surveillance test to restore operability (this also ensures the clutch has disengaged). **[CD-663D, CD-128A, CD-147X]**

4.4 Manually operating a MOV off of its electrically closed or open position does not make the valve inoperable, provided, it is electrically operated subsequent to this operation. Document the electrical operation in the Control Room Narrative Log as a retest of the valve and the reason why. **[70023106]**

5. **MOV MANUAL OPERATION [CD-769F]**

5.1 The SJ44 valves at Salem do not have locking type clutches. The breakers for the SJ44 valves should be open prior to manually operating the SJ44s. The declutch lever for the SJ44 will need to be held in position while turning handwheel.

5.2 MOVs may be manually operated with the power supply energized. At the discretion of the CRS, or when directed by procedure, the breaker may be opened prior to manually operating a MOV.

5.3 If the breaker is to be opened for MOV manual operation, and not directed by procedure then the position of this breaker must be tracked for configuration control.

5.4 Declare the MOV inoperable as applicable.

ATTACHMENT 1
SKILL OF THE CRAFT EVOLUTIONS
Page 2 of 3

1.2 **Skill of the Craft Evolutions- Common**

- Adding oil to pump oilers. Proper oil must be verified by approved methods.
- Opening/Closing of equipment doors to support logs, inspection and procedure performance.
- Resetting and testing alarm panels and annunciators.
- Replacing indicating light bulbs with SM/CRS concurrence and ensuring the correct replacement bulb is used.
- Replacing gas cylinders, with the exception of T/S related cylinders, using the precautions listed in the Industrial Safety Manual.
- Manual makeup to surge tanks and expansion tanks when auto makeup is not available.
- Routine adjustment to operating parameters such as level, temperature, flow, pressure, speed and Generator MVARs. Operating values must be maintained within procedural limits.
- Manual sump pump operations.
- Station Air Receiver blowdown.
- Strainer manual blowdown or handle rotation.
- Hooking up and disconnecting hoses, with the exception of CRDs
- Venting and Draining systems (pumps, seals, tanks, coolers, sight glasses and heat exchangers).
- Re-setting breaker overloads and thermals.
- Changing chart recorder paper.
- Advancing HVAC roll filters.
- Duties being performed under the Auxiliary Boiler License (e.g. cleaning the fire gun).
- Taking oil samples, such as D/G lube oil sumps with the exception of EHC (i.e. one valve basic samples).
- Lining up and/or securing heating steam to components.
- Engaging a MOV manually.
- Filling and/or adjusting level in various tanks.
- Installation of pneumatic gauges.
- Installation of fluid gauges that do not require height adjustment calculations or are being used for non-critical / test applications.

Facility: Hope Creek

Vendor: GE

Exam Date: 2019

Exam Type: SRO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		3
Group #		
K/A #	2.2.5	
Importance Rating		3.2

K/A Statement: Knowledge of the process for making design or operating changes to the facility.

Question: SRO #96

The plant is planning to install a new flow control system for HPCI during the next outage.

In accordance with LS-AA-104, "50.59 Review Process," a 10 CFR 50.59 Evaluation would determine if the new flow control system requires ___ (1) ___.

The administrative process to perform initial acceptance testing of HPCI with the new flow control system is controlled by ___ (2) ___.

A. (1) NRC **approval** prior to implementation
(2) OP-AA-108-110, EVALUATION OF SPECIAL TESTS OR EVOLUTIONS

B. (1) NRC **approval** prior to implementation
(2) OP-AA-103-103, OPERATION OF PLANT EQUIPMENT

C. (1) NRC **notification** prior to implementation
(2) OP-AA-108-110, EVALUATION OF SPECIAL TESTS OR EVOLUTIONS

D. (1) NRC **notification** prior to implementation
(2) OP-AA-103-103, OPERATION OF PLANT EQUIPMENT

Proposed Answer: **A**

Explanation (Optional): IAW LS-AA-104, "This procedure establishes the requirements for preparing, reviewing, approving, and documenting evaluations performed pursuant to the requirements of 10 CFR 50.59, "Changes, tests, and experiments," for determining if a facility or procedure change, test, or experiment requires **NRC approval prior to implementation**." Additionally, IAW OP-AA-108-110, evolutions that require the use of special tests in conjunction with existing procedures may also be classified as special evolutions. Specifically, Attachment 1, "Guidelines for Identifying Special Tests and Evolutions," lists the following as consideration for a special test or evolution:

> ECCS Operability: Any activity that may prevent the actuation of, or ability of the Core Spray system, Low Pressure Coolant Injection system, **High Pressure Coolant Injection System**, or the Automatic Depressurization System from performing their ECCS function.

A: **CORRECT**- See above explanation

B: **INCORRECT**- The purpose of the OP-AA-103-103 procedure is to provide clear policies regarding **who is authorized to manipulate or operate plant equipment**.

C: **INCORRECT**- NRC approval is the purpose, not notification.

D: **INCORRECT**- NRC approval is the purpose, not notification. The purpose of the OP-AA-103-103 procedure is to provide clear policies regarding who is authorized to manipulate or operate plant equipment.

Technical Reference(s):	LS-AA-104 50.59 Review OP-AA-108-110 Evaluation of Special Tests or Evolutions OP-AA-103-103 Operation of Plant Equipment	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective:	Given Access to Control Room References determine if a proposed action requires a License Amendment Given access to Control Room References Determine if an activity meets the criteria for a Special Test or Evolution	(As available)

Question Source:	Bank #125434	
	Modified Bank #	(Note changes or attach parent)
	New	

Question History: 2016 NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43 (3)

Comments

50.59 REVIEW PROCESS

1. **PURPOSE**

1.1. This procedure establishes the requirements for preparing, reviewing, approving, and documenting evaluations performed pursuant to the requirements of 10 CFR 50.59 "Changes, tests, and experiments," for determining if a facility or procedure change, test, or experiment requires NRC approval prior to implementation.

1.2. This procedure establishes requirements for submittal of periodic reports to the NRC describing changes, tests and experiments that were evaluated under the provisions of 10 CFR 50.59.

ATTACHMENT 1
Guidelines for Identifying Special Tests and Evolutions
Page 3 of 5

1.5.3. Activities that cause or could cause potentially significant reactivity changes may also be classified as special evolutions. Examples may include:

1. Maintenance of RCS control valve circuitry.
2. Maintenance of turbine pressure regulator controls and subsequent testing.
3. Placing condensate demineralizers in service following modification that could cause resin intrusion into the reactor.

2. **INSTRUCTION**

2.1 Does this procedure or evolution involve significant risk in either of the following areas?

2.1.1. Is equipment essential to any of the following safety functions required for the existing plant operating condition affected in a potentially adverse way?

1. Reactivity and reactor control: Any activity that may adversely affect a system or equipment that may change reactor power.
2. Reactor Protection System: Any activity that may adversely affect HCU, CRD equipment, SCRAM valves, SCRAM air solenoid valves, RPS, etc…
3. Decay Heat Removal: Any activity that may adversely affect the decay heat removal systems, such as SDC, RHR, RWCU, Fuel Pool Cooling, etc… (This includes system needed during an outage.)
4. ECCS Operability: Any activity that may prevent the actuation of, or ability of the Core Spray system, Low Pressure Coolant Injection system, High Pressure Coolant Injection System, or the Automatic Depressurization System from performing their ECCS function.

OP-AA-103-103
Revision 2
Page 1 of 2
LEVEL 3 – INFORMATION USE

OPERATION OF PLANT EQUIPMENT

1. **PURPOSE**

1.1. The purpose of this procedure is to provide clear policies regarding who is authorized to manipulate or operate plant equipment.

Facility: Hope Creek

Vendor: GE

Exam Date: 2019

Exam Type: SRO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		
	K/A #		2.2.18
	Importance Rating		3.9

K/A Statement: Knowledge of the process for managing maintenance activities during shutdown operations, such as risk assessments, work prioritization, etc.

Question: SRO #97

Which of the following is a responsibility of the SM/CRS IAW MA-AA-716-025, Scaffold Installation, Modification, and Removal Request Process?

A. Ensures prior to startup that there are no scaffolds installed that may adversely affect equipment in Seismic II/I or Safety Related Areas.

B. Evaluates, approves or rejects any scaffold specific deviations requested for scaffold erection.

C. Takes corrective action on deficient scaffolding for removal or modification as required.

D. Enters the appropriate information into the Scaffold Log, including identifying scaffold considered to be a missile hazard.

Proposed Answer: **A**

Explanation (Optional): Prior to Mode 2, the SM/CRS shall:

1. Review the Scaffold Log (EXCEL spreadsheet @ m:\shared\scaffold\site services\scaffold), to ensure there are no scaffolds installed that may adversely effect equipment in Seismic II/I or Safety Related Areas.
2. Notify the Scaffold Coordinator of any scaffold concerns and ensure corrective action is taken prior to commencing reactor operations.

A: Correct-SM/CRS (see attached) Reviews the Scaffold Log and notifies the Work Group Scaffold Coordinator and ensures that corrective actions are taken.

B: Incorrect- Engineering evaluates, approves or rejects deviations in the request for scaffolding

C: Incorrect- Work Group Scaffold Coordinator takes the corrective actions on scaffolding deficiencies.

D: Incorrect- Maintenance Supervisor Makes entries into Scaffold Log

Technical Reference(s): MA-AA-716-025 (Attach if not previously provided)

Scaffold Installation, Modification, and Removal Request Process

Proposed References to be provided to applicants during examination: none

Learning Objective: (SRO Only) From Memory state the SM/CRS responsibilities for review of the Scaffold Control Log before entering Operational Condition 2 IAW MA-AA-716-025 (As available)

Question Source: Bank #

Modified Bank # 34051 (Modifies distractors to delineate specific roles and have components in them that are plausible for a CRS)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43 (5)

Comments: Original Question:

Which of the following is a responsibility of the SM/CRS per IAW MA-AA-716-025, Hope Creek Scaffold Program?

A. Ensures prior to startup there are no scaffolds installed that may adversely effect equipment in Seismic II/I or safety-related areas.

B. Serves as a single point of contact for resolving Scaffold Program issues.

C. Ensures required scaffold inspections are performed prior to the use of scaffolds.

D. Tracks and updates the status of scaffolds.

MA-AA-716-025	Page 20 of 28
SCAFFOLD INSTALLATION, MODIFICATION, AND REMOVAL REQUEST PROCESS	Rev: 14

4.6 Pre-Startup Review

4.6.1. Prior to Mode ascension, the SM/CRS shall:

1. **REVIEW** the Scaffold Log to ensure there are no scaffolds installed that may adversely affect equipment in Seismic II/I or Safety Related Areas.

2. **NOTIFY** the Work Group Scaffold Coordinator of any scaffold concerns and ensure corrective action is taken prior to commencing reactor operations.

4.6.2. Work Group Scaffold Coordinator shall take corrective action on deficient scaffolding to remove or modify the scaffolding as required and/or when notified by the SM/CRS. **[CD-452Y]**

3.8 Nuclear Maintenance Supervisor/Designee

3.8.1. Responsible for maintaining Scaffold Log and ensuring the appropriate information is entered into it, including identifying scaffold considered to be a missile hazard. Notifying the Work Group Scaffold Coordinator when all work is complete and the scaffold is no longer required. **[C0391]**

3.3 Engineering

3.3.1. Responsible for evaluating, approving or rejecting any scaffold specific deviations requested for scaffold erection in accordance with Attachments 1 and/or 2.

3.3.2. Evaluation of any specific deviations resulting from inspections of Permanent Scaffold in Seismic II/I and Safety Related areas.

3.3.3. Technical review of scaffold activities covered by this procedure.

Facility: Hope Creek

Vendor: GE

Exam Date: 2019

Exam Type: SRO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		3
Group #		
K/A #		2.3.11
Importance Rating		4.3

K/A Statement: Ability to control radiation releases.

Question: SRO #98

Given:

- The reactor was manually scrammed.
- Multiple control rods are stuck full out.
- Main Steam Line Radiation levels reached 3xNFPB and the MSIVs were closed.
- When HPCI was initiated, a steam line break in the HPCI room occurred.
- All efforts to close the HPCI steam line isolation valves have failed.

Current plant conditions:

- Reactor Power is at 3%.
- Reactor water level is at -58" and being maintained with RCIC.
- Reactor pressure 800 psig and steady.
- SLC is injecting and SLC tank level is at 3000 gallons.
- HPCI Room Temperature is at 260°F and rising.
- RCIC Room Temperature is at 110°F and rising.
- All other Reactor Building Room Temperatures are at 90°F and steady.
- Offsite Release Rate on SPDS is at 9.2E+08 μCi/sec.
- No Dose Assessment is available at this time.
- There is visible confirmation of steam coming from the Reactor Building Blowout Panels.

What is (are) the required operator action(s)?

A. Depressurize the RPV and maintain cooldown rate below 100°F/hr.

B. Immediately perform a Rapid RPV Depressurization ONLY.

C. When a second Max Normal Op Temperature is exceeded, then perform an Emergency RPV Depressurization.

D. Immediately perform an Emergency RPV Depressurization.

Proposed Answer: D

Explanation (Optional): See attached EOP-103/4 BASES

A: **Incorrect-**.. This could be correct if there was no Rad release (EOP-103) and SLC tank level was <1100 gal.

B: **Incorrect -**. This would be correct IF ONLY HPCI/RCIC were available for injection. RCIC and **SLC are currently injecting**, and there is no indications that other pumps are unavailable.

C: **Incorrect-**.. This could be correct if there was no Rad release (EOP-103).

D: **Correct-**. Given Offsite Release Rate on SPDS is 9.2E+08 µCi/sec, you are above the 5.25E+08 µCi/sec limit in RG 1.1 and answer **YES** to step RR-6. You are given conditions to answer **YES** to step RR-7. You have SLC pumps injecting, and have to answer **NO** to step RR-12. The required operator actions are steps RR-8 and RR-9 (ED and place FW Sealing in service). (See attached EOP-104 FC).

Technical Reference(s): EOP-103/4 FC (Attach if not previously provided)

EOP103/4 BASES

EP-HC-325-103
Offsite Rad Conditons

Proposed References to be provided to applicants during examination: EALs and RALs (No EAL attachments)

Learning Objective: Given any step in the procedure, describe the reason for performance of that step and/or expected system response to control manipulations prescribed by the step. (As available)

Question Source: Bank #153571

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43 (4,5)

Comments:

10.5 RR-6 Can Rad Release be maintained below General Emergency Level

RR-7 Is Primary System discharging outside Containment and Rx Bldg

RR-8 Emergency RPV Depressurization is required, enter EOP-101 and execute it concurrently with the following steps.

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.

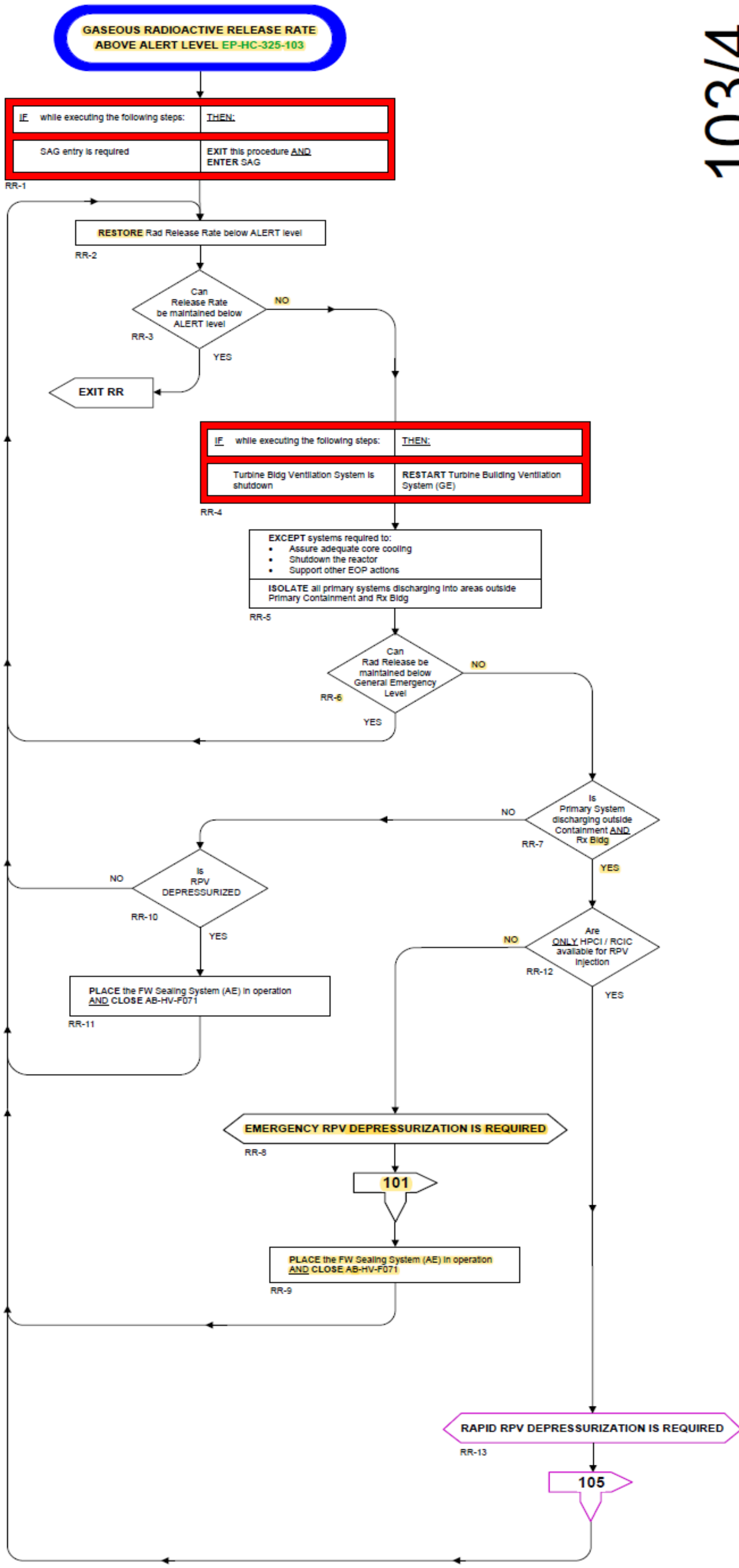
10.7 RR-9 PLACE the FW Sealing System (AE) in operation and close AB-HV-F071

RR-10 Is the RPV depressurized

RR-11 PLACE the FW Sealing System (AE) in operation and close AB-HV-F071

If the RPV is depressurized, the FW Sealing System is placed in service and MSL drain valve AB-HC-F071 is closed to limit radioactive release through leaking MSIVs/FW system.

103/4


GASEOUS RADIOACTIVE RELEASE RATE ABOVE ALERT LEVEL EP-HC-325-103
IF while executing the following steps: | THEN:
SAG entry is required | EXIT this procedure AND ENTER SAG
RR-1
RESTORE Rad Release Rate below ALERT level
RR-2
Can Release Rate be maintained below ALERT level
RR-3
NO
YES
EXIT RR
IF while executing the following steps: | THEN:
Turbine Bldg Ventilation System is shutdown | RESTART Turbine Building Ventilation System (GE)
RR-4
EXCEPT systems required to:
Assure adequate core cooling
Shutdown the reactor
Support other EOP actions
ISOLATE all primary systems discharging into areas outside Primary Containment and Rx Bldg
RR-5
Can Rad Release be maintained below General Emergency Level
RR-6
NO
YES
Is Primary System discharging outside Containment AND Rx Bldg
RR-7
NO
YES
Is RPV DEPRESSURIZED
RR-10
NO
YES
PLACE the FW Sealing System (AE) in operation AND CLOSE AB-HV-F071
RR-11
Are ONLY HPCI / RCIC available for RPV injection
RR-12
NO
YES
EMERGENCY RPV DEPRESSURIZATION IS REQUIRED
RR-8
101
PLACE the FW Sealing System (AE) in operation AND CLOSE AB-HV-F071
RR-9
RAPID RPV DEPRESSURIZATION IS REQUIRED
RR-13
105

Section R - Abnormal Rad Levels / Rad Effluent
R1 – Offsite Rad Conditions

Initiating Condition: Release of gaseous radioactivity resulting in offsite dose greater than 1,000 mrem TEDE or 5,000 mrem thyroid CDE

OPCON: All

EAL # — EMERGENCY ACTION LEVELS

RG1.1 — Release Rate

IF Plant Vent Effluent Offsite Gas Rad Release Rate reading of:

> 5.25E+08 µCi/sec

(SPDS Point B5097 **OR** the **SUM** of below listed Effluent Monitors)

- FRVS Vent NG (9RX680)
- NPV NG (9RX590)
- SPV NG (9RX580)
- HTV NG (9RX518)

(Notes 3, 4)

AND Dose Assessment results are NOT available

AND **≥ 15 minutes** have elapsed (Notes 1, 2)

RG1.2 — Dose Assessment

IF Dose assessment using actual meteorology indicates doses **> 1000 mrem TEDE** at or beyond the **MINIMUM EXCLUSION AREA** (Note 4)

RG1.2

IF Dose assessment using actual meteorology indicates doses **> 5000 mrem thyroid CDE** at or beyond the **MINIMUM EXCLUSION AREA** (Note 4)

RG1.3 — Field Measured Dose Rate

IF Field survey results indicate **EITHER** of the following at or beyond the **PROTECTED AREA** boundary:

- Closed window dose rates **> 1000 mR/hr** expected to continue for **≥ 60 min.**
- Analyses of field survey samples indicate I-131 concentration **> 3.85E-06 µCi/cc**

(Notes 1, 2)

THEN

Action Required: Refer to Attachment 4 **GENERAL EMERGENCY**

Note 1: The Emergency Coordinator should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded.

Note 2: If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded the specified time limit.

Note 3: If the effluent flow past an effluent monitor is known to have stopped, indicating that the release path is isolated, the effluent monitor reading is **NO** longer **VALID** for classification purposes.

Note 4: The pre-calculated effluent monitor values presented in EALs RA1.1, RS1.1 and RG1.1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

Facility: Hope Creek

Vendor: GE

Exam Date: 2019

Exam Type: SRO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		3
Group #		
K/A #		2.4.38
Importance Rating		4.4

K/A Statement: Ability to take actions called for in the facility emergency plan, including supporting or acting as emergency coordinator if required.

Question: SRO #99

Given:

- An event has occurred at the plant.
- The TSC (Technical Support Center) and EOF (Emergency Operations Facility) are manned but NOT activated.

IAW NC.EP-EP.ZZ-0102 “Emergency Coordinator Response”, which one of the following describes the individual responsible for escalating an emergency classification level from a SAE (Site Area Emergency) to a GE (General Emergency)?

A. The Emergency Duty Officer.

B. The Shift Manager.

C. The Emergency Response Manager.

D. The Site Vice President.

Proposed Answer: B

Explanation (Optional): The shift manager is the emergency coordinator (any escalation in emergency level) until the TSC is activated then the SM will turnover EC duties to the Emergency Duty Officer in the TSC and when the EOF is activated then the responsibility goes to the Emergency Response Manager. (see attached NC.EP-EP.ZZ-0102)

A: **Incorrect-** Correct if the TSC was activated and the EOF manned.

B: **Correct-** The SM is the Emergency Coordinator until the TSC is ACTIVATED. Until then the SM (as the Emergency Coordinator) is responsible for escalating an event.

C: **Incorrect-** Correct if the EOF was activated

D: **Incorrect-** Site VP is not a designated emergency coordinator.

Technical Reference(s): EP-EP.ZZ-0102 EC Response (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: ECG/E-Plan/Fire & Medical Questions (As available)

Question Source: Bank #116025

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43 (5)

Comments:

5.1 Shift Manager (SM) Actions:

NOTES
1. Should the Emergency Classification Level change, the SM shall implement actions based on the new Emergency Classification Level. 2. After EC turnover to the EDO, the SM shall continue to implement the SM designated steps in Section 5.1. 3. Since the Rad Alert alarm is located in the Control Room, the SM directs all emergency status change announcements.

Note: Initials/time each block as applicable

5.1.a	SM Emergency Actions – Initial	UE	A	SAE	GE
SM	IF EITHER one of the following has or is occurring: • Security Based Emergency • Catastrophic Event – large explosion, airliner impact, etc. resulting in a loss of command and control at the other station THEN **IMPLEMENT** Attachment 10 - Security Emergency Guideline (SEG), of this procedure	**Att 10** Init:____ Time:____	**Att 10** Init:____ Time:____	**Att 10** Init:____ Time:____	**Att 10** Init:____ Time:____
SM	For each change in Emergency Classification Level, **DIRECT / MAKE** emergency status announcements AND Security notifications as follows: • **IMPLEMENT** Attachment 4 for Unusual Event • **IMPLEMENT** Attachment 5 for ALERT or Higher (will be requested by EDO if TSC is activated) *For Security based emergency, SEG may require modifications of associated actions/announcements.*	**Att 4** Init:____ Time:____	**Att 5A** Init:____ Time:___	**Att 5B** Init:____ Time:___	**Att 5C** Init:____ Time:___

SM	Prior to TSC activation, IF desired or WHEN required, THEN **DIRECT IMPLEMENTATION** / **IMPLEMENT** Accountability IAW Attachment 3 of this procedure. *For Security based emergency, SEG may require modifications of associated actions/announcements.*	**Optional**	**Optional**	Init:____ Time:____	Init:____ Time:____

Note: Initials/time each block as applicable

5.1.b	**SM Subsequent Actions** (cont'd)	UE	A	SAE	GE
SM	Until relieved by the EDO, IF Remote Response Center (RRC) Guideline, SH.OP-AM.TSC-0002, REMOTE RESPONSE CENTER (RRC) OPERATIONS was implemented per directions in Attachment 10, THEN: **[CM-HC.2007-42]** • **PROVIDE** overall direction (command & control) and support to the RRC Leader (NETS X5135 or DID X2801). • **PROVIDE** periodic briefings to the RRC Leader. • **TURNOVER** command & control of the affected station's emergency response to a qualified SM or EDO from the affected station when the affected station has adequate resources.	Init:____ Time:____	Init:____ Time:____	Init:____ Time:____	Init:____ Time:____
SM	Until relieved by the EDO, as needed/requested for accident mitigation (25R) or life saving (75R), **AUTHORIZE** Emergency Dose. *(per OSC RP Response, NC.EP-EP.ZZ-0304)* • **ENSURE** RAC works with the OSC and considers dose savings to the public to expedite OSC team dispatch. • **ENSURE** backup actions are being planned; (OSC or TSC engineering staff). • **ENSURE** OSC Coordinator considers pre-staging teams at the control point while plans are being finalized.	Init:____ Time:____	Init:____ Time:____	Init:____ Time:____	Init:____ Time:____

Facility: Hope Creek

Vendor: GE

Exam Date: 2019

Exam Type: SRO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		3
Group #		
K/A #	2.4.25	
Importance Rating		3.7

K/A Statement: Knowledge of fire protection procedures.

Question: SRO #100

During a fire in the Turbine Building, a Fire Brigade Liaison is assigned by the Shift Manager.

Who, by title, can be assigned this role?

A. MCR/Plant Operator (PO).

B. Communicator #2 (CM2).

C. Shift Technical Advisor (STA).

D. Work Control Supervisor (WCS).

Proposed Answer: D

Explanation (Optional): **OP-AA-101-111** Roles and Responsibilities of on Shift Personnel-The SM should designate an appropriate NEO, RWEO, RO/PO or SRO (**cannot be concurrently assigned to fill the position of SM, CRS, STA, NCO(2), CM1, CM2 or OSCC** in Attachment 1 table) to function as the station Fire Brigade Liaison. They should also function as liaison for other emergencies (ex. chemical spill, toxic gas, environmental, etc.). **The purpose of the liaison is to provide real-time communication to the Control Room regarding the status of the event.**
IAW HC.FP-EO.ZZ-0001 Fire and Medical Emergency Response Attachment 1 The Fire Brigade Liaison:

- Enhances Fire Brigade and Control Room coordination.
- **Makes recommendations to the SM regarding what equipment needs to be removed from service to mitigate the fire and/or help stabilize the plant.**
- **Assist Fire Department in providing the Initial Status Report**
- Assist the Incident Commander (IC) in keeping Control Room advised of firefighting progress:
- **Perform actions as directed by Control Room such as removing affected equipment from service, aligning ventilation, etc.**
- Review pre-fire plan for affected fire area to assist in performing listed duties. The IC will have a copy of the pre fire plan at the fire scene

A: **Incorrect-** Assigned to the MCR. (see attached OP-AA-101-111 Attachment 1Note 3)

B: **Incorrect-** Assigned to the MCR for EP duty.(see attached OP-AA-101-111 Attachment 1Note 3)

C: **Incorrect** Assigned as STA.(see attached OP-AA-101-111 Attachment 1Note 3)

D: **Correct-** can be assigned the Fire Brigade Liaison and will be providing real time status of the event.

Technical Reference(s): OP-AA-101-111 Roles and Responsibilities of on shift personnel (Attach if not previously provided)
HC.FP-EO.ZZ-0001

Fire and Medical Emergency Response

Proposed References to be provided to applicants during examination: none

Learning Objective: Given plant conditions and/or access to control room references Determine the following: (As available)
a. The level of licensing required for the SM, CRS, and RO/PO.
b. Minimum shift manning requirements for all plant conditions.
c. Normal shift staffing levels.
d. When a person can serve a dual role as CRS/STA or SM/STA. IAW OP-AA-101-111

Question Source: Bank #163436

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43 (5)

Comments:

Attachment 1, Hope Creek Shift Complement

[CD 419Y, T/S 6.2.2]

	Position Title	**License Required**	**Normal Complement** (Note 5)	**Minimum Required Complement**		
				Emergency Plan Implementation	**T/S**	
					Cond. 1, 2, 3	**Cond. 4, 5**
CD-669X	SM*	SRO	1 (Note 1)	1	1	1
CD-669X	CRS	SRO	1 (Note 1)	1	1	0
(OP	STA	(Note 2)	1 (Note 1)	1	1	0
COND	RO/PO**	RO	2	2	2	1
1, 2 & 3)	NEO	None	4	2 (Note 6)	2	1
CD-121Y	RWEO	None	1	1	0	0
	OSCC	None	1 (Note 4)	1 (Note 4)	0	0
CD-252X	SHIFT ELECT	None	1 (each)	1 (each)	0	0
CD-252X	R.P. SUPPORT	None	2	2	1	1
CD-252X	I&C TECH	None	1 (each)	1 (each)	0	0
CD-252X	CHEM. TECH	None	1	1	0	0
	COMMUNICATOR	None	2	2 (Note 7)	0	0
	FIRE BRIGADE LIAISON (FBL)	None	1 (Note 3)	1 (Note 3)	1 (Note 3)	1 (Note 3)
	FIRE BRIGADE	IAW FP-AA-012 (Note 3)				
	DRO & TO***	Shared with Salem Station – see Note*** below				

* Shift Manager (SM) has the Control Room Command Authority which may be delegated to a CRS (in any condition) or, to an RO (in Condition 4 or 5) in the event the Shift Manager is absent from the Control Room. Delegation of Control Room Command Authority to an RO (in Condition 4 or 5) shall be recorded in the RO log.

** The assigned RO(s) are required to remain in the Control Room when on shift as necessary to meet the T/S Minimum Required Complement. One RO may be relieved, from time to time, by an SRO who may be the SRO filling the Control Room SRO and/or Control Room Command Authority.

*** OSC on-shift Stock Handlers normally fulfill the functions of Debris Removal Operator (DRO) and Towing Operator (TO) and are accounted for in the Salem Shift Compliment of Attachment 2 and Attachment 18 of OP-SA-112-101-1001. The DRO and TO positions support both Salem and Hope Creek Stations.

Attachment 1, Hope Creek Shift Complement (continued)

1. An individual can serve a dual role capacity as the CRS/STA or the SM/STA on shift provided the individual is SRO licensed, STA qualified and satisfies one of the following educational alternatives:

   1) A bachelor's degree in engineering from an accredited institution,

   2) A Professional Engineer's license,

   3) A bachelor's degree in engineering technology or physical science from an accredited institution including work in the physical, mathematical or engineering science. **[CD 669X, CD 117Y, CD 176A]**

2. The STA can be SRO licensed; however, this is not an NRC requirement. The STA shall have a bachelor's degree or equivalent in a scientific or engineering discipline and have specific training in the response and analysis of the plant for transients and accidents and in the unit design and layout including the capabilities of instrumentation and controls in the Control Room. **[CD 117Y, CD 669X, CD 176A]**

3. The SM should designate an appropriate NEO, RWEO, RO/PO or SRO (cannot be concurrently assigned to fill the position of SM, CRS, STA, NCO(2), CM1, CM2 or OSCC in Attachment 1 table) to function as the station Fire Brigade Liaison. They should also function as liaison for other emergencies (ex. chemical spill, toxic gas, environmental, etc.). The purpose of the liaison is to provide real-time communication to the Control Room regarding the status of the event.
   [80087166]

4. The Shift Maintenance Supervisor or an Operations Supervisor will be the initial OSC Coordinator until relieved by the Duty OSC Coordinator. This shall not be the same individual fulfilling the shift STA requirement or Fire Brigade Liaison position. If the STA is the scheduled WCC Supervisor, turnover of WCC Supervisor duties to the non-STA CRS may be required.

5. Except for the SM, the Hope Creek Shift Complement may be one less than the minimum requirements for a period not to exceed 2 hours to accommodate the unexpected absence of on duty shift complement members provided that immediate action is taken to restore the Shift Complement Composition to within the minimum requirements of the tables. This provision does not permit any shift complement position to be unmanned upon shift change due to an oncoming shift member being late or absent. The preferred action is to hold over the off-going shift personnel until a relief is available. **[CD 252X]**

6. One NEO, that is Self-Monitoring Qualified, is credited as Radiation Protection Support for the Emergency Plan (NUREG 0654, Table B 1 position **E3). **[CD 252X]**

7. The ERO positions of Primary and Secondary Communicator should be filled by ERO communicator qualified licensed and non-licensed operators, or operations staff personnel not assigned to any other minimum shift complement position. **[CD 162F]**

ATTACHMENT 1 – FIRE BRIGADE LIAISON DUTIES

NOTE:

The SM **shall RE-ASSIGN** the Fire Brigade Liaison as needed to support the needs of the reactor and plant during an emergency.

Fire Brigade Liaison does not have firefighting duties and does not require firefighter protective clothing or SCBA.

As applicable, the Fire Brigade Liaison ~~(FDL)~~ performs the following duties:

~~1.1.1.~~
- Enhances Fire Brigade and Control Room coordination.
- Makes recommendations to the SM regarding what equipment needs to be removed from service to mitigate the fire and/or help stabilize the plant.
- **ASSIST** Fire Department in providing the Initial Status Report:
  - Location, type and size of fire.
  - Equipment involved or potentially affected.
  - Any personnel injured.
  - Status of suppression systems.
  - Any assistance required
- **ASSIST** the Incident Commander (IC) in keeping Control Room advised of firefighting progress:
  - Priorities for protection of equipment are addressed.
  - Electrical equipment isolation is safe for fire-fighting.
  - Water supplies to containment and reactor buildings are coordinated as necessary.
  - Damage assessment and recovery efforts are coordinated with Fire Department activities.
- **PERFORM** actions as directed by Control Room such as removing affected equipment from service, aligning ventilation, etc.
- **REVIEW** pre-fire plan for affected fire area to assist in performing listed duties. The IC will have a copy of the pre fire plan at the fire scene