1. RO 203000A1.01 001/C/A/T2G1/RHR/DWSP/1/203000A1.01/4.2/4.3/RO/MODIFIED 11/17/07

Given the following conditions:

- Unit 2 has experienced a Loss of Coolant Accident (LOCA).
- Drywell sprays are required in accordance with 2-EOI-2 flowchart.

Which ONE of the following plant conditions must exist prior to opening BOTH the Residual Heat Removal (RHR) SYS I Inboard AND Outboard Drywell Spray Valves?

A. RPV level must be greater than (-)155 inches (Emergency Range) with ONLY the the CONT SPRAY VLV SEL SWITCH in SELECT.

B. RPV level must be greater than (-)162 inches (Post Accident Range) with ONLY the CONT SPRAY VLV SEL SWITCH in SELECT.

C.✓ RPV level is greater than (-)183 inches (Post Accident Range) with ONLY the CONT SPRAY VLV SEL SWITCH in SELECT.

D. RPV level is less than (-)200 inches (Post Accident Range) with ONLY the 2/3 CORE HEIGHT KEYLOCK BYPASS SWITCH in BYPASS.

K/A Statement:

203000 RHR/LPCI: Injection Mode
A1.01 - Ability to predict and/or monitor changes in parameters associated with operating the RHR/LPCI: INJECTION MODE (PLANT SPECIFIC) controls including: Reactor water level

K/A Justification: This question satisfies the K/A statement by requiring the candidate to use specific values of reactor water level to determine the conditions which allow diverting RHR from a LPCI Injection lineup to containment control.

Level of Knowledge Justification: This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

References: 2-OI-74, OPL171.044, TP-33

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modified from OPL171.044 #22
REFERENCE PROVIDED: None

Plausibility Analysis:

In order to answer this question correctly the candidate must determine the following:

1. Drywell sprays being required infers that DW pressure is >2.45 psig.
2. Based on the RPV level conditions given in the available answers, determine whether a CAS signal has been generated due to the LOCA.
3. Which switch(s) must be manipulated to override a CAS signal with the existing conditions.

NOTE: All of these answers are plausible based on minimal procedural guidance given in EOI Appendix 17B. Experience has shown that both switches are manipulated by novice operators regardless of conditions to facilitate Drywell sprays as required. This is not a procedure violation, but demonstrates a lack of specific knowledge of required conditions.

C - correct. OPL 171.044, TP 33 shows that under the conditions only the CONT SPRAY VLV SEL SWITCH in SELECT is required to allow spraying containment. Contact K14A/B is closed if level is above 2/3's core height (-183") as referenced on the Post Accident range (-268" to +60"). Contact K61A/B is closed due to an accident signal (level <-122" on the Wide range level instruments). Therefore to complete the logic only the SELECT switch is required to be closed which energizes relays K58A/B (Seal In) and K50A/B which allows spraying containment.

A - incorrect. With RPV level > -155 inches (-155" bottom of scale on the emergency range is operationally considered top of active fuel) adequate core cooling is assured by submergence and therefore spraying containment is allowed. If candidate determines that top of active fuel verses 2/3 core height is necessary, it becomes plausible. Also level must be above -183" not -155"

B - incorrect. With RPV level > -162 inches (top of active fuel) adequate core cooling is assured. If candidate determines that top of active fuel versus 2/3's core height is necessary, it becomes plausible. Also level must be above -183" not -162"

D - incorrect. With RPV level < -200 inches, you are below 2/3's core height which requires both the CONT SPRAY VLV SEL SWITCH in SELECT and 2/3 CORE HEIGHT KEYLOCK BYPASS switch in BYPASS.

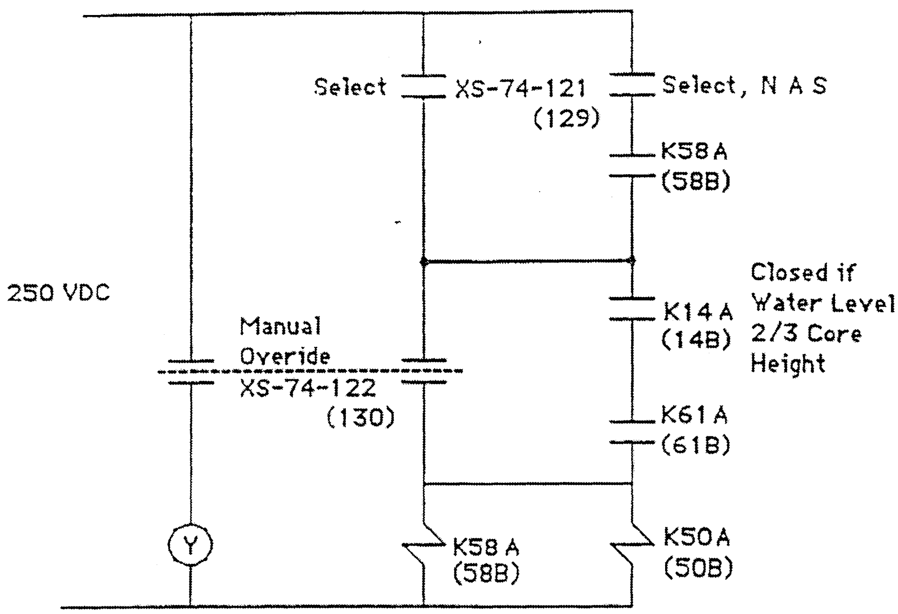
Amplification: In regard to specific comments/questions,concerning valve numbers versus noun names at BFN, this poses a problem for systems with multiple loops such as RHR, Core Spray and Recirc. TVA does NOT use conventional GE Master Parts List (MPL) numbers like most BWR plants. The same valve NAME on different loops will have a different NUMBER. In this question it doesn't matter which "loop" we are referencing since the interlock in question is the same. Therefore, using the noun name is appropriate without the number. In other cases where there is only one valve by that NAME, using the number AND the name will reduce confusion.

Regarding RPV level instrumentation, all GE BWRs have multiple ranges of instrumentation to cover the large height of the reactor vessel and maintain accuracy under various operating conditions within the vessel. This design feature, unique to BWRs, creates a significant challenge to our operators. All of the various ranges of RPV level instrumentation have control functions that input to various systems such as RPS, PCIS, LOCA logic and ECCS initiation logic. As such, providing the candidate with level indications, with or without the corresponding name of the range in question, is all part of determining whether that candidate possesses the required knowledge of RPV level instrumentation and its effect on plant operation.

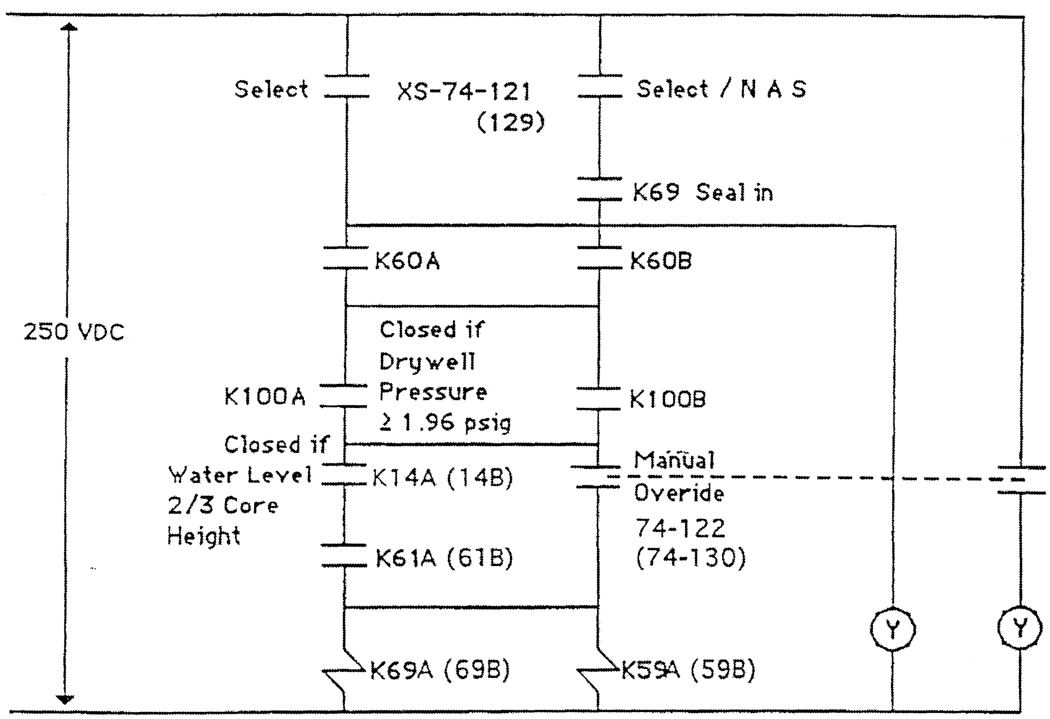
Browns Ferry has the following level ranges:

Narrow range (Normal Control range): 0" to +60"
Wide range (Emergency Systems range): -155" to +60"
Shudown Vessel Floodup (Floodup range): 0" to +400"
Post Accident Flood range (Post Accident range): -268" to +32"

The 2/3 core height logic is developed using the Post Accident Flood range instrumentation, whereas the accident signal logic is developed using the Wide range instrumentation.



Aux Relay Select Logics to Valves 74-59 (73) and 74-57 (71)



Aux Relay Select Logics for Containment/Torus Spray Valves

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modified from OPL171.044 #22

3.5 INTERLOCKS (continued)

7. The RHR spray/cooling valves, 2-FCV-74-57(71), receive an auto closure signal in the presence of a LPCI initiation signal and they are interlocked to prevent opening if the in-line torus spray valve, 2-FCV-74-58(72), is not fully closed. The in-line valve interlock can be by-passed if the following conditions exist.

   (1) Reactor level is >2/3 core height and a LPCI initiation signal is present and the select reset switch is in the SELECT position.

   The requirements for >2/3 core height and a LPCI initiation signal may be by-passed using the keylock bypass switch, 2-XS-74-122/30.

8. If primary containment cooling is desired with reactor level at <2/3 core height, the keylock bypass switch is required to be placed in BYPASS <u>before</u> the select reset switch is placed in SELECT to ensure relay logic is made up.

9. The RHR torus spray valves, 2-FCV-74-58(72), have the same in-line valve interlocks as those outlined in Step 3.5A.8 for the torus spray/cooling valves. Additionally these valves have an interlock preventing opening unless drywell pressure is ≥1.96 psig which cannot be bypassed.

10. The RHR torus cooling/test valves, 2-FCV-74-59(73), receive an auto closure signal in the presence of a LPCI initiation signal. Auto closure may be bypassed by the same conditions/actions outlined in Step 3.5A.8.

11. The RHR containment spray valves, 2-FCV-74-60(74) and 61(75), have in-line valve interlocks similar to these described in Step 3.5A.8 through 3.5A.10 for the RHR torus spray valves 2-FCV-74-57(58) and 71(72).

12. If 2-FCV-74-59(73) LOCA CLOSURE TIME light (2-IL-74-59Y Loop I; 2-IL-74-73Y Loop II) on Panel 2-9-3 is extinguished due to its associated valve being opened, that Loop is inoperable for LPCI.

13. If 2-HS-74-148(149) RHR SYSTEM I (II) MIN FLOW INHIBIT switch is in the INHIBIT position, the pumps on that loop do not have automatic minimum flow protection.

6. **INITIATE** Drywell Sprays as follows:

   a. **VERIFY** at least one RHRSW pump supplying each EECW header. _____

   b. IF.....EITHER of the following exists:

      - LPCI Initiation signal is NOT present,

        OR

      - Directed by SRO,

      THEN...**PLACE** keylock switch 2-XS-74-122(130), RHR SYS I(II) LPCI 2/3 CORE HEIGHT OVRD, in MANUAL OVERRIDE. _____

   c. **MOMENTARILY PLACE** 2-XS-74-121(129), RHR SYS I(II) CTMT SPRAY/CLG VLV SELECT, switch in SELECT. _____

   d. IF.....2-FCV-74-53(67), RHR SYS I(II) LPCI INBD INJECT VALVE, is OPEN,
      THEN...**VERIFY CLOSED** 2-FCV-74-52(66), RHR SYS I(II) LPCI OUTBD INJECT VALVE. _____

   e. **VERIFY OPERATING** the desired System I(II) RHR pump(s) for Drywell Spray. _____

   f. **OPEN** the following valves:

      - 2-FCV-74-60(74), RHR SYS I(II) DW SPRAY OUTBD VLV _____
      - 2-FCV-74-61(75), RHR SYS I(II) DW SPRAY INBD VLV. _____

   g. **VERIFY CLOSED** 2-FCV-74-7(30), RHR SYSTEM I(II) MIN FLOW VALVE. _____

   h. IF.....Additional Drywell Spray flow is necessary,
      THEN...**PLACE** the second System I(II) RHR Pump in service. _____

   i. **MONITOR** RHR Pump NPSH using Attachment 2. _____

   j. **VERIFY** RHRSW pump supplying desired RHR Heat Exchanger(s). _____

2. RO 205000K4.02 001/MEM/T2/G1/RHR//205000K4.02/3.7/3.8/RO/BANK 11/27/07 RMS

Given the following conditions on Unit 2:

- Reactor level (+)20 inches
- Reactor pressure 90 psig
- Drywell pressure 1.7 psig

Which ONE of the following describes which modes of Residual Heat Removal (RHR) are available for use (consider interlocks only)?

A. Low Pressure Coolant Injection (LPCI), Drywell Sprays, and Shutdown Cooling.

B. Suppression Pool Cooling, Suppression Pool Sprays, and Shutdown Cooling.

C.✔ Suppression Pool Cooling, Low Pressure Coolant Injection, and Shutdown Cooling.

D. Low Pressure Coolant Injection (LPCI), Drywell Sprays, and Supplemental Fuel Pool Cooling.

K/A Statement:

205000 Shutdown Cooling
K4.02 - Knowledge of SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE) design feature(s) and/or interlocks which provide for the following: High pressure isolation: Plant-Specific

K/A Justification: This question satisfies the K/A statement by requiring the candidate to use specific plant conditions to determine which interlocks apply to those conditions including the High Pressure isolation of RHR SDC mode.

References: OPL171.044 rev 15, pg 31, C - Instrumentation and Interlocks, 2-OI-74-rev 133, step 8.8[11]

Level of Knowledge Justification: This question is rated as MEM due to the requirement to recall or recognize discrete bits of information.

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REFERENCE PROVIDED: None

Plausibility Analysis:

In order to answer this question correctly the candidate must determine the following:

1. Drywell pressure <1.96 psig prohibits use of containment sprays (DW and SP).
2. RPV pressure <450 psig allows the use of LPCI Injection.
3. RPV pressure of 90 psig allows the use of Shutdown Cooling.

NOTE: All given answers are plausible since they all contain at least one acceptable lineup with the given conditions.

C - correct. The lack of an accident signal (an accident signal is either +2.45 psig High Drywell pressure and <450 psig reactor pressure or -122" low reactor water level) allows Suppression Pool Cooling. Reactor pressure <450 psig allows Low Pressure Coolant Injection. Reactor pressure at 90 psig is below the high pressure isolation setpoint of 100 psig for Shutdown Cooing and Drywell pressure is below the PCIS group II isolation setpoint of +2.45 psig; therefore, Shutdown Cooling is available.

A - incorrect. LPCI and Shutdown Cooling would be allowed by interlocks. Drywell Sprays will not function <1.96 psig in the Drywell.

B - incorrect. Suppression Pool and Shutdown Cooling would be allowed by interlocks. Suppression Pool Sprays will not function <1.96 psig in the Drywell.

D - incorrect. LPCI and Supplemental Fuel Pool Cooling would be allowed by interlocks. Drywell sprays will not function, as previously stated.

Amplification: In responding to your specific questions, concerning "Consider interlocks only", the reason for that statement is that the given pressure is procedurally too high to place RHR in Shutdown Cooling, but the interlocks will allow it. Per OI-74, Section 8.8, "**VERIFY** Reactor pressure is less than 55 psig, OR if entering this procedure from RC/P of 2-EOI-1, pressure is less than 100 psig."

In question #1, the allowance or preclusion of Drywell Sprays are related to reactor vessel water level. In this question, it is a Drywell pressure setpoint, versus RPV water level, which is associated with answering the question correctly. If the candidate only assumes that the + 20 inches RPV water level will allow sprays, then the candidate will answer the question incorrectly.

Additionally, a specific instrument is not required to answer this question, since the level that is given is not above or below any initiation or trip setpoint.

In order to properly comply with the K/A statement, the candidate's knowledge of the high pressure INTERLOCK associated with RHR Shutdown Cooling must be evaluated, not the procedural limitations. A pressure above the procedure limit but BELOW the interlock for was given for that purpose.

8.8 Initiation/Operation of Loop I(II) Shutdown Cooling (continued)

[10.2] **IF** RECIRC PUMP 2B(2A) SUCTION VALVE, 2-FCV-68-77(1) is to be closed, **THEN**

PERFORM the following:

[10.2.1] **VERIFY** TRIPPED, RECIRC DRIVE 2B(2A) NORMAL FEEDER, 2-HS-57-14(17). ☐

[10.2.2] **VERIFY** TRIPPED, RECIRC DRIVE 2B(2A) ALTERNATE FEEDER, 2-HS-57-12(15). ☐

[10.2.3] **CLOSE** tripped recirc pump suction valve using RECIRC PUMP 2B(2A) SUCTION VALVE, 2-HS-68-77(1). ☐

[11] **VERIFY** Reactor pressure is less than 55 psig, OR if entering this procedure from <u>RC/P</u> of 2-EOI-1, pressure is less than 100 psig. ☐

[12] **DIRECT** Instrument Mechanics to enable RHR SD CLG FLOW LOW annunciator, 2-XA-55-3D, Window 11 and **VERIFY** setpoint of 3700 gpm by programming recorder 2-FR-74-64, RHR SYS I/II FLOW, for RHR Loop to be placed in Shutdown Cooling. ☐

[13] **NOTIFY** Chemistry that RHRSW is to be placed in service and Shutdown Cooling is to be started. ☐

NOTES

1) For closed loop vents, venting is required for 1 minute.
2) Step 8.8[14] may be N/A'd, when the RHR Loop has been vented within 24 hours.

[14] **OPEN** the following RHR Loop I(II) vent valves until a solid stream of water is observed, **THEN CLOSE**:

A. Head (Containment) Spray Line through RHR SYS I HEAD SPRAY HI POINT (RHR SYS DW SPRAY) TELL-TALE VENT SOV, 2-SHV-074-0746(0747), AND ☐

B. HIGH POINT TELL TALE VENT HEAD SPRAY LINE (CONTAINMENT SPRAY), 2-FSV-74-138(139). [Rx Bldg, El 593' Fuel Pool Cooling Area (Rx Bldg, E, El 621')] ☐

INSTRUCTOR NOTES

c. Reactor pressure interlock 100 psig — TP-1 and 2

Two pressure switches monitor reactor system pressure at the "A" Reactor Recirculation Pump Suction and prevent opening (automatic-closure) the 74-48 & 47 isolation valves unless sensed pressure is <100 psig (greater than 100 psig is beyond RHR piping design). These press switches do NOT provide the alarm function — PS-68-93 PS-68-94

d. Drywell pressure 2.45 psig

(1) Pressure switches initiate LPCI mode of RHR in conjunction with < 450 psig reactor pressure, interlock Containment Cooling/spray valves closed, initiate PCIS Group 2 isolation and initiate a reactor scram.

(2) The DW pressure switches operate relays in the RHR Logic system (i.e. does not work thru Core Spray)

e. Reactor vessel water level -183 inches indicated on Post Accident Flooding Range

LIS used to provide a level permissive signal to Containment Cooling valves based on level inside the shroud. Permits containment spray only if level is >2/3 core height with LOCA signal present. — LIS-3-52 & 3-62A

- A keylock bypass switch can bypass the 2/3 core height interlock.

f. Drywell pressure 1.96 psig

Pressure switches used to provide pressure permissive signal to containment spray valves. Permits opening of containment spray only if pressure is significant in containment after accident.

- DWP decreasing to less than 1.96 psig will automatically close the spray valves if an accident signal is present.

Monitoring plant systems is critical during a transient when conditions change rapidly. SER 03-05

g. Reactor vessel low water level +2 inches (Lvl 3)

Level switches used to initiate: PCIS Group 2 isolation of Shutdown Cooling, CS System, RHR SDC isolation valves, RHR inbd isolation valves, RWCU isolation and a reactor scram.

INSTRUCTOR NOTES

(3) Can be opened when Rx press is $\leq$ 450 psig.

NOTE: Effect of Logic failure and valve operation

(4) Automatically opens on LPCI initiation signal when reactor pressure is < 450 psig.

(5) LPCI Injection Valve Open Signal Bypass Switch (Keylock switch on 9-3) can be utilized to bypass the open signal during execution of EOI's. Allows operator to manually (pnl 9-3) close the injection valve

NEW! 10A-S155A(B) Unit 1 and 2 ONLY at this time

Indicating light informs operators when open signal logic is bypassed.

(6) The normally open outboard injection valves (1-FCV-74-52,74-66;2-FCV-74-52, 74-66; 3-74-66) have added circuitry so that a fire cannot energize the closing coil and shut the valve (any close signal with the Control Room handswitch in NORMAL. Shorts out the closing coil and blows the control power fuses).

Modifications also disabled the local control "Close" pushbutton on 1-FCV-74-52/66, 2-FCV-74-52/66 and 3-FCV-74-66.

(7) Control Circuit

	1-74-52	1-74-66	2-74-52	2-74-66	3-74-52	3-74-66
Operate Interlock	74-53	74-67	74-53	74-67	74-53	74-67
Outgoing interlock	74-53	74-67	74-53	74-67	74-53	74-67
Normal/Emerg Sw		X	X		X	
Local Controls	X	X	X	X	X	X
Controls at Bkr		X	X		X	
Panel 9-3 controls	X	X	X	X	X	X
Local Indication lgts	X	X	X	X	X	X
Lights on Bkr		X	X		X	
Lights on Pnl 9-3	X	X	X	X	X	X

k. LPCI inboard injection valves

(74-53; 74-67)

Obj. V.C.5. TP-29, 30, and 31

(1) Normally closed - non-throttling

INSTRUCTOR NOTES

(2) Interlock prevents normal opening unless in-line valve (74-52; 74-66) is fully closed with reactor pressure > 450 psig

Operation of the valve at the breaker using the controls there will bypass the in-line and 450 psig interlock; prevents automatic opening and closure due to logic; and prevent any operation except from breaker

1-74-53 only
1-74-67 only
2-74-53 only
3-74-53 only

(3) Can be opened when Rx pressure is < 450 psig

NOTE: Effect of Logic failure and valve operation

(4) Automatically opens when Rx pressure < 450 psig with an LPCI initiation signal present and is interlocked open until LPCI initiation signal is cleared and reset.

NEW!
The Redundant logic has been removed.

(5) Only Respective Divisional LPCI Initiation logic will close the valve.

(6) Automatically close (both valves) if:

FCV 74-47 and 48 (S/D Cooling supply valves) open and a Group 2 isolation signal occurs

Automatic closure signal seals in (light indication). Can be reset (FCV 74-53/67 Shutdown Cooling isolation reset pushbuttons) when any of the conditions above are cleared.

Sys I-XS-74-126
Sys II-XS-74-132

Note that this closure signal will prevent opening if an LPCI signal is received.

(7) The normally closed inboard injection valves (2/3-FCV-74-53 and 74-67) have a new App ‘R’ Emergency Open Switch on the power supply board to bypass all interlocks and other circuitry (except the fully open limit switch) to open the valve.

INSTRUCTOR NOTES

(7) Separate bypass switch allows bypassing interlock from Valves 74-2/13 (74-25/36)

(8) Control Circuit

	1-74-57	1-74-71	2-74-57	2-74-71	3-74-57	3-74-71
Operate Interlock	74-58, 74-2/13	74-72, 74-25/36	74-58, 74-2/13	74-72, 74-25/36	74-58, 74-2/13	74-72, 74-25/36
Outgoing interlock	74-2/13	74-25/36	74-2/13	74-25/36	74-2/13	74-25/36
Normal/Emerg Sw		X	X		X	
Local Controls	X	X	X	X	X	X
Controls at Bkr		X	X		X	
Panel 9-3 controls	X	X	X	X	X	X
Local Indication lgts	X	X	X	X	X	X
Lights on Bkr		X	X		X	
Lights on Pnl 9-3	X	X	X	X	X	X
Bypass Switch	X	X	X	X	X	X

m. RHR Suppression Pool spray valves

(74-58; 74-72)
Obj. V.C.5.
TP-33, 36 and 37

(1) No automatic opening logic

(2) Interlock prevents normal opening if in-line valve not full closed (74-57; 74-71)

(3) Automatically closed and interlocked closed on LPCI initiation signal.

(4) The in-line valve interlock and/or the LPCI closure signal can be bypassed if the following exist:

(a) Reactor level $\geq$-183 inches and drywell pressure $\geq$ 1.96 psig and LPCI initiation signal and Select-Reset switch to SELECT position.

(b) Reactor level interlock and LPCI initiation signal may be bypassed by use of keylock bypass switch (XA 74-122/130)

INSTRUCTOR NOTES

(5) Amber light above the "SELECT" switch indicates:

Obj. V.B.11
Obj. V.C.6.

Switch in "Select or Normal after Select"

AND

DWP is $\geq$1.96 psig

AND

RPV level $\geq$-183" & have LPCI signal

OR

Keylock in Bypass position

(a) As long as the light remains "on", the valves may be opened and a LPCI signal will not close them.

(b) The in-line interlock valve does not have the DWP interlock, so it is possible to open the Spray valve (58/72) first and then open the in-line valve (57/71) without DWP being $\geq$1.96 psig

(6) Drywell pressure interlock prevents drawing vacuum on containment under accident condition.

This interlock cannot be bypassed.

(7) Control Circuit

	1-74-58	1-74-72	2-74-58	2-74-72	3-74-58	3-74-72
Operate Interlock	74-57	74-71	74-57	74-71	74-57	74-71
Outgoing interlock						
Normal/Emerg Sw						
Local Controls	X	X	X	X	X	X
Controls at Bkr						
Panel 9-3 controls	X	X	X	X	X	X
Local Indication lgts	X	X	X	X	X	X
Lights on Bkr						
Lights on Pnl 9-3	X	X	X	X	X	X

INSTRUCTOR NOTES

o. Containment Spray valves — (74-60/61; 74-74/75)

(1) No automatic opening logic — Obj. V.C.5; TP-33, 40, 41, 42, 43

(2) IN-line valve interlock prevents normal opening unless other valve fully closed

(3) Automatically closed/interlocked closed on LPCI signal

(4) Automatic closure signal and/or the in-line valve interlock may be bypassed if the following exist:

(a) Reactor level $\geq$-183 inches and drywell pressure $\geq$1.96 psig and LPCI initiation signal present and Select=Reset switch placed to SELECT position

(b) Reactor level interlock and LPCI initiation signal may be bypassed by use of keylock bypass switch (XS-74-122/130)

(5) Amber light above the "SELECT" switch indicates: — Obj. V.C.6

Switch in "Select or Normal after Select"

AND

DWP is $\geq$1.96 psig

AND

RPV level $\geq$-183" and have LPCI signal

OR

Keylock in Bypass position

(a) As long as the light remains "on", the valves may be opened and a LPCI signal will not close them.

(6) Drywell pressure interlock prevents drawing vacuum on containment under accident condition. — This interlock cannot by bypassed.

(7) Emergency position at breaker bypasses both of the normal control circuits (opening/closing/interlocks) — Obj. V.D.8; U2 & U3-74-60; U1-74-74

3. RO 206000K6.09 001/C/A/SYS/HPCI/4/206000K6.09//RO/SRO/BANK 11/27/07 RMS

Given the following plant conditions:

- Unit 2 reactor water level initially lowered to (-)69 inches.
- Conditions required entry into EOI-1, "RPV Control" and EOI-2, "Primary Containment Control."
- After water level recovery, the High Pressure Coolant Injection (HPCI) Pump Injection Valve (FCV-73-44) was manually closed and HPCI was placed in pressure control to remove decay heat.
- Subsequently, Condensate Storage Tank (CST) level drops below 6800 gallons.

Which ONE of the following describes the status of the HPCI system (assume no other operator actions have occurred)?

A. HPCI would be operating in pressure control with suction from the CST.

B. HPCI would be pumping to the CST with suction from the Suppression Pool.

C.✔ HPCI would be operating at shutoff head with suction from the Suppression Pool.

D. The HPCI turbine would trip on overspeed due to loss of suction during the transfer.

K/A Statement:

206000 HPCI
K6.09 - Knowledge of the effect that a loss or malfunction of the following will have on the HIGH PRESSURE COOLANT INJECTION SYSTEM : Condensate storage and transfer system: BWR-2,3,4

K/A Justification: This question satisfies the K/A statement by requiring the candidate to use specific plant conditions to determine the effect of low CST level on HPCI operation.

References: OPL171.042 Rev 19 Page 36 and TP-12

Level of Knowledge Justification: This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

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REFERENCE PROVIDED: None

Plausibility Analysis:

In order to answer this question correctly the candidate must determine the following:

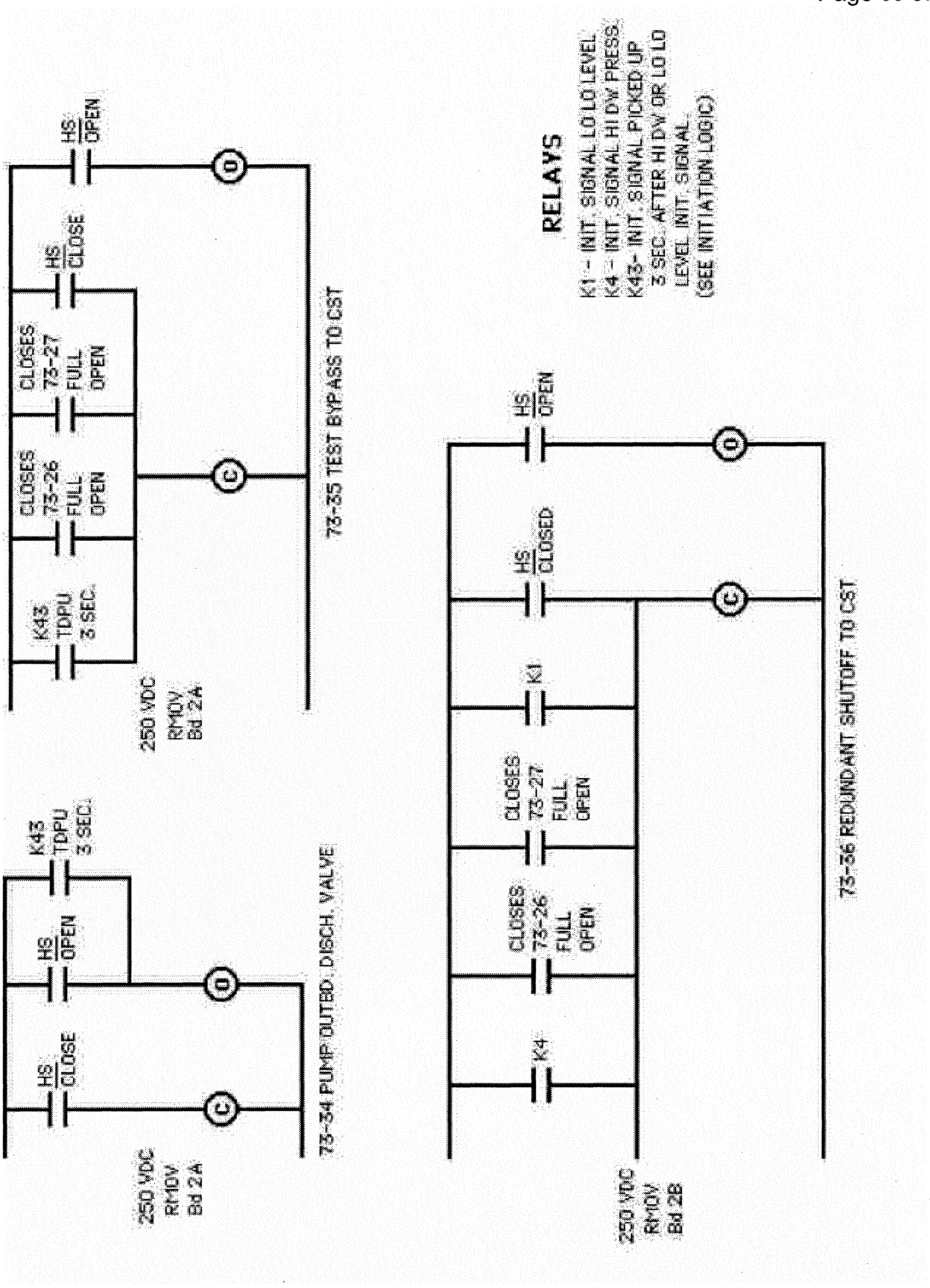
1. Recognize that the HPCI initiation signal is reset to allow HPCI to be placed in Pressure Control.
2. Recognize that the HPCI Pressure Control lineup if from the CST and back to the CST.
3. Recognize that the current CST level would initiate a suction swap to the Suppression Pool.
4. Recognize that HPCI would not receive a trip signal as the suction valves re-aligned.
5. Recognize that the CST Test Isolation Valve will auto close on low CST level.

C - correct: At ~7000 gallons in the CST, HPCI auto swaps from CST suction to Suppression Pool (Torus) suction. When this occurs the CST Test Return Isolation valve receives a close signal from the Torus suction valves opening; to prevent pumping the Torus to the CST. Therefore, with the HPCI injection valve previously closed, HPCI would be operating at shutoff head without minimum flow protection. Note - the minimum flow valve would be closed due to lack of a valid initiation signal.

A - incorrect: This assumes the low CST level has NOT initiated a suction swap to the Torus. This is plausible since most procedures list the setpoint for the auto swap as an elevation above sea level versus a "gallons" setpoint.

B - incorrect: This lineup would occur if the HPCI Test Isolation Valve did NOT receive a close signal following the suction swap logic initiation. This is plausible since the ONLY auto closure interlock of the HPCI Test Isolation Valve is under this specific condition.

D - incorrect: HPCI will not trip on low suction pressure under this specific condition. The Torus suction valves begin to open before the CST suction valve closes in a "make-before-break" fashion. This is plausible since closure of the suction path to HPCI typically results in a low suction trip.



TP-12: HPCI Valve Logic (2 of 3)

INSTRUCTOR NOTES

2. If during HPCI operation, suppression pool water level increases to 7" (5.2" on Unit 3) above zero or if CST level drops to 552'6" above sea level (7000 gallons), then HPCI pump suction valves from the suppression pool (73-26 and 73-27) open. (This will then cause the CST suction valve to close once the SP suction valves get full open).

   NOTE: There are normally 300,000 gallons available in the CST for HPCI and RCIC use.

3. A flow switch tapped in parallel with the HPCI system flow controller closes the minimum flow bypass valve to suppression pool (73-30) at 1255 gpm increasing; and opens it at 900 gpm decreasing, only if an auto start signal is present. Minimum flow valve closes on a Turbine Trip signal.

4. If either of the suppression pool suction line isolation valves (73-26 or 73-27) are full open then the HPCI test line to the CST valves (73-35 and 73-36) will close. Obj. V.B.4 Obj. V.C.4

5. If the HPCI turbine isolation valve (73-16) is fully closed, then gland seal condenser condensate pump discharge valves to clean radwaste (73-17A and 73-17B) will open if the gland seal condenser hotwell has high level.

6. If the HPCI turbine isolation valve (73-16) is fully closed, then HPCI turbine steam line drain pot discharge isolation valves to the main condenser (73-6A and 73-6B) will open.

7. If 73-16 is full closed, the auxiliary oil pump will not start from the control room. When 73-16 opens 10% and the control switch is in the start position, the auxiliary oil pump will run.

8. If the HPCI turbine steam line drain pot level reaches the high level setpoint, then the downstream trap bypass valve (73-5) will open. Unit 1 73-5 has been replaced with a manual valve. DCN 51221 Unit difference

4. RO 209001K5.04 001/MEM/T2G1/BASIS//209001K5.04//RO/SRO/MODIFIED 11/17/07

During EOI execution, when injection from low pressure systems is required to restore and maintain RPV level, the Core Spray System is NOT on the list of preferred systems for low pressure injection IF all control rods are NOT inserted.

Which ONE of the following describes the basis for this restriction?

A. Cold water from Core Spray creates a rapid pressure reduction and cooldown rates CANNOT be controlled.

B.✓ Core Spray injects directly onto fuel bundles inside the shroud which could damage fuel and cause a power excursion.

C. Core Spray injection creates a steam blanket at the top of the fuel bundles which inhibits heat transfer via steam flow past the fuel.

D. Core Spray does NOT provide sufficient flow to maintain adequate core cooling if an ATWS power level greater than or equal to 80% occurs.

K/A Statement:

209001 LPCS
K5.04 - Knowledge of the operational implications of the following concepts as they apply to LOW PRESSURE CORE SPRAY SYSTEM : Heat removal (transfer) mechanisms

K/A Justification: This question satisfies the K/A statement by requiring the candidate to recall the unique heat removal mechanisms of Core Spray and recall a condition where that mechanism can result in unfavorable consequences.

References: OPL171.205 rev 8, pg 60, 11.d and EOIPM Section 0-V-K

Level of Knowledge Justification: This question is rated as MEM due to the requirement to recall or recognize discrete bits of information.

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MODIFIED FROM OPL171.205 #9

REFERENCE PROVIDED: None

Plausibility Analysis:

In order to answer this question correctly the candidate must determine the following:

1. The bases behind the restriction of Core Spray injection during an ATWS emergency.

B - correct: The systems listed in Step C5-13 for use in controlling RPV water level comprise all those which inject outside of the core shroud. These are used, preferentially, because the flow path outside the core shroud mixes the relatively cold injected water with the warmer water in the lower plenum prior to it reaching the core. Core Spray injects outside the shroud spraying relatively cold water directly on the fuel which does not provide mixing of the boron (if injected) could lead to fuel damage or a power excursion.

A - incorrect: This is could cause a rapid pressure reduction but is not the overriding factor for not using Core Spray under these conditions. This is plausible since high volume Core Spray injection at close to the maximum injection pressure would cause a rapid pressure reduction.

C - incorrect: This phenomenom, referred to as Counter Current Flow Instability, is plausible but is only of significant concern with the core completely uncovered and is the basis for removing Spray Cooling from the EPG definition of Adequate Core Cooling.

D - incorrect. This is incorrect because Core Spray can provide enough flow to maintain adequate core cooling at any ATWS power level if water level is lowered IAW C5. This is plausible since Core Spray would not be able to maintain water level if level was not lowered in C5.

DISCUSSION: STEP C5-16 (Continued)

In comparison to Minimum Zero-Injection RPV Water Level (refer to the discussion of Step C3-3 in the C3, Steam Cooling, Bases), Minimum Steam Cooling RPV Water Level is slightly higher than Minimum Zero-Injection RPV Water Level. This is attributed to two key factors:

1. Injection of subcooled water requires that part of the energy that would be used to generate steam for cooling the uncovered portion of the core must now be expended in heating subcooled liquid to saturation temperature (Minimum Zero-Injection RPV Water Level is calculated assuming no injection into the RPV).

2. More steam is required to maintain clad temperature below 1500 °F as compared to the 1800 °F limit assumed for Minimum Zero-Injection RPV Water Level calculation.

The injection sources listed for use in controlling RPV water level comprise all of those that inject outside of the core shroud. These are used, preferentially, because the flowpath outside the core shroud mixes the relatively cold injected water with warmer water in the lower plenum prior to reaching the core. No priority between use of each listed system is intended, therefore the operator should use the most appropriate means available under current plant conditions.

EOI Appendices 5A, 5B, and 6A provide guidance to operate Condensate/Feedwater, CRD, and only Condensate respectively. These systems are preferred sources of injection since they are of high quality water and are used for RPV water level control during normal plant power operations. Feedwater and CRD both provide high pressure injection from either a steam or motor-driven supply, and Condensate by itself provides for low pressure injection.

EOI Appendices 5C and 5D provide guidance to operate RCIC and HPCI respectively. The operator is instructed to operate RCIC and HPCI with suction from the CST if available, to ensure that the highest quality water is used for injection into the RPV. The CST is the preferred suction source not only because of higher water quality, but also because the CST is not subjected to the temperature increase that the suppression pool is. For these reasons, defeating HPCI high suppression pool suction transfer logic in EOI Appendix 5D, allows the operator to maintain the CST as the suction source. EOI Appendix 5C provides direction to defeat the RCIC low RPV pressure isolation interlock, that allows operation of the RCIC turbine at low pressure. Even if RPV pressure is below the isolation setpoint, but above turbine stall pressure, RCIC can still provide some injection into the RPV.

EOI Appendices 6B and 6C provide guidance to operate LPCI Systems I and II respectively. The operator is instructed to only operate RHR in LPCI mode when suppression pool level is above <A.62>. Engineering calculations have determined that operation of RHR pumps below a suppression pool level of <A.62> may induce vortex formation at the system suction strainer.

DISCUSSION: STEP C5-30

This signal step informs the operator that actions to control RPV pressure control must immediately change because of present plant conditions.

When emergency RPV depressurization is required, the operator shall transfer RPV pressure control actions from the RC/P Section of EOI-1, RPV Control, to C2, Emergency RPV Depressurization.

This step has been reached in this procedure because previous attempts to maintain adequate core cooling have been unsuccessful, or plant conditions are such that emergency RPV depressurization is required, as indicated by a signal step in another EOI being concurrently executed.

If adequate core cooling cannot be assured, then plant conditions may be such that RPV water level is at or below TAF, and RPV pressure is high enough to prevent injection from low-head pumps. Therefore, emergency RPV depressurization is required for the purpose of maximizing injection flow from high-head pumps and to permit injection from low-head pumps.

Depressurizing the RPV is preferred over restoring RPV water level through the use of systems that inject inside the shroud because:

1. A large reactor power excursion may result from the in-shroud injection of relatively cold water.

2. Rapid depressurization, by itself, will shut down the reactor due to a substantial increase in voids.

3. Following the depressurization, reactor power will stabilize at a lower level.

DISCUSSION: STEP C5-38

Caution #5 applies throughout performance of Step C5-38. Caution #5 is identified at this step to highlight the potential for large power excursions and subsequent core damage if cold, unborated water is rapidly injected using injection sources within Step C5-38.

This action step directs the operator to use injection sources listed to restore and maintain RPV water level above <A.71>. System specific EOI Appendices provide step-by-step guidance for lining up and injecting into the RPV. Injection pressures (<A.1>) have also been provided as additional information to the operator.

Engineering calculations have determined that when RPV water level is at or above <A.71>, adequate core cooling is still assured. The value of <A.71> RPV water level is Minimum Steam Cooling RPV Water Level. Refer to discussion of Step C5-16 for more information on Minimum Steam Cooling RPV Water Level.

This step has been reached only when RPV water level cannot be restored and maintained above <A.71> using preferred systems. Therefore, use of additional systems is required that either inject inside the core shroud, are difficult to lineup, or take suction on sources of comparatively lower water quality. No priority between use of each listed system is intended, therefore, the operator should use the most appropriate means available under current plant conditions.

EOI Appendices 6D and 6E provide guidance to operate CS Systems I and II. CS provides relatively high quality water from the suppression pool and can provide injection into the RPV quicker than other sources listed in this step. However, reactor power excursions are more probable since CS injects directly into the core shroud at high flowrates. Therefore, extreme caution should be used for CS injection at this step.

Unlike directions given for use of motor driven pumps in EOI-1, RPV Control, CS System operation is not restricted by pump NPSH and Vortex limits (suppression pool level). Even though risk of equipment damage exists if NPSH and Vortex limits are exceeded, immediate and catastrophic pump failure is not expected should operation beyond these limits be required. Since prolonged operation under these conditions is most likely required before degraded system and pump performance may result, the undesirable consequences of uncovering the reactor core outweigh risk of equipment damage.

EOI Appendices 7C, 7E, and 7F provide guidance to inject RHR into the RPV from crossties to other units or through RHR Drain Pumps A and B. EOI Appendix 7G provides guidance to inject into the RPV with PSC Head Tank Pumps. All of these injection sources provide suppression pool water at low pressure, but are relatively complicated to line up.

d. The systems listed in Step C5-13 for use in controlling RPV water level comprise all those which inject outside of the core shroud. These are used, preferentially, because the flow path outside the core shroud mixes the relatively cold injected water with the warmer water in the lower plenum prior to it reaching the core.

Cautions 2,3,5,6 apply

NPSH and Supp PL lvl limits apply

e. The last bullet in the list of systems applies only if step C5-25 was executed for level control. Those systems listed in step C5-25 inject outside the shroud, or are of low quality water or are difficult to line up. If level restoration was performed using those systems, then it is appropriate to use them again when this step is reached.

12. Step C5-14 (Override)

This override applies to steps C5-15 through C5-18

If while performing the associated steps, reactor power rises above 5% and RPV water level is above -50" direction to return to step C5-6 and re-evaluate conditions to determine a new water level band.

13. Step C5-15

a. This step is reached only if power was below 5%. There is no driving condition to lower level. The RPV water level band is -180" to +51.

b. As in step C5-13, those systems listed inject outside the shroud.

Cautions 2,3,5,6 apply NPSH and vortex limits apply

14. Step C5-16 (Override)

a. Part One

(1) Step C5-18, which follows, directs restoration of RPV water level to the normal range after having injected sufficient boron to shut down the reactor.

5. RO 211000AK2.01 001/C/A/T2G1//63/V.B.5/211000AK2.01/2.9/3.1/RO/SRO/MODIFIED 11/17/07

Given the following plant conditions:

- Unit 1 is operating at 100% power.
- A fire inside the board causes a loss of the '1B' 480V Shutdown Board.
- Fire Protection reports that the fire cannot be extinguished.
- The Unit Supervisor (US) directs a manual scram.
- NOT ALL control rods insert, and the following conditions are noted:

- Reactor Power	15%
- Suppression Pool Temperature	108°F and rising

- The 'A' 4kV Shutdown Board deenergized due to an electrical fault when '1A' RHR pump was started for Suppression Pool cooling IAW EOI-2.

Which ONE of the following describes the action and method of injecting boron into the reactor?

Transfer the _____(1)_____ to alternate and inject Standby Liquid Control (SLC) using the _____(2)_____.

	(1)	(2)
A.✓	'1A' 480V Shutdown Board;	'1A' SLC Pump.
B.	'1B' 480V Shutdown Board;	'1B' SLC Pump.
C.	'1A' 480V Reactor MOV Board;	'1A' SLC Pump.
D.	'1B' 480V Reactor MOV Board;	'1B' SLC Pump.

K/A Statement:

211000 SLC
K2.01 - Knowledge of electrical power supplies to the following: SBLC pumps

K/A Justification: This question satisfies the K/A statement by requiring the candidate to correctly identify the power supplies to the SLC pumps.

References: OPL171.039 rev 16, pg 15, 4.b & c

Level of Knowledge Justification: This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to solve a problem. This requires mentally using this knowledge and its meaning to resolve the problem.

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MODIFIED FROM 211000K2.01

REFERENCE PROVIDED: None

Plausibility Analysis:

A - correct: '1A' and '1B' SLC pumps are fed from their respective 480V Shutdown Boards. '1B' is not available due to the loss of the '1B' 480V Shutdown Board. The 'A' 4kV Shutdown Board feeds the '1A' 480V Shutdown Board. When the 'A' 4kV Shutdown Board is lost due to an electrical fault, the '1A' 480V Shutdown Board is de-energized. Manually transferring '1A' 480V Shutdown Board to Alternate will restore power to the board and allow starting '1A' SLC Pump.

B - incorrect: Due to the loss of 'A' 4kV Shutdown Board, the '1A' 480V Shutdown Board has lost power. This is plausible if the candidate is not aware that the 480V Shutdown Board does not automatically transfer to Alternate the same as the 4kV Shutdown Board.

C - incorrect: The power supply to '1A' SLC is not '1A' 480V RMOV Board. It is plausible because '1A' 480V RMOV Board is safety related and powered from the same DG as '1A' 480V Shutdown Board.

D - incorrect: The '1B' 480V Shutdown Board is unavailable due to a fire. This answer is plausible if the operator does not know the correct power supply to SLC Pumps.

Amplification: A fire in a 480V Shutdown Board may not result in it being lost depending on the location and severity of the fire. It is necessary to state the the '1B' 480V S/D Board is lost so that the candidate can assess the fact that he cannot transfer the board to alternate.

INSTRUCTOR NOTES

4. SLC Pumps

a) Two 100% capacity, triplex, positive displacement piston pumps are installed in parallel.

Obj. V.B.5.c
Obj. V.C.4.d
Obj. V.D.4
Obj. V.E.4

b) 'A' pump is powered from 480V Shutdown Board A.

c) 'B' pump is powered from 480V Shutdown Board B.

Obj. V.B.5.c
Obj. V.C.4.d

d) Electrically interlocked so that only one pump will run at a time. This prevents system overpressurization.

Obj. V.B.3.f
Obj. V.C.2.f

e) The pumps are manually started from the main control room using the key-lock switch on panel 9-5, or locally, using the Test Permissive Transfer Switch at Panel 25-19.

f) A control room start signal will fire the explosive valves. A local start will not fire the explosive valves.

g) Either pump is capable of supplying a system flow of approximately 50 gpm at a system pressure of 1275 psig.

Obj. V.D.3.d
Obj. V.E.3.d

h) Each pump discharge has a relief valve, set at 1425 $\pm$ 75 psig, to protect the pump and the system from overpressurization.

Obj. V.B.3.f
Obj. V.C.2.f
Obj. V.D.3.e
Obj. V.E.3.e

i) Each pump contains internal suction and discharge check valves, which open at approximately 5 psid, allowing only forward flow through an idle pump. (INPO O&MR 341).

j) Pump motors are protected by an undervoltage trip.

5. Accumulators

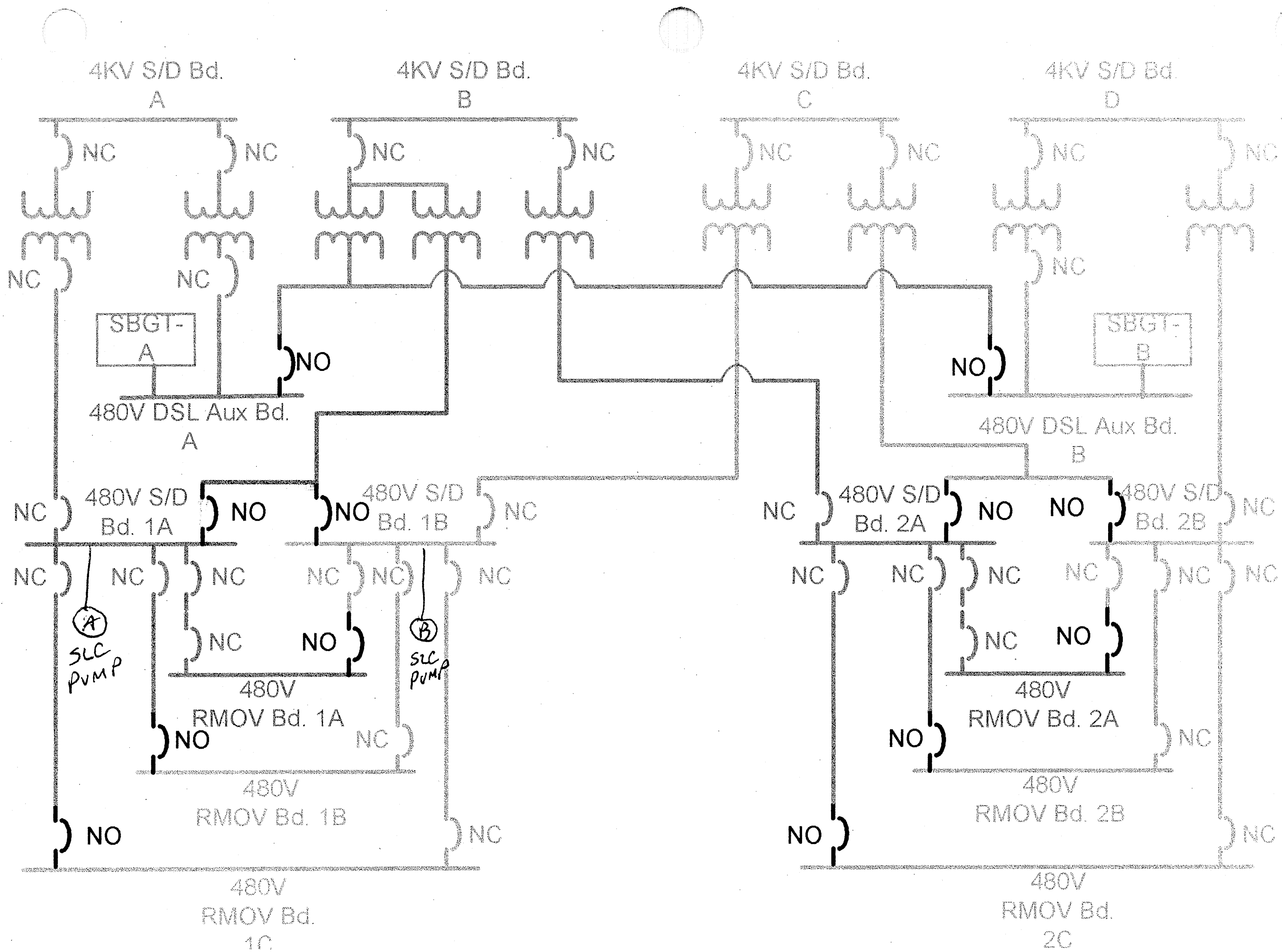
a) An accumulator is installed between each pump and its discharge check valve.

b) Dampens the pressure pulsations that are inherent with piston-type, positive-displacement pumps.

Obj. V.D.3.d
Obj. V.E.3.d

c) A steel vessel accumulator, containing a synthetic bladder, with one side charged to ~450 psig nitrogen gas and SLC solution on the other side.

TP-1: Unit 1-2 480V Power Distribution



6. RO 212000K6.03 001/MEM/T2G1/RPS//212000K6.03//RO/SRO/BANK 11/27/07 RMS

Given the following plant conditions:

- Reactor Water Level Normal Control Range Instrument, 2-LIS-3-203A, has failed downscale.

Which ONE of the following describes the Analog Trip System Response?

The trip relay will be ___(1)______ and the contact in the Reactor Protection System (RPS) Logic will be ______(2)_____.

	(1)	(2)
A.	de-energized;	closed
B.✓	de-energized;	open
C.	energized;	closed
D.	energized;	open

K/A Statement:

212000 RPS
K6.03 - Knowledge of the effect that a loss or malfunction of the following will have on the REACTOR PROTECTION SYSTEM : Nuclear boiler instrumentation

K/A Justification: This question satisfies the K/A statement by requiring the candidate to use specific plant conditions involving level instrumentation to determine the response of RPS logic components.

References: OPL171.003 rev 18, pg 28 (d), OPL171.028 rev 17, pg 14,3.e(3) and 2-730E915 series prints.

Level of Knowledge Justification: This question is rated as MEM due to the requirement to recall or recognize discrete bits of information.

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REFERENCE PROVIDED: None

Plausibility Analysis:

In order to answer this question correctly the candidate must determine the following:

1. Whether RPS relays are normally emergeized or de-energized.
2. Whether RPS contacts fed from relays are normally open or closed.
3. Recall which scram signal, if any, is fed from 2-LIS-3-203A.

B - correct: RPS relays are normally energized and closed. When 2-LIS-3-203A fails downscale it is essentially the same as a low water level scram signal in that channel. This will deenergize the associated relay which results in OPEN contacts.

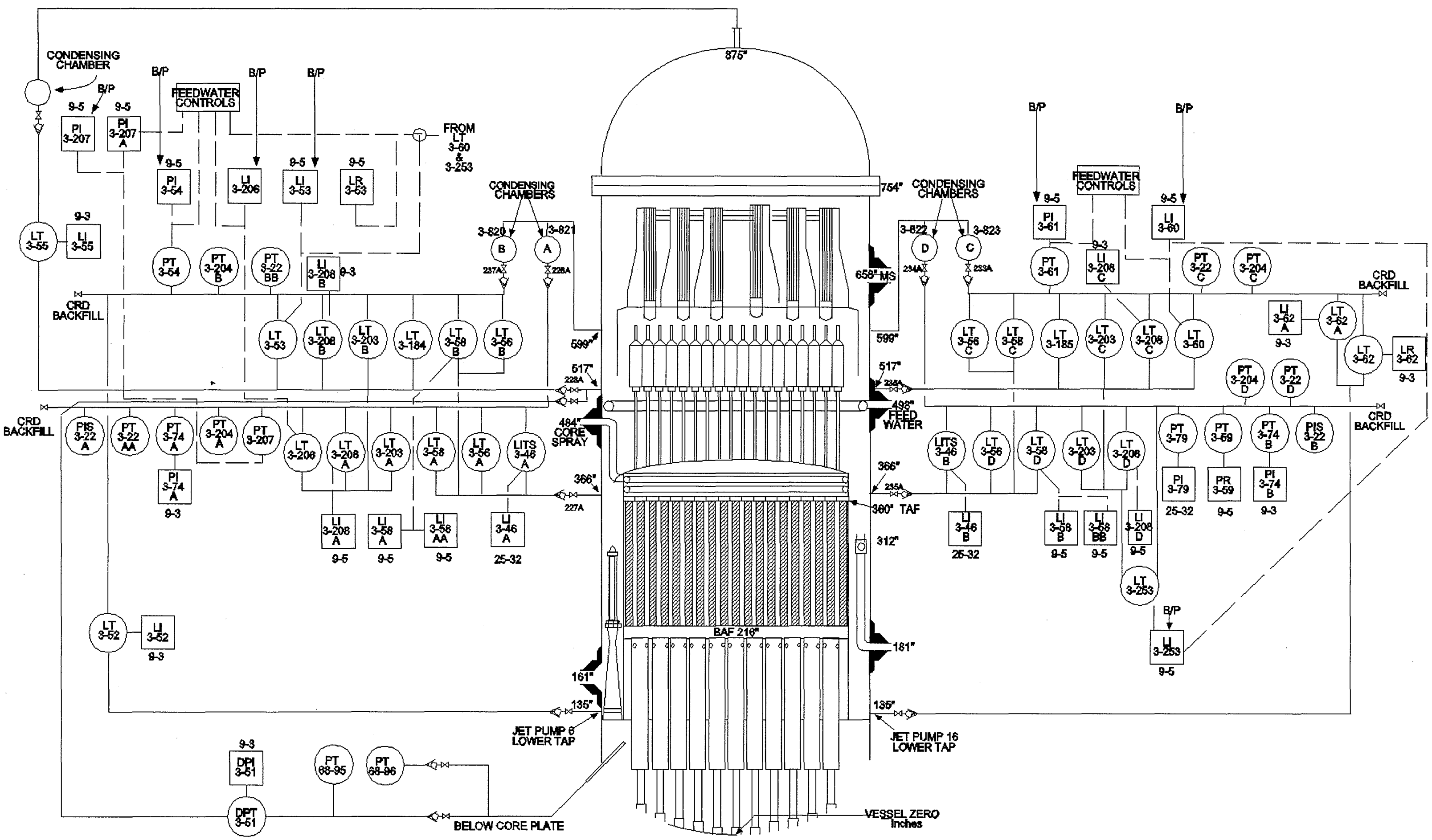
A - incorrect: This is plausible if the operator confuses PCIS logic relay response with RPS logic relay response and fails to recognize a valid trip has been generated by 2-LIS-3-203A.

C - incorrect: This is plausible if the operator fails to recognize a valid trip has been generated by 2-LIS-3-203A.

D - incorrect: This is plausible if the operator confuses PCIS logic relay response with RPS logic relay response.

Amplification: This a MEMORY level question and the justification has been corrected.

Transparancy
Revision 13



NORMAL CONTROL RANGE INSTRUMENTS (0 TO 60")	
INST. NUM.	FUNCTION
LT 3-184 &185	ADS CONFIRMATORY
LT 3-203 A,B,C & D	Rx SCRAM, PCIS 2,3,6 & 8
LT 3-208 A,B,C & D	HPCI, RCIC, Main & RFP TRIPS
LT 3-53, 60, 206, & 253	FEEDWATER LEVEL CONTROL
PT 3-54 , 3-61 & 207	FEEDWATER LEVEL CONTROL
PT 3-204 A,B,C, & D	RECIRC PUMP TRIP

EMERGENCY RANGE INSTRUMENTS (-155 TO +60")	
INST. NUM.	FUNCTION
LT 3-56 A,B,C & D	RECIRC PUMP TRIP, PCIS GP 1
LT 3-58 A,B,C & D	HPCI,RCIC,RHR,CS & ADS INIT.
LT 3-46 A & B	PNL 25-32 INDICATION
PT 3-74 A & B	RHR & CS LOGIC
PT 3-22 AA,BB,C & D	HIGH PRES SCRAM
PIS 3-22A	MECH VAC PUMP TRIP
PT 3-59	RECORDER PNL 9-5
PT 3-79	PNL 25-32 INDICATION

POST ACCIDENT RANGE INSTRUMENTS (+32 TO -268")	
INST. NUM.	FUNCTION
LT 3-52 & 62	CONTAINMENT SPRAY INTERLOCK

SHUTDOWN VESSEL FLOODING RANGE (0 TO +400")	
INST. NUM.	FUNCTION
LT 3-55	PANEL 9-3 INDICATION

TP-7: REACTOR VESSEL LEVEL/PRESSURE INSTRUMENTATION

INSTRUCTOR NOTES

(c) Also ensures that MCPR and LHGR are not violated during a feedwater controller maximum demand failure.

(2) High level alarm at +39 inches, Level 7

Defines upper end of normal operating region. If level is ≤ 39 inches, transients such as a recirculation pump trip will not cause level to increase to turbine trip point.

(3) Low level alarm at + 27 inches, Level 4

Defines lower end of normal operating region. If level is ≥ +27 inches, transients such as a reactor feedpump trip from full power will not cause level to decrease to reactor scram point with a runback of the recirculation pumps.

(4) Reactor scram at +2 inches, Level 3

(a) Assures the reactor will not be operated without sufficient water above the reactor core.

(b) In conjunction with the containment isolation signal of +2 inches, prevents fuel cladding perforation

(c) Prevents operation with separator skirts uncovered, which could result in carryunder.

(d) Set low enough to prevent spurious operation for normal operating transients.

(e) Starting SGTS establishes secondary containment.

(5) ATWS RPT/ARI trip at -45 inches, Level 2

Reduces core flow and bleeds off CRD HCU air header in event that low level was due to a failure to scram.

Also prevents operation of recirc pumps without adequate net positive suction head.

RPS LOGIC

INSTRUCTOR NOTES

(2) The third is used to produce manual SCRAM trip signals (trip channel A3).

(3) The channels for trip system B are designated B1, B2 and B3.

c. Both of the automatic channels in each trip system monitor critical reactor parameters. — Obj. V.B.5.c; Obj. V.D.4

(1) At least four channels for each monitored parameter are required for the trip system logic. — TP-3; Drawing 2-730E915RF-11; 2-730E915RF-12

(2) If either of the two channels sense a parameter which exceeds a setpoint, then this would place the associated trip system (A or B) into a tripped condition.

(3) To produce a SCRAM, both trip systems must be tripped. This is called a "one-out-or-two-taken twice" arrangement.

d. Each trip system logic may also be manually tripped. — TP-4

(1) Each Trip system contains manual SCRAM switches on Panel 9-5 which cause a trip in the respective trip system when actuated. — Drawing 2-730E915RF-11; 2-730E915RF-12

(2) The Reactor mode switch has contacts in both the A3 and B3 channels. Placing the reactor mode switch in SHUTDOWN will result in a trip of both trip systems.

(3) A trip in both channels A3 and B3 initiates a reactor SCRAM.

e. During normal operation — SER 3-05 Operator fundamentals

(1) All sensor and trip contacts essential to safety are closed.

(2) Channels, logics, and actuators are energized.

RPS LOGIC

INSTRUCTOR NOTES

(3) When a SCRAM signal is received, the logic relays deenergize to cause a SCRAM. — Drawing 2-730E915-13

(4) Loss of power to one RPS bus will result in a half-SCRAM. Loss of power to both RPS buses will result in a full SCRAM. — Obj. V.B.3, Obj. V.C.3, Obj. V.D.8

4. Reactor SCRAM Signals and Arrangement
Refer to OI-99 for the setpoints for each SCRAM. — Obj. V.B.6, Obj. V.C.4, SER 3-05 Operator fundamentals

a. Channel test switch

(1) Allows for testing each channel's trip function. — Drawing 2-730E915RF-11, 2-730E915RF-12

(2) Four, one per channel located on Panel 9-15 and 9-17 in Aux. Inst. Room. — REACTIVITY MANAGMENT — Discuss when switches can be used.

(3) Key-locked, two positions - NORMAL and TRIP

(4) TRIP de-energizes that channel's relays producing a half-SCRAM.

b. Turbine Stop Valves, 10 percent closure anticipates the pressure and neutron flux rise caused by the rapid closure of the Turbine Stop Valves. — TP-5, Drawing 2-730E915-9, 10, Obj. V.D.5

(1) Each of the four Turbine Stop Valves is equipped with two limit switches. One limit switch is assigned to RPS “A” and one to RPS “B”.

(2) These switches will provide a valve-closed signal to the RPS trip logic.

(3) The position switch contacts are arranged so that any two Stop Valves can be closed causing no more than a half-SCRAM. — Drawing 2-730E915RF-11, 2-730E915RF-12

(4) Closure (< 90% full open) of any combination of three Stop Valves will cause a full SCRAM in all cases.

PCIS LOGIC

C. Typical PCIS Isolation Logic

TP-1

1. A typical logic arrangement for the PCIS valves (except MSIVs) is shown in TP-1. This figure shows that two separate trip channels (A and B) are each provided with two sensor relay contacts (A/C and B/D).

PCIS de-energizes to isolate (except HPCI/RCIC)

Obj. V.B.1
Obj. V.C.1

a. This arrangement creates trip subchannels A1/A2 and B1/B2.

b. A trip of either sensor relay within a trip channel will cause opening of the associated contact and de-energization of the associated relay. This condition will create a "half isolation" signal within both logic channels but NO VALVE MOVEMENT.

HPCI/RCIC are energize to actuate

Obj. V.B.3
Obj. V.C.3

c. Should a trip of either sensor relay in the other trip channel occur, conditions will exist to de-energize the valve actuation relays in each logic channel, causing both isolation valves to close.

PCIS logic is arranged as follows:

A1 **OR** A2 **AND** B1 **OR** B2	= **Inboard AND Outboard** valve closure

Note: Most PCIS logic is assembled as above. The MSL drains however are an exception.

The MSL drain logic is as follows:

A1 **AND** B1 = I/B valve closure

A2 **AND** B2 = O/B valve closure

7. RO 215003A4.04 001/MEM/SYS/IRM/B6/215003A4.04//RO/SRO/MODIFIED 11/17/07

Given the following plant conditions:

- Unit 3 reactor startup preparations are in progress with NO rods withdrawn.
- Instrument Mechanics are performing the Intermediate Range Monitor (IRM) functional surveillance.
- No IRMs are currently bypassed.
- The Instrument Mechanic technician has placed the "INOP INHIBIT" toggle switch for the 'H' Channel IRM in the "INHIBIT" position.

Which ONE of the following describes the IRM trip function that is bypassed as a result of this action?

A. IRM "High Voltage Low" INOP TRIP is bypassed.

B. IRM "Loss of $\pm$ 24 VDC" INOP TRIP is bypassed.

C. IRM "Module Unplugged" INOP TRIP is bypassed.

D.✓ IRM "Mode Switch Out of Operate" INOP TRIP is bypassed.

K/A Statement:

215003 IRM
A4.04 - Ability to manually operate and/or monitor in the control room: IRM back panel switches, meters, and indicating lights

K/A Justification: This question satisfies the K/A statement by requiring the candidate to use specific component manipulations to correctly determine the response of the IRM system.

References: OPL171.020 rev. 10, pg 22, c.(1) & (2) and 3-OI-92A rev. 14, pg 7 of 15 P&L 3.0.J

Level of Knowledge Justification: This question is rated as MEM due to the requirement to recall or recognize discrete bits of information.

0610 NRC Exam
MODIFIED FROM OPL171.020 #28

REFERENCE PROVIDED: None

Plausibility Analysis:

In order to answer this question correctly the candidate must determine the following:

1. The function switch is bypassed by the "INOP INHIBIT" pushbutton.

NOTE: Each of the possible answers below will typically initiate an INOP trip of it's associated IRM channel, therefore each distractor is plausible.

D - correct: INOP/INHIBIT Pushbuttons (toggle sw for Unit 3) are pushed/flipped to bypass the INOP trip that results from taking mode switch S-1 out of "OPERATE." They are used to allow testing of other scram or rod block signals from the IRM drawer into RPS/RMCS without them being masked by the INOP trip.

A - incorrect: This INOP trip will still function.

B - incorrect: This INOP trip will still function.

C - incorrect: This INOP trip will still function.

INSTRUCTOR NOTES

E. Trips — TP-10

1. Rod blocks — Obj.V.D.7, V.B.5; Obj. V.C.3.,

Block	Setpoint	When Bypassed	Instructor Notes
Downscale	≤ 7.5	Range 1 or RUN	
High	≥ 90	RUN Mode	Obj. V.B.6. Obj. V.C.4.
INOP	-HV low (<90v) -Module unplugged -Function switch not in OPERATE -Loss of ±24VDC	RUN Mode	Obj. V.B.5
Detector Wrong Position	Detector Not Full IN	RUN Mode	Obj.V.B.13

2. Scrams — TP-11

Scrams	Setpoint	When Bypassed	Instructor Notes
High-High	≥ 116.4	In RUN Mode	Obj. V.B.7. Obj. V.C.5. Obj.V.D.8
INOP	-HV low (<90v) -Module unplugged -Function switch not in OPERATE -Loss of ±24VDC	In RUN Mode	

F. Controls Provided

1. Panel 9-5

   a. Recorder switches switch between IRM channels and APRM/RBM channels

   b. Range switches allow operator to select appropriate IRM range to maintain indications between 25 to 75 on 0-125 scale. 0-40 scale is no longer utilized.

INSTRUCTOR NOTES

(2) 'Standby' - same as operate, except gives Inop trip to yield maximum design protection before channel is removed from service.

(3) 'Zero 1' - Removes signal from output amplifier so that output amplifier, local meter and recorder can be zeroed.

(4) 'Zero 2' - Removes voltage from range switch. This deselects all ranges. This, in turn, causes no input to be sent to attenuator and allows setting the zero adjust on output amplifier.

(5) '125 '- Input is removed from attenuator same as Zero 2 position. A calibration signal is substituted which will yield 125 on the 125 scale. Used to set gain of output amplifier.

(6) '40' - Produces a 40 reading on the 125 scale.

c. INOP/INHIBIT Pushbuttons

Obj. V.B.6
Obj. V.C.4

(1) Pushed to bypass the INOP trip that results from taking mode switch S-1 out of "operate."

DCN W18726A replaced the INOP/INHIBIT Pushbutton with a toggle switch for the U-3 IRM drawers. (UNIT DIFFERENCE)

(2) Used to allow testing of other scram or rod block signals from the IRM drawer into RPS/RMCS without them being masked by the INOP trip.

8. RO 215004A3.03 001/C/A/T2G1/SRM/B8/215004A3.03//RO/SRO/BANK 11/27/07 RMS

Given the following plant conditions:

- A reactor startup is in progress following refueling, with all Reactor Protection System (RPS) Shorting Links removed.
- The reactor is approaching criticality.
- An electronic failure in the 'B' Source Range Monitor (SRM) drawer results in an SRM HIGH/HIGH output signal.

Which ONE of the following describes the plant response?

A. A Rod Out Block ONLY.

B. A Rod Out Block and 1/2 Scram.

C. A "SRM HIGH/HIGH" alarm ONLY.

D.✔ A Full Reactor Scram.

K/A Statement:

215004 Source Range Monitor
A3.03 - Ability to monitor automatic operations of the SOURCE RANGE MONITOR (SRM) SYSTEM including: RPS status

K/A Justification: This question satisfies the K/A statement by requiring the candidate to use specific plant conditions to correctly determine the response of the SRM with the shorting links removed.

References: 1-OI-92 rev. 6 pg 6 of 14, P&L 3.0.G, OPL171.019 rev 12, pg 23, g(1) & pg 24 g(2), OPL171.028 rev 17 TP-4 and OPL171.148 TP-10.

Level of Knowledge Justification: This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

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REFERENCE PROVIDED: None

Plausibility Analysis:

In order to answer this question correctly the candidate must determine the following:

1. The SRM response to a Hi/Hi output signal.
2. The SRM response to the same conditions with the shorting links removed.

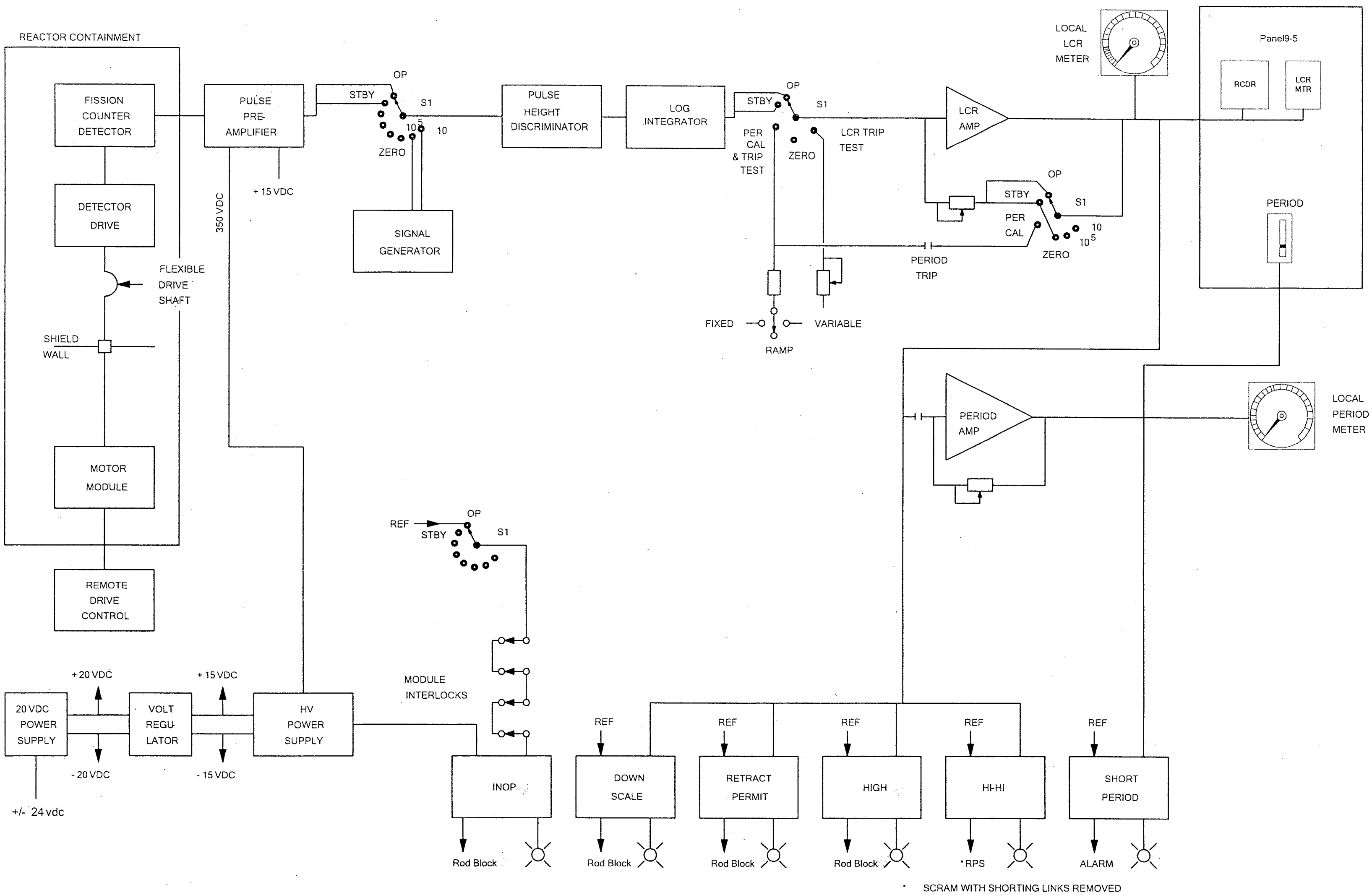
D - correct: The Reactor Protection System, in conjunction with the Neutron Monitoring System (SRM and IRM), has non-coincident trip logic IF all eight shorting links are removed. In this condition the SRM Hi-Hi will generate a scram. In addition, a Rod Out Block is also generated but will not be relevent following a full scram.

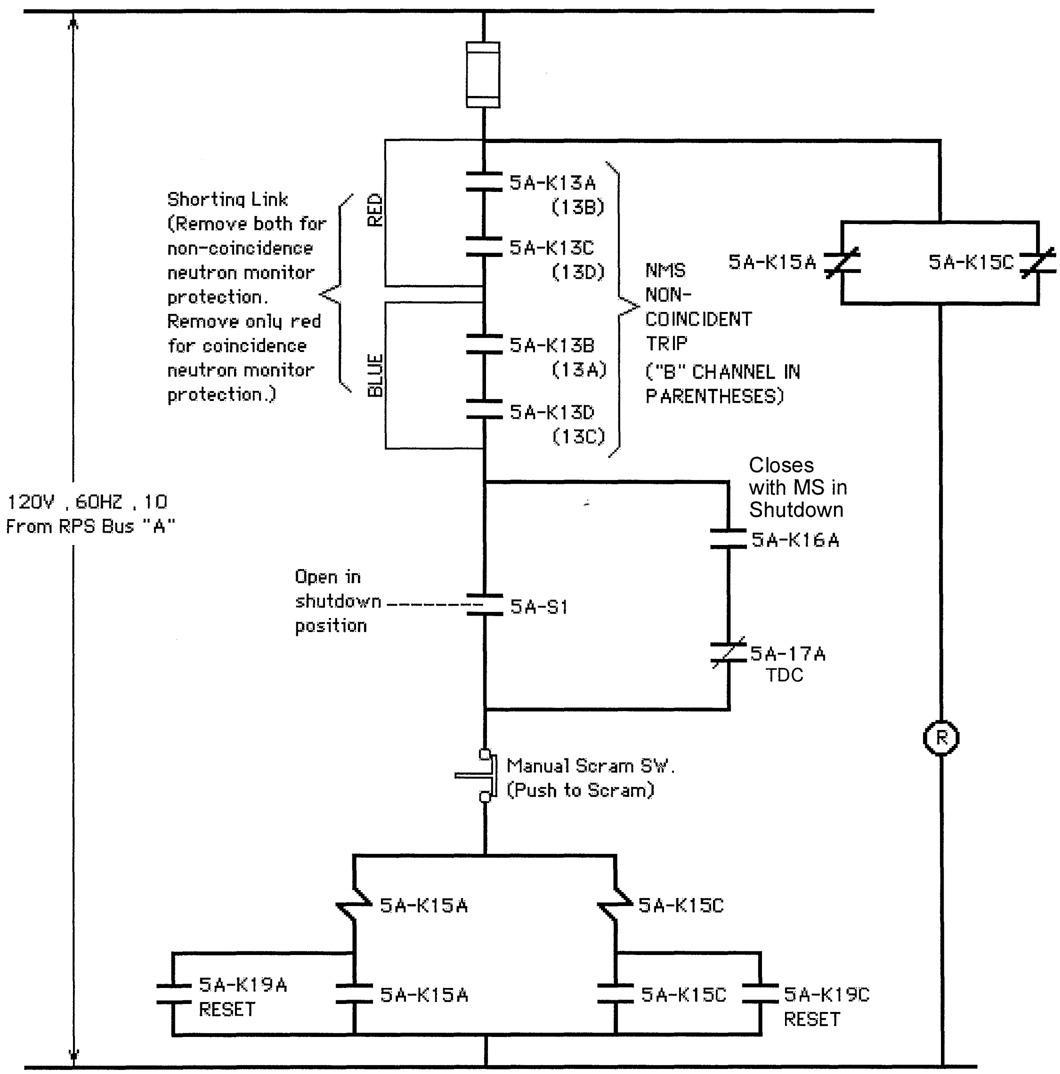
B - incorrect: This is plausible if the operator determines a scram signal is generated with the typical "1-out of-2 taken twice" logic. However, if all eight shorting links are removed, this condition will generate a full scram.

C - incorrect: In this condition the SRM Hi-Hi will generate a scram.

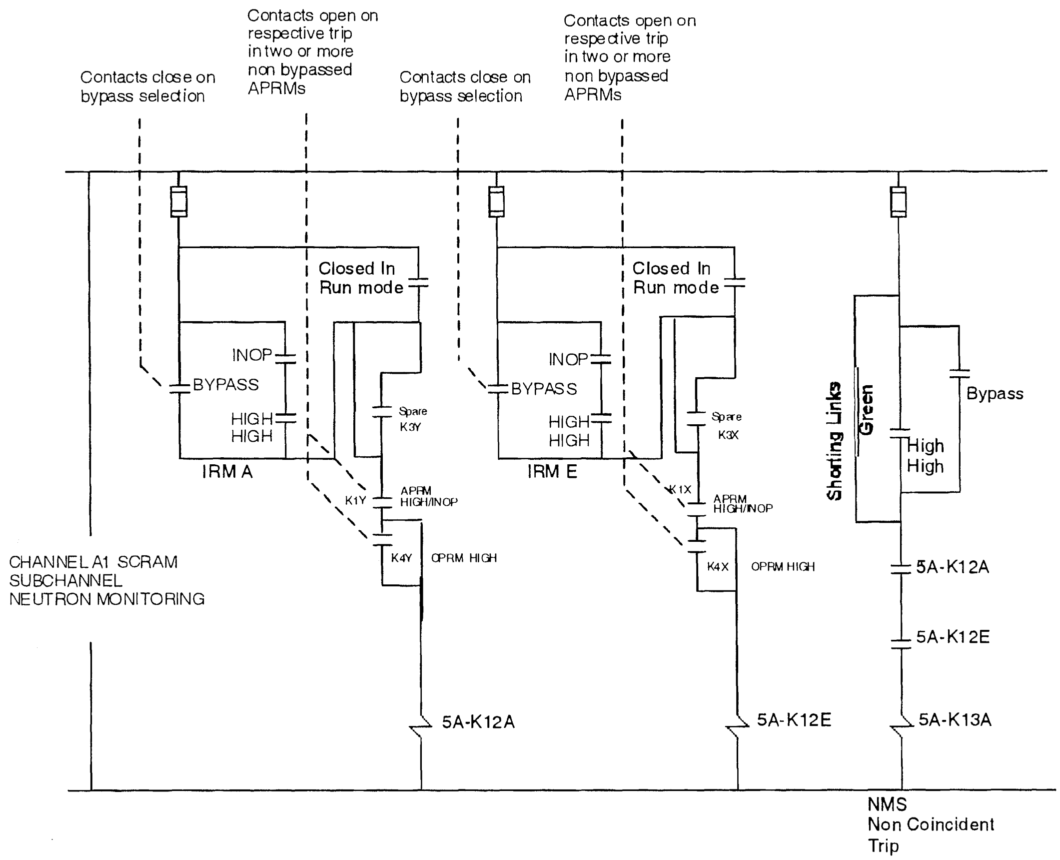
A - incorrect: This is plausible if the operator determines a scram signal is NOT generated with the shorting links removed.

TP-1: SOURCE RANGE MONITORING CHANNEL FUNCTIONAL BLOCK DIAGRAM





TP-4: RPS MANUAL TRIP CHANNEL A3 (TYPICAL FOR B3)



TP-10 TYPICAL RPS CONFIGURATION FOR NEUTRON MONITORING SYSTEM

3.0 PRECAUTIONS AND LIMITATIONS

A. To prevent a rod withdrawal block when withdrawing SRMs, SRM count rate is required to be above Retract Permit (145 counts per second) or all unbypassed IRM channels are set to Range 3 or above and indicating above their downscale trip point (7.5 on 125 scale).

B. Only one SRM channel can be bypassed at a time.

C. In order to prevent an inadvertent rod withdrawal block or Reactor scram (with shorting links removed) while operating the SRM BYPASS selector switch, 1-HS-92-7A/S3,

- Verify the previously bypassed channel returns to normal status by observing the applicable HIGH HIGH and HIGH or INOP status lights are extinguished prior to selecting any other channel to be bypassed.

- After bypassing a channel, the applicable BYPASSED status light should be illuminated prior to testing, operating, or working on that channel.

D. To prevent SRM detector drive damage, the CRD service platform should be locked in the stored position with key removed to allow free movement of SRMs.

E. In order to minimize their exposure, SRM detectors should be fully withdrawn from the core when IRMs are on range 3 or above and indicating above their downscale trip point.

F. Illustration 1 lists trip signals and associated actions for the Source Range Monitoring System.

G. The Reactor Protection System in conjunction with the Neutron Monitoring System (SRM and IRM) has non-coincident trip logic if all eight shorting links are removed. If only the yellow, green, and red shorting links (six total) are removed, the SRM High-High trips will be placed in a one-out-of-two taken twice logic.

H. The time required to drive a detector from full-out to full-in is approximately 3 minutes.

I. The INOP/INHIBIT switches, located on Panel 1-9-12 SRM drawers, bypass the SRM switch position out-of-operate trip. These switches are to be used only during testing of SRM channels.

J. [NRC/C] Upon return to service of 24 VDC Neutron Monitoring Battery A or B, Instrument Maintenance is required to perform functional tests on SRMs and IRMs that are powered from the affected battery board. [NRC IE Inspector Followup Item 86-40-03]

9. RO 215005A2.03 001/C/A/T2G1/PRNM/APRM/B7/215005A2.03//RO/SRO/BANK 11/27/07 RMS

Which ONE of the following describes the expected response due to a "FAULT" in an Average Power Range Monitor (APRM) channel and the required action(s) to address this condition?

An APRM channel ______(1)______ will result in an INOP trip to________(2)________.

To continue plant operation, bypassing the APRM ____(3)______ required.

	(1)	(2)	(3)
A.	Critical Fault;	ONLY the respective 2/4 logic module voter;	is NOT
B.	Non-critical Fault;	ONLYthe respective 2/4 logic module voter;	is NOT
C.	Non-critical Fault;	ALL four of the 2/4 logic modules voters;	is
D.✔	Critical Fault;	ALL four of the 2/4 logic modules voters;	is

K/A Statement:

215005 APRM / LPRM

A2.03 - 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 Inoperative trip (all causes).

K/A Justification: This question satisfies the K/A statement by requiring the candidate to use specific plant conditions to correctly determine equipment response and the corrective actions due to an APRM INOP trip.

References: OI-92B Precautions and Limitations 3.0.I, 1-ARP-9-5A rev. 11 window 25 and OPL171.148 rev 8, pg 31.

Level of Knowledge Justification: This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

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REFERENCE PROVIDED: None

Plausibility Analysis:

In order to answer this question correctly the candidate must determine the following:

1. The difference between a "Critical Fault" and "Non-critical Fault" with respect to PRNM response.
2. The operation of the 2/4 Logic Module Voter operation for each type of fault.
3. Based on the above, determine the appropriate course of action regarding the APRM channel in question.

D - correct: "Critical Faults" are those that affect the instrument's capability to perform its intended function and will cause an instrument INOP trip to all four 2/4 logic module voters and a trouble Alarm indication. Per the ARP it is allowable to bypass the APRM since no other APRMs are bypassed.

A - incorrect: This is plausible because a "Critical Fault" generates an INOP trip input to the 2/4 Logic Module Voters, but to all four modules, not just two. In addition, not placing the APRM in BYPASS because of a "Critical Fault" is inappropriate since any additional equipment failure could result in an unnecessary scram.

B - incorrect: A "Non-critical Fault" will not result in an INOP trip input. This is plausible because a "Non-critical Fault" generates a Trouble Alarm similar to a "Critical Fault". In addition, not placing the APRM in BYPASS because of a "Non-critical Fault" is appropriate.

C - incorrect: A "Non-critical Fault" will not result in an INOP trip input. This is plausible because a "Non-critical Fault" generates a Trouble Alarm similar to a "Critical Fault". In addition, placing the APRM in BYPASS would be correct IF an INOP trip was generated.

Amplification: will need clarification on you're "true/false" concern?

APRM
HIGH/INOP
OR OPRM TRIP

25

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Sensor/Trip Point:

APRM:

C. HIGH.
   1. (0.66W + 65%) Two Loop operation with Rx MODE Switch in Run.
   2. 14.0% with Rx Mode Switch not in RUN.
   3. 119% in any mode.
   4. (0.66(W-10%)+65%) Single loop operation with Mode Switch in RUN.

D. INOP.
   1. APRM Chassis Mode switch not in OPERATE.
   2. Loss of input power.
   3. Watchdog timer timed out.
   4. Critical self test fault detected.

OPRM TRIP: Any one of the three algorithms, period, growth, or amplitude for an operable OPRM Cell has exceeded its trip value conditions.

Sensor Location: Control Room Panel 1-9-14.

Probable Cause:

A. Flux level at or above setpoint.
B. Testing in progress.
C. Malfunction of sensor.
D. Control rod drop accident.
E. Thermal-Hydraulic instability detected

Automatic Action:

A. Input signal to all four Voters, no output to RPS.
B. Full Reactor scram if two sensors actuate.

Operator Action:

A. **VALIDATE** alarm by multiple indications ☐
B. With SRO permission, **BYPASS** initiating channel to reset the alarm. **REFER TO** 1-OI-92B. ☐
C. **IF** HIGH is indicated, **THEN** **DETERMINE** cause of inadvertent reactivity change. ☐

Continued on Next Page

b. Each LPRM instrument provides a brief description of the self-test faults which are divided into two categories, “**Critical**” and “**Non-Critical**” faults.

Obj, V,D,4

(1) **Critical** faults are those that affect the instrument's capability to perform its intended function and will cause an instrument INOP trip and a Trouble Alarm indication.

V.B.7 V.C.2 The Trouble Alarm is indicated in the Status Header for each instrument.

(2) **Non-critical** faults do not prevent the instrument from performing its intended function and will cause a Trouble Alarm indication only.

c. The LPRM instrument transmits its self-test status to its associated APRM and RBM instruments.

3.0 PRECAUTIONS AND LIMITATIONS (continued)

I. Each of the four APRM/OPRM channels input to the four Voters, such that when a signal is generated from an APRM/OPRM channel, all four Voters see and reflect that signal. Each Voter is directly associated with one RPS sub-channel.

When operating in a 2 out of 4 voting configuration, the first un-bypassed input will be seen as a single input with no trip outputs. When the second un-bypassed signal of the same type [The SAME TYPE inferring that one type is an APRM function and a different type is an OPRM function] is received it will also be seen by all four Voters resulting in a trip output from all four Voters consequently producing a full reactor scram.

J. Bypassing an APRM does not preclude testing a Voter, such that with an APRM in bypass, the Voters can still be tested and produce half scrams. Voters are not bypassed with the APRM joystick.

K. The Recirc Flow Indication and the Voters are never bypassed unless they are removed for testing. There is no bypass capability for the Recirc flow signal input or Voters.

L. A reactor scram will be produced when at least two of the SAME TYPE of trip inputs are received by the Voters:

Either: APRM HIGH/INOP {i.e., APRM High Flux/STP Flow Biased Scram/INOP}
OR: Any OPRM ABA, PBA, or GRBA algorithm trip conditions met.
The SAME TYPE inferring that one type is an APRM function and a different type is an OPRM function.

M. The new APRM modules contain an automatic power oscillation detection and suppression function (Oscillation Power Range Monitoring) which detects and protects against thermal hydraulic instabilities. OPRM monitors local cell area for thermal hydraulic core instabilities. There are 4 channels each containing 33 cells. Each cell contains up to 4 LPRM inputs per OPRM channel for power monitoring.

Oscillations are detected using any one of three algorithms; Period Based Algorithm, Growth Rate Based Algorithm, and, Amplitude Based Algorithm. When power oscillations are detected a trip signal inputs to the Voters which will in turn, send a trip output to the RPS sub-channels and will produce a trip signal. Two of these types of signals will produce a full reactor scram.

Illustration 1
(Page 1 of 6)

APRM/OPRM Trip Outputs and PRNMS Overview

APRM Trip Outputs

TRIP SIGNAL	SETPOINT	ACTION
APRM Downscale	5%	1. Rod Block if REACTOR MODE SWITCH in RUN.
APRM Inop	1. APRM Chassis Mode not in OPERATE (keylock to INOP). 2. Loss of Input Power to APRM. 3. Self Test detected Critical Fault in the APRM instrument. 4. Firmware Watchdog timer has timed out.	1. One Channel detected, no alarm or RPS output signal. 2. Two Channels detected, RPS output signal to all four Voters (Full Reactor Scram).
APRM Inop Condition	1. < 20 LPRMs in OPERATE, or < 3 LPRMs per level.	1. <20 LPRMs total or <3 per level results in a Rod Block and a trouble alarm on the display panel. This does not yield an automatic APRM trip, but does, however, make the associated APRM INOP.
APRM High	1. DLO $\leq (0.66W + 59\%)$ SLO $\leq (0.66\ (W\text{-}\Delta W) + 59\%)$ [W = Total Recirc drive flow in % rated]. 2. Neutron Flux Clamp Rod Block $\leq$ 113% 3. $\leq$ 10% APRM Flux.	1. Rod Block if REACTOR MODE SWITCH in RUN. 2. Rod Block if REACTOR MODE SWITCH in RUN. 3. Rod Block in all REACTOR MODE SWITCH positions except RUN.
APRM High High	1. a. DLO $\leq (0.66W + 65\%)$ SLO $\leq (0.66(W\text{-}\Delta W) + 65\%)$ [W = Total Recirc drive in % rated]. b. $\leq$ 119% APRM FLUX. 2. $\leq$ 14% APRM FLUX.	1. Scram 2. Scram in all REACTOR MODE SWITCH positions except RUN.
Recirc Flow Compare Recirc Flow Upscale	1. $\leq$ 5% mismatch between APRM Channels. 2. 107% Flow Monitor upscale.	1. Flow compare inverse video alarm. 2. Rod Block.

10. RO 217000K2.03 001/C/A/T2G1/RCIC/7/217000K2.03//RO/SRO/BANK 11/27/07 RMS

Given the following Unit 2 conditions:

- The Control Room has been evacuated.
- Reactor Core Isolation Cooling (RCIC) is controlling reactor water level.
- A loss of the Division I Emergency Core Cooling System Inverter (Div 1 ECCS Inverter) occurs.

Assuming no further operator action, which ONE of the following describes the RCIC system response?

The RCIC Flow Controller____.

A. lowers to minimum in auto ONLY.

B. raises to maximum in manual ONLY.

C.✓ lowers to minimum in either manual OR auto mode.

D. raises to maximum in either manual OR auto mode.

K/A Statement:

217000 RCIC
K2.03 - Knowledge of electrical power supplies to the following: RCIC flow controller

K/A Justification: This question satisfies the K/A statement by requiring the candidate to use specific plant conditions to correctly determine a loss of the logic power supply to the controller has occurred and the RCIC system response to that loss.

References: 2-OI-71 Precautions and Limitations 3.0.W, OPL171.040 rev 22, pg 34, 8.a(1) & (2)

Level of Knowledge Justification: This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

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REFERENCE PROVIDED: None

Plausibility Analysis:

In order to answer this question correctly the candidate must determine the following:

1. The current power supply to the RCIC controller while being operated from the Backup Control Panel, 25-32.
2. The Power Transfer Switch (XS-256-1) is NOT part of the RCIC initiation procedure in 2-AOI-100-2, "Control Room Abandonment".
3. The failure mode of the Yokogawa Flow Controller used for RCIC while operating from the Backup Control Panel, 25-32.

NOTE: Due to the widespread use, at BFN, and various failure modes of Yokogawa Flow Controllers, each of the four answers become plausible. These controllers can be set to fail "as-is", "fail high" or "fail low" depending on the system and application.

C - correct: Loss of Power to the Flow Controller (Div I ECCS ATU Inverter) results in the controller output dropping to zero milliamps and turbine speed would lower to minimum (600 rpm). However, on Unit **3**, the Div I ECCS Inverter also powers to EGM Control Box which would result in overspeed on Unit 3 ONLY.

A - incorrect: The RCIC controller at Panel 25-32 fails low in either AUTO or MANUAL.

B - incorrect: The RCIC controller at Panel 25-32 fails low.

D - incorrect: This is plausible due to the effect it would have on Unit 3, versus Unit 2.

Amplification: As stated in P&L 'W', the power supply was changed. It did change the answer in the original question; so the stem was modified to include the correct power supply. This is considered a BANK question because none of the distractors were modified.

RCIC PUMP SUCTION PRESSURE	PI-71-20A	0-50 psig	
RCIC TURBINE STEAM LINE PRESSURE	PI-71-4A	0-1500 psig	
RCIC TURBINE EXHAUST PRESSURE	PI-71-12A	0-50 psig	BFPER971133 Indicator could read from 0-200 rpm in standby readiness due to non-linearity in low RPM range
RCIC TURBINE SPEED	SI-71-42A	0-6000 rpm	
RCIC TURBINE STEAM FLOW	FI-71-1A FI-71-1B	0-80 lbm x1000	

2. Flow Controller (FIC-71-36A & B)

   a. One located on Panel 9-3 and one on Panel 25-32 (Remote Shutdown Panel).

   b. Power Supply to the Pnl 9-3 controller (FIC-71-36A) is the Div I ECCS Inverter — Obj. V.B.7.

   c. Power Supply to the Pnl 25-32 Controller (FIC-71-36B) is also the Div I ECCS Inverter.

   d. AT Pnl 25-32, there is a power transfer switch (XS-256-1) which, if placed in the Alternate position, will transfer both (36A & B) Flow Controller power supplies from the Div I ECCS Inverter to the Unit Preferred 120VAC Power Supply. — TP-13

3. Yokogawa Flow Controller

   a. AUTO - output signal is changed by changing the setpoint. Full Scale travel of setpoint is 40 seconds. Momentary depressing of either the raise or lower keys will cause ~0.7 gpm change (~1%).

8. Failure Modes

a. Loss of Power to the Flow Controller — Obj. V.C.4

(1) Div I ECCS ATU Inverter

(2) Loss of Power causes the controller to go to zero milliamp output and turbine speed would lower to minimum (~600 rpm). However, on U3, the Div I ECCS Inverter also powers to EGM Control Box which would result in overspeed on Unit 3 only. — Reference P&L 3.23 / 3.0.W

b. Loss of control air — Obj.V.B.7 Obj. V.C.4

(1) RCIC steam line steam trap bypass valve (FCV-71-5) fails closed (Unit 3)

(2) RCIC steam line condensate drain valves (FCV 71-6A and 6B) fail closed

(3) RCIC condensate pump Clean Radwaste discharge valves (FCV-71-7A and 7B) fail closed

c. Loss of electrical power to valves — Obj. V.B.7. Obj. V.C.4.

All motor-operated isolation valves remain in the last position upon failure of valve power. Solenoid operated valve FCV 71-5 (Unit 2) fails closed.

d. Loss of Power to Relay Logic — Obj. V.B.7. Obj. V.C.4

(1) If Bus A fails, the automatic initiation circuit and turbine trip solenoid will not operate. Channel A isolation logic circuit is lost. Power is lost to EG-M control box and this causes FCV-71-10 trip governor valve to go wide open (if RCIC is operating). - Unit 2 (Unit 3 EGM power is from DIV I Inverter) — UNIT DIFFERENCE

(2) If power is lost to the EGM Control Box, Springs will re-position the 71-10 servo to fully open the governor valve (Unit 2 only). — UNIT DIFFERENCE

4.2 Unit 2 Subsequent Actions (continued)

NOTES
1) Attachment 1 provides normal backup control stations and available communications.
2) Attachment 10 provides PAX extensions and locations.

[7] **ESTABLISH** communication with the following personnel and **DIRECT** attachments be completed as follows:

- U-2 Unit Operator complete Attachment 2, Part A. ☐
- U-2 Rx Bldg AUO complete Attachment 3, Part A. ☐
- U-2 Turb Bldg AUO complete Attachment 4, Part A. ☐

CAUTION
RCIC TURBINE STEAM SUPPLY VALVE, 2-FCV-71-8, transfer switch has been placed in EMERGENCY and will **NOT** trip on Reactor Water Level High (+51 inches). Failure to maintain level below this value may result in equipment damage.
RCIC will still trip on low suction pressure, high turbine exhaust pressure, mechanical overspeed, and trip push button on pnl 25-32.

[8] Upon completion of attachments, **RE-ESTABLISH** communication using the best available means and continue procedure. ☐

★ NO REQUIREMENT TO TRANSFER RCIC CONTROLLER POWER SUPPLY.

[9] **INITIATE** RCIC as follows:

[9.1] At Panel 2-25-32, **CHECK OPEN** 2-FCV-71-9 (Red Light above switch) RCIC TURB TRIP/THROT VALVE RESET, 2-HS-71-9D. ☐

[9.2] At 250V DC RMOV Bd 2B, compt. 5D, **PLACE** RCIC PUMP MIN FLOW VALVE EMER HAND SWITCH, 2-HS-071-0034C, in OPEN. (Unit 2 Turbine Building AUO) ☐

[9.3] At 250V DC RMOV Bd 2C, compt. 4B, **PLACE** RCIC TURB STM SUPPLY VALVE EMER HAND SWITCH, 2-HS-071-0008C, in OPEN. (Unit 2 Reactor Building AUO) ☐

4.2 Unit 2 Subsequent Actions (continued)

NOTE
RCIC Turbine should start and flow should stabilize at 600 gpm.

[9.4] At Panel 2-25-32, **CHECK** turbine speed 2100 rpm or above using RCIC TURBINE SPEED, 2-SI-71-42B. ☐

[9.5] At 250V DC RMOV Bd 2B, compt. 5D, **PLACE** RCIC PUMP MIN FLOW VALVE EMER HAND SWITCH, 2-HS-071-0034C, in CLOSE. (Unit 2 Turbine Building AUO) ☐

[9.6] At Panel 2-25-32, **ADJUST** flowrate as necessary using RCIC SYSTEM FLOW/CONTROL, 2-FIC-71-36B. ☐

[9.7] At Panel 2-25-32, **MAINTAIN** Reactor Water Level between +2 and +50 inches using RX WATER LEVEL A & B, 2-LI-3-46A & B. ☐

NOTE
The following step prevents HPCI operation and automatic opening of HPCI MAIN PUMP MINIMUM FLOW VALVE, 2-FCV-73-30.

[10] At 250V Reactor MOV Bd 2A, **PERFORM** the following:

[10.1] Compt. 3D, **VERIFY CLOSED** HPCI STEAM SUPPLY VALVE TO TURB FCV-73-16 (MO 23-14). ☐

[10.2] Compt. 3D, **PLACE** HPCI TURBINE STEAM SUP VLV TRANS, 2-XS-73-16, in EMERG. ☐

[10.3] **IF** desired to verify HPCI MIN FLOW BYPASS TO SUPPRESSION CHAMBER VALVE, 2-FCV-73-30, closed prior to opening breaker, **THEN**

DIRECT operator to verify locally. ☐

[10.4] Compt. 8D, **PLACE** HPCI MAIN PUMP MIN FLOW VLV FCV-73-30, breaker in OFF. ☐

3.0 PRECAUTIONS AND LIMITATIONS (continued)

Q. Suppression pool water temperature should not exceed 95°F without suppression pool cooling in service to restore temperature to less than or equal to 95°F within 24 hours.

R. RCIC Testing is NOT permitted with suppression pool water temperature above 105°F.

S. After RCIC steam lines have been hydrostatically tested, leak tested, or exposed to other conditions which could fill the 2-FCV-71-2 valve bonnet with water, 2-FCV-71-2 should be cycled to prevent overpressurization.

T. [II/F] Prior to initiating any event which adds, or has the potential to add, heat energy to the suppression chamber, the Unit Supervisor will evaluate the necessity of placing suppression pool cooling in service. This is due to the potential of developing thermal stagnation during sustained heat additions. [II-B-91-129]

U. Calculations have shown that 16 min. of RCIC operation without RHR operating in the Suppression Pool Cooling Mode will result in a one deg F rise in bulk suppression pool temperature.

V. [NER/C] Extended RCIC System operation may raise suppression chamber O_2 concentration above TRM 3.6.2 limits because of air-inleakage from RCIC Turbine Gland Seal System. [GE SIL 548]

W. Whenever the 1E ECCS ATU Inverter (Division I) becomes INOP, RCIC Is considered INOP. DCN W17726B changed power supply for RCIC flow controller from 1E Unit Preferred MMG set busses to the Unit 2 1E ECCS ATU Inverter (Division I).

X. The RCIC STEAM LINE OUTBD ISOLATION VLV hand switch, 2-HS-71-3A, must be held in the OPEN position until 2-FCV-71-3 is fully open because the open seal-in circuit has been removed per ECN P0161.

Y. [INPO/C] A buildup of corrosion products in the RCIC TURBINE CONTROL VALVE stem packing could result in speed oscillations, failure to control at the desired speed, and mechanical overspeed of the RCIC Turbine. During operation, RCIC Turbine parameters such as time to reach operating speed, speed stability, and governor response should be monitored to identify possible corrosion product buildup in the RCIC TURBINE CONTROL VALVE. [INPO SER 95004]

Z. (II/C)During routine plant evolutions, notify RADCON prior to making changes in the RCIC System which could cause a rise in area radiation levels. Confirm RADCON has implemented appropriate radiological controls/barriers for the expected RCIC System alignment prior to performing the alignment. (BFPER961778)

11. RO 218000K1.05 001/MEM/T2G1/100-2/5/218000K1.05//RO/SRO/BANK 11/27/07 RMS

Given the following plant conditions:

- The Unit 1 and 2 Control Rooms have been abandoned.
- ALL Panel 25-32 Safety Relief Valve (SRV) Transfer Switches have been placed in EMERGENCY.
- ALL Panel 25-32 SRV Control Switches have been verified in CLOSE.

Which ONE of the following describes the operation of the SRVs associated with the Automatic Depressurization System (ADS)?

The associated ADS valves will OPEN ___________________________________.

A. upon receipt of an ADS initiation signal.

B.✓ in Safety mode, if their respective pressure setpoints are exceeded.

C. in Relief mode, if reactor pressure exceeds the relief mode setpoint.

D. if their respective control switches, on Panel 9-3, are placed in OPEN.

K/A Statement:

218000 ADS
K1.05 - Knowledge of the physical connections and/or cause- effect relationships between AUTOMATIC DEPRESSURIZATION SYSTEM and the following: Remote shutdown system: Plant-Specific

K/A Justification: This question satisfies the K/A statement by requiring the candidate to use specific plant conditions to correctly determine their effect on MSRV operation during a Remote Shutdown condition.

References: OPL171.208 Rev. 5 page 8, 2-AOI-100-2 page 8 and OPL171.009 rev 10 TP-4.

Level of Knowledge Justification: This question is rated as MEM due to the requirement to recall or recognize discrete bits of information.

0610 NRC Exam

REFERENCE PROVIDED: None

Plausibility Analysis:

In order to answer this question correctly the candidate must determine the following:

1. Transferring the SRV control to Panel 25-32 disables the ADS function.
2. Transferring the SRV control to Panel 25-32 disables the Panel 9-3 control switch.
3. Transferring the SRV control to Panel 25-32 does NOT disable the Pressure Relief function.

B - correct: Transferring the SRV control to Panel 25-32 disables the ADS function and the Panel 9-3 control switch. The SRV's will still operate in safety mode but not relief mode, ie. the pressure switches that input into the logic are bypassed when control is transferred to panel 25-32.

A - incorrect: Transferring the SRV control to Panel 25-32 disables the ADS function.

C - incorrect: The pressure switches that input into the releif mode logic are bypassed when control is transfered to panel 25-32.

D - incorrect: Transferring the SRV control to Panel 25-32 disables the Panel 9-3 control switch.

Amplification: Changed distractor 'C' ('D' from previous NRC comments) to make it more plausible.

INSTRUCTOR NOTES

9. Trip reactor feed pumps as necessary to prevent tripping on high water level. — Obj. V.B.8; Obj. V.C.5

10. Start the diesel generators. (9-8 Switch starts respective units D/G only)

11. Verify each EECW header has one pump in service.

12. Announce to all plant personnel that the Control Room is being evacuated and all operators are to report to their assigned backup control stations.

13. Obtain hand held radios from the control room.

14. Proceed to the Backup Control Panel (25-32)

F. Subsequent Actions

See AOI-100-2 for details for actions
HU Tools: Procedure Use

1. If rods failed to fully insert and RPS did not deenergize, an operator is directed to pull RPS fuses. However, this is beyond the actual design bases.

Obj V.C.2
See AOI-100-2 Attachment 11

2. Transfer reactor pressure control to Panel 25-32 to allow for pressure control while the rest of the panel checklist is being completed.

Note: System Status prior to abandonment maintained by GOI-300-1 checklists.

3. Before any transfer switch is placed in EMERGENCY, its associated control switch must be verified to be in the proper position. Placing a transfer switch in the EMERGENCY position enables the local control switch, and the device will assume the condition called for by the local control switch. For example, if a transfer switch for an ADS valve is placed in EMERGENCY with the local control switch in OPEN, the ADS valve will open.

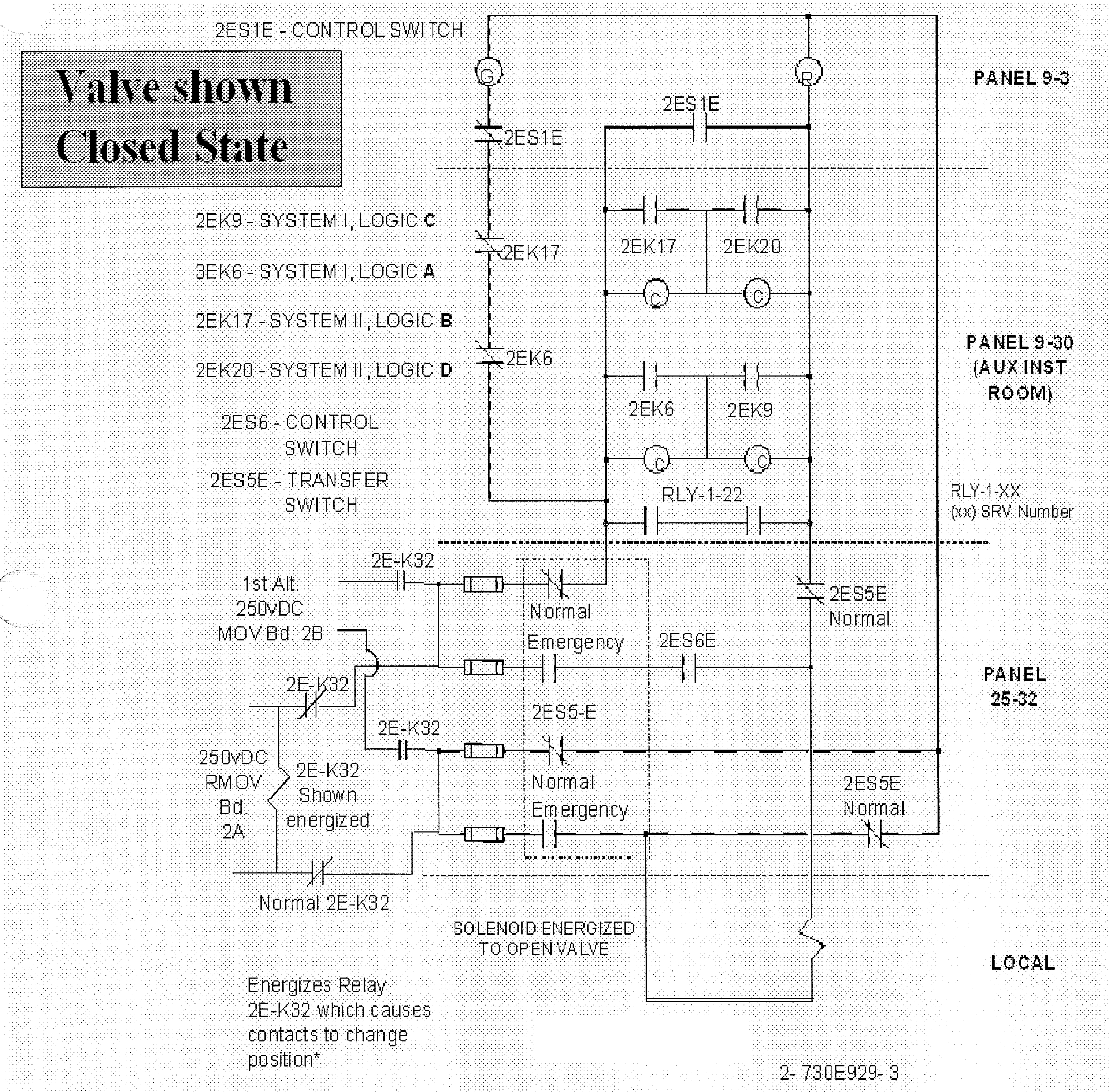
Obj. V.B.2
Obj. V.B.3.

a. Place the transfer switches for the ADS valves, and the disconnect switches for the non-ADS valves in EMERGENCY after making sure the control switches are in the AUTO position. This action disables the Control Room hand switches and the ADS function and is performed to prevent spurious blowdown of the primary system. The other 3 SRVs are disabled by opening their breakers on 250VDC RMOV board 2B(3B).

TP-1
Obj. V.B.7

Four ADS valves can be controlled from Panel 25-32. Six SRVs (Non-ADS) have only disconnect switches at Panel 25-32.

Obj. V.B.8
Obj. V.B.7



TP-4: ADS VALVE CONTROL CIRCUIT PCV 1-22

4.2 Unit 2 Subsequent Actions

[1] **IF** ALL control rods were **NOT** fully inserted **AND** RPS failed to deenergize, **THEN** (Otherwise N/A)

DIRECT an operator to Unit 2 Auxiliary Instrument Room to perform Attachment 11. ☐

NOTES

1) The following transfers Reactor Pressure Control to Panel 2-25-32 to allow for pressure control while completing the Panel Checklist.
2) Attachment 9, Alarm Response Procedure Panel 2-25-32, provides for any alarms associated with this instruction.

CAUTION

Failure to place control switch in desired position prior to transferring to emergency position may result in inadvertent actuation of the component.

[NER/C] Operation from Panel 2-25-32 bypasses logic and interlocks normally associated with the components. [GE SIL 326,S1]

[2] At Panel 2-25-32, **PLACE** the following MSRV control switches in CLOSE/AUTO:

Switch No.	Description	
2-HS-1-22C	MAIN STM LINE B RELIEF VALVE	☐
2-HS-1-5C	MAIN STM LINE A RELIEF VALVE	☐
2-HS-1-30C	MAIN STM LINE C RELIEF VALVE	☐
2-HS-1-34C	MAIN STM LINE C RELIEF VALVE	☐

Given the following plant conditions:

- Unit 2 is at rated power.
- A loss of '2A' 250 Volt RMOV Board has occurred.

Which ONE of the following describes the effect on the Unit 2 Automatic Depressurization System (ADS) valves and ADS logic?

A. BOTH Div I & II ADS logics are still operable.
ALL ADS valves will operate automatically.
ALL ADS valves can be manually operated.

B. BOTH Div I & II ADS logic is NOT operable.
NO ADS valves will operate automatically.
Four (4) ADS valves can still be operated manually.

C.✓ Div I ADS logic operable; Div II ADS logic is NOT operable.
ALL ADS valves will still actuate automatically.
ALL ADS valves can still be operated manually.

D. Div I ADS logic operable, Div II ADS logic is NOT operable.
ADS logic is ONLY capable of opening 4 ADS valves automatically.
Four (4) ADS valves can still be operated manually.

K/A Statement:

218000 ADS
2.1.24 - Conduct of Operations Ability to obtain and interpret station electrical and mechanical drawings

K/A Justification: This question satisfies the K/A statement by requiring the candidate to recall and interpret the electrical logic drawing of the ADS system to determine the effect of a loss of power to that logic.

References: OPL171.043 rev, 13 pg 16 f. through n. and TP-2

Level of Knowledge Justification: This question is rated as MEM due to the requirement to recall or recognize discrete bits of information.

0610 NRC Exam

REFERENCE PROVIDED: None

Plausibility Analysis:

In order to answer this question correctly the candidate must determine the following:

1. '2B' 250V RMOV Board supplies Div 1, '2A' 250V RMOV Board supplies Div II. This is the opposite of conventional logic.
2. '2A' 250V RMOV Board supplies only relay power and only in Div II.
3. '2B' 250V RMOV Board supplies BOTH Div I and Div II logic.
4. Four ADS valves have an alternate power supply that is NOT '2B' 250V RMOV Board.
5. 'PCV-1-22 & 1-30' are fed from '2A' 250V RMOV Board but auto transfer to an alternate power supply.

C - correct: The power supply for the LOGIC and the solenoid valves is 250VDC. 250V RMOV Bd 'A' supplies Power for relays in system II of ADS Logic. A Loss of 250V RMOV Bd 'A' would prevent system II actuation. All ADS valves would still Actuate via Division I logic.

A - incorrect: Div II logic is inoperable.

B - incorrect: Only Div II ADS logic is inoperable. All ADS valves will still operate out of the other division. This is plausible if the operator believes that '2A' vs '2B' 250V RMOV Board fed both division logics.

D - incorrect: All ADS valves would function in automatic and manually.

Duplicated and modified RO 218000G2.1.24 #1 due to it being used on the 0610 audit exam. Changed to a loss of '2A' 250V RMOV BD

f. The power supply for the LOGIC and the solenoid valves is 250VDC

g. 250V RMOV Bd B supplies LOGIC Power for both system I & II

h. A Loss of 250V RMOV Bd B would prevent actuation

i. 250V RMOV Bd A supplies Power for relays in system II of ADS Logic

All ADS valves with alternate power supplies can be manually operated from backup control panel (25-32)

j. A Loss of 250V RMOV Bd A would prevent system II actuation

k. PCV 1-22 is powered from 250V RMOV Board 2A with alternate supply from 250V RMOV Board 2B

See section F. Unit Differences for U-3 Power Supplies

l. PCVs 1-19, 1-31 are powered from 250V RMOV Board 2B. There is no alternate power to these valves

m. PCVs 1-5 and 1-34 are normally powered from 250V RMOV Board 2C with alternate power supply from Battery Board 1 panel

n. PCV 1-30 is normally powered from 250V RMOV Board 2A with a first alternate to 250V RMOV Board 2C and a second alternate to Battery Board 1 panel 7

o. Valves powered from 250V RMOV Bd 2C required alternate sources due to RMOV Board 2C not being environmentally qualified for a line break in secondary containment

p. The transfer occurs automatically when undervoltage relays (mounted on panel 2-25-32) sense a loss of power to 250V RMOV Bd 2

DCN 51106

B. Instrumentation

1. SRV discharge piping temperatures are measured by a multipoint recorder in the Control Room located on Panel 9-47 (range 0-600°F)

INSTRUCTOR NOTES

4) EOIs will direct the operator when this action is appropriate. Both keylocks must be placed in inhibit to prevent ADS blowdown — FLAGGING

5) ADS Logic can be inhibited by removing fuses in Panel 9-30 in Auxiliary Instrument Room.

6) The fuses for "A" logic are on terminal block "BB 104 & 105" (FU2-1-2EK3)

7) The "B" logic fuses are on terminal "AA94 & 95" (FU2-1-2EK13)

8) The time delay setting is chosen to be long enough so that HPCI has time to start and yet not so long that Core Spray and LPCI are unable to adequately cool the fuel if HPCI should fail to start — 3-WAY COMMUNICATIONS

e. The 100 psig and 185 psig ECCS interlocks are provided to ensure that there is a vessel level inventory medium available prior to initiating blowdown of steam from the vessel — Obj. V.B.4, Obj. V.C.3, Obj. V.C.4, Obj. V.D.3, Obj. V.E.4

6. ADS Trip Systems — Obj. V.B.4, Obj. V.C.5, Obj. V.D.5, Obj. V.E.4

a. Redundant trip systems from the same power supply

b. A/C interlock ensures ADS functions when needed

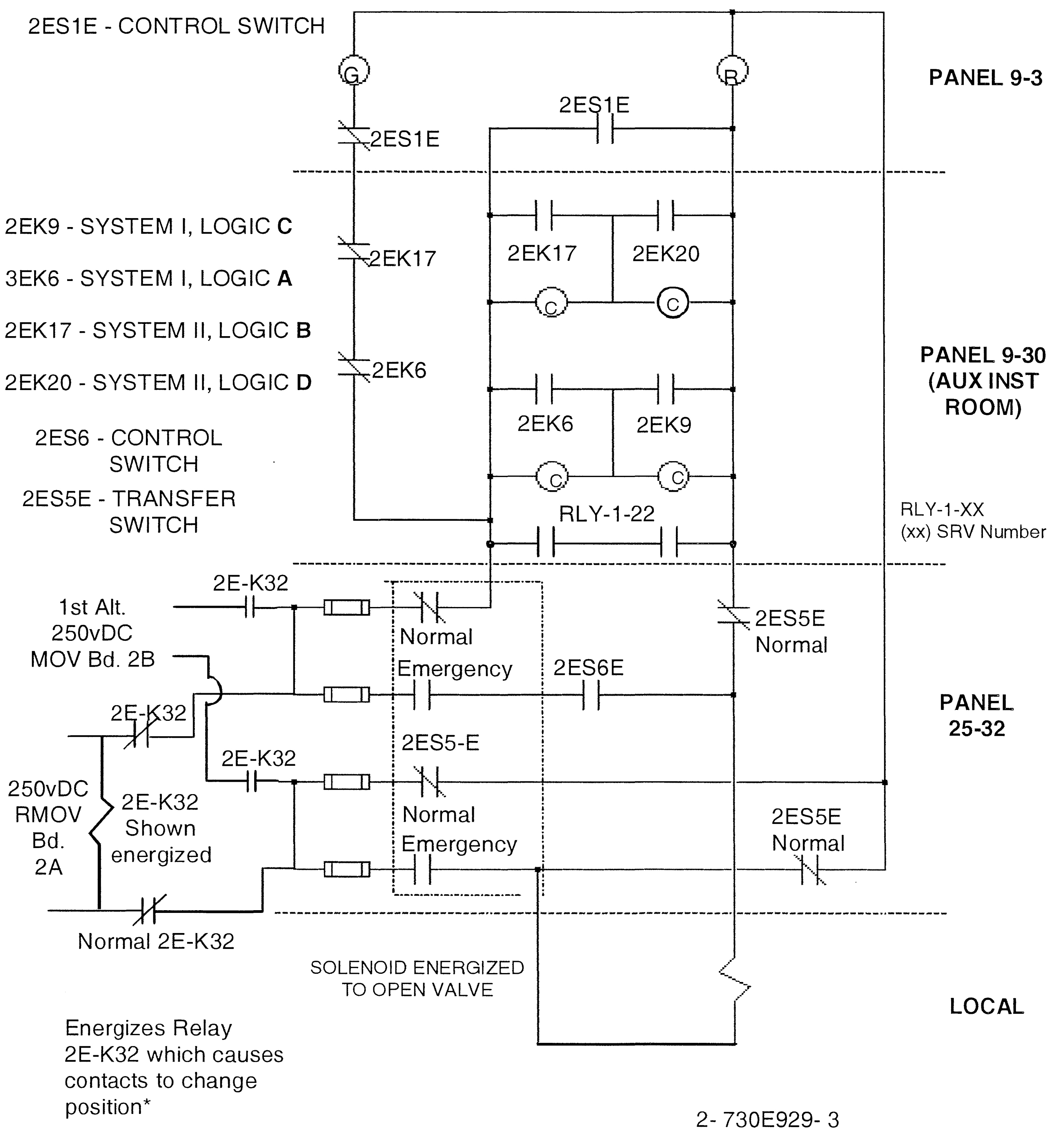
c. There are two channels in each trip system

1) A and C in System I

2) B and D in System II

d. Both channels of a trip system are required to function to initiate ADS from a given trip system

e. This two-channel interlock is called the A-C interlock and is provided to ensure that all signals to initiate ADS response are confirmed, thus preventing an ADS response from an erroneous or failed signal — Obj. V.B.5, Obj. V.C.5, Obj. V.D.5, Obj. V.E.5



TP-2 ADS Valve Control Circuit PCV-1-22

13. RO 223002A2.06 001/C/A/T2G1/PCIS//223002A3.01//RO/SRO/BANK

Given the following plant conditions:

- During performance of 2-SR-3.3.1.1.13, "Reactor Protection and Primary Containment Isolation Systems Low Reactor Water Level Instrument Channel B2 Calibration," 2-LIS-3-203D fails to actuate.
- It is determined that the failure is due to an inoperable switch and a replacement is NOT available for 4 days.
- The Shift Manager has determined that the proper action is to trip the inoperable channel ONLY.

Which ONE of the following describes how this is accomplished and the effect on Unit status?

A.✓ Remove the fuse associated with 2-LIS-3-203D. A half scram will result and NO Primary Containment Isolation Valves will realign.

B. Remove the fuse associated with 2-LIS-3-203D. A half scram will result and PCIS Groups 2, 3 and 6 Inboard Isolation Valves will close.

C. Place a trip into the Analog Trip Unit associated with 2-LIS-3-203D. NO half scram will result and NO Primary Containment Isolation Valves will realign.

D. Place a trip into the Analog Trip Unit associated with 2-LIS-3-203D. A half scram will result and PCIS Groups 2, 3 and 6 Outboard Isolation Valves will close.

K/A Statement:

223002 PCIS/Nuclear Steam Supply Shutoff
A2.06 - Ability to (a) predict the impacts of the following on the PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abn cond or ops. Containment instrumentation failures

K/A Justification: This question satisfies the K/A statement by requiring the candidate to use specific plant conditions to determine the effect of an instrumentation failure and the corrective actions required as a result of that failure.

References: 2-OI-99 Illustration 3 (page 6 of 11)

Level of Knowledge Justification: This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

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REFERENCE PROVIDED: None

Plausibility Analysis:

In order to answer this question correctly the candidate must determine the following:

1. Recognize the appropriate action is to ensure trip input by de-energizing the level switches.
2. Recognize the effect on RPS logic based on the trip input.
3. Recognize the effect on PCIS logic based on the trip input.

A - correct: Pulling the fuse associated with LIS-3-203D deenergizes th 'B2' Channel of RPS causing a 1/2 scram in RPS 'B' and a 1/4 isolation in PCIS. No PCIS valves will reposition.

B - incorrect: PCIS logic causes a "1/4-isolation" signal BUT no PCIV devices actuate. This is plausible because the action to ensure the trip input is correct. In addition, the "1/4-isolation" applies to the PCIS groups identified in the distractor.

C - incorrect: The method of inputting the trip is incorrect. Tripping an ATU cannot be ensured via a clearance. This is plausible because the RPS and PCIS response is correct.

D - incorrect: The method of inputting the trip is incorrect. Tripping an ATU cannot be ensured via a clearance. This is plausible because the "1/4-isolation" applies to the PCIS groups identified in the distractor. This distractor is also similar to answer "B;" except it is applied to outboard PCIVs.

Amplification: Basically, when a signal received from a sensor reaches the ATU, it is compared to the setpoint value programmed into the ATU. When the setpoint is exceeded, the ATU transmits a TRIP output to the associated logic, whether RPS or PCIS. Essentially converting an ANALOG input into a DIGITAL output. The ATU has a feature which allows it to generate a TRIP output for testing purposes. It could serve the same function as removing power from the input sensor, but is not considered reliable enough to comply with Tech Spec requirements for "placing a channel in the TRIP condition."

Illustration 3
(Page 6 of 11)

Actions to Place RPS Instruments in Tripped Conditions (TS Table 3.3.1.1-1)

DEVICE	FUSE	RELAY	PANEL	PRINT	ALARMS	REMARKS
2-LIS-3-203A RX WATER LEVEL LOW (Level 3) A1 CHANNEL Function: 4	2-FU1-3-203AA (5AF6A)	2-RLY-099-05AK06A 2-RLY-099-5A-K25A 2-RLY-064-16AK5A 2-RLY-064-16AK6A	9-15	2-730E915-9 2-730E927-7 2-45E671-26	2-XA-55-4A-2 RX VESSEL WTR LEVEL LOW HALF SCRAM 2-XA-55-5B-1 REACTOR CHANNEL A AUTO SCRAM	ALARMS AND 1/2 SCRAM IN CHANNEL A NO PCIS DEVICES ACTUATE. 1 channel actuated for secondary containment and CREV initiation
2-LIS-3-203B RX WATER LEVEL LOW (Level 3) B1 CHANNEL Function: 4	2-FU1-3-203BA (5AF6B)	2-RLY-099-05AK06B 2-RLY-099-5A-K25B 2-RLY-064-16AK5B 2-RLY-064-16AK6B	9-17	2-730E915-10 2-730E927-8 2-45E671-38	2-XA-55-4A-2 RX VESSEL WTR LEVEL LOW HALF SCRAM 2-XA-55-5B2 REACTOR CHANNEL B AUTO SCRAM	ALARMS AND 1/2 SCRAM IN CHANNEL B NO PCIS DEVICES ACTUATE. 1 channel actuated for secondary containment and CREV initiation
2-LIS-3-203C RX WATER LEVEL LOW (Level 3) A2 CHANNEL Function: 4	2-FU1-3-203CA (5AF6C)	2-RLY-099-05AK06C 2-RLY-099-5A-K25C 2-RLY-064-16AK5C 2-RLY-064-16AK6C	9-15	2-730E915-9 2-730E927-7 2-45E671-32	2-XA-55-4A-2 RX VESSEL WTR LEVEL LOW HALF SCRAM 2-XA-55-5B-1 REACTOR CHANNEL A AUTO SCRAM	ALARMS AND 1/2 SCRAM IN CHANNEL A NO PCIS DEVICES ACTUATE. 1 channel actuated for secondary containment and CREV initiation
2-LIS-3-203D RX WATER LEVEL LOW (Level 3) B2 CHANNEL Function: 4	2-FU1-3-203DA (5AF6D)	2-RLY-099-05AK06D 2-RLY-099-5A-K25D 2-RLY-064-16AK5D 2-RLY-064-16AK6D	9-17	2-730E915-10 2-730E927-8 2-45E671-44	2-XA-55-4A-2 RX VESSEL WTR LEVEL LOW HALF SCRAM 2-XA-55-5B2 REACTOR CHANNEL B AUTO SCRAM	ALARMS AND 1/2 SCRAM IN CHANNEL B NO PCIS DEVICES ACTUATE. 1 channel actuated for secondary containment and CREV initiation

NOTE:

Device Function corresponds to the TS Table 3.3.1.1 Functions.

Illustration 2
(Page 1 of 10)

Actions to Place PCIS in Tripped Condition

NOTE

Water level designators (1-8)are listed for relationship to the applicable device only.

(T.S. Tables 3.3.6.1-1,3.3.6.2-1, & 3.3.7.1-1)

DEVICE	FUSE	RELAY	PANEL	PRINT	ALARM	REMARKS
2-LIS-3-203A RX WATER LEVEL LOW (Level 3)	2-FU1-3-203AA (5A-F6A)	5AK6A 5AK25A 16AK5A 16A6A	9-15	2-730E915-9 2-730E927-7 2-45E671-26	2-XA-55-4A-2 RX VESSEL WTR LEVEL LOW HALF SCRAM 2-XA-55-5B-1 REACTOR CHANNEL A AUTO SCRAM	ALARMS AND 1/2 SCRAM IN CHANNEL A. CAUSES 1/4 ISOLATION IN PCIS GROUPS 2,3, 6 AND 8. NO PCIS DEVICES ACTUATE.
2-LIS-3-203B RX WATER LEVEL LOW (Level 3)	2-FU1-3-203BA (5A-F6B)	5AK6B 5AK25B 16AK5B 16AK6B	9-17	2-730E915-10 2-730E927-8 2-45E671-38	2-XA-55-4A-2 RX VESSEL WTR LEVEL LOW HALF SCRAM 2-XA-55-5B-2 REACTOR CHANNEL B AUTO SCRAM	ALARMS AND 1/2 SCRAM IN CHANNEL B. CAUSES 1/4 ISOLATION IN PCIS GROUPS 2,3, 6 AND 8. NO PCIS DEVICES ACTUATE.
2-LIS-3-203C RX WATER LEVEL LOW (Level 3)	2-FU1-3-203CA (5A F6C)	5AK6C 5AK25C 16AK5C 16AK6C	9-15	2-730E915-9 2-730E927-7 2-45E671-32	2-XA-55-4A-2 RX VESSEL WTR LEVEL LOW HALF SCRAM 2-XA-55-5B-1 REACTOR CHANNEL A AUTO SCRAM	ALARMS AND 1/2 SCRAM IN CHANNEL A. CAUSES 1/4 ISOLATION IN PCIS GROUPS 2,3, 6 AND 8. NO PCIS DEVICES ACTUATE.
2-LIS-3-203D RX WATER LEVEL LOW (Level 3)	2-FU1-3-203DA (5A-F6D)	5AK6D 5AK25D 16AK5D 16AK6D	9-17	2-730E915-10 2-730E927-8 2-45E671-44	2-XA-55-4A-2 RX VESSEL WTR LEVEL LOW HALF SCRAM 2-XA-55-5B-2 REACTOR CHANNEL B AUTO SCRAM	ALARMS AND 1/2 SCRAM IN CHANNEL B. CAUSES 1/4 ISOLATION IN PCIS GROUPS 2,3, 6 AND 8. NO PCIS DEVICES ACTUATE.
Table 3.3.6.1-1: Function 2a and 5h Table 3.3.6.2-1: Function 1 Table 3.3.7.1-1: Function 1						

14. RO 223002A3.01 002/C/A/T2G1/ADS/B5/223002A2.06//RO/SRO/BANK

Accident conditions on Unit 1 have resulted in an EOI-directed Emergency Depressurization. Reactor pressure is currently 106 psig. ALL systems function as designed.

Regarding the High Pressure Coolant Injection (HPCI) System, which ONE of the following statements describes the current status of the respective indications on the Containment Isolation Status System (CISS) Panel and Panel 9-3 (assume ALL instruments/controls actuate at the setpoint)?

The 'amber' HPCI AUTO-ISOL LOGIC A/B lights, on Panel 9-3, are ____(1)____.

The 'amber' (HPCI) PCIS LOGIC A/B INTITIATION lights, on the CISS Panel, are ____(2)____.

The 'green' (HPCI) PCIS LOGIC A/B SUCCESS lights, on the CISS Panel, are ____(3)____.

	(1)	(2)	(3)
A.	Extinguished.	Extinguished.	Extinguished.
B.	Illuminated.	Illuminated.	Illuminated.
C.	Illuminated.	Extinguished.	Illuminated.
D.✓	Extinguished.	Illuminated.	Illuminated.

K/A Statement:

223002 PCIS/Nuclear Steam Supply Shutoff
A3.01 - Ability to monitor automatic operations of the PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF including: System indicating lights and alarms.

K/A Justification: This question satisfies the K/A statement by requiring the candidate to use specific plant conditions to correctly determine the effect a loss of indication has on MSRV operability.

References: OPL171.042 rev, 19 pg 38 of 67

Level of Knowledge Justification: This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

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REFERENCE PROVIDED: None

Plausibility Analysis:

In order to answer this question correctly the candidate must determine the following:

1. Unit differences between the setpoints for HPCI Isolations between Unit 1 and Units 2/3.
2. HPCI Isolation Logic differences in relation to the amber AUTO-ISOL LOGIC A/B lights, on Panel 9-3.
3. Differences between HPCI and RCIC Isolation logic, for Low Reactor Pressure Isolation, in relation to the amber AUTO-ISOL LOGIC A/B lights, on Panel 9-3.

D - correct: This question hinges upon understanding NOT ONLY Unit differences; but, also logic differences between HPCI and RCIC, which are very similar systems. Their logic is often confused. First, the candidate must determine if the Low Reactor Pressure Isolation exists (This isolation actuates when dropping BELOW the setpoint). The given stem conditions of "Unit 1" and "Reactor pressure is currently 106 psig" complicate the decision; as, the setpoint is above the procedural isolations for Units 2 and 3, but below the procedural setpoint for Unit 1.

Once the determination of a valid isolation signal is made, then the candidate must determine how this affects operator indications. The Nuclear Steam Supply Valves to the HPCI system will, in fact, automatically isolate (CLOSE). The nuance here is that although the isolation logic functions, the 'amber' HPCI AUTO-ISOL LOGIC A/B lights, at the HPCI control station on Panel 9-3, will NOT illuminate. This plays on a system difference, in that RCIC has the same exact lights ('amber' RCIC AUTO-ISOL LOGIC A/B) in the same relative location, which WILL illuminate on the Low Reactor Pressure Isolation. This is often a point of confusion to experienced operators.

The Containment Isolation Status System (CISS) Panel amber/green lights that correspond to (HPCI) PCIS LOGIC A/B INTITATION / SUCCESS will BOTH illuminate to provide the operator a visual cue that HPCI has BOTH a demand for isolation and that it has isolated successfully.

A - incorrect: If the candidate assumes that the Unit 1 HPCI Isolation is 105 psig (Unit 2 /3) versus 110 psig, with the given stem addition of "(assume ALL instruments / controls actuate at the setpoint)," they will come to the conclusion that the isolation has NOT yet occurred. The result would be that this particular distractor will be chosen.

B - incorrect: If the candidate confuses HPCI and RCIC logic light indications for the exact same scenario (i.e., 'amber' RCIC AUTO-ISOL LOGIC A/B would be illuminated), then this particular distractor will be chosen.

C - incorrect: If the candidate confuses the set of 'amber' (HPCI) PCIS LOGIC A/B INTITATION with the set of 'amber' HPCI AUTO-ISOL LOGIC A/B lights, then this particular distractor will be chosen.

INSTRUCTOR NOTES

(2) Suction valve from CST (73-40) opens if it was closed. (Provided at least one torus suction valve is closed).

(3) HPCI pump discharge valves to feedwater system (73-34 and 73-44) receive open signal.

Mgmt Exp 06-003: Once HPCI T&P'd iaw EOI App. 4, must evaluate per CAUTION #5 prior to re-start. Read 06-003.

(4) Test line isolation valves (73-35 and 73-36) close if they were open.

EFFECTIVE COMMUNICATIO N

(5) The steam supply valve to the turbine (73-16) opens.

(6) The auxiliary oil pump and GSC blower are started by initiation signal.

(7) As oil pressure increases, the turbine control and stop valves open when hydraulic pressure is available to admit steam to the turbine.

(8) The minimum flow bypass valve (73-30) will be shut automatically when HPCI flow becomes adequate.

(9) The turbine control system will maintain turbine speed to provide constant flow.

2. HPCI Isolation

TP-10
Obj. V.B.2.c
Obj. V.C.2.c
Obj. V.D.6
Obj. V.E.7

a. Conditions causing HPCI isolation

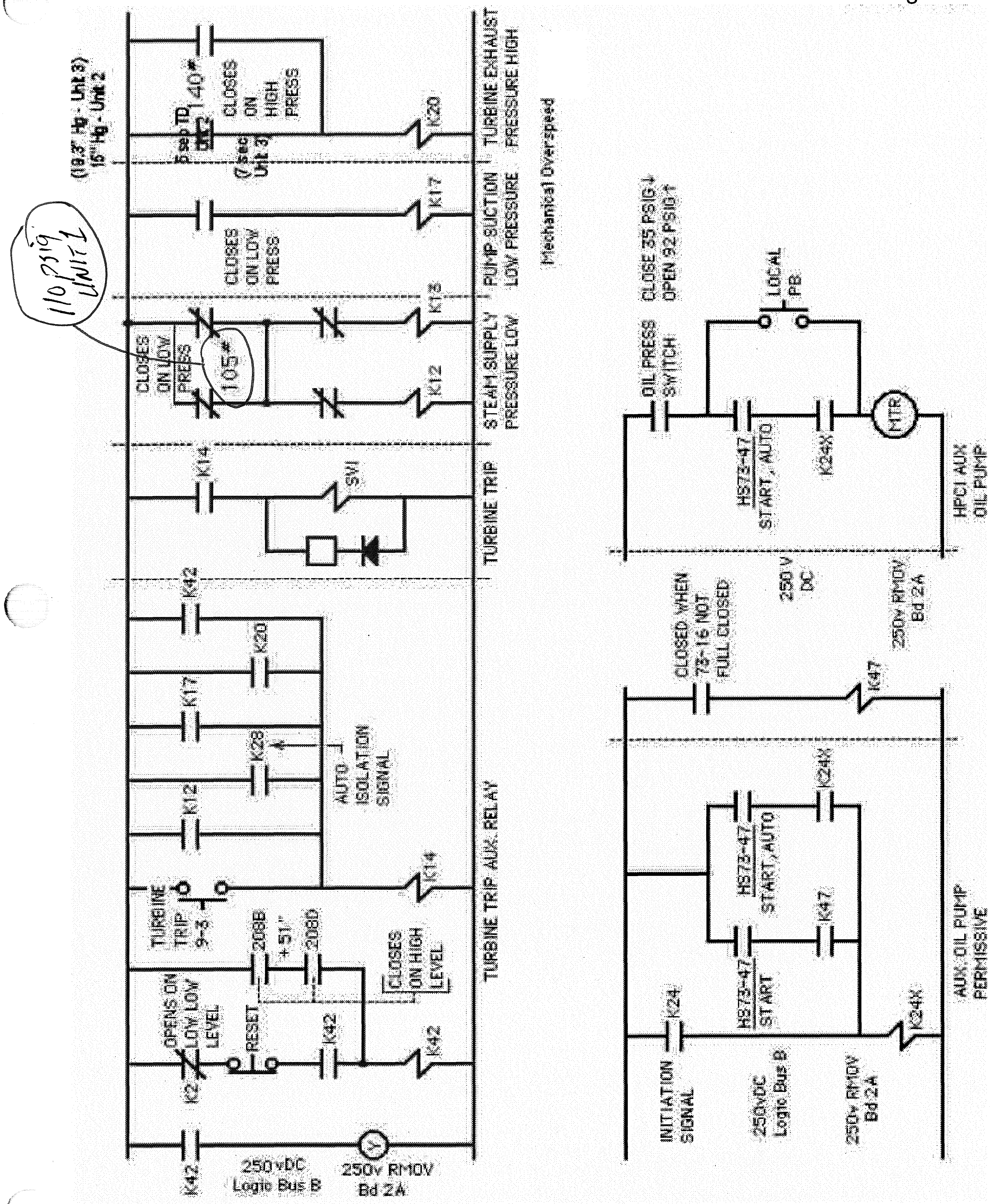
(1) Low reactor pressure 105 psig (one out of two twice). 110 psig for Unit 1. DCN 51237

Only signal that does not seal in
Unit Difference

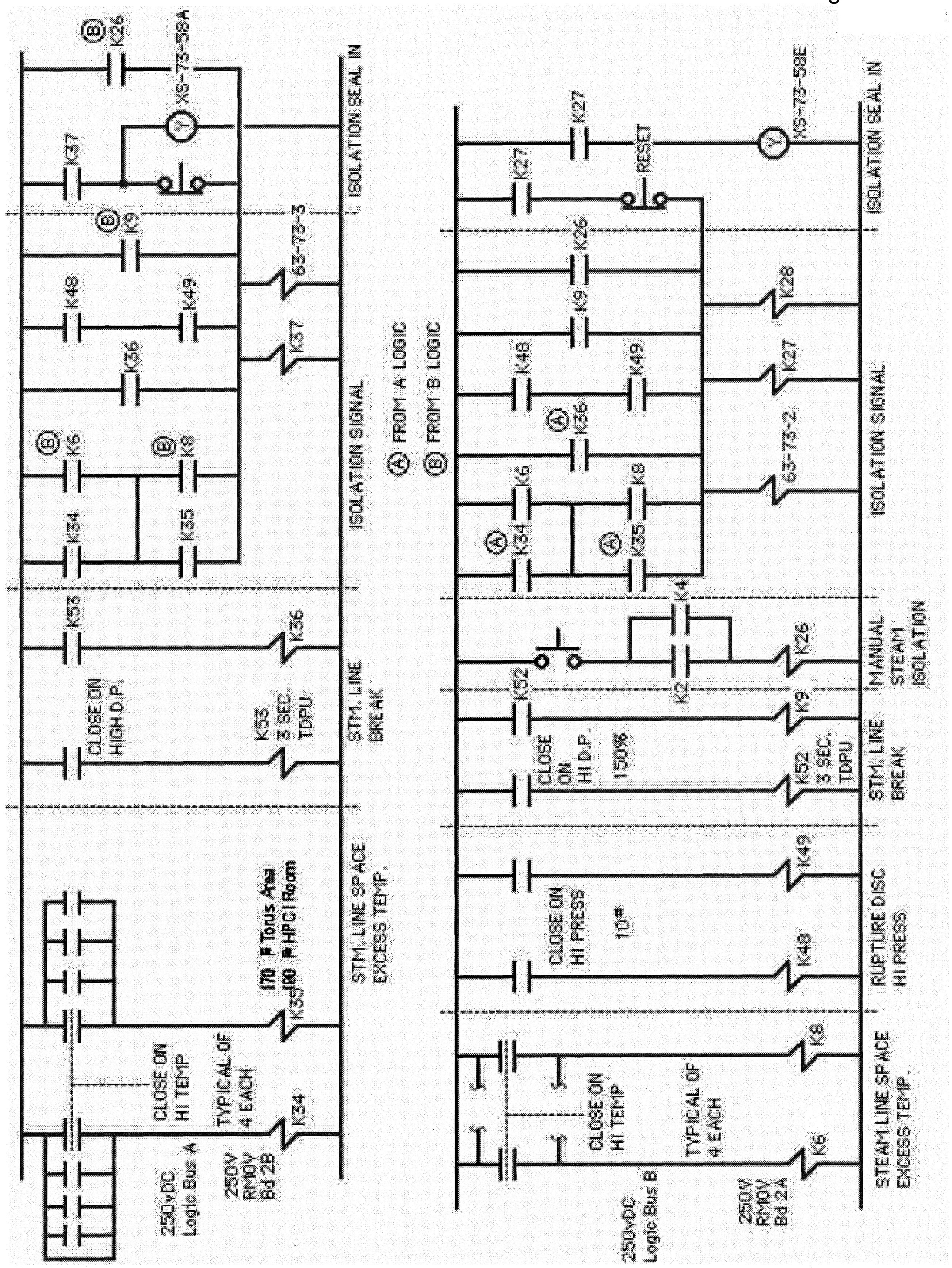
(2) High HPCI area temperature $\geq$170°F (Torus Area) or $\geq$190°F (HPCI Pump Room) (one out of two twice)

SER 3-05

(3) High HPCI steam line flow (200%) (3-second time delay) (one out of two)



TP-9: HPCI Turbine Trip Logic



TP-10: HPCI Isolation Logic

15. RO 239002A3.03 001/C/A/T2G1/MAIN STEAM/C/A/239002A3.03//RO/SRO/BANK

Given the following plant conditions:

- The reactor is operating at 100% power and 1000 psig.
- A Turbine Control Valve malfunction resulted in Safety Relief Valve (SRV) '1-4' lifting and failing to reseat.

Which ONE of the following describes the expected SRV tailpipe temperature?

REFERENCE PROVIDED

A. 212°F

B.✓ 290°F

C. 345°F

D. 545°F

K/A Statement:

239002 SRVs
A3.03 - Ability to monitor automatic operations of the RELIEF/SAFETY VALVES including: Tail pipe temperatures

K/A Justification: This question satisfies the K/A statement by requiring the candidate to use specific plant conditions to determine the expected tailpipe temperature of an open MSRV using steam tables.

References: Steam Tables

Level of Knowledge Justification: This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

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REFERENCE PROVIDED: Steam Tables

Plausibility Analysis:

In order to answer this question correctly the candidate must:

1. Use the Steam Table Mollier Diagram to determine the correct process and temperature for an open MSRV.

NOTE: This question is typical for a GFES examination, however the K/A provides little lattitude for a question with discriminatory value based on reading a multi-point recorder. In addition, with it's direct connection to an issue identified following the accident at TMI, the importance of understanding this process becomes self-evident.

B - correct: This is a throttling process and is therefore isenthalpic.

A - incorrect: This temperature is indicative of saturation temperature for steam at tailpipe pressure (atmospheric).

C - incorrect: 345°F would be incorrectly determined if the candidate considered the process to be isenthalpic to the saturation line, then followed the constant superheat line to atmospheric pressure.

D - incorrect: This temperature is indicative of saturation temperature for reactor pressure. The operator may assume that due to the location of the tailpipe temperature detector that the indication would be consistent with the saturation temperature of the reactor coolant system.

Amplification: Regarding Distractor "D", the reason it is plausible is because this is inline with what the operators at TMI expected to see and why they assummed the PORV was closed when it was still open.

EXAMINATION REFERENCE PROVIDED TO CANDIDATE

Combustion Engineering Steam Tables

16. RO 239002A4.08 001/MEM/T2G1/32A-1/4/239002A4.08//RO/SRO/BANK

Given the following plant conditions:

- Unit 2 is operating at 100% power.
- A complete loss of Drywell Control Air occurs (BOTH headers).
- NEITHER crosstie with Containment Atmosphere Dilution (CAD), nor plant Control Air, can restore system pressure.

Which ONE of the following statements describes the effect on pneumatically operated valves inside the Primary Containment in accordance with 2-AOI-32A-1, "Loss of Drywell Control Air"?

A. ALL MSRV's can still be cycled five times due to the air system check valve arrangement.

B.✓ ADS MSRVs can still be cycled five times due to the air system accumulator and check valve arrangement.

C. Inboard MSIVs can be opened one time due to the air system accumulator and check valve arrangement to allow using the condenser as a heat sink.

D. Inboard MSIVs will remain OPEN due to spring pressure until closed by a valid isolation signal due to the air system accumulator and check valve arrangement.

K/A Statement:

239002 SRVs
A4.08 - Ability to manually operate and/or monitor in the control room: Plant air system pressure: Plant-Specific

K/A Justification: This question satisfies the K/A statement by requiring the candidate to use specific plant conditions to determine the operability of MSRVs following a loss of pneumatic supply.

References: 2-AOI-32A-1

Level of Knowledge Justification: This question is rated as MEM due to the requirement to recall or recognize discrete bits of information.

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REFERENCE PROVIDED: None

Plausibility Analysis:

In order to answer this question correctly the candidate must determine the following:

1. Recognize that ADS MSRV accumulators allow for five cycle operations.
2. Recognize that Inboard MSIVs are capable of being CLOSED one time, but not OPENED one time and that the MSIVs fail closed on loss of control air.

B - correct: The ADS MSRV air accumulators are provided to ensure that the valves can be held open following failure of the air supply to the accumulators, and they are sized to contain sufficient air for a minimum of five valve operations. Operations of the ADS MSRV should be limited to 5 times.

A - incorrect: Only ADS MSRVs have accumulators sufficient to cycle five times. The remaining MSRVs will not function without a pneumatic supply.

C - incorrect: The MSIVs fail closed on a loss of control air. This question is plausible if the candidate misunderstands the purpose of the MSIV accumulator.

D - incorrect: The MSIVs are designed to fail closed on a loss of control air. Spring pressure and air close the MSIV. Also a loss of contol air will cause the MSIV to drift close due to spring pressure. This distractor is plausible if the candidate thinks that spring pressure assists in opening the MSIV.

LOK changed to MEMORY.

4.0 OPERATOR ACTIONS

4.1 Immediate Actions

None

4.2 Subsequent Actions

[1] **IF** ANY EOI entry condition is met, **THEN**

ENTER the appropriate EOI(s). ☐

NOTES

1) The MSIV air accumulators are designed to provide for one closing actuation following loss of air supply. Once closed the valve is held closed by the springs.
2) The ADS MSRV air accumulators are provided to assure that the valves can be held open following failure of the air supply to the accumulators, and they are sized to contain sufficient air for a minimum of five valve operations. Operations of the ADS MSRV should be limited to 5 times.
3) Nitrogen Tanks supply pressurized nitrogen to the Drywell Control Air System via the DWCA SUPPLY REGULATORS 2-PREG-32-49A and 2-PREG-32-49A (lead regulator will be set at 100 psig and backup regulator set at 5-8 psig lower)
4) DWCA NITROGEN REG STATION BYPASS VLV, 2-BYV-032-0141 can be used to maintain approximately 98 psig in DWCA Receiver Tanks A & B when required by plant conditions

[2] **CHECK** Drywell Control Air System operating properly.

REFER TO 2-OI-32A. ☐

[3] **IF** Operation with DWCA Nitrogen Regulattion Bypass Valve Open/Throttled is required, **THEN**

REFER TO 2-OI-32A. ☐

17. RO 259002A4.03 002/C/A/T2G1/FW OI-3//259002A4.03//RO/NEW 02/12/2008

Which ONE of the following describes the indications and controls available when the Unit Operator transfers the Unit 1 Reactor Fedwater Startup Level Controller from MANUAL to AUTO?

The Auto/Manual ____(1)____ light extinguishes and the ___(2)___ light illuminates.

The controller is now controlling in ___(3)___ Element with the controller output changed from ______(4)______.

	(1)	(2)	(3)	(4)
A.	Amber;	Blue;	Single;	Demand output to Level Setpoint output.
B.	Amber;	Blue;	Three;	Level Setpoint output to Demand output.
C.	Blue;	Amber;	Single;	Level Setpoint output to Demand output.
D.	Blue;	Amber;	Three;	Demand output to Level Setpoint output.

K/A Statement:

259002 Reactor Water Level Control
A4.03 - Ability to manually operate and/or monitor in the control room: All individual component controllers when transferring from manual to automatic modes.

K/A Justification: This question satisfies the K/A statement by requiring the candidate to understand the process and indications for transferring FWLC from manual to auto.

References: 1-OI-3 Illustration 1

Level of Knowledge Justification: This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

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New question for RO 259002A4.03#1 to better meet the K/A

REFERENCE PROVIDED: None

Plausibility Analysis:

In order to answer this question correctly the candidate must determine the following:
1. The indications when in auto and the operation of the controller.

A - correct: The AUTO/MANUAL pushbutton is a momentary backlit pushbutton with a split lens backlight. AUTO is blue and MANUAL is amber. Pushing this button toggles the controls from AUTO to MANUAL, and vice versa. In AUTO, the output is the LEVEL SETPOINT. In MANUAL, the Operator has manual control (demand) with RAISE/LOWER pushbuttons. Also, the Startup Level controller only controls in SINGLE ELEMENT when in AUTO (ref. Illustration 1, Sec. 3.0 description).

B - incorrect: The Startup Level controller only controls in SINGLE ELEMENT when in AUTO. This is plausible because the desired FWLC control method is THREE ELEMENT. Also, the controller will be in LEVEL SETPOINT output.

C - incorrect: The blue light will be lit and NOT the amber. This is plausible because of the differences between the units. Unit's 2 & 3 have different controllers. Also, the controller will be in LEVEL SETPOINT output.

D - incorrect: The blue light will be lit and NOT the amber. Same plausiblity as 'C'. The Startup Level controller only controls in SINGLE ELEMENT when in AUTO.

Amplification: Changed question to better meet the K/A

Illustration 1
(Page 2 of 3)

RFW Startup Level Controller Panel Display Station

2.0 COMPONENTS (REFER TO FIGURE 1)

RAMP ENABLE	This is a maintained backlit amber pushbutton that will change the RAISE/LOWER pushbuttons from "One Shot" to "Ramp" control. This button is pushed for ramp control
LOWER	This is a momentary non-backlit pushbutton.
MANUAL/AUTO	This is a momentary backlit pushbutton with a split lens backlight. AUTO is Blue and MANUAL is amber. Pushing this button toggles the controls from AUTO to MANUAL and vice versa. In AUTO control the Output is the Level Setpoint output, and in MANUAL the Operator manual control with RAISE/LOWER pushbuttons.
RAISE	This is a momentary non-backlit puhbutton.
MANUAL RAMP MODE	Ramps the RFW Startup Level Controller demand by 2% per second by using RAISE/LOWER pushbuttons.
MANUAL "ONE-SHOT" MODE	Changes the RFW Startup Level Controller demand by 0.1% per push of the RAISE/LOWER pushbuttons
AUTO RAMP MODE	Ramps the RFW Startup Level Controller Setpoint value by 0.25 inches per second.
AUTO-"ONE-SHOT" MODE	Changes the RFW Startup Level Controller Setpoint value by 0.20 inches per push of the RAISE/LOWER puhbuttons.

3.0 DESCRIPTION

TheRFW SU LVL CONT, 1-LIC-3-53, is used from 0 psig to approximately 350 psig RX Pressure. The Panel Display Station(PDS) varies the position of the RFW START-UP LCV, 1-LCV-003-0053, to control Feedwater flow to the reactor. When AUTO on the PDS is selected, then Reactor Water Level is controlled in the Automatic Single Element Mode with the level setpoint adjusted by the Unit Operator utilizing the RAISE/LOWER push-buttons on the PDS. When MANUALon the PDS is selected, then RFW START-UP LCV, 1-LCV-003-0053, position is changed manually by the Unit Operator using the RAISE/LOWER push-buttons. The RFW START-UP LCV, 1-LCV-003-0053 can also be positioned locally by handwheel.

18. RO 261000K3.03 001/C/A/T2G1/CONT/PRI/V.B.8/261000K3.03//RO/SRO/NEW 10/16/07

Unit 2 has experienced a LOCA with the following plant conditions:

- Drywell pressure is 51 psig and rising.
- Suppression Chamber (Torus) water level is 18.5 feet.
- An Emergency Depressurization has been conducted.
- The Torus is being vented through Standby Gas Treatment (SGT) 'A' train.
- Standby Gas Trains 'B' and 'C' are INOPERABLE due to common mode failures.

IF Standby Gas Train 'A' were to suffer the same common mode failure under these conditions, which ONE of the following would be the next sequential, EOI-directed step to exhaust the primary containment atmosphere ?

Vent the ____(1)_____ in accordance with 2-EOI Appendix 13, "Emergency Venting Primary Containment," ____________________(2)________________________.

REFERENCE PROVIDED

	(1)	(2)
A.✓	Torus;	via the Hardened Wetwell Vent System.
B.	Torus;	allowing the Primary Containment Vent Ducts to fail.
C.	Drywell;	via the Hardened Wetwell Vent System.
D.	Drywell;	allowing the Primary Containment Vent Ducts to fail.

K/A Statement:

261000 SGTS
K3.03 - Knowledge of the effect that a loss or malfunction of the STANDBY GAS TREATMENT SYSTEM will have on following: Primary containment pressure: Mark-I&II

K/A Justification: This question satisfies the K/A statement by requiring the candidate to use specific knowledge of the relationship between SGT and the inerting process.

References: OPL171.016, *Primary and Secondary Containment Systems*, 2-EOI-2, *Primary Containment Control,* 2-EOI Appendix 13, *Emergency Venting Primary Containment*

Level of Knowledge Justification: This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

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REFERENCE PROVIDED: 2-EOI-2, *Primary Containment Control*

Plausibility Analysis:

In order to answer this question correctly the candidate must determine the following:

1. Drywell pressure is approaching the 55 psig pressure limit.
2. Recognize which available vent path does NOT require SGT to be OPERABLE.
3. Recognize that venting the Suppression Chamber is preferred over the Drywell to facilitate scrubbing of radioactive fission products.
4. Recognize which vent paths are available to the Torus and Drywell.
5. Recognize the implications of Torus Water Level as it relates to the chosen vent path.

A - correct: First, let it be recognized that in any case that you are venting the Drywell OR Torus, you are, in effect, venting the respective other. In this particular case, the impact of losing the last available Standby Gas Train removes the viability of utilizing Appendix 12, *Primary Containment Venting*. Additionally, based on given stem conditions of Drywell Pressure and Torus Water Level, the Pressure Suppression Pressure (PSP) curve (EOI-2, Curve 6) has been exceeded; and Torus pressure CANNOT be maintained in the "safe area of the curve." Also identified in stem conditions, an Emergency Depressurization has been conducted, providing the allowance to travel farther down the Primary Containment Pressure "PC/P" leg of EOI-2.

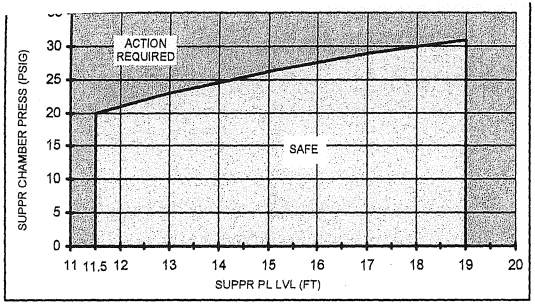
Once current Torus Water Level is applied to the equation, the candidate would be

OFFSITE RADIOACTIVITY RELEASE RATE (APPX 13)" versus the other optional path "VENT THE DW IRRESPECTIVE OF OFFSITE RADIOACTIVITY RELEASE RATE (APPX 13)." Without Appendix 13 available to the candidate, they must now ascertain which vent path is applicable. NO additional failures are provided in the stem conditions which would preclude the subsequent venting of the Torus; utilizing the Appendix 13 designated path. This designated path for the Torus is the Hardened Wetwell Vent System, as opposed to the Appendix 13 designated Drywell path, which is allowing the Primary Containment Vent Ducts to fail.

B - incorrect: Although venting the Torus would be the directed path per EOI-2 (by given stem conditions), the procedurally-designated path for the Torus is the Hardened Wetwell Vent System, as opposed to the designated Drywell path, which is allowing the Primary Containment Vent Ducts to fail. Destructive venting of the Suppression Chamber is physically possible; but, is NOT procedurally authorized. As detailed above, the candidate is ONLY directed, in EOI-2, to "VENT THE SUPPR CHMBR IRRESPECTIVE OF OFFSITE RADIOACTIVITY RELEASE RATE (APPX 13)." It does NOT specify a particular path.

C - incorrect: Although venting the Drywell is a directed path in EOI-2, it is NOT yet required (by given stem conditions). Additionally, the procedurally-designated path for the Drywell is allowing the Primary Containment Vent Ducts to fail, as opposed to the designated path for the Torus, which is the Hardened Wetwell Vent System. If the Drywell path is chosen, the candidate's ONLY direction, once again, is to "VENT THE DW IRRESPECTIVE OF OFFSITE RADIOACTIVITY RELEASE RATE (APPX 13)." It does NOT specify a particular path.

D - incorrect: This is a fully-functional, procedurally-allowed vent path for the Drywell; directed by EOI-2 and detailed in Appendix 13. If the candidate does NOT apply ALL of the given stem conditions, primarily Torus Water Level LESS THAN 20 feet, an error will be made. Complicating this decision for the candidate is another Decision Block, in EOI-2, inquiring as to "CAN THE SUPPR CHMBR BE VENTED?"



W BLOWERS

S NOT REQUIRED
CONTINUOUS

IS REQUIRED
)

 # 2 PUMP NPSH AND VORTEX LIMITS

INITIATE DW SPRAYS USING ONLY PUMPS NOT REQUIRED TO ASSURE ADEQUATE CORE COOLING BY CONTINUOUS INJ (APPX 17B)

PC/P-9

2

CAN SUPPR CHMBR PRESS BE MAINTAINED WITHIN THE SAFE AREA OF CURVE 6

PC/P-10

NO

YES

EMERGENCY RPV DEPRESSURIZATION IS REQUIRED
(EOI-1, RC/P-4; C1-2; C1-21; C5-1)

PC/P-11

MAINTAIN SUPPR CHMBR PRESS BELOW 55 PSIG

PC/P-12

BEFORE
SUPPR CHMBR PRESS RISES TO 55 PSIG
CONTINUE

PC/P-13

13

IS SUPPR PL LVL BELOW 20 FT

PC/P-14

NO

YES

13

VENT THE SUPPR CHMBR IRRESPECTIVE OF OFFSITE RADIOACTIVITY RELEASE RATE (APPX 13)

PC/P-15

13

CAN THE SUPPR CHMBR BE VENTED

PC/P-16

NO

YES

13

VENT THE DW IRRESPECTIVE OF OFFSITE RADIOACTIVITY RELEASE RATE (APPX 13)

PC/P-17

13

CAN SUPPR CHMBR PRESS BE MAINTAINED BELOW 55 PSIG

PC/P-18

NO

YES

EOI-2

2-EOI APPENDIX-13

EMERGENCY VENTING PRIMARY CONTAINMENT

LOCATION: Unit 2 Control Room

ATTACHMENTS: 1.Tools and Equipment
2.Vent System Overview
3.Hardened Vent Flow Path

(√)

1. **NOTIFY** SHIFT MNGR./SED of the following:

   - Emergency Venting of Primary Containment is in progress. ____
   - Off-Gas Release Rate Limits will be exceeded. ____

2. **VENT** the Suppression Chamber as follows (Panel 9-3):

   a. IF.......EITHER of the following exists:

      - Suppression Pool water level CANNOT be determined to be below 20 ft,

      OR

      - Suppression Chamber CANNOT be vented,

      THEN.....**CONTINUE** in this procedure at Step 3. ____

   b. **PLACE** keylock switch 2-HS-64-222B, HARDENED SUPPR CHBR VENT OUTBD PERMISSIVE, in PERM. ____

   c. **CHECK** blue indicating light above 2-HS-64-222B, HARDENED SUPPR CHBR VENT OUTBD PERMISSIVE, illuminated. ____

   d. **OPEN** 2-FCV-64-222, HARDENED SUPPR CHBR VENT OUTBD ISOL VLV. ____

   e. **PLACE** keylock switch 2-HS-64-221B, HARDENED SUPPR CHBR VENT INBD PERMISSIVE, in PERM. ____

   f. **CHECK** blue indicating light above 2-HS-64-221B, HARDENED SUPPR CHBR VENT INBD PERMISSIVE, illuminated. ____

   g. **OPEN** 2-FCV-64-221, HARDENED SUPPR CHBR VENT INBD ISOL VLV. ____

2. (continued from previous page)

   h. **CHECK** Drywell and Suppression Chamber Pressure lowering. ____

   i. **MAINTAIN** Primary Containment Pressure below 55 psig using 2-FCV-64-222, HARDENED SUPR CHBR VENT OUTBD ISOL VLV, as directed by SRO. ____

3. IF ..... Suppression Chamber vent path is NOT available, THEN ... **VENT** the Drywell as follows:

   a. **NOTIFY** SHIFT MNGR./SED that Secondary Containment integrity failure is possible. ____

   b. **NOTIFY** RADCON that Reactor Building is being evacuated due to imminent failure of Primary Containment vent ducts. ____

   c. **EVACUATE** ALL Reactor Buildings using P.A. System. ____

   d. **START** ALL available SGTS trains. ____

   e. **VERIFY CLOSED** 2-FCV-64-36, DW/SUPPR CHBR VENT TO SGT (Panel 9-3). ____

   f. **VERIFY OPEN** the following dampers (Panel 9-25):

      - 2-FCO-64-40, REACTOR ZONE EXH TO SGTS ____

      - 2-FCO-64-41, REACTOR ZONE EXH TO SGTS. ____

   g. **VERIFY CLOSED** 2-FCV-64-29, DRYWELL VENT INBD ISOL VALVE (Panel 9-3 or Panel 9-54). ____

   h. **DISPATCH** personnel to Unit 2 Auxiliary Instrument Room to perform the following:

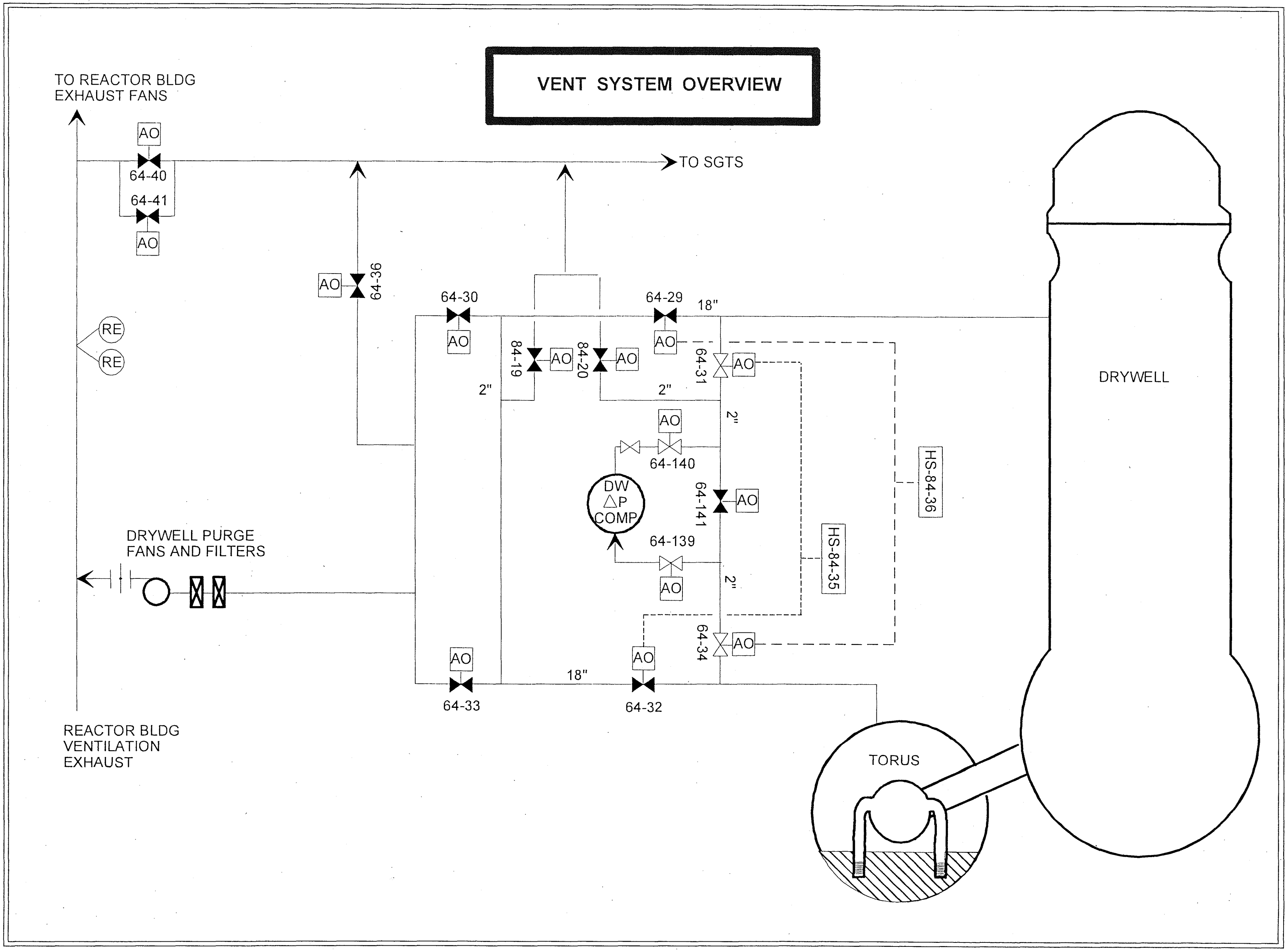
      1) **REFER TO** Attachment 1 and **OBTAIN** one 12-in. Banana Jack Jumper from EOI Equipment Storage Box. ____

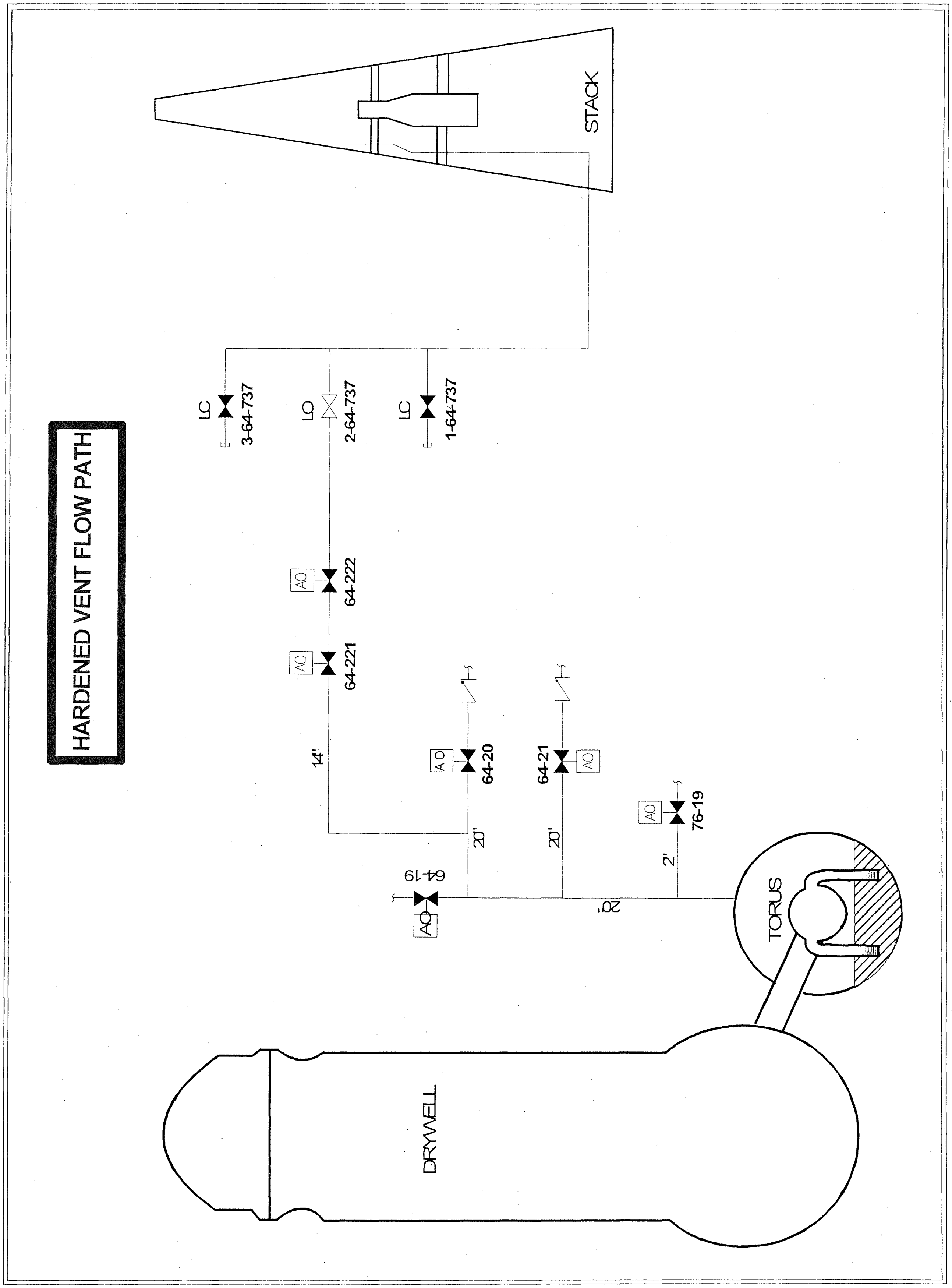
      2) **LOCATE** terminal strip DD in Panel 9-43, Front. ____

      3) **JUMPER** DD-76 to DD-77 (Panel 9-43). ____

      4) **NOTIFY** Unit Operator that jumper for 2-FCV-64-30, DRYWELL VENT OUTBD ISOLATION VLV, is in place. ____

   i. **VERIFY OPEN** 2-FCV-64-30, DRYWELL VENT OUTBD SOLATION VLV (Panel 9-3). ____


VENT SYSTEM OVERVIEW
TO REACTOR BLDG EXHAUST FANS
TO SGTS
AO
64-40
64-41
64-36
64-30
64-29
18"
84-19
84-20
64-31
2"
64-140
DW △P COMP
64-141
64-139
HS-84-36
HS-84-35
64-34
64-33
64-32
RE
DRYWELL PURGE FANS AND FILTERS
REACTOR BLDG VENTILATION EXHAUST
DRYWELL
TORUS


HARDENED VENT FLOW PATH
STACK
LC
3-64-737
LO
2-64-737
LC
1-64-737
AO
64-222
AO
64-221
14"
AO
64-20
64-21
AO
20"
20"
AO
76-19
2"
64-19
AO
20"
TORUS
DRYWELL

19. RO 262001K4.04 001/C/A/SYS/ACDIST/3/262001K4.04//RO/SRO/BANK

Given the following plant electrical distribution alignment:

- 4kV Shutdown Bus 1 '43' Switch is in MANUAL.
- ALL 4kV Shutdown Board '43' Switches are in AUTO.
- A fault on '1A' 4kV Unit Board de-energizes 4kV Shutdown Bus 1.

Which ONE of the following represents how and when the '1A' 4kV Shutdown Board alternate supply breaker will auto close?

A ___(1)___ transfer occurs when _______(2)_______ voltage decays to less than 30%.

	(1)	(2)
A.	fast;	4kV Shutdown Bus 1
B.✓	slow;	4kV Shutdown Bus 1
C.	fast;	'1A' 4kV Shutdown Board
D.	slow;	'1A' 4kV Shutdown Board

K/A Statement:

262001 AC Electrical Distribution
K4.04 - Knowledge of A.C. ELECTRICAL DISTRIBUTION design feature(s) and/or interlocks which provide for the following: Protective relaying

K/A Justification: This question satisfies the K/A statement by requiring the candidate to use specific plant conditions and times to correctly determine the response of the AC distribution system to a fault which initiates protective relaying.

References: OPL171.036 rev 11, pg, 31, I.7

Level of Knowledge Justification: This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

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REFERENCE PROVIDED: None

Plausibility Analysis:

In order to answer this question correctly the candidate must determine the following:

1. Whether the transfer is a fast transfer or slow transfer.
2. Whether the low voltage is sensed on the line side of the breaker or the load side of the breaker.

NOTE: The plausibility of the distractors is based on determining the answers to the above questions.

B - correct: With Shutdown Bus transfer scheme in manual the Shutdown Bus will not fast transfer when a fault occurs on 1A 4kV Unit Board. This will cause a loss of voltage to the line side of the normal feeder to the 'A' 4kV Shudown Board. Therefore a slow transfer will occur on the 'A' 4kV Shudown Board when voltage decays to 30%.

A - incorrect: Fast Transfers are MANUAL only. This is plausible because the undervoltage sensing location is correct.

C - incorrect: Fast Transfers are MANUAL only. The undervoltage is sensed on the Shutdown BUS side of the breaker.

D - incorrect: The undervoltage is sensed on the Shutdown BUS side of the breaker. This is plausible because the transfer scheme is correct.

Note - removed the 'S' from the '43S' switch. Any '43' switch is an auto/manual transfer switch. The 'S' or 'Sx' designation is not required to answer this question correctly.

7. Shutdown Board Transfer Scheme

a. The only automatic transfer of power on a shutdown board is a delayed (slow) transfer. In order for the transfer to take place, the bus transfer control switch (43Sx) must be in AUTOMATIC.

Obj. V.B.8.c
Obj. V.C.2.c
Obj. V.D.8.c
Procedural Adherence when transferring boards

(1) Undervoltage is sensed on the line side of the normal feeder breaker.

(2) Voltage is available on the line side of the alternate feeder breaker.

(3) The normal feeder breaker then receives a trip signal.

(4) A 52b contact on the normal supply breaker shuts in the close circuit of the alternate feeder breaker, indicating that the normal breaker is open.

(5) A residual voltage relay shuts in the close circuit of the alternate supply breaker, indicating that board voltage has decayed to less than 30 percent of normal.

(6) The alternate supply breaker then closes.
The shutdown board transfer scheme is NORMAL seeking. If power is restored to the line side of the normal feeder breaker, and if the 43Sx switch is still in AUTOMATIC, then a "slow" transfer back to the normal supply will occur. This will cause momentary power loss to loads on the bus and ESF actuations are possible.

**b Manual High Speed (Fast Transfer)

Obj. V.B.8.c
Obj. V.C.2.c
Review INPO SOER 83-06

To fast transfer a shutdown board perform the following:

20. RO 262002A1.02 001/C/A/T2/G1/UNIT PREFFERRED/C/A 2.5/2.9/262002AA1.02/BF05301/RO/SRO/BANK

Given the following plant conditions:

- Unit 3 is in a normal lineup.
- The following alarm is received:
  - UNIT PFD SUPPLY ABNORMAL
- It is determined that the alarm is due to the Unit 3 Unit Preferred AC Generator Overvoltage condition

Which ONE of the following describes the result of this condition?

Unit 3 Breaker 1001 _____(1)_____; Unit 2 Breaker 1003 ______(2)_____; and the Motor-Motor-Generator (MMG) set ________________(3)__________________.

	(1)	(2)	(3)
A.	trips OPEN;	is interlocked OPEN;	automatically shuts down.
B.	is interlocked OPEN;	trips OPEN;	automatically shuts down.
C.✓	trips OPEN;	is interlocked OPEN;	continues to run without excitation.
D.	is interlocked OPEN;	trips OPEN;	continues to run without excitation.

K/A Statement:

262002 UPS (AC/DC)
A1.02 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.

K/A Justification: This question satisfies the K/A statement by requiring the candidate to correctly apply a specific operating condition of the UPS MMG Set to the correct response of the system to that condition.

References: OPL171.102, Rev.6, pg 20, 21, TP-2 and 3-ARP-9-8B, Rev.9, tile 35

Level of Knowledge Justification: This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to solve a problem. This requires mentally using this knowledge and its meaning to resolve the problem.

0610 NRC Exam

REFERENCE PROVIDED: None

Plausibility Analysis:

In order to answer this question correctly the candidate must determine the following:

1. The 1001 and 1003 breakers from an MMG set will trip on overvoltage or underfrequency at the output of the MMG.
2. Unit 2 MMG Breakers are interlocked to prevent alternate power to unit 1 and 3 at the same time.
3. When an overvoltage condition exists at the Generator Output, the 1001 breaker from the MMG Set trips.
4. Excitation is lost and the MMG Set continues to run.
5. The "Hold to build up voltage" switch must be depressed to restore voltage.

C - correct: The 1001 and 1003 breakers from an MMG set will trip on overvoltage or underfrequency at the output of the MMG. Also Unit 2 MMG Breakers are interlocked to prevent alternate power to unit 1 and 3 at the same time.

A - incorrect: The MMG set does not automatically shut down. This is plausible because the breaker lineup is correct.

B - incorrect: The MMG set does not automatically shut down. This is plausible although the breaker lineup is backwards.

D - incorrect: The breaker lineup is backwards. This is plausible because the MMG Set will continue to run without excitation.

Note - The operator could incorrectly assume that the Unit 3 1001 bkr is interlocked to prevent paralleling battery boards 2 & 3 (ref TP-2).

(d) Another Unit's MMG set

The second alternate is from another unit's MMG set output. Unit 2 MMG is the second alternate for either Unit 1 or Unit 3; Unit 3 is the second alternate for Unit 2. Transfers to this source are done manually at Battery Board 2 panel 11.

b. MMG Sets (Unit 2&3) — Obj. V.B.2.b; TP-11

(1) The MMG is normally driven By the AC motor, powered from 480V Shutdown Board A. Should this supply fail, the AC motor is automatically disconnected and the DC motor starts, powered from 250V Battery Board. The DC motor has an alternate power supply from another 250V Battery Board. Transfer to the alternate DC source is manual. Underfrequency on the generator output will trip the DC motor. Transfer of the MMG set back to the AC motor is manual.

Obj.V.D.2.c
Obj.V.D.2.d/j
Obj V.E.2.c
Obj.V.E.2.d/i
Obj V.B.2.h
Obj.V.C.3.e
Obj.V.D.2.j
Obj.V.E.2.i

(2) The 1001 and 1003 breakers from an MMG set will trip on overvoltage or underfrequency at the output of the MMG. Also Unit 2 MMG Breakers are interlocked to prevent alternate power to unit 1 and 3 at the same time.

(3) When an under frequency or overvoltage condition exists at the Generator Output the following occurs

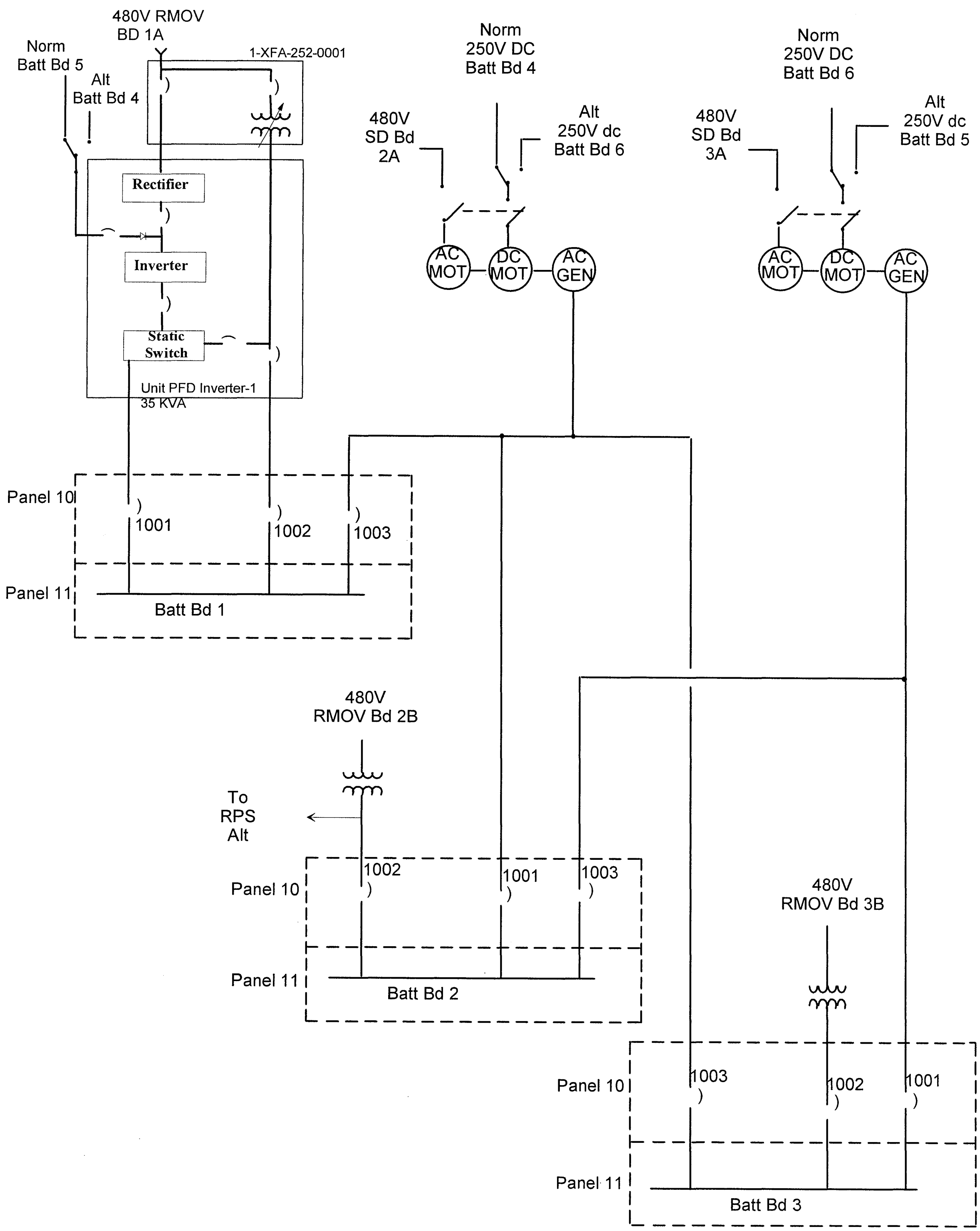
Obj. V.B.2.h
Obj. V.C.3.e
Obj. V.D.2.j
Obj. V.E.2.i

(a) BB panel 10 breakers from the MMG Set trip.

U2	1001 (U2)	1003 (U1&3)
U3	1001 (U3)	1003 (U2)

(b) Excitation is lost and the MMG Set continues to run. (The Hold to build up voltage switch must be depressed to restore voltage.)

OPL171.102



TP-2: POWER SUPPLIES TO UPS BATTERY BOARD CABINETS

UNIT PFD SUPPLY ABNORMAL	35

(Page 1 of 1)

Sensor/Trip Point:

Relay SE - loss of normal DC power source.
Relay TS - DC Xfer switch transfers to Emergency DC Power Source.
Regulating Transformer Common Alarm.
1-INV-252-001, INVT-1 System Common Alarm.

Sensor Location: EL 593' 250V DC Battery Board 2

Probable Cause:

A. Loss of normal DC power source
B. DC power transfer.
C. Relay failure
D. INVT-1 System Common Alarms
  1. Fan Failure Rectifier
  2. Over temperature Rectifier
  3. AC Power Failure to Rectifier
  4. Low DC Voltage
  5. High DC Voltage
  6. Low DC Disconnect
  7. Fan Failure Inverter
  8. Alternate Source Failure
  9. :Low AC Output Voltage
  10. High Output Voltage
  11. Inverter Fuse Blown
  12. Static Switch Fuse Blown
  13. Over Temperature Inverter

E. PFD Regulating XFMR Common Alarms
  1. Transformer Over temperature
  2. Fan Failure
  3. CB1 Breaker Trip
  4. CB2 Breaker Trip

Automatic Action:

A. Auto transfer to DC Power Source on Rectifier failure.
B. Auto transfer to Alternate AC supply (Regulated Transformer) on Inverter failure.

Operator Action:

A. **IF** 120V AC Unit Preferred is lost, **THEN** **REFER TO** 1-AOI-57-4, Loss of Unit Preferred. ☐
B. **REFER TO** appropriate portion of 0-OI-57C, 208V/120V AC Electrical System. ☐

References:

0-45E641-2	1-45E620-11	1-3300D15A4585-1
10-100467	0-20-100756	20-110437

21. RO 263000K1.02 001/MEM/T2G1/250VDC/3/263000K1.02//RO/SRO/BANK

Which ONE of the following statements describes the operation of '2B' 250VDC Battery Charger?

A. The normal power supply to '2B' 250VDC Battery Charger is '2A' 480V Shutdown Board.

B. '2B' 250VDC Battery Charger can supply, directly from unit 2 Battery Board Room, any of the six Unit & Plant 250VDC Battery Boards.

C. Load shedding of the '2B' 250VDC Battery Charger CANNOT be bypassed until the Load Shed signal has been reset.

D.✓ Load shedding of the '2B' 250VDC Battery Charger can be bypassed by placing the "Emergency ON Select Switch" in the "EMERGENCY ON" Position.

K/A Statement:

263000 DC Electrical Distribution
K1.02 - Knowledge of the physical connections and/or cause- effect relationships between D.C. ELECTRICAL DISTRIBUTION and the following: Battery charger and battery

K/A Justification: This question satisfies the K/A statement by requiring the candidate to use specific knowledge of battery charger operation.

References: OPL171.037 rev 10, pg 11, 12 & 31 and 0-OI-57D.

Level of Knowledge Justification: This question is rated as MEM due to the requirement to recall or recognize discrete bits of information.

0610 NRC Exam

REFERENCE PROVIDED: None

Plausibility Analysis:

In order to answer this question correctly the candidate must determine the following:

1. Normal and Alternate power to '2B' 250VDC Battery Charger.
2. Loads capable of being supplied by '2B' 250VDC Battery Charger.
3. Load Shedding Logic and bypass capability.

D - correct: 250V DC battery chargers 1, 2A and 2B will load shed upon receipt of a Unit 1 or Unit 2 accident signal and any Unit 1/2 shutdown board being supplied by its respective diesel generator or cross tied to a Unit 3 shutdown board and a unit three Diesel Generator. The load shedding feature can be bypassed by placing the "Emergency" switch on the charger to the "EMERG" position

A - incorrect: This is plausible because '2A' 480V Shutdown Board is the Normal supply to '2A' 250VDC Battery Charger.

B - incorrect: This is plausible because '2B' 250VDC Battery Charger is capable of supplying any of the six 250V Battery Boards, but NOT directly from Unit 2 Battery Board Room.

C - incorrect: This is plausible because the '2B' 250V Battery Charger is considered a spare charger and the candidate may draw the conclusion that the spare charger doesen't have the ablility to defeat the load shed logic.

Amplification: The actual switch position in 'D' is "Emergency ON" in procedure 0-OI-57D. The position in the lesson plan is NOT correct.

Changed distractor 'C' for plausiblity.

(2) The Plant/Station Batteries (4, 5, and 6) are Class Non-1E and are utilized primarily for U-2, U-1, and U-3 respectively --for normal loads — Obj V.B.1, Obj. V.C.1, Obj. V.D.1

(3) Battery (4) Room is located on Unit 3 in the Turbine Building on Elev. 586

(4) Battery (5 & 6) Rooms are located on the Turbine Floor, Elev. 617

(5) The boards and chargers for the Unit Batteries are located in Battery Board Rooms adjacent to the batteries they serve, with the spare charger being in the Unit 2 Battery Board room. (Battery Boards 5 & 6 and their associated chargers are located adjacent to the batteries, but are in the open space of the turbine floor.)

c. 250V Plant DC components

(1) Battery charger

(a) The battery chargers are of the solid state rectifier type. They normally supply loads on the 250V Plant DC Distribution System. Upon loss of power to the charger, the battery supplies the loads.

(b) The main bank chargers only provide float and equalize charge when tied to their loads. The chargers **are not** placed on fast charge (high voltage equalize) with any loads attached.

(c) They can recharge a fully discharged battery in 12 hours while supplying normal loads.

(d) Battery charger power supplies are manual transfer only. — Follow Procedure

Obj. V.B.2
Obj. V.C.2
Obj .V.D.2

<u>250V Battery Charger</u>	<u>Normal Source</u>	<u>Alternate Source</u> (Charger Service bus)
1	480V SD Bd 1A Comp 6D	480V Common Bd 1 Comp 3A
2A	480V SD Bd 2A Comp 6D	480V Common Bd 1 Comp 3A
2B	480V SD Bd 2B Comp 6D	480V Common Bd 1 Comp 3A
3	480V SD Bd 3A Comp 6D	480V Common Bd 1 Comp 3A

4	480V SD Bd 3B Comp 6D	480V Common Bd 1 Comp 3A
5	480V Com Bd 1 Comp 5C	(no alternate)
6	480V Com Bd 3 Comp 3D	(no alternate)
2B spare charger DC output can be directed to any of four feeders. Three DC outputs can be connected to battery board 1, 2, or 3. The fourth output is connected to a new output transfer switch (located in battery board room 4) which charges batteries 4, 5, or 6 plant batteries. A mechanical interlock permits closing only one output feeder at a time. (A slide bar is utilized in battery board room 2 and a Kirk key interlock is used in battery board room 4)		

TP-2 & TP-7

Attention to Detail

BB-3 supplies 250V RMOV Boards 3A, 1B, 2B.

The 250V DC Shutdown Board Battery System

Five 120-cell batteries (one battery and battery charger for each Shutdown Board and one spare battery charger).One battery and charger for each Unit 1 and Unit 2 4KV Shutdown Board, one battery and charger for Unit 3 4KV Shutdown Board 3EB, and one spare portable charger.

The Units 1 and 2 Shutdown Board Batteries are located in the Reactor Building at Elev. 621, adjacent to the A and C 4KV Shutdown Board Rooms Batteries A and B are at the Unit 1 end (A Shutdown Board Room), Batteries C and D are at the Unit 2 end (C Shutdown Board Room). . 3EB Shutdown Board Battery is located on the south end of Unit 3 Diesel Generator Building at Elev. 583.

The chargers are similar to those used in the 250V DC Unit and Plant System. Each charger has only one power supply. A, B, C, and D are powered from Units 1 or 2 480V Reactor MOV Boards; 3EB is supplied from 3EA 480V Diesel Auxiliary Board. All supply breakers are manual. (The portable charger has a plug-in type power lead for use with a 480V receptacle.)

480 S/D BD	Normal source	Alternate source
1A S/D Bd.	Batt. Dist. Panel SB-A	Unit Batt. Bd. 2
1B S/D Bd.	Batt. Dist. Panel SB-C	Unit Batt. Bd. 3
2A S/D Bd.	Batt. Dist. Panel SB-B	Unit Batt. Bd. 1
2B S/D Bd.	Batt. Dist. Panel SB-D	Unit Batt. Bd. 3
3A S/D Bd.	Unit Batt Bd 1	Unit Batt Bd 2
3B S/D Bd.	Unit Batt Bd 3	Unit Batt Bd 1

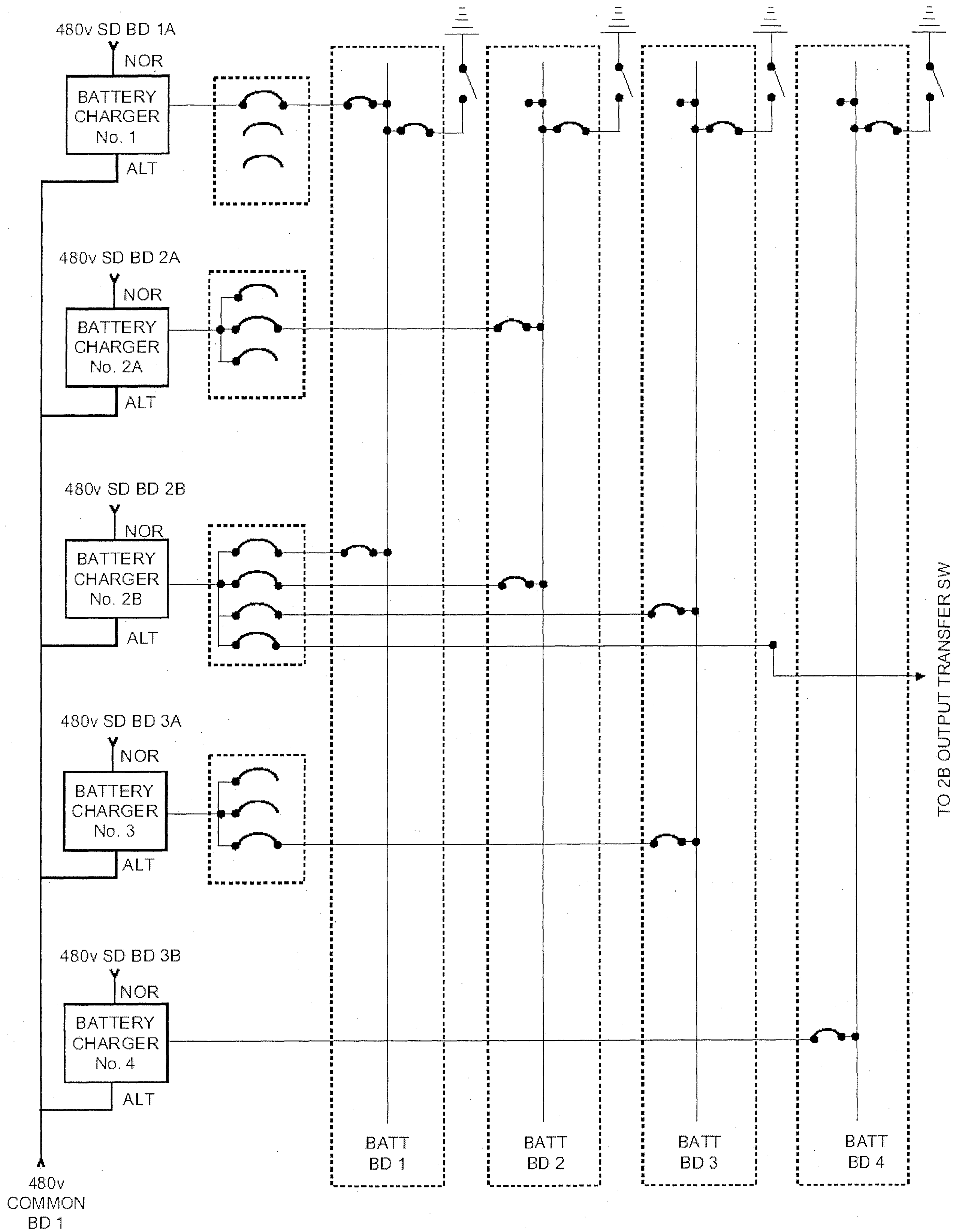
480V Shutdown Boards 2B, 3A and 3B have remote transfer capabilities for 250V DC Control Power During an Appendix R Event. The transfer switch for 2B is located on 4KV S/D Bd D, and the switches for 3A & 3B are in Elv. 593 Elect. Board Room on Unit 3.

125VDC diesel battery system

This DC system consists of eight batteries, sixteen chargers. Located in the room with the diesel generator. One of its chargers is also in the room; other is immediately outside.

1) Control and logic power
2) Governor booster pumps
3) Generator relay protection
4) Generator field flashing
5) Motor-driven fuel pump

Review enabling objectives.



TP-2 250V DC Power Distribution

3.0 PRECAUTIONS AND LIMITATIONS (continued)

E. All safety requirements concerning smoking, fires or sparks should be observed when in the Battery-Battery Board Rooms because of potential accumulation of hydrogen in flammable amounts.

F. 250V Unit Battery Charger 1,2A,2B and 3 Emergency ON select switch bypasses battery charger emergency load shed contacts. Placing the select switch in Emergency ON reestablishes charger operations with an accident signal present and Diesel Generator voltage available. Battery Charger 4 supply breaker, 480V Shutdown Board 3B, Compt 6D, receives a trip signal from the load shed logic and the breaker must be manually re-closed after a 40 second time delay to restore the charger to service. The annunciation circuit for the 250V Unit Battery Charger 3 does NOT work when the EMER/OFF/ON Select Switch is in the EMER Position.

G. [II/C] Neutron monitoring battery chargers are NOT stand alone power supplies and shall only be operated while connected to the neutron monitoring batteries. [BFPER 940862]

H. Within 30 minutes after the loss of the normal charger to a 250V Unit Battery another charger shall be placed in service to that battery and load reduced so that the battery is NOT discharging.

I. [NRC/C] Upon return to service of 24V DC Neutron Monitoring Battery A or B, Instrument Maintenance must perform functional tests on SRMs and IRMs that are powered from the affected battery board (In that the IRMs and SRMs are normally inoperable after entering RUN mode due to lack of testing, these tests are N/A for the IRMs and the SRMs if the Unit is in RUN Mode and the IRMs and SRMs are inoperable). Prior to calling the IRMs and SRMs operable, the tests have to be performed. [NRC IE Inspect Follow-up Item 86-40]

J. To return equipment to service following a failure or trip, the shutdown section of this instruction should be performed on the equipment failed. The initial conditions may NOT be applicable in this case.

K. [NRC/C] The transfer of 250VDC control power to a 4kV Shutdown Board with a diesel generator operating may cause an inadvertent start of a RHRSW pump. [LER 88021/25]

22. RO 264000K5.06 001/C/A/T2G1/82 - DG/9/264000K5.06//RO/SRO/BANK

Given the following plant conditions:

- Unit 2 is operating at 100% Power.
- No Equipment is Out of Service.
- A large-break LOCA occurs in the drywell and the following conditions exist:

  - Drywell Pressure peaked at 28 psig and is currently at 20 psig.
  - Reactor Pressure is at 110 psig.
  - Reactor Water Level is at (-)120 inches.
  - Offsite power is available.

Which ONE of the following describes the associated equipment and proper loading sequence?

A.✔ '2B' RHR Pump and '2B' Core Spray Pump start 7 seconds after the accident signal is received.

B. '2C' RHR Pump and '2C' Core Spray Pump start 7 seconds after the accident signal is received.

C. '2A' RHR Pump starts immediately and '2A' Core Spray Pump starts 7 seconds after the accident signal is received.

D. RHR Service Water Pumps, lined up for Emergency Equipment Cooling Water (EECW), start 14 seconds after the accident signal is received.

K/A Statement:

264000 EDGs
K5.06 - Knowledge of the operational implications of the following concepts as they apply to EMERGENCY GENERATORS (DIESEL/JET) : Load sequencing

K/A Justification: This question satisfies the K/A statement by requiring the candidate to use specific plant conditions and times to correctly determine the effect of load sequencing on plant equipment supplied by the Emergency Generators.

References: OPL171.038 rev 16, pg 37 & 38 and 2-OI-74.

Level of Knowledge Justification: This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

0610 NRC Exam

REFERENCE PROVIDED: None

Plausibility Analysis:

In order to answer this question correctly the candidate must determine the following:

1. Load Sequencing is NVA (Normal Voltage Available) and **NOT** DGVA (D/G Voltage Available).
2. Based on Item 1 above, the proper load sequencing with a Common Accident Signal (CAS) on Unit-2 alone and **NOT** in addition to a CAS on Unit 1.

A - correct: An accident signal with normal power available (NVA) causes the loads to sequence on in 7 second intervals (begining with the 'A' pumps) starting at time 0 after the accident signal is received. With NVA the EECW pumps start 28 seconds after the accident signal is reveived.

B - incorrect: This is plausible because 2-OI-74 P&L 3.2.B defines the start time as 7 second **"intervals"**.

C - incorrect: This is plausible because this is the sequence that they would start if a diesel generator was tied to a board (DGVA).

D - incorrect: This is plausible because RHRSW pumps all start at 14 seconds if load sequencing is DGVA.

Amplification: Candidates ARE required to know the time, in seconds, for ECCS loading sequence under accident conditions.

INSTRUCTOR NOTES

(8) An OVERSPEED trip stops injector rocker arm motion, causing the engine to stop immediately. This will also cause a START FAILURE condition to be sensed by the control and alarm circuits.

(9) If the overspeed trip lever is reset, the engine will start again. This may be prevented by opening the LOGIC breaker and pushing both local engine stop push buttons, then waiting 15 minutes for all of the shutdown sequence timers to time out before resetting the trip lever.

(10) To reset the overspeed trip rotate the limit switch fully clockwise by hand and pull down on the reset lever. This should latch and keep the limit switch positioned clockwise.

(11) A local emergency stop may be required if the normal remote or local electrical stop signals fail to begin the engine stop sequence. This may be performed by pulling the fuel rack injector lever (located between the diesel and the engine control cabinet) outward from the engine, and holding until the engine comes to a stop. If the DG had been started using the local slow start procedure, a "start failure" signal may be sent to the redundant start signal, which would cause the engine to try and restart until the redundant start circuit locks out after 5.5 seconds elapse.

Obj.V.D.17
Obj.V.E.17

(12) The redundant start may be canceled before the start circuit locks out by opening the logic breaker and pushing both engine stop push-buttons. Note that pulling the engine driven fuel pump shutoff plunger will not stop the diesel since the electric fuel pump will still be supplying fuel. (OI-82)

3. Accident Operation

a. Accident signal received (**CASx**)

(1) Signals diesel generators to start.

INSTRUCTOR NOTES

(2) Opens diesel output breakers if shut. Obj.V.B.9

b. If normal voltage is available, load will sequence on as follows: (**NVA**) Obj.V.C.6 Obj.V.D.15 Obj.V.E. 15

Time After Accident	S/D Board A	S/D Board C	S/D Board B	S/D Board D
0	RHR/CS A			
7		RHR/CS B		
14			RHR/CS C	
21				RHR/CS D
28	RHRSW*	RHRSW*	RHRSW*	RHRSW*

*RHRSW pumps assigned for EECW automatic start

c. **If normal voltage is NOT available: (DGVA)** Obj.V.B.9 Obj.V.C.6

(1) After 5-second time delay, all 4kV Shutdown Board loads except 4160/480V transformer breakers are automatically tripped.

(2) Diesel generator output breaker closes when diesel is at speed.

(3) Loads sequence as indicated below

Time After Accident	S/D Board A	S/D Board B	S/D Board C	S/D Board D
0	RHR A	RHR C	RHR B	RHR D
7	CS A	CS C	CS B	CS D
14	RHRSW *	RHRSW *	RHRSW *	RHRSW *

*RHRSW pumps assigned for EECW automatic start

d. Certain 480V loads are shed whenever an accident signal is received in conjunction with the diesel generator tied to the board. (see OPL171.072)

3.2 LPCI (continued)

B. Upon an automatic LPCI initiation with normal power available, RHR Pump 2A starts immediately and 2B, 2C, 2D sequentially start at 7 second intervals. Otherwise, all RHR pumps start immediately once diesel power is available (and normal power unavailable).

C. Manually stopping an RHR pump after LPCI initiation disables automatic restart of that pump until the initiation signal is reset. The affected RHR pump can still be started manually.

3.3 Shutdown Cooling

A. Prior to initiating Shutdown Cooling, RHR should be flushed to Radwaste until conductivity is less than 2.0 micromho/cm with less than 0.1 ppm chlorides (unless directed otherwise by 2-AOI-74-1, Loss of Shutdown Cooling). If CS&S has been aligned as the keep fill source for two days or more a chemistry sample should be requested and results analyzed to determine if flushing is required.

B. When in Shutdown Cooling, reactor temperature should be maintained greater than 72°F and only be controlled by throttling RHRSW flow. This is to assure adequate mixing of reactor water.

1. [NER/C] Reactor vessel water temperatures below 68°F exceed the temperature reactivity assumed in the criticality analysis. [INPO SER 90-017]

2. [NER/C] Maintaining water temperature below 100°F minimizes the release of soluble activity. [GE SIL 541]

C. Shutdown Cooling operation at saturated conditions (212°F) with 2 RHR pumps operating at or near combined maximum flow (20,000 gpm) could cause Jet Pump Cavitation. Indications of Jet Pump Cavitation are as follows:

1. Rise in RHR System flow without a corresponding rise in Jet Pump flow.

2. Fluctuation of Jet Pump flow.

3. Louder "Rumbling" noise heard when vessel head is off.

Corrective action for any of these symptoms would be to reduce RHR flow until the symptom is corrected.

23. RO 300000K2.02 001/MEM/T2G1/CA//300000K2.02/2.8/2.8/RO/SRO/BANK

Which ONE of the following lists the current power supplies to the Control and Service Air Compressor motors?

A. 'A' and 'B' are fed from 480V Common Board #1
'C' and 'D' from 480V Shutdown Boards '1B' & '2B', respectively
'G' from 4KV Shutdown Board 'B' and 480V Shutdown Board '2A'
'E' from 480V Common Board #1

B. 'A' and 'D' are fed from 480V Common Board #1
'B' and 'C' from 480V Shutdown Boards '1B' & '2B', respectively
'G' from 4KV Shutdown Board 'B' and 480V RMOV Board '2A'
'F' from 480V Common Board #3

C. 'A' is fed from 480V Shutdown Board '1B'
'B' and 'F' from 480V Common Board #3
'C' from 480V Shutdown Board '1A'
'D' from 480V Shutdown Board '2A'
'G' from 4KV Common Board #2

D.✓ 'A' is fed from 480V Shutdown Board '1B'
'B' and 'C' from 480V Common Board #1
'D' from 480V Shutdown Board '2A'
'G' from 4KV Shutdown Board 'B' and 480V RMOV Board '2A'
'E' from 480V Common Board #3

K/A Statement:

300000 Instrument Air
K2.02 - Knowledge of electrical power supplies to the following: Emergency air compressor

K/A Justification: This question satisfies the K/A statement by requiring the candidate to use specific knowledge of the power supplies of ALL air compressors.

References: OPL171.054 rev 12, pg 9, 10, 14 & 30

Level of Knowledge Justification: This question is rated as MEM due to the requirement to recall or recognize discrete bits of information.

0610 NRC Exam

REFERENCE PROVIDED: None

Plausibility Analysis:

In order to answer this question correctly the candidate must determine the following:

1. Power supplies to the six air compressors.

NOTE: Regarding plausibility, all the power supplies listed in the distractors are capable of supplying power to an air compressor.

D - correct: The air compressor power supplies are as follows:
'A' is fed from 480V Shutdown Board '1B'
'B' and 'C' from 480V Common Board #1
'D' from 480V Shutdown Board '2A'
'G' from 4KV Shutdown Board B and 480V RMOV Board '2A'
'E' from 480V Common Board #3

A - incorrect: B, G & E are correct. A, C & D are NOT correct.

B - incorrect: F & G are correct. A, B, C, & D are NOT correct.

C - incorrect: A, D & F are correct. B, C & G are NOT correct.

X. Lesson Body

A. Control Air System

1. **The purpose of the Control Air System is to process and distribute oil-free control air, dried to a low dew point and free of foreign materials. This high-quality air is required throughout the plant and yard to ensure the proper functioning of pneumatically operated instruments, valves, and final operators.

   ** SOER 88-1
   Obj. V.E.1

2. Basic Description of Flow Path

   TP-1

   a. The station control air system has 5 air compressors, each designed for continuous operation.

      Obj. V.E.3
      Obj. V.D.1

   b. Common header (fed by air compressors **A-D** and **G**)

      (1) The control air system is normally aligned with the **G** air compressor running and loaded. The existing **A-D** air compressors are aligned with one in second lead, one in third lead, and at least one compressor in standby.

      The **G** air compressor will be discussed later in this section of the lesson plan.

      (2) 3 control air receivers

      (3) 4 dual dryers One for each unit's control air header (units 1, 2 & 3 through their 4-inch headers) and One standby dryer supplies the standby, 3- inch common control air header for all three units

      normally aligned to all three units

      (4) Outlet from large service air receiver is connected to the control air receivers through a pressure control valve 0-FCV-33-1, which will automatically open to supply service air to the control air header if control air pressure falls to 85 psig.

   c. 4-inch control air header (1 per unit) is supplied from each unit dryer and backed up by a common, 3-inch standby header.

      TP-1

3. Control Air System Component Description

   a. Four Reciprocating Air Compressors **A-D** (2-stage, double acting, Y-type) are located El 565, U-1 Turbine Building.

      (1) Supply air to the control air receivers at 610 scfm each at a normal operating pressure of 90 - 101 psig.

      (2) 480V, 60 Hz, 3-phase, drive motors

      (3) Power supplies

      A from **480V Shutdown Board 1B**

D from **480V Shutdown Board 2A**

B from **480V Common Board 1**

C from **480V Common Board 1**

(a) Control air compressors which are powered from the 480 VAC shutdown boards are tripped automatically due to: Obj. V.B.1. Obj. V.C.1.

i. under voltage on the shutdown board.

ii. load shed logic during an accident signal concurrent with a loss of offsite power.

<u>**NOTE**</u>: The compressors must be restarted manually after power is restored to the board.

(b) Units powered from common boards also trip due to under voltage.

(4) Lubrication provided from attached oil system via gear-type oil pump

(a) Compressor trips on lube oil pressure < 10 psig or lube oil temperature >180 °F Obj. V.B.2. Obj. V.C.2. Obj. V.E.12 Obj. V.D.10

(b) Compressor cylinder is a non lubricated type

(5) Cooling water is from the Raw Cooling Water system with backup from EECW

(a) Compressor oil cooler, compressor inter-cooler, after cooler and cylinder water jackets

(b) Compressor inter-cooler and after cooler moisture traps drain moisture to the Unit 1 station sump.

<u>**NOTE**</u>: Cooling water flows to the compressors are regulated such that the RCW outlet temperature is maintained between 70° F and 100° F. Outlet temperatures should be adjusted low in the band (high flow rates) during warm seasons (river temps. ≥ 70°F). Outlet temperatures should be adjusted high in the band during the cooler seasons (river temps ≤ 70°F) to reduce condensation in the cylinders. Obj. V.B.2. Obj. V.C.2. Obj. V.E.12

(c) Compressor auto trips if discharge temperature of air > 310° F.

b. Unloaders Obj. V.D.10

(b) Should both the primary and the backup controllers fail, all four compressors will come on line at full load until these pressure switches cause the compressors to unload at 112 psig.

Cutout switch setpoints are set at 112 psig to prevent spurious operation when **G** air compressor running

(c) When air pressure drops below the high pressure cutoff setpoint (110.8 psig), the compressors will again come on line at full load until the high pressure cutoff switches cause the compressors to unload.

d. Relief valves on the compressors discharge set at 120 psig protects the compressor and piping.

e. **G** Air Compressor - centrifugal type, two stage

(1) Located 565' EL Turbine Bldg., Unit 1 end. Control Air Compressor **G** is the primary control air compressor and provides most of the control air needed for normal plant operation.

(2) Rated at 1440 SCFM @ 105 psig.

(3) Power Supply

(a) **4 kV Shutdown Board B** supplies power to the compressor motor.

(b) **480 V RMOV Bd. 2A** Supplies the following:

- Pre lube pump
- Oil reservoir heater
- Cooling water pumps
- Panel(s) control power
- Auto Restart circuit

(c) Except for short power interruptions on the **480v RMOV Bd**, Loss of either of these two power supplies will result in a shutdown of the **G** air compressor.

(4) A complete description of the **G** Air compressor controls and indications can be found in 0-OI-32. (The **G** and the **F** air compressor indications and Microcontrollers are similar).

Cover OI illustrations

(a) **UNLOAD MODULATE AUTO DUAL** handswitch is used to select the mode of operation for the compressor

TP-8

3. Component Description — Obj. V.E.6

   a. Compressors **E** and **F** (EL 565, U-3 Turbine Building) are designated for service air. — Obj. V.D.4

   b. The **F** air compressor is rated for approximately 630 SCFM @ 105 psig, centrifugal type, 2 stages

   c. The power supply for both compressors is **480VAC Common Board 3**.

   d. **F/G** air compressor comparison

      (1) Controls are similar to that of the **G** air compressor. There is no 4KV breaker control on the **F** air compressor control panel. — TP-16 Obj.V.E.7 Obj. V.D.5

      (2) Control system modulates discharge air pressure in the same manner as is done on the **G** air compressor. — Set to control at approx. 95 psig - Relief Valve is set to lift at ≈ 115 psig.

      (3) Air system is similar to the **G** air compressor. A difference is that the 2 stages of compression are driven by one shaft for the **F** air compressor. On the **G** air compressor, there is a separate drives; one for each of 3 compression stages. — TP-17

      (4) Oil system similar to that on the **G** air compressor with exception of location of components and capacity. **E** compressor has an electric oil pump that runs whenever control power is on. — TP-18

      (5) Cooling system is similar to that on the **G** air compressor with exception of flow rate, location, and capacity of components. — TP-19

      (6) Loss of power will result in **F** air compressor trip, loss of the pre lube pump, and the cooling water pumps.

      (7) Restart of the compressor can be accomplished once the compressor has come to a full stop and any trip conditions cleared and reset.

   e. Alarms/Trips

      (1) The Alert and Shutdown setpoints for the **F** air compressor are listed in 0-OI-33. — See for latest setpoints

24. RO 300000K3.01 001/C/A/T2G1/SGT/B10B/300000K3.01/3.2/3.4/RO/SRO/BANK

A LOCA has occurred on Unit 1 and the Drywell is being vented to the Standby Gas Treatment System (SBGT), when a loss of the Control Air System occurs.

Which ONE of the following describes the impact of the loss of air, on the operation of DRYWELL VENT INBD ISOL VALVE (1-FCV-64-29), and PATH B VENT FLOW CONT VALVE (1-FCV-84-19)?

Both vent valves 1-FCV-64-29 & 1-FCV-84-19 will fail CLOSED ________________ ____.

A. but, will automatically re-open as soon as INST GAS SELECTOR VALVE (1-PCV-84-33) automatically swaps to CAD Nitrogen Storage Tank 'A.'

B. and must be manually re-opened as soon as INST GAS SELECTOR VALVE (1-PCV-84-33) automatically swaps to CAD Nitrogen Storage Tank 'A.'

C.✓ but, will automatically re-open as soon as CAD SYSTEM A N2 SHUTOFF VALVE (0-FCV-84-5) is manually opened from Main Control Room Panel 1-9-54.

D. and must be manually re-opened as soon as CAD SYSTEM A N2 SHUTOFF VALVE (0-FCV-84-5) is manually opened from Main Control Room Panel 1-9-54.

K/A Statement:

300000 Instrument Air
K3.01 - Knowledge of the effect that a loss or malfunction of the (INSTRUMENT AIR SYSTEM) will have on the following: Containment air system.

K/A Justification: This question satisfies the K/A statement by requiring the candidate to use specific plant conditions to determine the effect on the containment air system due to a loss of Control Air.

References: 1-EOI Appendicies 8G and 12, 1-AOI-32-2 P&L 3.0.D

Level of Knowledge Justification: This question is rated as MEM due to the requirement to recall or recognize discrete bits of information.

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REFERENCE PROVIDED: None

Plausibility Analysis:

In order to answer this question correctly the candidate must determine the following:

1. Whether the vent valves automatically swap to be supplied by CAD/Nitrogen tank 'A' or must be manually aligned.
2. Whether the CAD supply to DW Control Air automatically swaps or must be manually aligned.
3. Failure positions of the Air-Operated Vent Valves 1-FCV-84-19 and 1-FCV-64-29.

C - correct: On a loss of Control Air, INST GAS SELECTOR VALVE (1-PCV-84-33) automatically repositions to align CAD tank 'A' (nitrogen) supply to 1-FCV-84-19 and 1-FCV-64-29; to allow venting containment (if necessary, during this event). Even though the 84-33 valve automatically re-aligns, there is a manual operation that must take place to actually align Nitrogen from CAD Tank 'A' to the supply line itself. This manual operation is the opening of CAD SYSTEM 'A' N2 SHUTOFF VALVE (0-FCV-84-5) from Control Room Panel 1-9-54; which is in an odd location (around the back side of the panel and often overlooked). Once this manual action is complete, then and only then will the Air-Operated Vent Valves 1-FCV-84-19 and 1-FCV-64-29 re-open (automatically) to re-establish the vent path from containment. This also involves a recent plant modification, wherein air compressors were previously tied into this air supply line and maintained it pressurized. The air compressors became obsolete, from a parts standpoint, and were thus removed from the system. Now, CAD is the ONLY supply to pressurize the line when Control Air is lost.

A - incorrect: This is plausible because the vent valves in question will automatically re-open AND the INST GAS SELECTOR VALVE (1-PCV-84-33) does, in fact, automatically swaps to CAD Nitrogen Storage Tank 'A.' This centers upon the candidate identifying the impact of the recent plant modification in combination with the potentially obscure operation of having to manually open CAD SYSTEM 'A' N2 SHUTOFF VALVE (0-FCV-84-5) from Control Room Panel 1-9-54.

B - incorrect: This is plausible because the vent valves in question will fail closed (and are required to be in the OPEN position to continue venting containment) AND the INST GAS SELECTOR VALVE (1-PCV-84-33) does, in fact, automatically swaps to CAD Nitrogen Storage Tank 'A.' This centers upon the candidate identifying the impact of the recent plant modification, the operation of having to manually open CAD SYSTEM 'A' N2 SHUTOFF VALVE (0-FCV-84-5), and the vent valve re-positioning schemes.

D - incorrect: This is plausible because the vent valves in question will fail closed (and are required to be in the OPEN position to continue venting containment) AND the potentially obscure operation of having to manually open CAD SYSTEM 'A' N2 SHUTOFF VALVE (0-FCV-84-5) from Control Room Panel 1-9-54.

Notes - Changed question substantially for plausibility purposes. Agree LOK is memory.

2.0 SYMPTOMS (continued)

- REACTOR CHANNEL A(B) AUTO SCRAM annunciator, (1-XA-55-5B, Window 1(2)) in alarm.
- MAIN STEAM RELIEF VLV AIR ACCUM PRESS LOW annunciator, (1-XA-55-3D, Window 18) in alarm.

3.0 AUTOMATIC ACTIONS

A. U-1 TO U-2 CONT AIR CROSSTIE, 1-PCV-032-3901, will CLOSE to separate Units 1 & 2 when control Air Header Control Air Header pressure reaches 65 psig lowering at the valve.

B. UNIT 2 TO UNIT 3 CONTROL AIR CROSSTIE, 2-PCV-032-3901, will CLOSE to separate Units 2 and 3 when Control Air Header pressure reaches 65 psig lowering at the valve.

C. CAD SUPPLY PRESS REGULATOR, 1-PCV-084-0706, will select nitrogen from CAD Tank A at ≤ 75 psig Control Air pressure to supply the following:

1. SUPPR CHBR VAC RELIEF VALVE, 1-FSV-064-0020
2. SUPPR CHBR VAC RELIEF VALVE, 1-FSV-064-0021

D. INST GAS SELECTOR VALVE, 1-PCV-084-0033, will select nitrogen from CAD Tank A to supply the following:

1. DRYWELL OR SUPPRESS CHMBR EXHAUST TO SGTS, 1-FSV-084-0019
2. DRYWELL VENT INBD ISOL VALVE, 1-FSV-064-0029
3. SUPPR CHMBR VENT INBD ISOL VALVE, 1-FSV-064-0032

E. INST GAS SELECTOR VALVE, 1-PCV-084-0034, will select nitrogen from CAD Tank B to supply the following:

1. DRYWELL OR SUPPRESS CHMBR EXHAUST TO SGTS, 1-FSV-084-0020
2. DRYWELL INBD ISOLATION VLV, 1-FSV-064-0031
3. SUPPR CHBR INBD ISOLATION VLV, 1-FSV-064-0034.

4.2 Subsequent Actions (continued)

NOTE
CNDS BSTR PMPS DISCH BYPASS TO COND 1C, 1-FCV-002-0029A and CNDS BSTR PMPS DISCH BYPASS TO COND 1B, 1-FCV-002-0029B both fail CLOSED on a loss of control air.

[3] **IF** there is **NOT** a flow path for Condensate system, **THEN**

STOP the Condensate Pumps and Condensate Booster Pumps. **REFER TO** 1-OI-2. ☐

[4] **IF** any Outboard MSIV closes, **THEN**

PLACE the associated handswitch on Panel 1-9-3 in the CLOSE position. ☐

NOTE
RSW STRG TNK ISOLATION, 0-FCV-25-32, fails CLOSED on loss of control air.

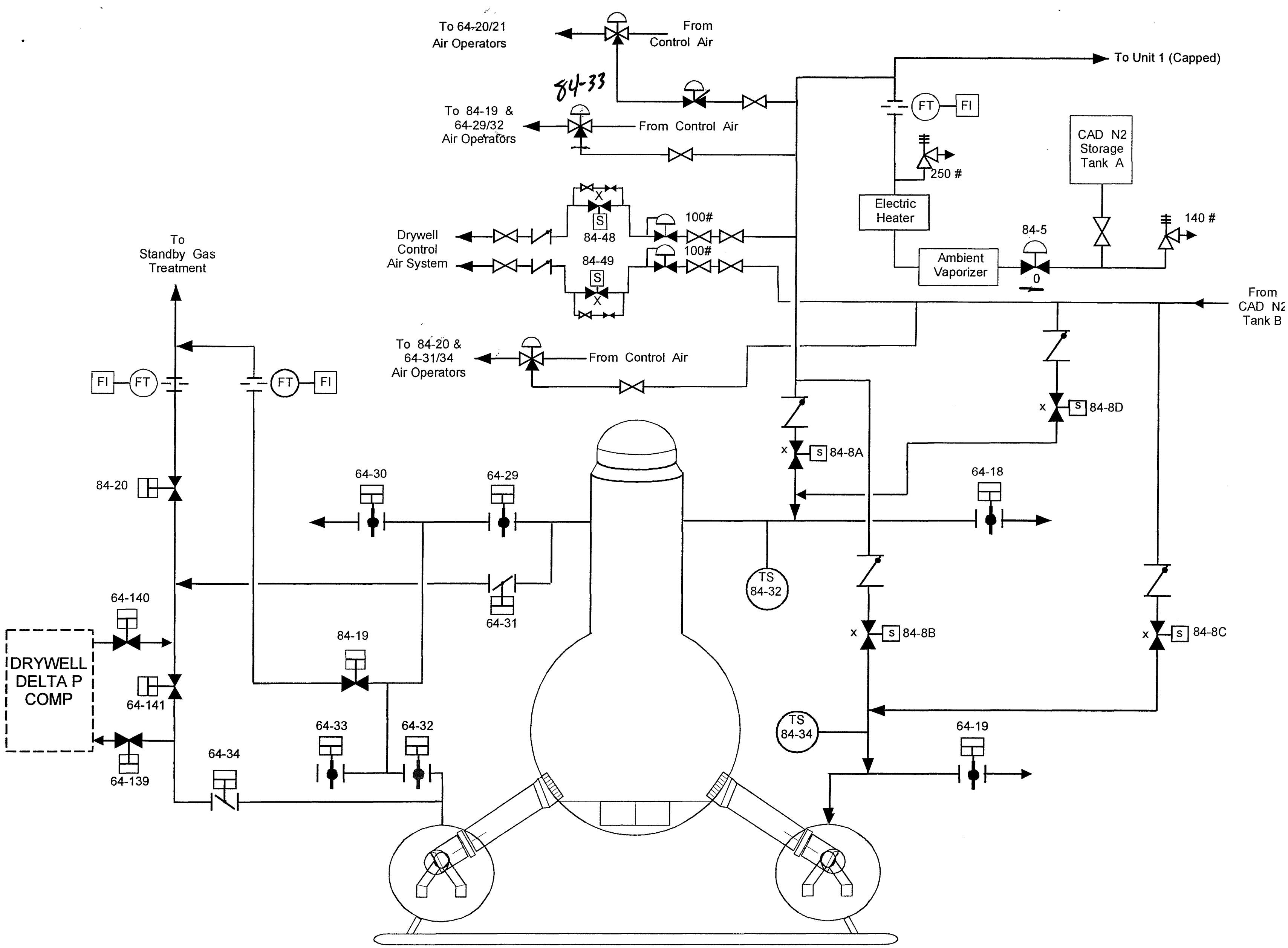
[5] **START** a High Pressure Fire Pump. **REFER TO** 0-OI-26. ☐

[6] **OPEN** CAD SYSTEM A N2 SHUTOFF VALVE, 0-FCV-84-5, at Panel 1-9-54. ☐

[7] **OPEN** CAD SYSTEM B N2 SHUTOFF VALVE, 0-FCV-84-16, at Panel 1-9-55. ☐

[8] **CHECK** RCW pump motor amps and **PERFORM** Steps 4.2[8.1] through 4.2[8.5]to reduce RCW flow:

TP-3: CONTAINMENT ATMOSPHERE DILUTION SYSTEM



f. **VERIFY** 1-FIC-84-20, PATH A VENT FLOW CONT, is indicating approximately 100 scfm. ______

g. **CONTINUE** in this procedure at step 12. ______

10. **VENT** the Drywell using 1-FIC-84-19, PATH B VENT FLOW CONT, as follows:

a. **VERIFY CLOSED** 1-FCV-64-141, DRYWELL DP COMP BYPASS VALVE (Panel 1-9-3). ______

b. **PLACE** keylock switch 1-HS-84-36, SUPPR CHBR/DW VENT ISOL BYP SELECT, to DRYWELL position (Panel 1-9-54). ______

c. **VERIFY OPEN** 1-FCV-64-29, DRYWELL VENT INBD ISOL VALVE (Panel 1-9-54). ______

d. **PLACE** 1-FIC-84-19, PATH B VENT FLOW CONT, in AUTO with setpoint at 100 scfm (Panel 1-9-55). ______

e. **PLACE** keylock switch 1-HS-84-19, 1-FCV-84-19 CONTROL, in OPEN (Panel 1-9-55). ______

f. **VERIFY** 1-FIC-84-19, PATH B VENT FLOW CONT, is indicating approximately 100 scfm. ______

g. **CONTINUE** in this procedure at step 12. ______

11. **VENT** the Drywell using 1-FIC-84-20, PATH A VENT FLOW CONT, as follows:

a. **VERIFY CLOSED** 1-FCV-64-141, DRYWELL DP COMP BYPASS VALVE (Panel 1-9-3). ______

b. **PLACE** keylock switch 1-HS-84-35, SUPPR CHBR / DW VENT ISOL BYP SELECT, to DRYWELL position (Panel 1-9-54). ______

c. **VERIFY OPEN** 1-FCV-64-31, DRYWELL INBD ISOL VALVE (Panel 1-9-54). ______

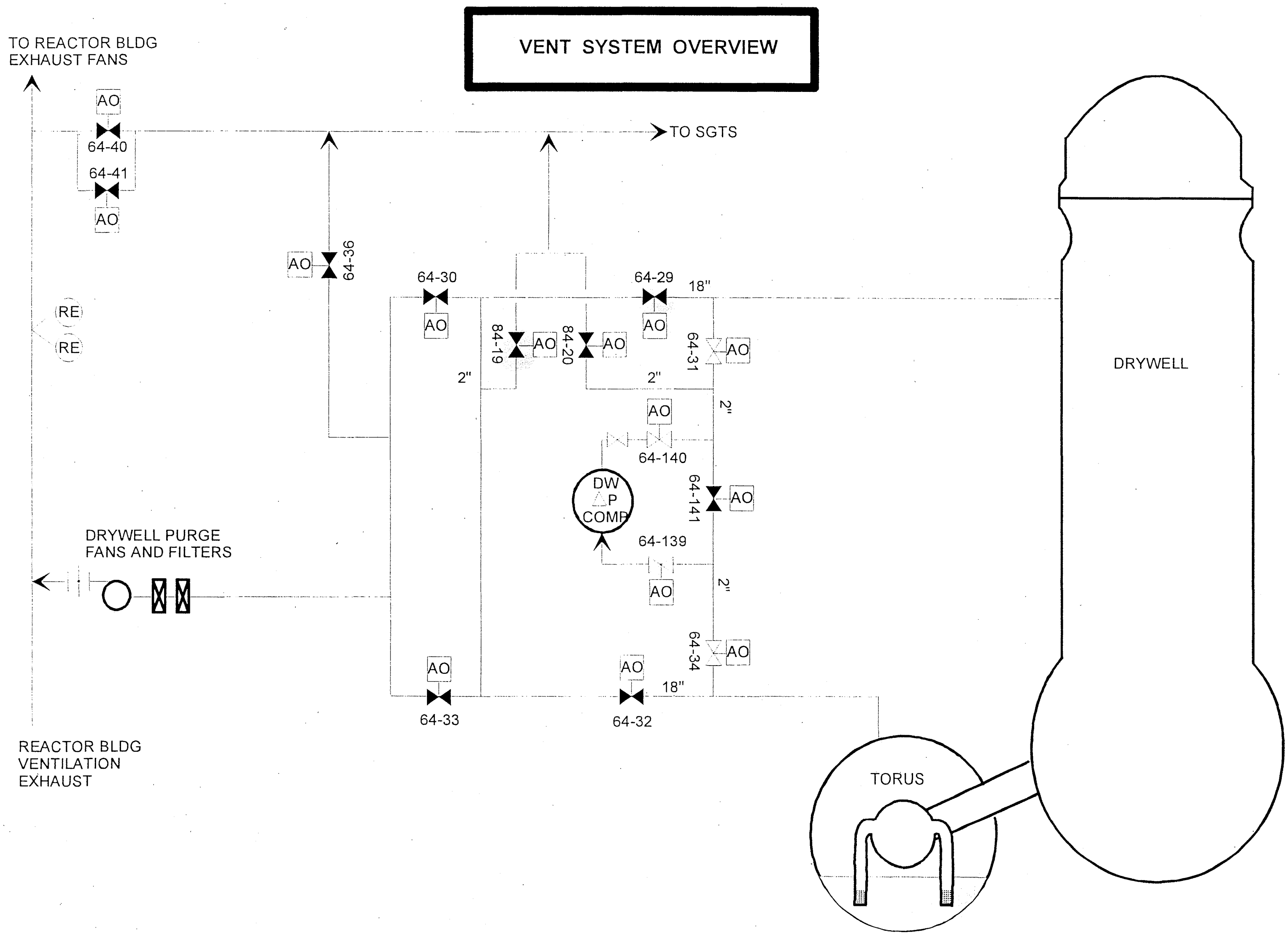
d. **VERIFY** 1-FIC-84-20, PATH A VENT FLOW CONT, in AUTO with setpoint at 100 scfm (Panel 1-9-55). ______

e. **PLACE** keylock switch 1-HS-84-20, 1-FCV-84-20 ISOLATION BYPASS, in BYPASS (Panel 1-9-55). ______

f. **VERIFY** 1-FIC-84-20, PATH A VENT FLOW CONT, is indicating approximately 100 scfm. ______

PRIMARY CONTAINMENT VENTING

ATTACHMENT 1



LOCATION:	Unit 1 Control Room	
ATTACHMENTS:	None	(√)

1. **OPEN** the following valves:

   - 0-FCV-84-5, CAD A TANK N2 OUTLET VALVE (Unit 1, Panel 1-9-54) ____

   - 0-FCV-84-16, CAD B TANK N2 OUTLET VALVE (Unit 1, Panel 1-9-55). ____

2. **VERIFY** 0-PI-84-6, VAPOR A OUTLET PRESS, and 0-PI-84-17, VAPOR B OUTLET PRESS, indicate approximately 100 psig Panel 1-9-54 and Panel 1-9-55). ____

3. **PLACE** keylock switch 1-HS-84-48, CAD A CROSS TIE TO DW CONTROL AIR, in OPEN (Panel 1-9-54). ____

4. **CHECK OPEN** 1-FSV-84-48, CAD A CROSS TIE TO DW CONTROL AIR, (Panel 1-9-54). ____

5. **PLACE** keylock switch 1-HS-84-49, CAD B CROSS TIE TO DW CONTROL AIR, in OPEN (Panel 1-9-55). ____

6. **CHECK OPEN** 1-FSV-84-49, CAD B CROSS TIE TO DW CONTROL AIR (Panel 1-9-55). ____

7. **CHECK** MAIN STEAM RELIEF VLV AIR ACCUM PRESS LOW, 1-PA-32-31, alarm cleared (1-XA-55-3D, Window 18). ____

8. IF .... MAIN STEAM RELIEF VLV AIR ACCUM PRESS LOW, 1-PA-32-31, annunciator is or remains in alarm (1-XA-55-3D, Window 18),

   THEN........... **DETERMINE** which Drywell Control Air header is depressurized as follows:

   a. **DISPATCH** personnel to Unit 1, RB, El 565 ft, to **MONITOR** the following indications for low pressure:

      - 1-PI-084-0051, DW CONT AIR N2 SUPPLY PRESS indicator, for CAD A (RB, El. 565, by Drywell Access Door), ____

      - 1-PI-084-0050, DW CONT AIR N2 SUPPLY PRESS indicator, for CAD B (RB, El. 565, left side of 480V RB Vent Board 1B). ____

25. RO 400000A2.02 001/C/A/T2G1/RBCCW//400000A2.02/3.8/4.1/RO/SRO/NEW 01/20/2008

Unit 2 is operating at 100% power, and the following annunciators are received:

- DRYWELL EQPT DR SUMP TEMP HIGH (2-XA-55-4C, Window 16)
- RBCCW SURGE TANK LEVEL LOW (2-XA-55-4C, Window 13)
- RBCCW PUMP DISCH HDR PRESS LOW (2-XA-55-4C, Window 12)
- RBCCW 2-FCV-70-48 CLOSED (2-XA-55-4C, Window 19)

Which ONE of the following describes the impact on Reactor Building Closed Cooling Water (RBCCW) System based on these indications, and the correct actions required?

A. RBCCW flow to the RWCU Non-Regenerative Heat Exchanger has been lost. Enter 2-AOI-70-1, "Loss of RBCCW" and immediately trip the RWCU pumps. Isolate the RWCU syst.

B. RBCCW flow to the Drywell has been lost. Enter 2-AOI-70-1, "Loss of RBCCW", Scram the Reactor, and immediately trip the Recirc Pumps.

C. Makeup to the RBCCW Surge Tank in accordance with the Alarm Response Procedure; then, place the Spare RBCCW Pump in service and re-open RBCCW SECTIONALIZING VALVE (2-FCV-70-48).

D.✓ RBCCW flow to the RWCU Non-Regenerative Heat Exchanger has been lost. Enter 2-AOI-70-1, "Loss of RBCCW" and remove the RWCU pumps from service IF RBCCW suction temperature cannot be maintained below 105°F.

K/A Statement:

400000 Component Cooling Water
A2.02 - Ability to (a) predict the impacts of the following on the CCWS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation: High/low surge tank level

K/A Justification: This question satisfies the K/A statement by requiring the candidate to use specific plant conditions to determine the effect of a leak into the RBCCW system and determine which procedure addresses this condition.

References: 2-AOI-70-1, 4.2[7], 1-ARP-9-4C rev. 0015 window 12, 13, 16 & 19

Level of Knowledge Justification: This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

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MODIFIED FROM OPL171.047 #9

REFERENCE PROVIDED: None

Plausibility Analysis:

In order to answer this question correctly the candidate must determine the following:

1. The impact on RBCCW based on given conditions.
2. What actions would be required to mitigate the problem.

D - correct: With the given alarms it can be deduced that there has been a loss of the RBCCW non-essential loop. The appropriate action would be to enter 2-AOI-70-1 and shutdown the RWCU pumps (when the RBCCW suction temperature exceeds 105^{o}F).

A - incorrect: RBCCW has been lost to the Non-regenerative Heat Exchanger but it is not appropriate to immediatley trip the RWCU pumps. This is plausible due to the immediate actions in 2-AOI-70-1 which have the operator trip the RWCU pumps in the event an RBCCW trips and can't be restored.

B - incorrect: It cannot be determined from the given conditions that Drywell cooling has been lost. 2-FCV-70-47 isolates the Drywell; NOT 2-FCV-70-48. This is plausible because the given conditions indicate the leak is possibly located inside the Drywell, however it is not appropriate to Scram the Reactor and trip the Recirc pumps at this time.

C - incorrect: This is plausible because all of the actions are inaccordance with approved plant procedures but placing the Spare pump inservice is not applicable for this event.

Note - rewrote the stem and 3 of the distractors.

4.2 Subsequent Actions (continued)

NOTE

Opening RBCCW TVC Bypass valves may cause an EECW pump start due to low pressure.

[7] **IF** RBCCW PUMP SUCTION HDR TEMP, 2-TIS-70-3 (Panel 2-9-4), cannot be maintained below 105°F, **THEN**

PERFORM the following (otherwise **N/A**):

- **SLOWLY OPEN** bypass valves around RBCCW TCVs. ☐
- **PLACE** Spare RBCCW Heat Exchanger in service. **REFER TO** 2-OI-70. ☐
- **VERIFY** RWCU System removed from service. ☐

NOTE

It may be necessary to place the RHR System in the Supplemental Fuel Pool Cooling Mode to maintain Fuel Pool temperature below 150°F (Tech. Specs. 3.10.C.2, TRM 3.9.2).

- **SHUT DOWN** Fuel Pool Cooling System. **REFER TO** 2-OI-78. ☐

RBCCW PUMP
DISCH. HDR
Press LOW
2-PA-70-15

12

(Page 1 of 1)

Sensor/Trip Point:

2-PS-70-15 — 50 psig (57 psig on local pump discharge pressure gauge)

Sensor Location:
Panel 25-8
Elevation 593
Column R-13 R-LINE

Probable Cause:
A. Failed pump.
B. Broken coupling.
C. Sensor malfunction.

Automatic Action: Closes 2-FCV-70-48, non-essential loop, closed cooling water sectionalizing MOV.

Operator Action:

A. **VERIFY** 2-FCV-70-48 CLOSING/CLOSED. ☐
B. **VERIFY** RBCCW pumps A and B in service. ☐
C. **VERIFY** RBCCW surge tank low level alarm is reset. ☐
D. **DISPATCH** personnel to check the following: ☐
- RBCCW surge tank level locally. ☐
- RBCCW pumps for proper operation. ☐

E. **REFER TO** 2-AOI-70-1 for RBCCW System failure and 2-OI-70, for starting spare pump. ☐

References: 45N620-4 45N779-6 2-47E610-70-1
FSAR Sections 10.6.4 and 13.6.2

RBCCW SURGE TANK LEVEL LOW 2-LA-70-2B
13

(Page 1 of 1)

Sensor/Trip Point:

2-LS-70-2B 4 inches below center line of tank

Sensor Location: On the RBCCW surge tank in the MG Set Room, El 639'

Probable Cause:

A. Normal leakage.
B. Drain valves open.
C. Abnormal leakage.

Automatic Action: None

Operator Action:

A. **ADD** water to the RBCCW Surge Tank for approximately one minute or until low level alarm resets using the following: ☐
- RBCCW SYS SURGE TANK FILL VLV, 2-FCV-70-1 (Panel 2-9-4) OR ☐
- FCV-70-1 BYPASS VLV, 2-HCV-2-1369 (locally). ☐

B. **IF** alarm does **NOT** resets, **THEN** **CHECK** tank locally. ☐
C. **IF** unable to maintain RBCCW Surge Tank level, **THEN** **REFER TO** 2-AOI-70-1. ☐
D. **IF** necessary to add water more than once per shift, **THEN** **CHECK** Drywell floor drain system for excessive operation AND **INSPECT** system outside Drywell for leakage. ☐

References: 45N614-4 2-47E610-70-1 45N620-4
FSAR Sections 10.6.4 and 13.6.2

DRYWELL
EQPT DR SUMP
TEMP HIGH
2-TA-77-14

16

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Sensor/Trip Point:

2-TS-077-0014 160°F

Sensor Location: Elevation 550 at Drywell equipment drain sump. 2-TM-77-14 on Panel 9-19 in Auxiliary Instrument Room.

Probable Cause:

A. Recirculation loop A or B drain and miscellaneous reactor drains have excessive leakage.
B. Loss of RBCCW.
C. Reactor refueling bellows excessive leakage.
D. CRD equipment service drain excessive leakage.

Automatic Action:

A. Starts sump pump.
B. Opens recirculation valve 2-FCV-77-14A and closes discharge valve 2-FCV-77-14B (on Panel 2-9-4).

Operator Action:

NOTE

DRYWELL EQPT DRAIN SUMP TEMP, 2-TIS-77-14 pump shutoff setpoint (Blue Pen) may be adjusted in the range of 115 to 125 degrees F. Adjustment of this setpoint with respect to seasonal changes in river temperature may minimize Drywell Equipment Drain Sump Recirculation Mode initiation frequency and/or minimize pump run time.

A. **CHECK OPEN** 2-FCV-77-14A ☐
B. **CHECK CLOSED** 2-FCV-77-14B. ☐
C. **CHECK** Drywell equipment drain sump pump running. ☐
D. **VERIFY** temperature lowers and alarm resets. ☐
E. **IF** RBCCW has been lost, **THEN** **REFER TO** 2-AOI-70-1. ☐
F. **IF** alarm does **NOT** resets after 15 minutes, **THEN** **NOTIFY** Shift Manager. ☐

References: 45N620-4 2-47E610-77-1 2-47E852-2

FSAR Sections 10.16.6 and 13.6.2

RBCCW 2-FCV-70-48 CLOSED 2-ZA-70-48 19

(Page 1 of 1)

Sensor/Trip Point:

Limit Switch on 2-FCV-70-48 — 2-FCV-70-48 FULLY CLOSED

Sensor Location: Elevation 593'
RBCCW System routing valve to the nonessential loop.

Probable Cause:

A. Manual operation of valve handswitch.
B. Low pump discharge header pressure ≤ 50 psig.
C. Re-energizing after a loss of normal AC power in conjunction with accident signal on affected or adjacent unit.
D. Failed pressure switch, 2-PS-070-0015, RBCCW PUMP DISCH HDR.

Automatic Action: None

Operator Action:

A. **IF** cause is accident signal AND loss of offsite power, **THEN** **REFER TO** 0-AOI-57-1A and 2-EOI-1 FLOWCHART and 2-EOI-2 FLOWCHART. ☐
B. **OPEN** 2-FCV-70-48, RBCCW Sectionalizing Valve, when conditions permit. ☐
C. **IF** unable to reopen 2-FCV-70-48, **THEN** if desired, **REMOVE** RWCU from service. **REFER TO** 2-OI-69. ☐
   1. **NOTIFY** CHEMISTRY if RWCU is removed from service (Reference TRM 3.4.1). ☐

D. **REFER TO** 2-AOI-70-1. ☐

References: 2-47E610-70-1 45N779-6 45N620-4
FSAR Sections 10.6.4 and 13.6.2

26. RO 400000G2.4.31 001/C/A/T2G1/RBCCW//4000002.4.30//RO/SRO/NEW 10/16/07

Unit 3 is at rated power with the following indications:

- RECIRC PUMP MTR 'B' TEMP HIGH (3-XA-9-4B, Window 13).
- RBCCW EFFLUENT RADIATION HIGH (3-XA-9-3A, Window 17).
- RBCCW SURGE TANK LEVEL HIGH (3-XA-9-4C, Window 6).
- RX BLDG AREA RADIATION HIGH (3-XA-9-3A, Window 22).
- Recirc Pump Motor '3B' Winding and Bearing Temperature Recorder (3-TR-68-84) is reading 170°F and rising.
- RBCCW Pump Suction Header Temperature Indicator (3-TIS-70-3) is reading 104°F and rising.
- RWCU NON-REGENERATIVE HX DISCH TEMP HIGH (3-XA-9-4C, Window 17).
- Area Radiation Monitor RE-90-13A and RE-90-14A are in alarm reading 55 mr/hr and rising.

Which ONE of the following describes the action(s) that should be taken in accordance with plant procedures?

REFERENCE PROVIDED

Enter 3-EOI-3, "Secondary Containment Control" and ______________________________

______________________________.

A.✓ Trip and isolate '3B' Recirc Pump.
Enter 3-AOI-68-1A "Recirc Pump Trip/Core Flow Decrease OPRMs Operable."

B. Trip and isolate '3B' Recirc Pump.
Immediately commence a shutdown in accordance with 3-GOI-100-12A, "Unit Shutdown."

C. Trip RWCU Pumps and isolate the RWCU system.
Commence a normal shutdown in accordance with 3-GOI-100-12A, "Unit Shutdown."

D. Trip RWCU pumps and isolate RWCU system.
Close RBCCW Sectionalizing Valve 3-FCV-70-48 to isolate non-essential loads and maximize cooling to '3B' Recirc Pump.

K/A Statement:

400000 Component Cooling Water
2.4.31 - Emergency Procedures / Plan Knowledge of annunciators alarms and indications, and use of the response instructions.

K/A Justification: This question satisfies the K/A statement by requiring the candidate to use specific plant conditions to determine the corrective actions required due to an emergency involving RBCCW based on annunciators and indications.

References: 3-EOI-3 flowchart, 3-AOI-68-1A rev 4, 4.2[8.4], 3-ARP-9-3A rev 36 window 17 and 3-ARP-9-4B rev 36 window 13

Level of Knowledge Justification: This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

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REFERENCE PROVIDED: 3-EOI-3 flowchart

Plausibility Analysis:

In order to answer this question correctly, the candidate must determine the following:

1. EOI Entry is required solely based on Area Radiation Monitor Alarms.
2. Location of the leak is from the 3B Recirc Pump.
3. RWCU temperature indications are due to insufficient cooling by RBCCW, not a RWCU leak.
4. Appropriate actions per 3-EOI-3 are to isolate the leak and monitor radiation levels.
5. Justification for Unit Shutdown and Cooldown are due to the Recirc Loop being isolated at rated temperature and pressure (pipe hanger and support issue), and NOT Directed by 3-EOI-3.

A - correct: EOI-3 entry is required due RE-90-13A & 14A in alarm. Once the 3B Recirc pump is isolated the appropriate action is to enter 3-AOI-68-1A which will eventually require the unit to be shutdown IAW GOI-100-12A due to not being able to maintain the required delta T between the isolated recirc loop and the steam dome temperature.

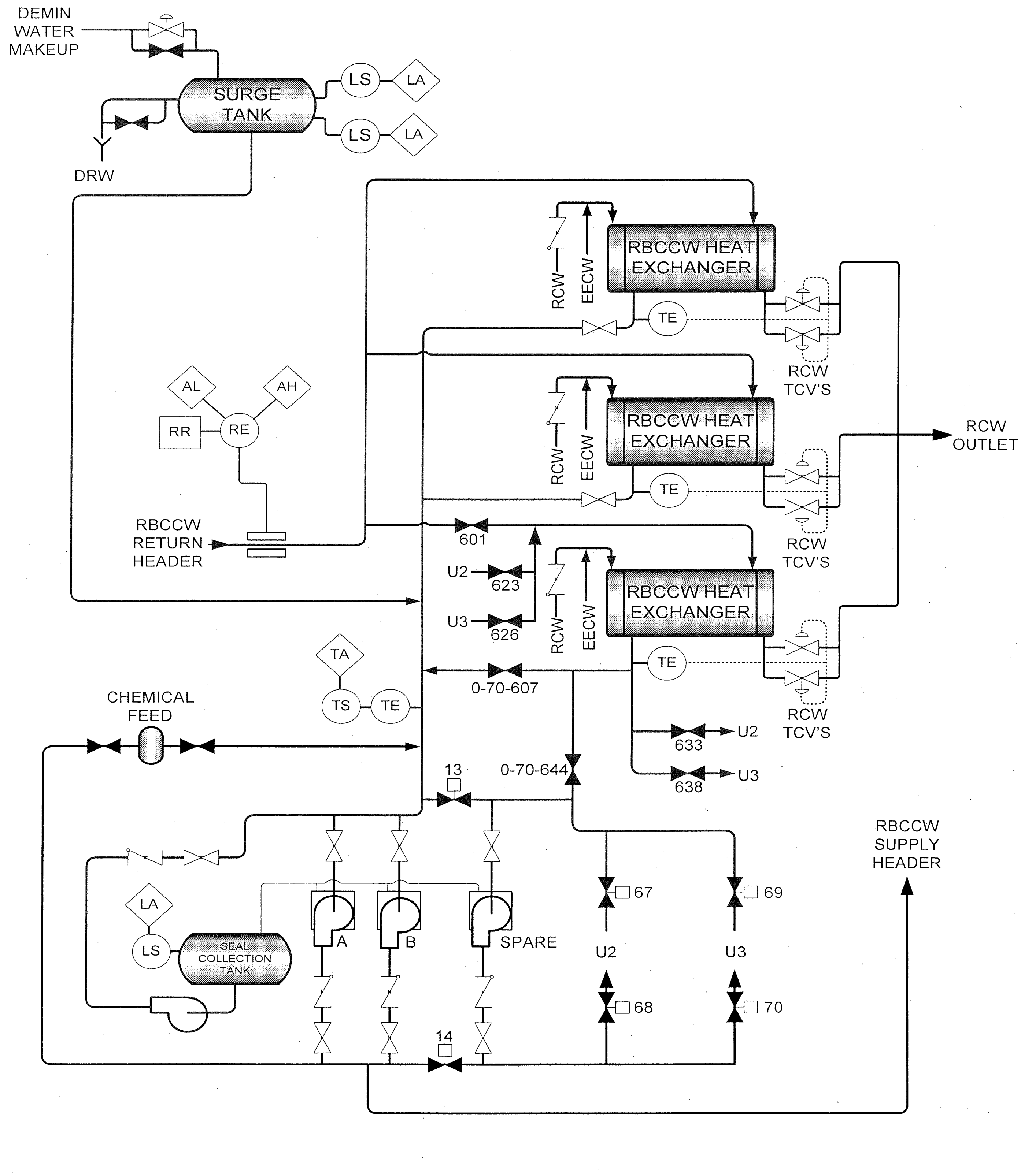
B - incorrect: An immediate shutdown is not required at this time. See 'A'. This is plausible because the location of the leak and requirement for isolation are correct.

C - incorrect: This is plausible if the candidate incorrectly determines that RWCU is causing the temperature issues with 3B Recirc Pump and not vice versa. In addition to the justification above, commencing a shutdown in accordance with 3-EOI-3 is not appropriate until ARMs indicate greater than 1000 mr/hr.

D - incorrect: This is plausible if the candidate incorrectly determines that RWCU is causing the temperature issues with 3B Recirc Pump and not vice versa. If RWCU was the leak location, the RBCCW temperature would NOT be high enough to provide the given indications. The leak would have to have occurred in the NRHX which is below the indicated RBCCW temperature.

Note - changed the RBCCW suction temp from 140°F to 104°F. A manual Scram is required at 110°F per AOI-70-1. Also changed distractor 'B' to enter AOI-68-1A which would be the correct actions under the conditions.

Changed the Stem, took Enter 3-EOI-3, "Secondary Containment Control" out of the distractors and put it in the Stem.



TP-1: RBCCW SYSTEM FLOW DIAGRAM

EOI - 3

TABLE 4
SECONDARY CONTAINMENT AREA RADIATION

AREA	APPLICABLE RADIATION INDICATORS	MAX NORMAL VALUE MR/HR	MAX SAFE VALUE MR/HR	POTENTIAL ISOLATION SOURCES
RHR SYS I PUMPS	90-25A	ALARMED	1000	FCV-74-47, 48
RHR SYS II PUMPS	90-28A	ALARMED	1000	FCV-74-47, 48
HPCI ROOM	90-24A	ALARMED	1000	FCV-73-2, 3, 81 FCV-73-44
CS SYS I PUMPS RCIC ROOM	90-26A	ALARMED	1000	FCV-71-2, 3, 39
CS SYS II PUMPS	90-27A	ALARMED	1000	NONE
TORUS GENERAL AREA	90-29A	ALARMED	1000	FCV-73-2, 3, 81 FCV-74-47, 48 FCV-71-2, 3
RB EL 565 W	90-20A	ALARMED	1000	FCV-69-1, 2, 12 SDV VENTS & DRAINS
RB EL 565 E	90-21A	ALARMED	1000	SDV VENTS & DRAINS
RB EL 565 NE	90-23A	ALARMED	1000	NONE
TIP ROOM	90-22A	ALARMED	100,000	TIP BALL VALVE
RB EL 593	90-13A, 14A	ALARMED	1000	FCV-74-47, 48
RB EL 621	90-9A	ALARMED	1000	FCV-43-13, 14
RECIRC MG SETS	90-4A	ALARMED	1000	NONE
REFUEL FLOOR	90-1A, 2A, 3A	ALARMED	1000	NONE

4.2 Subsequent Actions (continued)

CAUTION

The temperature of the coolant between the dome and the idle Recirc loop should be maintained within 75°F of each other. If this limit cannot be maintained, a plant cool down should be initiated. Failure to maintain this limit and not cool down could result in hangers and/or shock suppressers exceeding their maximum travel range. [GE SIL 251, 430 and 517]

[8] **IF** Recirc Pump was tripped due to dual seal failure, **THEN** (Otherwise N/A)

[8.1] **VERIFY** TRIPPED, RECIRC DRIVE 3A(3B) NORMAL FEEDER, 3-HS-57-17(14). ☐

[8.2] **VERIFY** TRIPPED, RECIRC DRIVE 3A(3B) ALTERNATE FEEDER, 3-HS-57-15(12). ☐

[8.3] **CLOSE** tripped recirc pump suction valve using, RECIRC PUMP 3A(3B) SUCTION VALVE, 3-HS-68-1(77). ☐

[8.4] **IF** it is evident that 75°F between the dome **AND** the idle Recirc loop cannot be maintained, **THEN**

COMMENCE plant shut down and cool down. Refer to 3-GOI-100-12A. ☐

[9] **NOTIFY** Reactor Engineer to **PERFORM** the following:

- Tech Specs 3.4.1 ☐
- 3-SR-3.4.1(SLO), Reactor Recirculation System Single Loop Operation ☐
- 0-TI-248, Core Flow Determination in Single Loop Operation ☐

[10] [NER/C] **WHEN** the Recirc Pump discharge valve has been closed for at least five minutes (to prevent reverse rotation of the pump) [GE SIL-517], **THEN** (N/A if Recirc Pump was isolated in Step 4.2[8])

OPEN Recirc Pump discharge valve as necessary to maintain Recirc Loop in thermal equilibrium. ☐

RBCCW EFFLUENT RADIATION HIGH 3-RA-90-131A 17
(Page 1 of 2)

Sensor/Trip Point:

	HI	HI-HI
RE-90-131D	(NOTE 2)	(NOTE 2)

Hi alarm from recorder
Hi-Hi alarm from drawer
(2) Chemlab should be contacted for current setpoints per 0-TI-45.

Sensor Location: RE-90-131A RBCCW HX Rx Bldg, El 593, R-20 S-LINE

Probable Cause: HX tube leak into RBCCW system.

Automatic Action: None

Operator Action:

A. **DETERMINE** cause of alarm by observing following:
   1. RBCCW and RCW EFFLUENT RADIATION recorder, 3-RR-90-131/132 Red pen on Panel 3-9-2. ☐
   2. RBCCW EFFLUENT OFFLINE RAD MON, 3-RM-90-131D on Panel 3-9-10. ☐

B. **NOTIFY** Chemistry to sample RBCCW for total gamma activity to verify condition. ☐

C. **START** an immediate investigation to determine if source of leak is RWCU Non-regenerative, Fuel Pool Cooling, Reactor Water Sample or RWCU Recirc Pump 3A or 3B Seal Water heat exchanger(s). ☐

D. [NER/C] **CHECK** Following for indication of Reactor Recirculation Pump Seal Heat Exchanger leak:
   1. LOWERING in reactor Recirculation pump 3A(3B) No. 1 or 2 SEAL, 3-PI-68-64A or 3-PI-68-63A (3-PI-68-76A or 3-PI-68-75A) on Panel 3-9-4. ☐
   2. Temperature rise on CLG WTR FROM SEAL CLG TE-68-54, on RECIRC PMP MTR 3A WINDING AND BRG TEMP temperature recorder, 3-TR-68-58, on Panel 3-9-21. ☐
   3. Temperature rise on CLG WTR FROM SEAL CLG TE-68-67, on RECIRC PMP MTR 3B WINDING AND BRG TEMP temperature recorder, 3-TR-68-84, on Panel 3-9-21. ☐

Continued on Next Page

RBCCW EFFLUENT RADIATION HIGH 3-RA-90-131A, Window 17
(Page 2 of 2)

Operator Action: (Continued)

E. **IF** it is determined the source of leakage is from Reactor Recirc Pump A(B), **THEN**
   1. **ISOLATE** Reactor Recirculation Loop A(B) per 3-OI-68, as applicable. ☐

NOTE

Cooldown is required to prevent hangers or shock suppressors from exceeding their maximum travel range.

   2. **WHEN** primary system pressure is less than 125 psig, **THEN** **ISOLATE** RBCCW System to preclude damage to RBCCW piping. [IEN 89-054, GE SIL-459] ☐

References: 3-45E620-3 3-47E610-90-3 GE 3-729E814-3

RECIRC PUMP MTR B TEMP HIGH 3-TA-68-84	13

(Page 1 of 1)

Sensor/Trip Point: Alarm is from 3-TR-68-84, Panel 3-9-2

3-TE-68-73A RECIRC PMP MTR 3B-THR BRG UPPER FACE (190°F)
3-TE-68-73C RECIRC PMP MTR 3B-THR BRG LOWER FACE (190°F)
3-TE-68-73E RECIRC PMP MTR 3B-UPPER GUIDE BRG (190°F)
3-TE-68-73N RECIRC PMP MTR 3B-LOWER GUIDE BRG (190°F)
3-TE-68-73G RECIRC PMP MTR 3B-MOTOR WINDING A (216°F)
3-TE-68-73J RECIRC PMP MTR 3B-MOTOR WINDING B (216°F)
3-TE-68-73L RECIRC PMP MTR 3B-MOTOR WINDING C (216°F)
3-TE-68-73T RECIRC PMP MTR 3B-SEAL NO. 2 CAVITY(180°F)
3-TE-68-73U RECIRC PMP MTR 3B-SEAL NO. 1 CAVITY(180°F)
3-TE-68-67 RECIRC PMP MTR 3B-CLG WTR FROM SEAL CLG (140°F)
3-TE-68-70 RECIRC PMP MTR 3B-CLG WTR FROM BRG (140°F)

Sensor Location: Temperature elements are located on recirculation pump motor, Elevation 563.12, Unit 3 drywell.

Probable Cause:

A. Possible bearing failure.
B. Possible motor overload.
C. Insufficient cooling water.
D. Possible seal failure.
E. High drywell temperature.

Automatic Action: None

Operator Action:

A. **CHECK** following on Panel 3-9-4: ☐
- RBCCW PUMP SUCTION HDR TEMP temperature indicating switch, 3-TIS-70-3 normal (summer 70-95°F, winter 60-80°F). ☐
- RBCCW PRI CTMT OUTLET handswitch, 3-HS-70-47A (3-FCV-70-47) OPEN. ☐

B. **CHECK** the temperature of the cooling water leaving the seal and bearing coolers < 140°F on RECIRC PMP MTR 3B WINDING AND BRG TEMP temperature recorder, 3-TR-68-84 on Panel 3-9-21. ☐

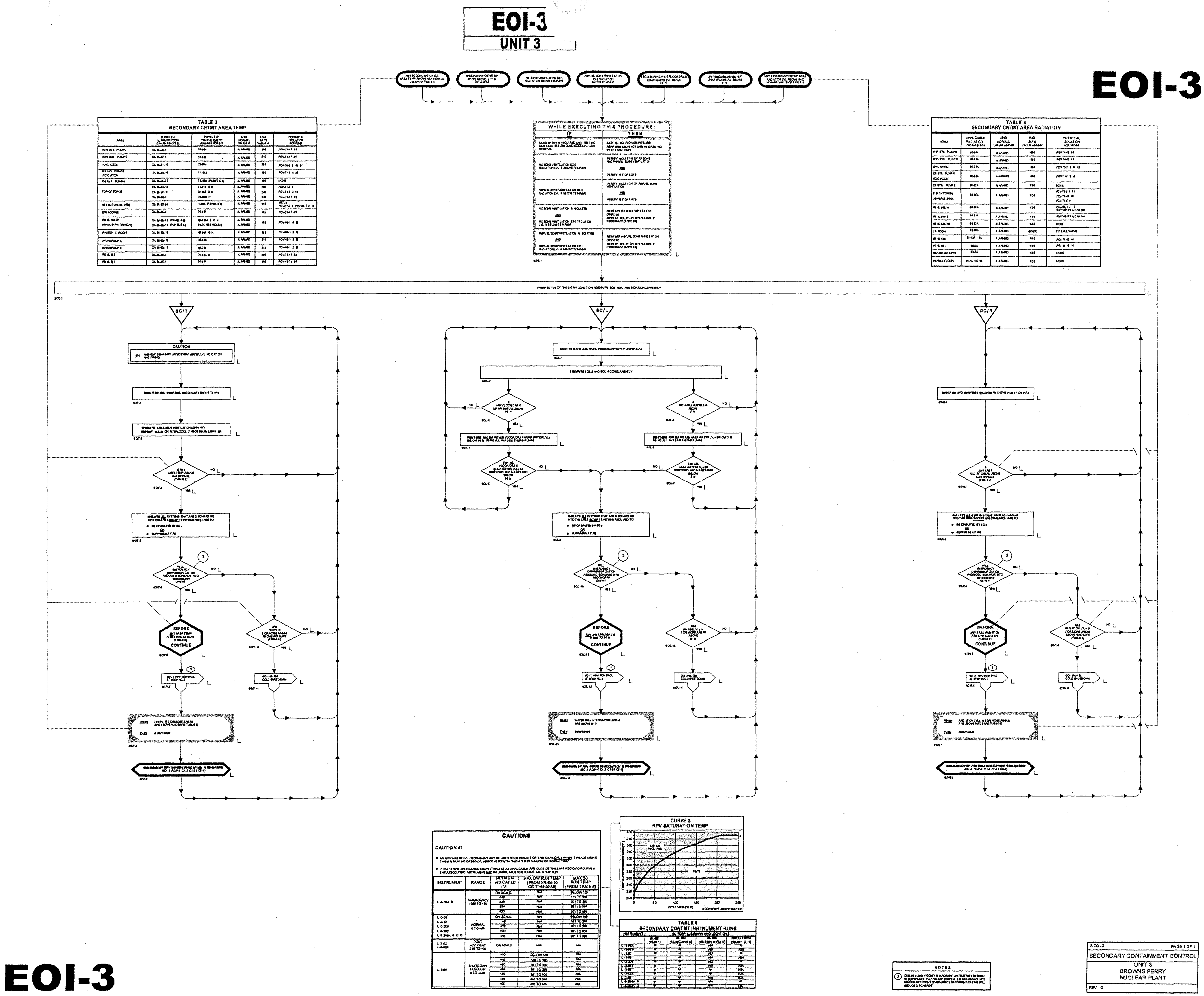
C. **LOWER** recirc pump speed until Bearing and/or Winding temperatures are below the alarm setpoint. ☐

D. **CONTACT** Site Engineering to **PERFORM** a complete assessment and monitoring of all seal conditions particularly seal leakage, temperature, and pressure of all stages for Recirc Pump seal temperatures in excess of 180°F. ☐

References:

3-45E620-5	3-47E610-68-1	Tech Spec 3.4.1
GE 731E320RE	3-SIMI-68B	FSAR Section 13.6.2

EXAMINATION REFERENCE PROVIDED TO CANDIDATE


EOI-3
UNIT 3
EOI-3
TABLE 3
SECONDARY CNTMT AREA TEMP
WHILE EXECUTING THIS PROCEDURE:
IF
THEN
TABLE 4
SECONDARY CNTMT AREA RADIATION
SC/T
SC/L
SC/R
CAUTION
BEFORE
CONTINUE
YES
NO
CAUTIONS
CAUTION #1
CURVE 8
RPV SATURATION TEMP
TABLE 6
SECONDARY CNTMT INSTRUMENT RUNS
NOTES
EOI-3
3-EOI-3
PAGE 1 OF 1
SECONDARY CONTAINMENT CONTROL
UNIT 3
BROWNS FERRY
NUCLEAR PLANT
REV. 0

27. RO 201003K3.03 001/MEM/T2G2/85-3/B11/201003K3.03/3.6/3.7/RO/SRO/BANK

Given the following plant conditions:

- AOI 85-3, "CRD System Failure," directs a manual scram based on low reactor pressure.

Which ONE of the following PROCEDURAL reactor pressure limits should be adhered to in this case and what is the basis for the limit?

A. 445 psig reactor pressure.
This is the lowest pressure required to move the control rods based on the weight of the control rod blade.

B. 800 psig reactor pressure.
This ensures the Technical Specification scram insertion time limit is not violated in the event of an automatic scram.

C.✓ 900 psig reactor pressure.
This would be the lowest pressure a scram can be ensured due to the loss of accumulators.

D. 940 psig reactor pressure.
This would be the lowest pressure a scram can be ensured in the event of a loss of CRD pumps.

K/A Statement:

201003 Control Rod and Drive Mechanism
K3.03 - Knowledge of the effect that a loss or malfunction of the CONTROL ROD AND DRIVE MECHANISM will have on following: Shutdown margin

K/A Justification: This question satisfies the K/A statement by requiring the candidate to use specific knowledge of CRD mechanism limitations and the basis for that limitation related to the ability to effect and maintain shutdown margin.

References: 1/2/3-AOI-85-3, sec 4.1 and OPL171.006 rev 9 pg 36, 38 & TP-9

Level of Knowledge Justification: This question is rated as MEM due to the requirement to recall or recognize discrete bits of information.

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REFERENCE PROVIDED: None

Plausibility Analysis:

In order to answer this question correctly the candidate must determine the following:

1. The minimum pressure allowed by 1/2/3-AOI 85-3, CRD System Failure.
2. The basis for that minimum pressure.

C - correct: <900 psig reactor pressure is the AOI limit for inserting a manual reactor scram with a failure of the CRD system. Below 900 psig (actual pressure is 800 psig) a reactor scram on pressure alone may not be within the design basis requirements.

A - incorrect: This is plausible because the entire statement is accurate, but is NOT the pressure specified by 1/2/3-AOI 85-3, CRD System Failure and is NOT the correct reason for inserting a scram.

B - incorrect: This is plausible because the entire statement is accurate, but is NOT the pressure specified by 1/2/3-AOI 85-3, CRD System Failure.

D - incorrect: This is plausible because 940 psig is the other pressure (charging water pressure) refered to in AOI-85-3.

Note - Changed the Stem to say what is the basis for the limit vs. 'WHY?'. Changed distractor 'A' to say 'to move a control rod blade based on the weight of the blade.' Changed distractor 'B' to say 'ensures the Technical Specification scram insertion time limit is not violated in the event of an automatic scram.' Changed distractor 'D' to '940 psig' and 'in the event of a loss of CRD pumps' instead of 'this would be the lowest pressure a scram can be ensured due to the loss of accumulators.'

4.1 Immediate Actions (continued)

[3] **IF** Reactor Pressure is **LESS THAN** 900 PSIG

AND

ANY ONE of the following conditions exist:

- In service CRD Pump tripped and NEITHER CRD Pump can be started,

  OR

- Charging Water Pressure can **NOT** be restored and maintained above 940 PSIG,

THEN

PERFORM the following: (Otherwise **N/A**)

[3.1] **MANUALLY SCRAM** Reactor, **PLACE** the reactor mode switch in the shutdown position immediately. ☐

[3.2] **REFER TO** 3-AOI-100-1. [Item D20] ☐

F_{net} = (Rx Pressure x Under-Piston Area) - (Rx Pressure x Area of Index Tube + Weight of Blade + Friction)

Note: 4 in^2 upward force - 1.2 in^2 downward force = 2.8 in^2

F_{net} = (1000 psig x 4.0 in^2) - [1000 psig x (4.0 in^2 - 1.2 in^2)] - 255 lbs - ~ 500 lbs

F_{net} = 4000 - 2800 - 255 - 500

F_{net} = 445 lbs (Upward)

c. Single failure proof - There is no single-mode failure to the hydraulic system which would prevent the drive from scramming.

d. Accumulator versus reactor vessel pressure scrams

(1) TP-9 represents a plot of 90 percent scram times versus reactor pressure. TP-9

(a) Reactor pressure only

(b) Accumulator pressure only

(c) Combined reactor and accumulator pressure

(2) Scram times are measured for only the first 90% of the rod insertion since the buffer holes at the top end of the stroke slow the drive.

(3) Reactor-pressure-only scram

(a) As can be seen from TP-9, the drive cannot be scrammed with reactor pressure ≤ 400 psig.

(b) The net initial upward force available to scram the drive can be calculated as follows.

e. Average scram times (normal drive) — TP-9

(1) Technical Specifications state that scram times are to be obtained without reliance on the CRD pumps.

(2) Consequently, the charging water must be valved out on the drive to be tested.

(3) Maximum scram time for a typical drive occurs at 800 psig reactor pressure.

(4) This is why Technical Specifications specify that scram times are to be taken at 800 psig or greater reactor pressure.

f. Abnormal scram conditions

(1) Scram outlet valve failure to open

(2) Drive will slowly scram on seal leakage as long as accumulator charging water pressure stays greater than reactor pressure.

(3) If the accumulator is not available, the drive will not scram (this is a double failure).

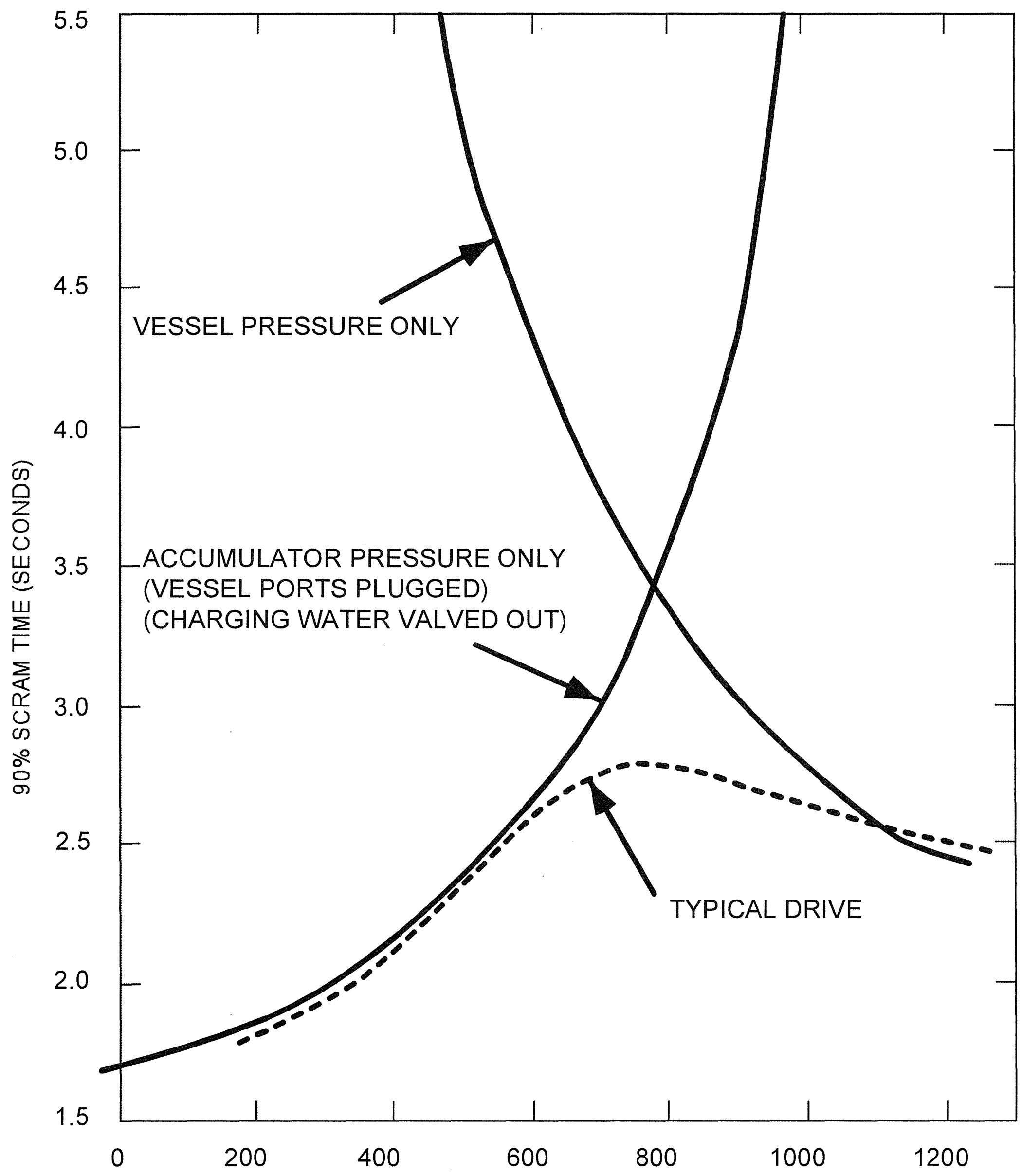
g. Control Rods failure to Insert After Scram — Obj. V.D.11

(1) This condition could be due to hydraulic lock.

(2) Procedure has operator close the Withdraw Riser Isolation valve. Connect drain hose to Withdraw Riser Vent Test Connection on the affected HCU. Slowly open Withdraw Riser Vent. When inward motion has stopped, close Withdraw Riser Vent.

See 2-OI-85 & 2-EOI App-1E for detailed operations

Self Check
Peer Check



TP-9: Scram Time Versus Reactor Pressure

28. RO 201006K4.09 001/MEM/T2G2/RWM//201006K4.09/3.2/3.2/RO/SRO/NEW 11/16/07 RMS

Which ONE of the following describes how the Rod Worth Minimizer (RWM) System INITIALIZATION is accomplished?

A. INITIALIZATION occurs automatically when the RWM is unbypassed.

B.✓ INITIALIZATION must be performed manually when the RWM is unbypassed.

C. INITIALIZATION occurs automatically every 5 seconds while in the transition zone.

D. INITIALIZATION must be performed manually when power drops below the Low Power Setpoint (LPSP).

K/A Statement:

201006 RWM
K4.09 - Knowledge of ROD WORTH MINIMIZER SYSTEM (RWM) (PLANT SPECIFIC) design feature(s) and/or interlocks which provide for the following: System initialization: P-Spec(Not-BWR6)

K/A Justification: This question satisfies the K/A statement by requiring the candidate to use specific of which plant condition would INITIALIZE the RWM.

References: 1/2/3-OI-85, OPL171.024 rev 13, pg 19 - f.(2)

Level of Knowledge Justification: This question is rated as MEM due to the requirement to recall or recognize discrete bits of information.

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REFERENCE PROVIDED: None

Plausibility Analysis:

In order to answer this question correctly the candidate must determine the following:

1. When RWM INITIALIZATION is required.
2. How RWM INITIALIZATION is accomplished.

B - correct: Initialization must be performed whenever the RWM has been taken off line, as occurs whenever the RWM program is aborted or manually bypassed.

A - incorrect: This is plausible because initialization is required when the RWM is unbypassed; but, this must be done manually.

C - incorrect: This is plausible because the RWM automatically initiates a "scan/latch" to determine the correct latched rod group; but, this is NOT the same as INITIALIZATION.

D - incorrect: The RWM does NOT require manual INITIALIZATION when the Low Power Setpoint (LPSP) is reached IF it has NOT been manually bypassed or had a program abort.

Notes - Changed distractors by removing the phrase 'using the INITIALIZATION push button'. Which button/buttons are depressed is irrelevant because it will be performed with the procedure in hand.

INSTRUCTOR NOTES

(2) The MANUAL indicator light will then be lit and all error and alarm indications that were on prior to bypass will be blanked out on the RWM system displays. Obj. V.B.6

(3) A manual bypass will also light the RWM and PROGR indicator on the RWM-COMP-PROGR-BUFF pushbutton.

f. SYSTEM INITIALIZE pushbutton switch/indicator

(1) The SYSTEM INITIALIZE switch is depressed to initialize the RWM system.

(2) Initialization must be performed whenever the RWM has been taken off line, as occurs whenever the RWM program is aborted or manually bypassed.

(3) Therefore, following any program abort or bypass, the SYSTEM INITIALIZE switch must be depressed before the program can be run again.

(4) The SYSTEM INITIALIZE window lights white while the switch is held down.

g. SYSTEM DIAGNOSTIC switch/indicator

(1) This switch can be pressed at any time after the system has been initialized to request that the system diagnostic routine be performed.

(2) The RWM program will thereupon be initiated and will perform the routine, which consists of applying and then removing in sequence the insert and withdraw blocks (nominal 10 second frequency).

(3) The operator can verify the operability of the rod block circuits by observing that the INSERT BLOCK and WITHDRAW BLOCK alarm lights come on and then go off as the blocks are

NOTE: Rod insert and withdrawal permit lights will go off when block is applied.

3.3 **Rod Worth Minimizer (RWM) (continued)**

N. For group limits only, RWM recognizes the Nominal Limits only. The Nominal Limit is the insert or withdraw limit for the group assigned by RWM. The Alternate Limit is no longer recognized by the RWM as an Acceptable Group Limit.

O. During RWM latching, the latched group will be the highest numbered group with 2 or less insert errors and having at least 1 rod withdrawn past its insert limits.

1. With Sequence Control ON, latching occurs as follows: (Normally, startups will be performed with Sequence Control ON)

    a. RWM will latch down when all rods in the presently latched group have been inserted to the group insert limit and a rod in the next lower group is selected.

    b. RWM will latch up when a rod within the next higher group is selected, provided that no more than two insert errors result.

2. With Sequence Control OFF, latching occurs as follows:

    a. For non-repeating groups, latching occurs as described above, OR

    b. For repeating groups, latching occurs to the next setup or set down based on rod movement as opposed to rod selection.

P. Latching occurs at the following times:

1. System initialization.
2. Following a "System Diagnostic" request.
3. When operator demands entry or termination of "Rod Test."
4. When power drops below LPAP.
5. When power drops below LPSP.
6. Every five seconds in the transition zone.
7. Following any full control rod scan when power is below LPAP.
8. Upon demand by the Operator (Scan/Latch Request function).
9. Following correction of insert or withdraw errors.

8.18 Reinitialization of the Rod Worth Minimizer

[1] **VERIFY** the following initial conditions are satisfied:

- The Rod Worth Minimizer is available to be placed in operation ☐
- Integrated Computer System (ICS) is available ☐
- The Shift Manager/Reactor Engineer has directed reinitialization of the Rod Worth Minimizer ☐

[2] **REVIEW** all Precautions and Limitations in Section 3.3. ☐

[3] **VERIFY** RWM SWITCH PANEL, 1-XS-85-9025 in NORMAL. ☐

[4] **CHECK** the Manual/Auto Bypass lights are extinguished. ☐

[5] **DEPRESS AND HOLD** INOP/RESET pushbutton. ☐

[6] **CHECK** all four lights (RWM/COMP/PROG/BUFF) are illuminated. ☐

[7] **RELEASE** INOP/RESET pushbutton and **CHECK** all four lights extinguished. ☐

[8] **SIMULTANEOUSLY DEPRESS** OUT OF SEQUENCE/SYSTEM INITIALIZE pushbutton and INOP/RESET pushbutton to place the Rod Worth Minimizer in service. ☐

[9] **IF** Rod Worth Minimizer will **NOT** initialize, **THEN**

DETERMINE alarms on RWM Display Screen and **CORRECT** problems. ☐

[10] **IF** unable to correct problems and initialize RWM, **THEN**

NOTIFY Reactor Engineer. ☐

29. RO 202001K6.09 001/C/A/T2G2/68 - RECIRC/24/202001K6.09//RO/SRO/BANK

Given the following plant conditions:

- Unit 3 is operating at 35% power with 'A' & 'C' Reactor Feed Pumps (RFP) running and 'B' RFP idling.
- Both Recirculation Pump speeds are 53%.
- The 'A' RFP trips, resulting in the following conditions:
  - REACTOR WATER LEVEL ABNORMAL (2-ARP-9-5A Window 8).
  - REACTOR CHANNEL 'A' AUTO SCRAM (2-ARP-9-5B Window 1).
  - REACTOR CHANNEL 'B' AUTO SCRAM (2-ARP-9-5B Window 2).
- Indicated Reactor Water Level drops to (-) 43" before 'B' RFP is brought on line to reverse the level trend and level is stabilized at 33".

Which ONE of the following describes the steady state speed of both Recirculation Pumps?

A.✓ Running at 28% speed.

B. Running at 45% speed.

C. Running at 53% speed.

D. Tripped on ATWS/RPT signal.

K/A Statement:

202001 Recirculation
K6.09 - Knowledge of the effect that a loss or malfunction of the following will have on the RECIRCULATION SYSTEM : Reactor water level

K/A Justification: This question satisfies the K/A statement by requiring the candidate to use specific plant conditions and times to determine the effect of a change in reactor water level on the Recirculation System.

References: 3-OI-68, OPL171.007, OPL171.012

Level of Knowledge Justification: This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

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REFERENCE PROVIDED: None

Plausibility Analysis:

In order to answer this question correctly the candidate must determine the following:

1. Did plant conditions exceed the Recirc Runback setpoint.
2. Which Runback is appropriate for the given conditions.

A - correct: Total Feed Flow would drop below 19% when the 'A' RFP trips thus impacting reactor water level control and initiating a Recirc Runback to 28%.

B - incorrect: This is plausible because a Recirc Runback DID occur, but the 45% speed given in the distractor is the typical speed the Recirc Pumps run at during startup, not following a RFP trip.

C - incorrect: This is plausible based on the initial power level being close enough to create doubt on Total Feed Flow resulting from the trip of one RFP.

D - incorrect: This is plausible because ATWS/RPT signals are associated with low RPV level, however the setpoint is -45 inches and level only lowered to -43 inches.

Amplification: This transient has been run on the simulator and can be duplicated if necessary.

Note - Changed the power level in the Stem from 55% to 35% so that desired response is achievable. Change the water level from -10" to -43" in the 4th bullet for plausibilty.

b. The buoyancy of the steam bubbles has a similar effect. The effects can be seen in the natural circulation line, and shows the power versus flow with no recirculation pumps running.

1) As core power rises along either line, the rise in core flow rate eventually stops, and will actually lower if power continues to raise. This is a result of the additional flow resistance created by the additional two-phase flow.

2) The rises due to density differences and buoyancy are not as pronounced at higher power levels because higher steam production tends to cause a pressure drop across the core, which retards core flow rise.

c. The 100% rod line (or the 100% load line) is based on equilibrium xenon conditions and is defined by the core configuration that will result in rated core thermal power when the core flow is raised to rated flow. Other flow control lines are defined similarly; for example, the 80% rod line will yield 80% of rated core thermal power at rated core flow.

d. The minimum expected flow control line defines the relationship between power and flow when control rods are pulled until 19% feed flow is exceeded. At that point the 28% limiter is removed, and flow could be raised to maximum.

e. The pump constant speed line (Pts G to D or H to I) is the rated recirculation pump speed line. The shape of the line will be followed if core power is changed with no change in recirculation pump speed (e.g., control rod movement or end-of-cycle coast-down). Upon a power reduction, a core flow will raise due to a reduction in the fuel channel void fraction and a subsequent reduction in two-phase flow resistance. The reverse is true for a power rise.

f. The recirculation pump and jet pump NPSH limit lines are designed to prevent cavitation of the jet pumps and the recirculation pumps. Notice that at rated core flow, cavitation of the recirculation pumps or the jet pumps will not occur if feedwater flow is above 19% of rated.

As the pressure drops in the throat of the jet pump, the temp at which boiling starts will lower. If the boiling point drops to or below the actual temp of the water, then cavitation will occur.

g. The minimum power line is a flow interlock line to prevent cavitation of the jet pumps. The recirculation pumps cannot be operated above 28% pump speed until measured feedwater flow is greater than 19% of rated.

1) Similarly, the recirculation pump will run back to minimum speed (28%) if the measured feedwater flow becomes less than 19% while the recirculation pump speed is greater than minimum.

2) This assures adequate subcooling (and therefore adequate NPSH) for all normal modes of pump operation.

h. The Increased Core Flow area is bounded by points D, G, H, and I and allows a maximum flow of 107.62 x 10^6 lbm/hr at 100% power. As power is lowered from 100% to 40.4%, flow rises from 105% to 112%. This area can be used to lengthen the cycle when all rods are fully withdrawn.

Normally operate at 107-109% rod line at 100% power (initial target rod pattern)

i. The Maximum Extended Load Line region is defined by the area above the 100% Rod Line and below the Maximum Extended Load Line Limit Analysis (MELLLA) Line. This region is part of the Normal Operating Region. The MELLLA Line has been determined by analysis to be the most limiting power/flow condition at which the unit will operate at steady state conditions. The purpose of this line and region is to ensure that operation at the 100% Rod Line is easily achieved at full power and to take advantage of the neutron spectral shift which occurs at higher power/flow lines and which raises fuel economy. The MELLLA region is bounded at 113.6% rod line and 81% core flow.

Spectral Shift techniques raise fast neutron flux during early core life. The fast neutrons react with U_{238} to form Pu_{239}. Pu_{239} will fission with thermal neutrons. This extends fuel life and cycle length by "making" fuel during the cycle.

3.0 PRECAUTIONS AND LIMITATIONS (continued)

10. The out of service pump may NOT be started unless the temperature of the coolant between the operating and idle Recirc loops are within 50°F of each other. This 50°F delta T limit is based on stress analysis for reactor nozzles, stress analysis for reactor recirculation components and piping, and fuel thermal limits. [GE SIL 517 Supplement 1]

11. The out of service pump may NOT be started unless the reactor is verified outside of regions 1, 2 and 3 of the Unit 3 Power to Flow Map (ICS or Station Reactor Engineering, 0-TI-248).

12. The temperature of the coolant between the dome and the idle Recirc loop should be maintained within 75°F of each other. If this limit cannot be maintained a plant cooldown should be initiated. Failure to maintain this limit and NOT cooldown could result in hangers and/or shock suppressers exceeding their maximum travel range. [GE SIL 251, 430 and 517]

M. Recirc Pump controller limits are as follows:

1. When any individual RFP flow is less than 19% and reactor water level is below 27 inches, speed limit is set to 75%(~1130 RPM speed) and if speed is greater than 75%(~1130 RPM speed), Recirc speed will run back to 75%(~1130 RPM speed).

2. When total feed water flow is less than 19% (15 sec TD) or Recirc Pump discharge valve is less than 90% open, speed limit is set to 28% (~480 RPM speed) and if speed is greater than 28%(~480 RPM speed), Recirc speed will run back to 28%(~480 RPM speed).

3.0 PRECAUTIONS AND LIMITATIONS (continued)

R. The power supplies to the MMR and DFR relays are listed below.

VFD 3A	
I&C BUS A (BKR 215)	3-RLY-068-MMR3/A & DFR3/A
ICS PNL 532 (BKR 30)	3-RLY-068-MMR2/A & DFR2/A
UNIT PFD (BKR 615)	3-RLY-068-MMR1/A & DFR1/A
VFD 3B	
I&C BUS B (BKR 315)	3-RLY-068-MMR3/B & DFR3/B
ICS PNL 532 (BKR 26)	3-RLY-068-MMR2/B & DFR2/B
UNIT PFD (BKR 616)	3-RLY-068-MMR1/B & DFR1/B

S. A complete list of Recirc System trip functions is provided in Illustration 4. The RPT breakers between the recirc drives and pump motors will open on any of the following:

1. Reactor dome Pressure ≥ 1148 psig (ATWS/RPT). (Both pressure switches in Logic A or both pressure switches in Logic B will cause RPT breakers to trip both pumps.) (2 out of 2 taken once logic)

2. Reactor Water Level ≤ -45" (ATWS/RPT). (Both level switches in Logic A or both level switches in Level B will cause RPT breakers to trip both pumps.) (2 out of 2 taken once logic)

3. Turbine trip or load reject condition, when ≥ 30% power by turbine first stage pressure (EOC/RPT).

T. The ATWS/RPT A(B) logic to trip the RPT breakers is defeated if the ATWS/RPT/ARI A(B) manual logic is armed using the arming collar on Panel 3-9-5. B(A) logic would still be functional and trip the RPT breakers if the setpoints are reached. If both manual push-buttons on 3-9-5 are armed, ATWS/RPT automatic logic is totally defeated (no RPT breaker trip will occur if the ATWS/RPT trip setpoints are reached). EOC/RPT logic and ATWS/ARI logic will function without regard to the position of the arming collars. ATWS/RPT/ARI logic can be reset 30 seconds after setpoints are reset.

30. RO 215001A1.01 001/MEM/T2G2/TIP//215001A1.01//RO/SRO/NEW 10/16/07

Given the following Unit 2 conditions:

- Preparations are in progress to commence a startup following a refueling outage.
- The Drywell Close-out inspection is in progress with three personnel inside the Drywell.
- Reactor Engineers have begun running Traversing In-Core Probe (TIP) traces.
- RX BLDG AREA RADIATION HIGH (2-XA-9-3A, Window 22) is in alarm.
- Area Radiation Monitor 2-RA-90-1D indicates 16 mr/hr.

Which ONE of the following describes the required actions?

A.✓ This is an expected indication while running TIP traces, therefore TIP operation may continue.

B. Verify the TIP mechanism automatically shifts to REVERSE mode and begins a full retraction.

C. Manually retract the detector in FAST speed to the In-Shield position and close the ball valve.

D. Evacuate the Unit-2 Drywell and Reactor Building. Allow the TIP trace to run automatically to completion.

K/A Statement:

215001 Traversing In-core Probe
A1.01 - Ability to predict and/or monitor changes in parameters associated with operating the TRAVERSING IN-CORE PROBE controls including: Radiation levels: (Not-BWR1)

K/A Justification: This question satisfies the K/A statement by requiring the candidate to determine the operating limitations of the TIP system with respect to high radiation.

References: 2-OI-94 Precautions & Limitations

Level of Knowledge Justification: This question is rated as MEM due to the requirement to recall or recognize discrete bits of information.

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REFERENCE PROVIDED: None

Plausibility Analysis:

In order to answer this question correctly the candidate must determine the following:

Running TIP traces will cause radiation levels inside the Reactor Building and the Drywell to increase due to the high activity associated with the TIP detectors as they are withdrawn from their shield and inserted into the appropriate TIP Tube which penetrates the Drywell and Reactor Vessel. As a result, limitations are provided to ensure personnel are protected from unnecessary exposure during performance of TIP traces. Although the given radiation alarms and conditions are typical during operation of the TIP System, with personnel inside the Drywell there are special considerations which must be addressed because of these radiation conditions which are a DIRECT result of TIP System operation. In order to correctly answer this question the candidate must recognize that high radiation conditions will exist inside the Drywell, but will be limited to areas inside the Reactor Vessel pedestal which is not typically entered during a close-out inspection. In addition, TIP operation with personnel inside the Drywell is permissable with appropriate approval and notification of personnel in the Drywell.

C - correct:

A - incorrect: This is plausible because that limitation is placed on TIP operation, but ONLY when TIP operation is NO longer required. The TIP detector can be stored in the Indexer in-between traces using the same TIP Machine for ALARA concerns.

B - incorrect: This is plausible because specific permission and controls are required to allow this condition, but it is allowable.

D - incorrect: This is plausible because the TIP response to a PCIS isolation is correct, but it is not a Group 6 isolation.

3.0 PRECAUTIONS AND LIMITATIONS

A. [NER/C] Verification of a digit in CORE LIMIT and DETECTOR POSITION windows prior to or during TIP insertion ensures TIPs retain the ability to determine its proper position. This will prevent malfunctions which could damage the TIP detector. [GE SIL-166]

B. To prevent accidental exposure to personnel, immediately evacuate the area if the TIP drive area radiation monitor alarms.

C. [NER/C] Always observe READY light illuminated prior to inserting detector. [GE SIL-166]

D. [NER/C] **DO NOT** move CHANNEL SELECT switch with detector inserted past Indexer position (0001). The common channel interlock can be defeated in this manner resulting in detector and equipment damage. [GE SIL-092]

E. [NER/C] Should detector fail to shift to slow speed when it enters the core, the LOW switch should be turned on, switched to manual mode, and the detector withdrawn. [GE SIL-166]

F. [NER/C] Length of time detector is left in core should be minimized to limit activation of detector and cable. [GE SIL-166]

G. [NER/C] When TIP System operation is not desired, detectors should be retracted and stored in chamber shield with ball valves closed. [GE SIL-166] Storage of detector in Indexer (0001) is allowed only for ALARA concerns and to prevent unnecessary masking of multiple inputs to annunciator RX BLDG AREA RADIATION HIGH 2-RA-90-1D (2-XA-55-3A, Window 22).

H. [NER/C] Upon receipt of a PCIS signal (low reactor water level or high drywell pressure), any detector inserted beyond its shield chamber should be verified to automatically shift to reverse mode and begin withdrawal. Once in shield, ball and purge valves close. [GE SIL-166] Ball valve cannot be reopened until PCIS is reset on Panel 2-9-4 and manual reset of TIP ISOLATION RESET pushbutton 2-HS-94-7D/S2 located on Panel 2-9-13.

I. A detector should not be abruptly stopped from fast speed to off without first switching to slow speed.

J. [NER/C] Drive Control Units (DCU) should be monitored during withdrawal to prevent any chamber shield withdrawal limit from being overrun. Detectors should be stopped manually at shield limit if auto stop limit switch should fail and verify ball valve closes. [GE SIL-166]

K. Only one TIP at a time should be operated when maintenance is being performed in TIP drive area.

3.0 PRECAUTIONS AND LIMITATIONS (continued)

L. [NRC/C] **DO NOT** operate TIPswith personnel inside TIP Room or in vicinity of TIP tubing and Indexers in Drywell. Requirement may be waived with approval of Shift Manager and site RADCON manager or designee. In this instance, RADCON is required to establish such controls as are necessary to prevent access to TIP tubing and Indexer areas to preclude unnecessary exposure to personnel working in Drywell. RADCON Field Operations Shift Supervisor is required to be notified prior to operation of TIP System. [NRC Information Notice 88-063, Supplement 2]

M. No channel should be indexed to common channel 10 unless all other channels are not indexed to channel 10 and all their READY lights are illuminated.

N. [NER/C] **DO NOT** turn MODE switch to OFF on Drive Control Unit if detector is outside shield chamber unless personnel safety requires it. [GE SIL-166] This removes power preventing automatic withdrawal on PCIS signal and causing ball valves to close on cable or detector. Tip Ball Valves CANNOT fully close and shear valves may have to be actuated.

O. CHANNEL SELECT switches on Drive Control Units should always be rotated in clockwise direction when selecting channels.

P. Connector on shear valve indicator circuit should not be removed while testing shear valve explosive charges or performing shear valve maintenance with detector inserted. This will cause an automatic detector withdrawal.

Q. Continuous voice communication should be maintained between TIP operator or maintenance personnel in control room and drive mechanism area while maintenance is being performed and TIP detector driving is necessary.

R. Each applicable ball valve should be opened prior to operating that TIP machine.

S. TIP Drive Mechanisms and Indexers should have continuous purge supply unless required to be removed from service for maintenance.

T. During outages when containment is deinerted for personnel access, TIP Indexer purge supply should be transferred from nitrogen to Control Air for personnel safety.

U. Detector damage is possible if TIP ball valve is left open, or is opened during DRYWELL PRESSURE TEST. (GE SIL-166)

31. RO 216000K1.10 001/MEM/T2G2/PR.INSTR/9/216000K1.10//RO/SRO/BANK

Which ONE of the following indicates how raising recirculation flow affects the Emergency System (Wide Range) 3-58A/58B and Normal Control Range Level Indicators (e.g., LI-3-53) on Panel 9-5?

Emergency System Range indication will _________ and the Normal Control Range indication will __________.

A. indicate lower; indicate lower.

B. NOT be affected; indicate higher.

C. indicate higher; NOT be affected.

D.✓ indicate lower; NOT be affected.

K/A Statement:

216000 Nuclear Boiler Inst
K1.10 - Knowledge of the physical connections and/or cause- effect relationships between NUCLEAR BOILER INSTRUMENTATION and the following: Recirculation flow control system

K/A Justification: This question satisfies the K/A statement by requiring the candidate to use specific knowledge of the effect of changes in Recirculation flow on reactor water level instrumentation.

References: OPL171.003

Level of Knowledge Justification: This question is rated as MEM due to the requirement to recall or recognize discrete bits of information.

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REFERENCE PROVIDED: None

Plausibility Analysis:

Normal Control Range level instruments are calibrated under hot conditions with Recirc pumps running. Based on that and the location of the variable leg taps of these instruments, changes in Recirc flow do not affect their indication. Emergency Systems Range level instruments are claibrated under cold conditions with no Recirc pumps running. As Recirc flow changes; the pressure sensed by the variable leg also changes, thereby causing a change in indicated water level. The magnitude and direction of the change in indication depends on perspective. Specifically, the initial and final conditions used to quantify the change.

D - correct:

A - incorrect: This is plausible because Emergency System Range instruments will read lower, but the Narrow Range instruments will not.

B - incorrect: This is plausible because Narrow Range instruments may read slightly higher at colder conditions, but this does NOT apply to Recirc flow changes.

C - incorrect: This is plausible because Narrow Range instruments are not effected by Recirc Flow changes, but Emergency System Range instruments will read lower.

INSTRUCTOR NOTES

d. Four ranges of level indication

(1) Normal Control Range (Narrow Range)

Obj. V.B.5

(a) 0 to +60 inch range covering the normal operating range (analog) with +60" up to +70" digital and 0" down to – 10" digital readings.

Obj. V.B.6
TP-3 shows only analog scale

(b) Referenced to instrument zero

(c) Four of these instruments are used by Feedwater Level Control System (FWLCS). The level signal utilized by the FWLCS is not directed through the Analog Trip System.

i. Temperature compensated by a pressure signal

Obj. V.B.11.
Obj. V.B.13.

ii. Most accurate level indication available to the operator

iii. Calibrated for normal operating pressure and temperature

(d) These indicators and a recorder point (average of the four) are located on Panel 9-5.

NOTE: An air bubble or leak in the reference leg can cause inaccurate readings in a non-conservative direction resulting in a mismatch between level indicators.

LER 85-006-02
(See LP Folder)
(Section X.C.1.j. provides more detail)

This problem is particularly prevalent after extended outages when starting up from cold shutdown conditions and at low reactor pressures.

INSTRUCTOR NOTES

(e) Four other narrow range instruments are located in the control room, two above the FWLCS level indicators on panel 9-5 (3-208A & D), one above HPCI (3-208B)and one above RCIC (3-208C)on panel 9-3.

Associated with RFPT/Main Turbine and HPCI/RCIC trip instruments

(2) Emergency Systems Range (Wide Range) 2 Analog meters and 2 Digital meters.

(a) -155 to +60 inches range covering normal operating range and down to the lower instrument nozzle return

(b) Referenced to instrument zero

(c) Four MCR indicators on Panel 9-5 monitor this range of level indication.

(d) Calibrated for normal operating pressure and temperature

(e) The level signal utilized by the Wide Range instruments have safety related functions and are directed through the Analog Trip System.

(f) Level indication for this range is also provided on the Backup Control Panel (25-32).

Obj. V.B.12.

(3) Shutdown Vessel Flood Range (Flood-up Range)

(a) 0 to +400 inches range covering upper portion of reactor vessel

(b) Referenced to instrument zero

Calibrated for cold conditions (<212°F, 0 psig)

(c) Provides level indication during vessel flooding or cool down.

INSTRUCTOR NOTES

Transient flashing effects can cause indicated level to oscillate or be erratic. As the reference leg refills, the indicated level approaches a more accurate water level indication. The RVLIS mod decreases the time necessary for this refill to occur

j. Normal Control Range (Narrow Range) and Emergency Systems Range (Wide Range) Level Discrepancies

(1) Narrow Range level instrumentation is calibrated to be most accurate at rated temperature and pressure (particularly the instruments for FWLCS, since they are temperature compensated). At cold conditions the non-FWLCS instruments read high (not temperature compensated).

(2) Wide Range instruments are also calibrated for rated temperature and pressure

(a) The indicated level on the Wide Range (9-5) is also affected by changes in the subcooling of recirculation water and the amount of flow at the lower (variable leg) tap.

Obj. V.B.15

(b) At rated conditions with minimum recirculation flow the Wide Range instruments are accurate. As recirculation flow is increased past the lower tap it has a significant velocity head and some friction loss which reduces the pressure on the variable leg to the differential pressure instrument, resulting in an indicated level lower than actual. This could be as much as 10-15 inches error when at rated flow and power.

(c) Due to calibration for rated conditions and no density compensation at cold conditions these instruments read high.

32. RO 219000K2.02 001/C/A/T2G2/OI-74//219000K2.02//RO/SRO/BANK

Given the following plant conditions:

- Unit 2 is at 100% rated power with Residual Heat Removal (RHR) Loop II in Suppression Pool Cooling mode to support a High Pressure Coolant Injection (HPCI) Full Flow Test surveillance.
- Unit 1 experiences a LOCA which results in a Common Accident Signal (CAS) initiation on Unit 1.

Which ONE of the following describes the current status of Unit 2 RHR system and what actions must be taken to restore Suppression Pool Cooling on Unit 2?

A. ALL four RHR pumps receive a trip signal. Place RHR Loop II in Suppression Pool Cooling IMMEDIATELY.

B. '2A' and '2C' RHR Pumps are tripped. '2B' and '2D' pumps are unaffected. NO additional action is required.

C✓ ALL four RHR pumps receive a trip signal. Place RHR Loop II in Suppression Pool Cooling after a 60-second time delay.

D. '2B' and '2D' RHR Pumps are tripped. '2A' and '2C' pumps are unaffected. Place RHR Loop I in Suppression Pool Cooling IMMEDIATELY.

K/A Statement:

219000 RHR/LPCI: Torus/Pool Cooling Mode
K2.02 - Knowledge of electrical power supplies to the following: Pumps

K/A Justification: This question satisfies the K/A statement by requiring the candidate to use specific plant conditions and times to determine which RHR pumps can be used for Suppression Pool Cooling.

References: 2-OI-74, OPL171.044

Level of Knowledge Justification: This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

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REFERENCE PROVIDED: None

Plausibility Analysis:

In order to answer this question correctly the candidate must determine the following:

1. Response of Unit 2 RHR pumps due to a Unit 1 CAS initiation.
2. Recognize the difference between a Single Unit CAS and Simultaneous Unit CAS.
3. Recognize that Preferred and Non-preferred Emergency Core Cooling System (ECCS) Pumps do NOT apply with the given conditions.

Unit 1 Preferred RHR pumps are **1A** and **1C.** Unit 2 Preferred RHR pumps are **2B** and **2D.** LOCA signals are divided into two separate signals, one referred to as a Pre Accident Signal (PAS) and the other referred to as a Common Accident Signal (CAS). If a unit receives a CAS, then all its respective RHR and Core Spray pumps will sequence on based upon power source to the SD Boards. All RHR and Core Spray pumps on the non-affected unit will trip (if running) and will be blocked from **manual** starting for 60 seconds. After 60 seconds all RHR pumps on the **non-affected** unit may be **manually** started. The **non-preferred** pumps on the **non-affected** unit are also prevented from **automatically** starting until the affected unit's accident signal is clear. The **preferred pumps** on the **non-affected** unit are locked out from automatically starting until the affected unit accident signal is clear **OR** the **non-affected** unit receives an accident signal.

C - correct:

A - incorrect: This is plausible because all four RHR pumps on Unit 2 will trip, but they are locked out from manual start for 60 seconds based on Diesel Generator and/or Shutdown Board loading concerns.

B - incorrect: This is plausible based on RHR Loop II being the preferred pumps for Unit 2.

D - incorrect: This is plausible if taken from the perspective of Unit 1 operation, NOT Unit 2 operation.

Appendix A
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Unit 1 & 2 Core Spray/RHR Logic Discussion

2.2 ECCS Preferred Pump Logic

Concurrent Accident Signals On Unit 1 and Unit 2

With normal power available, the starting and running of RHR pumps on a 4KV Shutdown Board already loaded by the opposite unit's Core Spray, RHR pumps, and RHRSW pumps could overload the affected 4KV Shutdown Boards and trip the normal feeder breaker. This would result in a temporary loss of power to the affected 4KV Shutdown Boards while the boards are being transferred to their diesels. To prevent this undesirable transient, Unit 2 RHR Pumps 2A and 2C are load shed on a Unit 1 accident signal and Unit 1 Pumps 1B and 1D will be load shed on a Unit 2 accident signal. Unit 2 Core Spray Pumps 2A and 2C are load shed on a Unit 1 accident signal and Unit 1 Core Spray Pumps 1B and 1D will be load shed on a Unit 2 accident signal. This makes the Preferred ECCS pumps Unit 1 Division I Core Spray and RHR Pumps and Unit 2 Division 2 Core Spray and RHR Pumps. Conversely, the Non-preferred ECCS pumps are Unit 1 Division 2 Core Spray and RHR Pumps and Unit 2 Division 1 Core Spray and RHR Pumps.

The preferred and non-preferred ECCS pumps are as follows:

UNIT 1 & 2

PREFERRED ECCS Pumps

CS 1A, CS 1C, RHR 1A, RHR 1C
CS 2B, CS 2D, RHR 2B, RHR 2D

NON-PREFERRED ECCS Pumps

CS 1B, CS 1D, RHR 1B, RHR 1D
CS 2A, CS 2C, RHR 2A, RHR 2C

UNIT 3

Unit 3 does not have ECCS Preferred/Non-Preferred Pump Logic.

Accident Signal On One Unit

With an accident on one unit, ECCS Preferred pump logic trips all running RHR and Core Spray pumps on the non-accident unit.

INSTRUCTOR NOTES

Note:

Presently Unit 1 Accident signal will not affect Unit 2 due to DCN H2735A that lifted wires from relays. Unit 2 will still affect Unit 1. However, the following represents modifications to the inter-tie logic as it will be upon Unit 1 recovery.

Obj. V.B.13.
Obj. V.C.3
Obj. V.C.7
Obj. V.D.6
Obj. V.E.II

(1) Unit 1 Preferred RHR pumps are **1A** and **1C**

(2) Unit 2 Preferred RHR pumps are **2B** and **2D**

(3) Unit 2 initiation logic is as follows:Div 1 RHR logic initiates Div 1 pumps (A and C), and Div 2 logic initiates Div 2 pumps (B and D)

f. Accident Signal

(1) LOCA signals are divided into two separate signals, one referred to as a Pre Accident Signal (PAS) and the other referred to as a Common Accident Signal (CAS).

Obj. V.B.13.
Obj. V.C.3
Obj. V.C.7
Obj. V.D.6
Obj. V.E.II

- PAS

  -122" Rx water level (Level 1)

  OR

  2.45 psig DW pressure

- CAS

  -122" Rx water level (Level 1)

  OR

  2.45 psig DW pressure **AND** <450 psig Rx pressure

Note:
It should be clear that the only difference between the two signals is the inclusion of Rx pressure in the CAS signal. The PAS signal is an anticipatory signal that allows the DG's to start on rising DW pressure and be ready should a CAS be received.

(2) If a unit receives an accident signal, then all its respective RHR and Core Spray pumps will sequence on based upon power source to the SD Boards.

(3) All RHR and Core Spray pumps on the non-affected unit will trip (if running) and will be blocked from **manual** starting for 60 seconds.

INSTRUCTOR NOTES

(4) After 60 seconds all RHR pumps on the **non-affected** unit may be **manually** started.

(5) The **non-preferred** pumps on the **non-affected** unit are also prevented from **automatically** starting until the affected unit's accident signal is clear.

Operator diligence required to prevent overloading SD boards/DG's

(6) The **preferred pumps** on the **non-affected** unit are locked out from automatically starting until the affected unit accident signal is clear **OR** the **non-affected** unit receives an accident signal.

g. 4KV Shutdown Board Load Shed

Obj. V.C.8.

(1) A stripping of motor loads on the 4KV boards occurs when the board experiences an undervoltage condition. This is referred to as a 4KV Load Shed. This shed prepares the board for the DG ensuring the DG will tie on to the bus unloaded and without faults.

(2) The Load Shed occurs when an undervoltage is experienced on the board i.e. or if the Diesel were tied to the board (only source) and one of the units experienced an accident signal which trips the Diesel output breaker.

(3) Then, when the Diesel output breaker interlocks are satisfied, the DG output breaker would close and, if an initiation signal is present (CAS) the RHR, CS, and RHRSW pumps would sequence on

(4) Following an initiation of a Common Accident Signal (which trips the diesel breaker), if a subsequent accident signal is received from another unit, a second diesel breaker trip on a "unit priority" basis is provided to ensure that the Shutdown boards are stripped prior to starting the RHR pumps and other ECCS loads

(5) When an accident signal trip of the diesel breakers is initiated from one unit (CASA or CASB), subsequent CAS trips of all eight diesel breakers are blocked.

Occurs due to actuation of the diesel breaker TSCRN relay

33. RO 226001A4.12 001/MEM/T2G2/PC/P//226001A4.12/3.8/3.9/RO/SRO/BANK

A LOCA inside containment results in the following plant conditions:

- Drywell pressure is 20 psig
- Drywell temperature is 210°F
- Suppression chamber pressure is 18 psig
- Suppression chamber temperature is 155°F
- Suppression pool level is (+) 2 inches
- Reactor water level is (+) 30 inches

Which ONE list of parameters below must ALWAYS be addressed to determine when it is appropriate to spray the drywell?

A. - Suppression Chamber temperature
- Drywell pressure
- Drywell temperature

B. - Suppression Chamber pressure
- Drywell temperature
- Suppression Pool level

C.✓ - Drywell pressure
- Drywell temperature
- Reactor water level

D. - Reactor water level
- Suppression Chamber temperature
- Drywell pressure

K/A Statement:

226001 RHR/LPCI: CTMT Spray Mode
A4.12 - Ability to manually operate and/or monitor in the control room: Containment/drywell pressure

K/A Justification: This question satisfies the K/A statement by requiring the candidate to use specific knowledge of which containment parameters are used to determine when Containment Sprays can be used.

References: 1/2/3-EOI-2 Flowchart

Level of Knowledge Justification: This question is rated as MEM due to the requirement to recall or recognize discrete bits of information.

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REFERENCE PROVIDED: None

Plausibility Analysis:

In order to answer this question correctly the candidate must determine the following:

1. Drywell temperature and pressure are always required to ensure Curve 5 limits are not exceeded.
2. RPV level is always required to verify adequate core cooling is assured prior to diverting RHR flow for Drywell sprays.
3. Suppression Pool level is always required to verify Suppression Chamber to Drywell vacuum breakers are uncovered.
4. Suppression Chamber pressure is ONLY required when initiating Drywell Sprays from flowpath PC/P.
5. Suppression Chamber temperature is NOT required to initiate Drywell Sprays.

C - correct:

A - incorrect: This is plausible because DW temp and press are required, but Suppression Chamber (Torus) temp is NOT.

B - incorrect: This is plausible because DW temp and SP level are required, but SC press is ONLY required when initiating DW Sprays using PC/P.

D - incorrect: This is plausible because RPV level and DW press are required, but Suppression Chamber (Torus) temp is NOT.

WHEN SUPPR CHMBR PRESS EXCEEDS 12 PSIG,

THEN **CONTINUE** IN THIS PROCEDURE

PC/P-5

IS SUPPR PL LVL BELOW 18 FT

NO

YES

PC/P-6

ARE DW TEMP AND DW PRESS WITHIN THE SAFE AREA OF CURVE 5

NO

YES

PC/P-7

SHUT DOWN RECIRC PUMPS AND DW BLOWERS

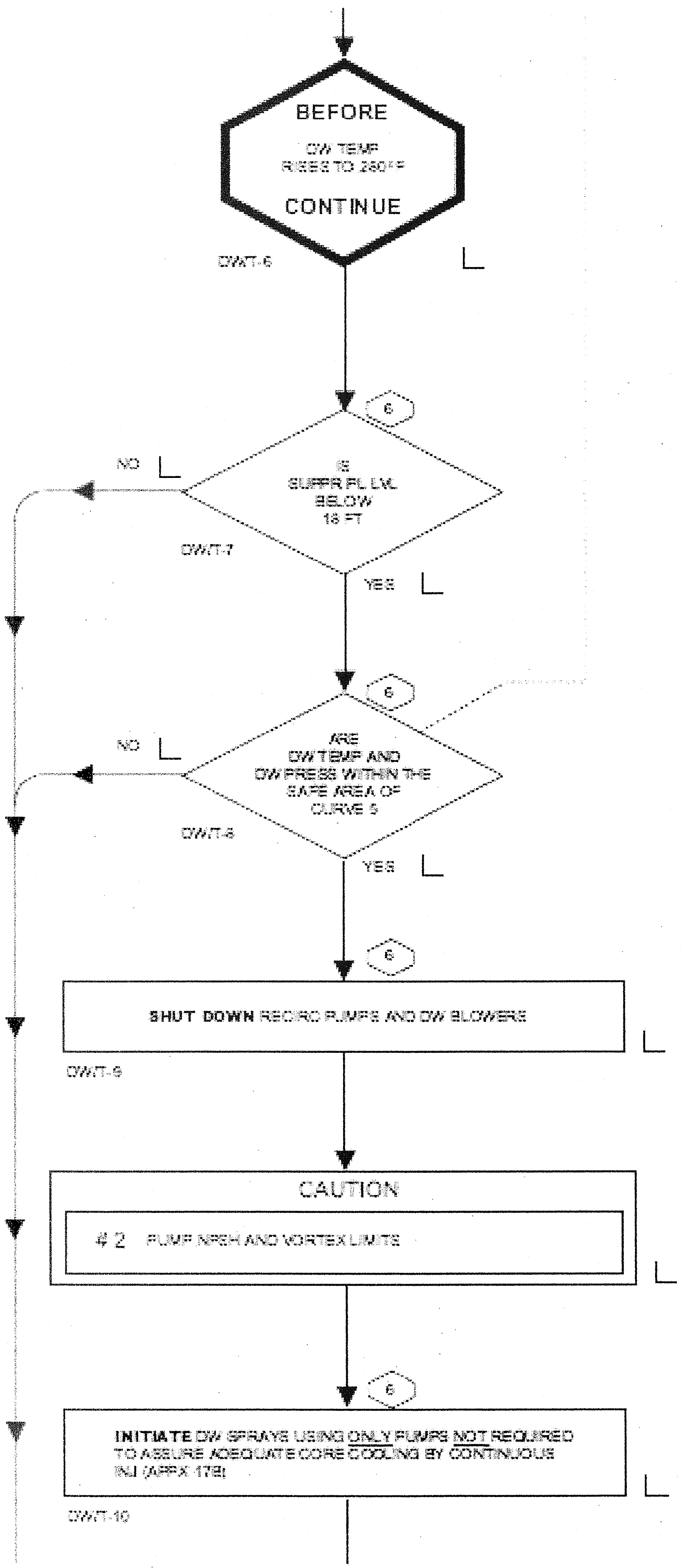
PC/P-8

CAUTION

\# 2 PUMP NPSH AND VORTEX LIMITS

INITIATE DW SPRAYS USING <u>ONLY</u> PUMPS <u>NOT</u> REQUIRED TO ASSURE ADEQUATE CORE COOLING BY CONTINUOUS INJ (APPX 17B)

PC/P-9


BEFORE
DW TEMP
RISES TO 280°F
CONTINUE
DW/T-6
6
IS
SUPPR PL LVL
BELOW
18 FT
NO
YES
DW/T-7
6
ARE
DW TEMP AND
DW PRESS WITHIN THE
SAFE AREA OF
CURVE 5
NO
YES
DW/T-8
6
SHUT DOWN RECIRC PUMPS AND DW BLOWERS
DW/T-9
CAUTION
2 PUMP NPSH AND VORTEX LIMITS
6
INITIATE DW SPRAYS USING ONLY PUMPS NOT REQUIRED TO ASSURE ADEQUATE CORE COOLING BY CONTINUOUS INJ (APPX 17B)
DW/T-10

34. RO 234000G2.4.50 001/C/A/T2G2///234000G2.4.50//RO/SRO/NEW 10/16/07

Given the following plant conditions:

- Fuel movement is in progress for channel changeout activities in the Fuel Prep Machine.
- Gas bubbles are visible coming from the de-channeled bundle.
- An Area Radiation Monitor adjacent to the Spent Fuel Storage Pool begins alarming.

Which ONE of the following describes the action (s) to take?

Immediately STOP fuel handling, then______________________________________.

A. evacuate ALL personnel from the Refuel Floor.

B. notify RADCON to monitor & evaluate radiation levels.

C.✓ evacuate non-essential personnel from the Refuel Floor.

D. obtain Reactor Engineering Supervisor's recommendation for movement and sipping of the damaged fuel assembly.

K/A Statement:

234000 Fuel Handling Equipment
2.4.50 - Emergency Procedures / Plan Ability to verify system alarm setpoints and operate controls identified in the alarm response manual

K/A Justification: This question satisfies the K/A statement by requiring the candidate to use specific plant conditions to determine the corrective actions involving Fuel Handling equipment under emergency conditions.

References: 1/2/3-AOI-79-1 & 79-2, 1/2/3-ARP-9-3A (W1)

Level of Knowledge Justification: This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

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REFERENCE PROVIDED: None

Plausibility Analysis:

In order to answer this question correctly the candidate must determine the following:

1. Whether indications are consistent with fuel damage or inadvertant criticality.
2. Based on the answer to Item #1 above, enter the appropriate AOI.
3. Immediate Operator Actions for the selected procedure, AOI-79-1, Fuel Damage During Refueling.

C - correct: Non-essential personnel evacuation is an IMMEDIATE action in AOI-79-1, Fuel Damage During Refueling.

A - incorrect: This is plausible because evacuation of ALL personnel is an IMMEDIATE action in AOI-79-**2,** Inadvertent Criticality During In-Core Fuel Movements. However, non-essential personnel evacuation is an IMMEDIATE action in the AOI-79-1l.

B - incorrect: This is plausible because RADCON notification is a subsequent action in AOI-79-1. However, non-essential personnel evacuation is an IMMEDIATE action.

D - incorrect: This is plausible because Reactor Engineer recommendations are a subsequent action in AOI-79-1. However, non-essential personnel evacuation is an IMMEDIATE action.

FUEL POOL FLOOR AREA RADIATION HIGH 2-RA-90-1A 1	Sensor/Trip Point: RI-90-1B RI-90-2B RI-90-3B	 For setpoints **REFER TO** 2-SIMI-90B.

(Page 1 of 1)

Sensor Location:	RE-90-1B	El 664'	R-11 P-LINE
	RE-90-2B	El 664'	R-10 U-LINE
	RE-90-3B	El 639'	R-10 Q-LINE

Probable Cause:

A. Change in general radiation levels.
B. Refueling accident.
C. Sensor malfunction.

Automatic Action: None

Operator Action:

A. **CHECK** 2-RI-90-1A, 2-RI-90-2A and 2-RI-90-3A on Panel 2-9-11. ☐
B. **NOTIFY** refuel floor personnel. ☐
C. **IF** Dry Cask loading/unloading activities are in progress, **THEN NOTIFY** Cask Supervisor. ☐
D. **IF** airborne levels rise by 100 DAC **AND** RADCON confirms, **THEN REFER TO** EPIP-1. ☐
E. **REFER TO** 2-AOI-79-1 or 2-AOI-79-2 as applicable. ☐
F. **IF** this alarm is not valid, **THEN REFER TO** 0-OI-55. ☐
G. **IF** this alarm is valid, **THEN**
MONITOR the other parameters that input to it frequently. These other parameters will be masked from alarming while this alarm is sealed in. ☐
H. **ENTER** 2-EOI-3 Flowchart. ☐

References: 0-47E600-13 2-47E610-90-1 2-45E620-3
GE 730E356 Series, TVA Calc NDQ00902005001/EDC63693

1.0 PURPOSE

This instruction provides the symptoms, automatic actions and operator actions for a fuel damage accident.

2.0 SYMPTOMS

A. Possible annunciators in alarm:

1. FUEL POOL FLOOR AREA RADIATION HIGH (2-XA-55-3A, window 1).
2. AIR PARTICULATE MONITOR RADIATION HIGH (2-XA-55-3A, window 2).
3. RX BLDG, TURB BLDG, RF ZONE EXH RADIATION HIGH (2-XA-55-3A, window 4).
4. REACTOR ZONE EXHAUST RADIATION HIGH (2-XA-55-3A, window 21).
5. RX BLDG AREA RADIATION HIGH (2-XA-55-3A, window 22).
6. REFUELING ZONE EXHAUST RADIATION HIGH (2-XA-55-3A, window 34).

B. Gas bubbles visible, in the Spent Fuel Storage Pool and/or Reactor Cavity, attributed to physical fuel damage.

C. Known dropped or physically damaged fuel bundle.

D. Portable CAM in alarm.

E. Radiation level on the Refuel Floor is greater than 25 mr/hr and cause is unknown.

4.0 OPERATOR ACTIONS

4.1 Immediate Actions

[1] **STOP** all fuel handling. ☐

[2] **EVACUATE** all non-essential personnel from Refuel Floor. ☐

4.2 Subsequent Actions

CAUTION

The release of iodine is of major concern. If gas bubbles are identified at any time, Iodine release should be assumed until RADCON determines otherwise.

[1] **VERIFY** secondary containment is intact.
(**REFER TO** Tech Spec 3.6.4.1) ☐

[2] **IF** any EOI entry condition is met, **THEN**

ENTER the appropriate EOI(s). ☐

[3] **VERIFY** automatic actions. ☐

[4] **NOTIFY** RADCON to perform the following:

- **EVALUATE** the radiation levels. ☐
- **MAKE** recommendation for personnel access. ☐
- **MONITOR** around the Reactor Building Equipment Hatch, at levels below the Refuel Floor, for possible spread of the release. ☐

[5] **REFER TO** EPIP-1 for proper notification. ☐

4.2 Subsequent Actions (continued)

[6] **MONITOR** radiation levels, for the affected areas, using the following radiation recorders and indicators:

A. 2-RR-90-1 (points 1 and 2), 2-MON-90-50 (Address 11), 2-RR-90-142 and 2-RR-90-140 (Panel 2-9-2). ☐

B. 2-RM-90-142, 2-RM-90-140, 2-RM-90-143 and 2-RM-90-141 Detectors A and B (Panel 2-9-10). ☐

C. 2-RI-90-1A and 2-RI-90-2A (Panel 2-9-11). ☐

D. 0-CONS-90-362A (Address 09, 10, 08) for Unit 1, 2, 3-RM-90-250, respectively (Panel 1-9-44). ☐

[7] **IF** possible, **MONITOR** portable CAMs & ARMs.

[8] **REQUEST** Chemistry to perform 0-SI-4.8.B.2-1 to determine if iodine concentration has risen. ☐

[9] **NOTIFY** Reactor Engineering Supervisor, or his designee, and **OBTAIN** recommendation for movement and sipping of the damaged fuel assembly. ☐

[10] **OBTAIN** Plant Managers approval prior to resuming any fuel transfer operations. ☐

[11] **WHEN** condition has cleared AND if required, **THEN**

RETURN ventilation systems, including SGTS, to normal. **REFER TO** 2-OI-30A, 2-OI-30B, 0-OI-30F, 0-OI-31, and 0-OI-65. ☐

4.0 OPERATOR ACTIONS

4.1 Immediate Actions

[1] **IF** unexpected criticality is observed following control rod withdrawal, **THEN**

REINSERT the control rod. ☐

[2] **IF** all control rods CANNOT be fully inserted, **THEN**

MANUALLY SCRAM the reactor. ☐

[3] **IF** unexpected criticality is observed following the insertion of a fuel assembly, **THEN**

PERFORM the following: ☐

[3.1] **VERIFY** fuel grapple latched onto the fuel assembly handle AND immediately **REMOVE** the fuel assembly from the reactor core. ☐

[3.2] **IF** the reactor can be determined to be subcritical **AND** no radiological hazard is apparent, **THEN**

PLACE the fuel assembly in a spent fuel storage pool location with the least possible number of surrounding fuel assemblies, leaving the fuel grapple latched to the fuel assembly handle. ☐

[3.3] **IF** the reactor CANNOT be determined to be subcritical **OR** adverse radiological conditions exist, **THEN**

TRAVERSE the refueling bridge and fuel assembly away from the reactor core, preferably to the area of the cattle chute, AND **CONTINUE** at Step 4.1[4]. ☐

[4] **IF** the reactor CANNOT be determined to be subcritical **OR** adverse radiological conditions exist, **THEN**

EVACUATE the refuel floor. ☐

35. RO 245000K6.04 001/C/A/T2G2/OI-35//245000K6.04//RO/SRO/NEW 11/28/07 RMS

Given the following plant conditions:

- Unit 2 is operating at 100% power.
- Main Generator is at 1150 MWe.
- The Chattanooga Load Coordinator requires a 0.95 lagging power factor.
- Generator hydrogen pressure is 65 psig.

Which ONE of the following describes the required action and reason if Generator hydrogen pressure drops to 45 psig?

Reduce ____.

REFERENCE PROVIDED

A. generator load below 800 MWe. Pole slippage will NOT occur at this generator load.

B✓ generator load below 800 MWe. Sufficient cooling capability still exists at this hydrogen pressure.

C. excitation to obtain a power factor of unity to maintain current generator load. Pole slippage will NOT occur at this power factor.

D. excitation to obtain a power factor of unity to maintain current generator load. Sufficient cooling capability still exists at this hydrogen pressure.

K/A Statement:

245000 Main Turbine Gen. / Aux.
K6.04 - Knowledge of the effect that a loss or malfunction of the following will have on the MAIN TURBINE GENERATOR AND AUXILIARY SYSTEMS : Hydrogen cooling

K/A Justification: This question satisfies the K/A statement by requiring the candidate to use specific plant conditions to determine the effect of a loss of hydrogen cooling on Main Generator operation.

Reference Provided: Generator Capability Curve without axis labeled

Level of Knowledge Justification: This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

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REFERENCE PROVIDED: Generator Capability Curve without the axis labeled.

Plausibility Analysis:

In order to answer this question correctly the candidate must determine the following:

1. Current operating point on the Generator Capability Curve based on given conditions.
2. Recognize that pole slippage is only a concern when operating with a significant leading power factor.
3. Recognize that pole slippage is a result of under excitation, not excessive generator load.
4. Recognize that generator hydrogen pressure is directly related to cooling capability.

B - correct:

A - incorrect: This is plausible because generator load is properly reduced; but, the basis for the reduction is NOT related to slipping poles.

C - incorrect: This is plausible because reducing excitation DOES reduce heat generation within the generator; but, NOT sufficiently to prevent generator damage. However, pole slippage is NOTa concern at a unity power factor.

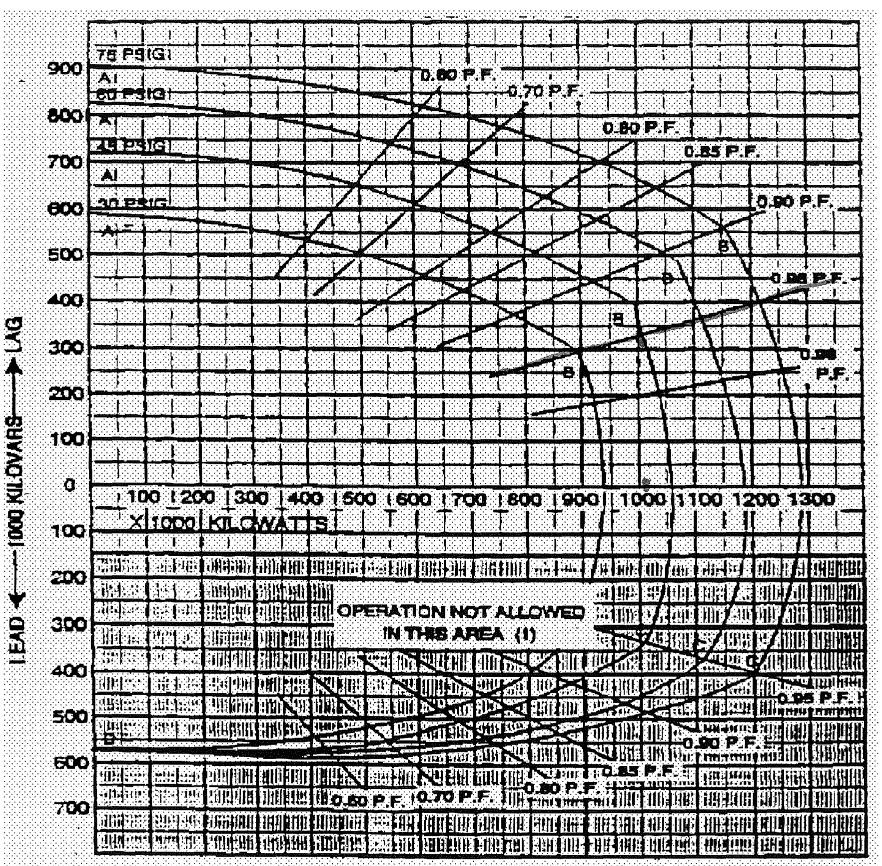
D - incorrect: This is plausible because reducing excitation DOES reduce heat generation within the generator; but NOT sufficiently to prevent generator damage. In addition, insufficient hydrogen pressure exists at the current generator load even wih a power factor of unity.

Illustration 6
(Page 1 of 1)

Generator Kilovar Limitations (Capability Curve)

NOTES

1) Operation in this area allowed only during calibration or testing.
2) **DO NOT** exceed the limits of the capability curve.
3) Maximum LAGGING MVARs are as follows. These values are based on turbine and generator load limitations. More limiting values based on transmission load limits and operational concerns can be found in the Switchyard Manual, 0-GOI-300-4.
   - 400 MVAR per unit for 1-Unit Operation
   - 300 MVAR per unit for 2-Unit Operation
   - 270 MVAR per unit for 3-Unit Operation
4) Curve AB is limited by FIELD heating.
5) Curve BC is limited by ARMATURE heating.
6) Curve CD is limited by ARMATURE CORE END heating.

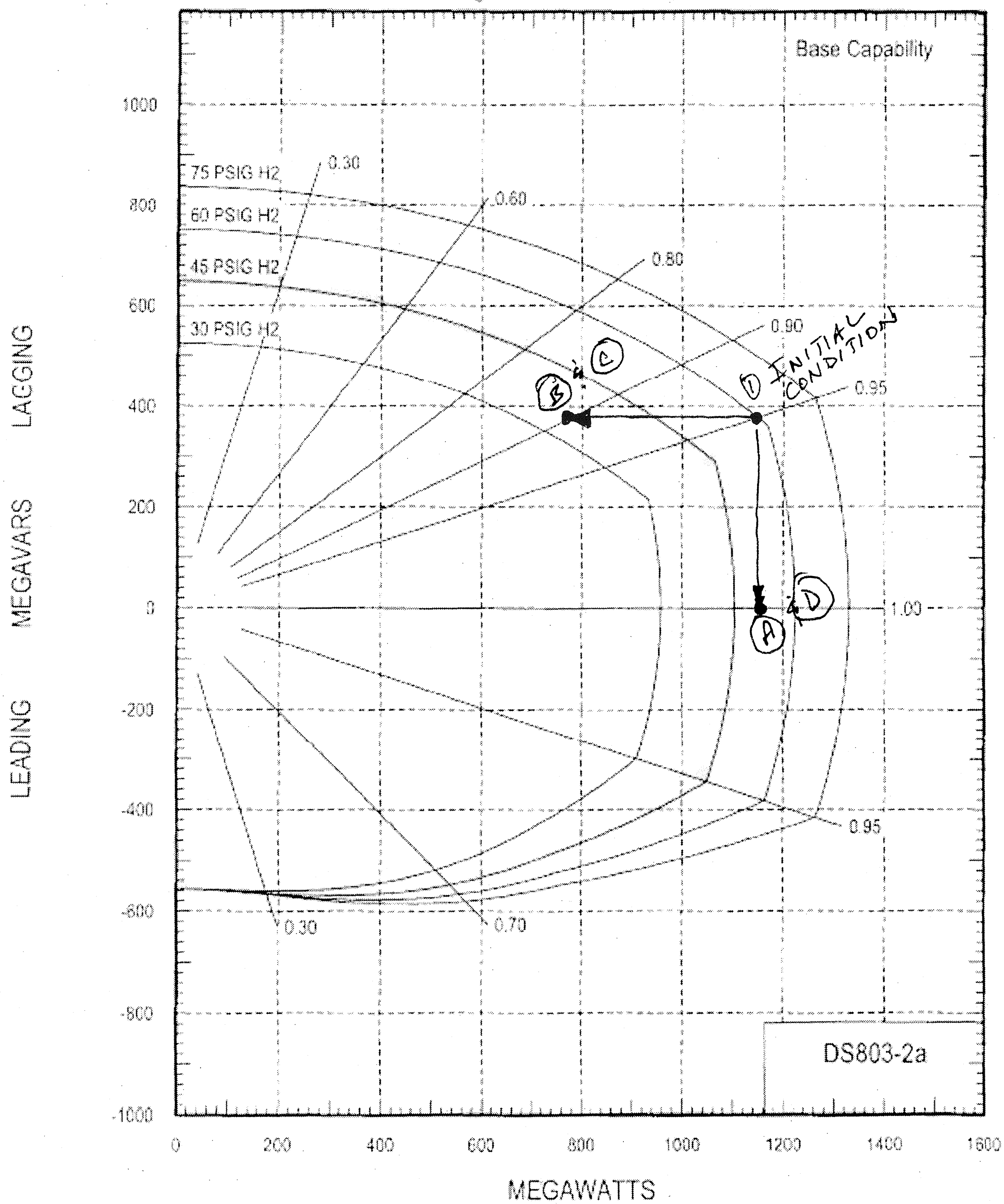


ESTIMATED REACTIVE CAPABILITY CURVES

4 Pole 1800 RPM 1330000 kVA 22000 Volts 0.950 PF

0.580 SCR 75.00 PSIG H2 Pressure 510 Volts Excitation

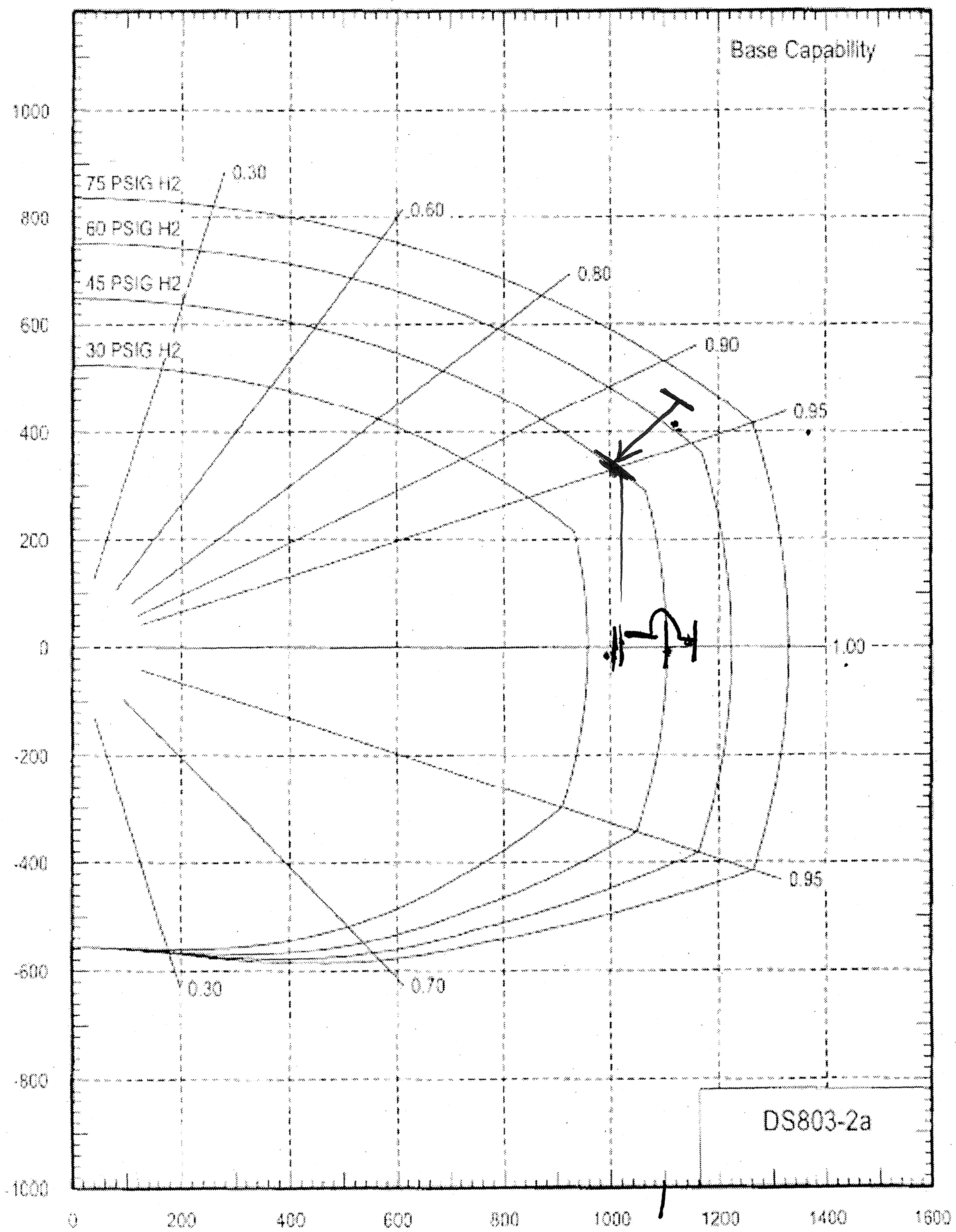
46 Deg. C Cold Gas 617 Ft. Altitude



EXAMINATION REFERENCE PROVIDED TO CANDIDATE

ESTIMATED REACTIVE CAPABILITY CURVES

4 Pole 1800 RPM 1330000 kVA 22000 Volts 0.950 PF
0.580 SCR 75.00 PSIG H2 Pressure 510 Volts Excitation
46 Deg. C Cold Gas 617 Ft. Altitude



36. RO 268000A2.01 002/MEM/T2G2///268000A2.01//RO/SRO/NEW 2/19/08

Given the following plant conditions:

- BFN is in the process of discharging the Waste Sample Tank to the Unit 1 Condensate Storage Tank.
- While moving resin containers, a forklift operator accidently punctures the discharge line 4 feet downstream of 0-RR-90-130 (Radwaste Effluent Radiation Monitor).
- Water immediately begins spraying out of the rupture.

Which ONE of the following describes the expected system response?

The discharge will ________(1)_______ because the________(2)_________.

A. (1) automatically terminate without any additional operator action;

(2) Radwaste Discharge Isolation Valve will auto close on HIGH discharge flow.

B.✓ (1) continue until manually terminated per 0-OI-77a, "Waste Collector/Surge System Processing;"

(2) Radwaste Discharge Isolation Valve does NOT auto close on high discharge flow.

C. (1) automatically terminate without any additional operator action;

(2) Radwaste Discharge Isolation Valve will auto close on LOW discharge flow.

D. (1) continue until manually terminated per 0-OI-77a, "Waste Collector/Surge System Processing;"

(2) Radwaste Discharge Isolation Valve will ONLY auto close on high discharge radiation.

K/A Statement:

268000 Radwaste
A2.01 - Ability to (a) predict the impacts of the following on the RADWASTE ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: System rupture

K/A Justification: This question satisfies the K/A statement by requiring the candidate to use specific plant conditions to determine the response of the RADWASTE system due to a rupture of a plant system and the procedures used to mitigate that condition.

References:

Level of Knowledge Justification: This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

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REFERENCE PROVIDED: None

Plausibility Analysis:

In order to answer this question correctly the candidate must determine the following:

Recognize that the given location of the rupture places it downstream of the Radwaste Discharge Isolation Valve. Using that knowledge, recognize the conditions which cause an automatic isolation of that valve. The Radwaste Discharge Isolation Valve (FCV-77-61) auto closes on radiation monitor upscale, downscale or INOP. The valve will ALSO close on low CCW flow or the specific unit's cooling tower gate "A" position. The same is true of the High or Low flowrate isolation valves downstream of the 77-61 valve. These valves are named "High" or "Low" based on the anticipated flowrate of the radwaste discharge, which is based on the chemistry sample taken prior to the discharge. Neither valve automatically closes on high or low flow. No provision is made to automatically isolate the radwaste discharge line based on a system rupture. This must be accomplished manually.

B - correct:

A - incorrect: The Radwaste Discharge Isolation Valve will NOT automatically close under the given conditions.

C - incorrect: The Radwaste Discharge Isolation Valve will NOT automatically close under the given conditions.

D - incorrect: Although the discharge must be manually terminated, the Radwaste Discharge Isolation Valve will automatically close on other conditions besides high radiation.

Instructor Notes

(2) Discharge Control Station

(a) Purpose - used to control rate of release, direct to discharge canal or cooling tower blowdown, and provides system isolation.

(b) Interlocks — TP-6

i. FCV 77-61 - Radwaste discharge isolation auto closure on Radiation monitor $\geq$ upscale isolation setpoint, downscale, or Inop, Unit specific 1A gate is not full open, or two CCWP's are not operating

Caution Order is placed on Assoc. Unit 77-61 valve when in helper mode.

Obj. V.D.7

ii. FCV 77-58B radwaste high flow rate discharge isolation valve auto closure on Radiation monitor $\geq$ upscale isolation setpoint, downscale, or Inop. All 3 unit 1A gates are closed and cooling tower blowdown flow is $\leq$ 60,000 gpm.

Instructor Notes

iii. FCV 77-58A radwaste low flow rate discharge isolation valve auto closure on (1)Radiation monitor $\geq$ upscale isolation setpoint, downscale, or Inop (2)All 3 unit 1A gates closed, and Cooling tower blowdown flow $\leq$ 60,000 gpm

iv. FCV 77-279 radwaste isolation valve to cooling tower blowdown auto closure on (1)Radiation monitor $\geq$ upscale isolation setpoint, downscale, or Inop (2)cooling tower blowdown flow $\leq$ 60,000 gpm

(c) Flow rate is set by 0-SI-4.8.A.1-1 and is controlled by FCV-77-59A or B by use of controls on panel 25-17.

(d) Recorders

i. Flow recorder provides indication of release rate and permanent record of flow rate and total discharge release time.

37. RO 272000K5.01 001/C/A/SYS/HWC/B9/272000K5.01//RO/SRO/BANK

Given the following plant conditions:

- The Hydrogen Water Chemistry (HWC) System is in the "Operator Determined Setpoint" mode.
- Hydrogen flow is set at 14 SCFM.

Which ONE of the following describes the plant response IF reactor power is reduced?

Main Steam Line radiation levels will______(1)__________ the lowering of reactor power due to __________________(2)__________________.

(1)	(2)
A.✓ rise in opposition to;	a rise in volatile Ammonia production.
B. lower in response to;	a reduction in Nitrogen concentration.
C. lower in response to;	a reduction in Hydrogen concentration.
D. rise in opposition to;	a rise in Nitrite and Nitrate production.

K/A Statement:

272000 Radiation Monitoring
K5.01 - Knowledge of the operational implications of the following concepts as they apply to RADIATION MONITORING SYSTEM : Hydrogen injection operation's effect on process radiation indications: Plant-Specific

K/A Justification: This question satisfies the K/A statement by requiring the candidate to use plant conditions to determine the effect on radiation levels due to specific operating conditions of the Hydrogen Injection system.

References:

Level of Knowledge Justification: This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

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REFERENCE PROVIDED: None

Plausibility Analysis:

In order to answer this question correctly the candidate must determine the following:

1. The mechanism by which Hydrogen injection causes Main Steam Line (MSL) radiation levels to rise higher than normal.
2. The effect on hydrogen concentration in the reactor at reduced feedwater flow with Hydrogen Injection flowrate unchanged.
3. The effect on ammonia production due to a reduction of oxygen concentration in the reactor.

A - correct:

B - incorrect: This is plausible because nitrogen concentration DOES decrease with power level. However, due to the reduction in oxygen concentration, the reduction in nitrite and nitrate production allows more nitrogen available to combine with the excess hydrogen and form volitile ammonia.

C - incorrect: This is plausible because of the TYPICAL response of the HWC system when operated in Automatic mode. In addition, understanding that hydrogen injection flowrate is constant may not lead to an understanding of that relationship to a reduction in feedwater flow where hydrogen is injected.. However, in Operator Demand mode the hydrogen concentration increases due to the reduction in feedwater flow.

D - incorrect: This is plausible because MSL radiation levels DO rise in opposition to lowering power. However, the reduction in available oxygen causes a reduction in nitrite and nitrate production, which allows more volatile ammonia formation.

h. O_2 source

(1) Air in-leakage - oxygen in the air leaks into the low pressure parts of the steam cycle

LP turbine blading to discharge of condensate pumps is less than atmospheric pressure

(2) Some air in-leakage is removed by the SJAE's but some dissolves in the condensate

(3) Radiolysis reaction -

$$2\,H_2O \xrightarrow{\text{n and } \gamma \text{ radiation}} 2\,H_2 + O_2$$

One way O_2 is produced

i. O_2 removal

(1) Radiation induced recombination of H_2 and O_2

$$2H_2O \xleftrightarrow{\text{n and gamma radiation}} 2H_2 + O_2$$

Reaction is an equilibrium reaction

(2) Carry-over with the steam

j. Hydrogen addition

(1) When an excess of hydrogen is injected to the feed water, the reaction is driven to the left and less oxygen (and peroxide) is produced

This is the basis of H_2 injection

(2) The chemical environment becomes less oxidizing

(3) Elements exposed to the coolant will assume chemical forms using less oxygen and/or more hydrogen

(4) Solubility and volatility may be affected by the change in "oxidation state" of the element

This is how MSL radiation levels will increase which will be discussed later

G. Operation

Be very careful when selecting functions from different screens. Use self checking.

1. Normally controlled from the HWC Main Control Panel

a. When using the Operator Interface Unit (OIU) function buttons, be aware that the same function key will cause different actions on different screens

INPO SER 3-05

b. Operation of the HWC PLC

(1) Hydrogen controller

Flow controller operates in 2 modes

(a) Automatic/Power Determined Setpoint Mode - changes hydrogen injection flow in response to changes in reactor power. Used for normal operation of the HWC System and when reducing hydrogen injection related dose rates to support maintenance, chemistry or radcon activities while the plant is operating

Procedure Use

(b) Automatic/Operator Determined Setpoint Mode - changes hydrogen injection flow in response to the setpoint being manually entered by the operator. Normally used when initially pressurizing, purging and placing the HWC System in service or if Power Determined setpoint is unavailable

Hydrogen flow stays constant, regardless of power changes, until operator manually enters a new setpoint

(2) Oxygen controller - Automatic/Hydrogen Determined Setpoint

Only mode used for oxygen control

n. Supply Facility Trip - A shutdown signal is generated when either the hydrogen or oxygen gas supply facility trips

o. Hydrogen or Feedwater Flow Signal Failed - A shutdown signal is generated when the hydrogen flow signal or the feedwater flow signal is less than 2 mA or greater than 22 mA

I. Radiological Effects of HWC On MSL's

Obj. V.B.6
Obj. V.B.4
Obj. V.D.6
Obj. V.E.6

1. The primary source of background MSL radiation levels during reactor operation is due to the decay of nitrogen-16 (N^{16})

a. N^{16} has a half-life of 7.1 seconds

b. A 6.13 or 7.12 Mev gamma is emitted on N^{16} decay

6.13 Mev gamma is more common

2. Major source of nitrogen in a BWR is O^{16} (η,ρ) N^{16} reaction

$O^{16} + \eta \rightarrow N^{16} + \rho$

3. When using normal water chemistry methods, a major portion of the N^{16} present in the reactor coolant combines with the free oxygen to form water soluble nitrites (NO_2) and nitrates (NO_3)

a. These compounds are circulated through the reactor coolant systems and are ultimately removed by the RWCU System

b. A smaller fraction of the N^{16} is carried over in the steam in the form of nitrogen gas (N_2) and ammonia (NH_3)

Predominate contributor to background radiation levels

4. H_2 injection alters the N^{16} carryover ratio

a. Concentrations of NO_3, NO_2, and NO decrease

b. Concentration of NH_3 increases

Ammonia

(1) A gas

(2) High water solubility

5. The net production of N^{16} is not influenced by hydrogen injection

6. The increased dose rates are due to the increased ease with which N^{16} gets out of the reactor and into the steam pipes when in the NH_3 form

We can maintain up to 2.7 ppm injection concentration.

7. The initial U2 run was the first week in Nov. 1999. Up to 90 scfm hydrogen was injected. Average MSL radiation level increased approximately 5 times normal

MSL 'B' was highest at 5.2 times normal

8. Addition of noble metals to reactor water

a. Noble metals decompose during reactor startup or shutdown

Rubidium and Iridium

b. During this time it produces a thin layer of noble metal on wetted surfaces

c. The ECP on these surfaces are reduced significantly during subsequent operation

d. This leaves a stoichiometric excess of hydrogen

e. Now the amount of hydrogen injection can be reduced which will lower MSL radiation levels

This NOTE

c. Consequences of Event

No effects were noted. However there is the potential for rapid recombination in the Offgas Charcoal beds. Additionally excessive hydrogen increases the risk for explosion or fire. .

2. High radiation on reduction in power event at Monticello

a. Event description

On December 13, 1997 with Monticello at approximately 75 percent power, workers entered the main condenser room to repair a leaking root valve and found the dose rate 2.5 times greater than expected

Reactor power had been reduced from 100 percent power to 75 percent power for ALARA purposed and hydrogen water chemistry injection rate had been reduced from the normal 40 scfm to 8 scfm

Dose rate encountered was significantly higher that expected

Encountered - 4,800 mrem/hr
Expected - 2,000 mrem/hr

Job was stopped to evaluate the situation and management decided to have power reduced further

At 60 percent power, dose rates were about 3,200 mrem/hr and the job was completed

b. Cause of event

Lack of understanding of the radiological effect of reducing reactor power under HWC conditions. As reactor power is decreased, less N-16 is produced, and steam line dose rates decrease. However, power level changes also change feedwater flow rate and hydrogen concentration

Questioning Attitude could have prevented this.

Reactor power and feedwater hydrogen concentration both affect steam line dose rates. At a constant hydrogen injection rate, as power is decreased, feedwater flow rate decreases, and hydrogen concentration increases

An increase in hydrogen concentration increases the ammonia concentration although hydrogen injection rates were reduced, the hydrogen concentration increase that occurred when feedwater flow rate decreased with reactor power was not accounted for and resulted in higher than expected dose rates

c. Consequences of Event:

Rx power needed to be lowered to 60% vice 75% planned. This resulted in unplanned lost generation.

Work Planning

Work had to be performed in a 3200mr/hr vice a planned 2000 mr/hr field. This resulted in unplanned exposure.

Potential lost opportunity to plan and perform work that required Rx power to be lowered to 60 % power vice the planned reduction to 75% power

38. RO 290003A3.01 001/MEM/T2G2/HVAC/4/290003A3.01/3.3/3.5/RO/SRO/BANK

Given the following plant conditions:

- High radiation has been detected in the air inlet to the Unit 3 Control Room.
- Radiation Monitor RE-90-259B is reading 250 cpm.

Which ONE of the following describes the Control Room Emergency Ventilation (CREV) System response?

___(1)___ CREVS Unit(s) will automatically start ______(2)______.

(1) (2)

A. NEITHER; at the current radiation level.

B. BOTH; with suction from the normal outside air path to Elevation 3C.

C. Selected; and will continue to run until Control Bay Ventilation is restarted; then, it will automatically stop.

D.✔ Selected; Standby CREVS Unit will begin to auto-start; but, will ONLY run if the selected CREVS Unit fails to develop sufficient flow.

K/A Statement:

290003 Control Room HVAC
A3.01 - Ability to monitor automatic operations of the CONTROL ROOM HVAC including: Initiation/reconfiguration

K/A Justification: This question satisfies the K/A statement by requiring the candidate to use specific plant conditions to determine the effect on CREV initiation logic.

References:

Level of Knowledge Justification: This question is rated as MEM due to the requirement to recall or recognize discrete bits of information.

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REFERENCE PROVIDED: None

Plausibility Analysis:

In order to answer this question correctly the candidate must determine the following:

The Control Room Emergency Ventilation (CREV) system automatically starts on high radiation sensed by Radiation Monitor RE-90-259B. The actual stepoint is 221 cpm. The radiation level that is given in the stem was chosen because the Tech Spec required setpoint if 270 cpm. This requires the candidate to determine the difference for operational validity. THere are two CREV units. One is selected as "Primary" and the other "Standby". On initiation, both CREV units receive a start signal, however the "Standby" unit's signal is delayed 30 seconds to allow the "Primary" unit enough time to start. If the start is successful, the flow from the "Primary" unit will remove a start permissive to the "Standby" unit and it will abort it's startup. CREV units must be manually secured once the initiating conditions have cleared.

D - correct:

A - incorrect: This is plausible because the Tech Spec initiation setpoint is 270 cpm, which is less than the given radiation level. However, the actual CREV initiation setpoint is 221 cpm.

B - incorrect: This is plausible since both CREV units receive a start signal on a valid initiation. However, the CREV unit NOT selected will experience a 30 second time delay on initiation and will only complete its start sequence if the selected CREV unit fails to start.

C - incorrect: This is plausible because the start sequence is correct. However, once initiated, CREV must be manually secured. There is no automatic shutdown capability, only trips.

INSTRUCTOR NOTES

7. Control Room Emergency Ventilation (CREV) is designed to supply and process the outdoor air needed for pressurization during isolated conditions. There are 2 CREV units rated at 3000 cfm each. A CREV unit consists of Motor-driven fan, (power supply is from 480V RMOV Bd 1A for CREV Fan A; RMOV Bd 3B for CREV Fan B), HEPA filter (common), charcoal filter assemblies located in the CREVS Equipment Room, charcoal heater, and inlet isolation damper and a backflow check outlet damper. They are designed to maintain a positive pressurization to 1/8" w.g. minimum to the control room.

Tech. Spec. 3.7.3
Obj.V.B.2/ V.B.5/ V.C.6 /V.C.7
(Old CREV Units abandoned in place as Auxiliary Pressurization Systems)
TP-4
2-47E2865-4

a. A CREV may be started manually from control room Panel 2-9-22 if local control switch is in AUTO position via a 3 position, spring-return to center switch. (STOP-AUTO-START). Actuates only the CREVS unit & associated damper, not the isolation dampers.

Red indicating lights on panel 3-9-21 to provide indication of CREV Fan A and/or B running on Unit 3. Annunciators are on panel 9-6 for all units.

b. There is also a 2 position maintained contact, one per train, AUTO-INITIATE/ TEST switch which is used to perform system level actions for that train (primarily testing). It provides the same response as auto start.

c. Local start at local control station in relay room is done using a 2 position maintained, one per train, AUTO-TEST switch. Isolation dampers do not operate automatically if started from local panel.

d. Automatic start signals are:

Obj. V.B.1/V.B.2

(1) High radiation of 221 cpm above background (270 cpm Tech Specs) in air inlet ducts to control room from (Radiation monitor RE 90-259A Units 1 & 2, Radiation monitor RE 90-259B Unit 3). Either monitor starts selected CREV unit.

Obj. V.C.1
Obj. V.C.17

T. S. 3.3.7.1

(2) Reactor zone ventilation systems radiation high $\geq$72 MR/hr

INSTRUCTOR NOTES

(3) Refuel zone ventilation systems radiation high $\geq$72 MR/hr

(4) Low reactor water level at +2 inches above instrument zero

(5) High primary containment pressure $\geq$2.45 psig

e. On receipt of a start signal, normal outside air paths (see below) to elevation 3C are isolated. The selected CREV unit starts once the inlet damper is full open. This supplies pressurizing air to the Unit 1, 2 and 3 control rooms. One CREV unit can supply all three control rooms, so the STBY CREV unit will not normally start. Once started, the CREV unit will continue to run until manually secured by first clearing the high radiation signals and the PCIS signals (otherwise equipment cycling will occur)

f. Control bay (EL 617) isolation is accomplished by five pneumatic and motor-operated low leakage dampers which isolate all normal air intakes and exhausts for EL 617.

(1) FCO-31-150B, fresh air make-up duct to Units 1 and 2 Control Room and Relay Room AHU.

(2) FCO 31-150G, 3C elevation relief vent isolation

(3) FCO-31-150E, exhaust from Unit 1 toilet, locker, and other rooms at elevation 617.

(4) FCO-31-150D, mounted in fresh air makeup duct to Unit 3 Control Room AHU.

(5) FCO-31-150F, exhaust from unit 3 toilet, locker, and other rooms at elevation 617.

The inlet damper is normally closed & fails closed. Damper opening takes ~70 seconds. While in the intermediate position both red & green lights will be lit on 2-9-22.
The unit heater will energize 10 sec. after the damper is full open to allow the fan to come up to speed. High Rad or PCIS signal will energize relays in Div I (CR1-A) and Div II (CR1-B). Contacts from the CR1 relays are used to energize solenoids to isolate the M.C.R. normal intake dampers (150B,D,E,F, and G)

INSTRUCTOR NOTES

g. Manual initiation of the emergency mode of operation can be performed from the control room by operation of the AUTO-INITIATE/TEST switch (putting switch in INITIATE/TEST position) at Pnl 2-9-22. This operation results in energizing the CR1 relay for that train/division and isolation of the control room dampers. Only one solenoid must be energized to close these dampers. Therefore, either test switch will initiate a damper isolation.

Adherence to procedures
INPO SER 03-05
Obj.V.B.4/V.B.5

h. One switch alone in the INITIATE/TEST position will NOT result in full functionality of the CREVS units. If that train is not the selected unit, then operation of that train will be delayed by approx. 30 seconds, waiting for the selected train. This delay will result in the operator waiting to see the result of his operation of the switch. The operator will see the amber light lit, indicating energization of the CR 1 relay and the solenoid for isolation damper closing, but will see no activity of the CREVS unit until the delay timer has timed out.

Normally “A” is selected unit via switch in CREV room. If “A” is inoperable, switch 0-XSW-031-7214 SYSTEM PRIORITY SELECTOR SWITCH is placed in TRAIN B position to start it without time delay.

i. If the selected train’s switch is put in the INITIATE/TEST position that train will immediately enter its initiation sequence, with the damper’s red light being lit as well as the green, indicating travel of the damper toward the open position. However, should there be any failure of the selected unit; the standby unit will not start. This is because the CR1 relay for the standby unit was not actuated.

j. Therefore, when manually initiating emergency operation of the new CREVS units, it is important to put the AUTO-INITIATE/TEST switches of BOTH trains to the INITIATE/TEST position.

INSTRUCTOR NOTES

k. Again, to secure operation, the AUTO-INITIATE/TEST switches must both be returned to the AUTO position and then the STOP-AUTO-START switches turned to the STOP position, to reset the CR1 relays in both divisions.

l. Trips for the units, which are effective at all times, are the following:

  (1) Fan overload

  (2) Unit low flow, less than approx. 2700 cfm -- trip is delayed for 10 seconds after fan start.

  (3) High heater discharge temperature, approx. 220°F

  (4) Low heater delta temperature(between unit inlet and heater discharge), indicating that the heater is not getting the relative humidity below 70 % -- trip is delayed for approx. 15 seconds after the heater is energized.

m. When any of these trip signals are received, the following will occur: — Obj.V.B.4/ V.B.5

  (1) The heater will be immediately deenergized.

  (2) The fan will continue to run and the damper will remain open for approx. 30 seconds, to dissipate the heat from the heater. (In the case of fan overload, the fan will trip immediately.)

  (3) The inlet damper will be deenergized, and when no longer fully open, the fan will be deenergized. The damper requires approx. 20 seconds to close, while fan coast down is approx. 60-90 seconds.

INSTRUCTOR NOTES

n. In addition to the trips shown above, loss of power to the inlet damper will trip the unit. In this case, the heater is immediately tripped and the fan is deenergized when the damper is no longer fully open. This action results in (slightly) faster tripping of the heater to avoid heat dissipation problems.

o. Flow switches are provided, one for each division/unit, to start the standby unit if the selected unit does not start or trips off. The selected unit not starting is sensed by low differential pressure across the common HEPA filter in the Unit 2 vent tower. Low differential pressure exists when a fan is not operating; this signal will normally be present. The circuit for each unit is such that its initiation sequence is begun upon either of the following:

PDIS 7316 at Unit 2 Vent Tower Intake Plenum

(1) Unit is selected as primary unit and CR1 relay for that division is energized.

(2) Other unit is selected as primary unit, low differential pressure exists across the common HEPA filter, and CR1 relay for that division has been energized for approx. 30 seconds.

p. With this circuit design, when an accident signal is initially received, the selected unit will enter its initiation sequence immediately and the other unit will enter its initiation sequence approx. 30 seconds later. Once the selected unit fan has been started (taking approx. 75 seconds -- 70 for the damper and 5 for the fan), the low differential pressure signal will no longer be present in the standby unit circuitry and its damper will return to the fail-close position.

INSTRUCTOR NOTES

q. If the selected unit fails to start properly, it will itself be turned off by the trips noted above, and the standby unit will continue in its initiation sequence. The time delay for startup of the standby unit will be selected to ensure that regardless of the primary unit failure, both fans will not be running at the same time.

r. If the selected unit starts properly, but then trips at a later time, the standby unit will only be missing the low differential pressure signal to receive its start signal. The standby unit will start when the selected system has completed its shutdown process and the fan has been deenergized.

s. To secure from emergency operation, the high rad signals and the PCIS signals must first be cleared (otherwise equipment cycling will occur). These signals must be cleared on both divisions to not have the standby unit start up when the selected unit is secure. The STOP-AUTO-START switches in the control room should then be moved to the STOP position <u>for both units</u>. This will reset / deenergize the CR1 relays in both divisions, reopen the control room isolation dampers and remove the start signal from the operating CREVS unit. The CREVS unit heater will then be deenergized, with the fan continuing to run and the damper held open for approx. 30 seconds, and the damper closing and the fan turned off as discussed earlier.

Obj. V.B.2.

39. RO 295001G2.1.14 001/MEM/T1G1/68 - RECIRC/2/295001G2.1.14//RO/SRO/NEW 10/16/07

Given the following plant conditions:

- You are the At-The-Controls (ATC) operator on Unit 1 .
- Unit 1 is operating at full power when 1A Recirculation pump tripped.
- The Unit Supervisor has directed you to carry out the actions of 1-AOI-68-1A, "Recirc Pump Trip/Core Flow Decrease OPRMs Operable."

Which ONE of the following describes the required operator action(s) that CANNOT be carried out from your watch station?

A.✓ Perform 1-SR-3.4.1(SLO), "Reactor Recirculation System Single Loop Operation."

B. REFER TO ICS screens VFDPMPA(VFDPMPB) and VFDAAL(VFDBAL) to assist in determining the cause of the Recirc Pump trip/core flow decrease.

C. IMMEDIATELY take actions to insert control rods to less than 95.2% loadline AND REFER TO 0-TI-464, "Reactivity Control Plan Development and Implementation."

D. CHECK parameters associated with the Recirc Drive and Recirc Pump/Motor 1A(1B) on ICS and RECIRC PMP MTR 1A & 1B WINDING & BRG TEMPS, 1-TR-68-71 to determine the cause of the Recirc Pump trip.

K/A Statement:

295001 Partial or Complete Loss of Forced Core Flow Circulation / 1 & 4
2.1.14 - Conduct of Operations Knowledge of system status criteria which require the notification of plant personnel

K/A Justification: This question satisfies the K/A statement by requiring the candidate to use specific plant conditions to determine the required actions which require notification of plant personnel outside of the control room due to a partial loss of Recirculation flow.

References: 1-AOI-68-1

Level of Knowledge Justification: This question is rated as MEM due to the requirement to recall or recognize discrete bits of information.

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REFERENCE PROVIDED: None

Plausibility Analysis:

In order to answer this question correctly the candidate must determine the following:

1. Which actions are required to be performed by 1-AOI-68-1.
2. Which of the actions determined from Item 1 above CANNOT be carried out by the ATC operator.

A - correct: This duty is carried out by Reactor Engineering once notified by the Unit Operator.

B - incorrect: This is plausible because the BOP operator or STA typically carry out this action. However, utilizing ICS screens is available at the ATC watch station and within his required duties.

C - incorrect: This is plausible because the BOP operator typically inserts control rods while the ATC operator executes 1-AOI-68-1 and acts as Peer Checker if possible. However, manipulating control rods is part of the ATCwatch station duties.

D - incorrect: This is plausible because the BOP operator or STA typically carry out this action. However, utilizing ICS screens is available at the ATC watch station and within his required duties.

4.2 Subsequent Actions (continued)

NOTE
1) Step 4.2[3] through Step 4.2[18.3] apply to any core flow lowering event.
2) Power To Flow Map is maintained in 0-TI-248, Station Reactor Engineer and on ICS.

[3] **IF** Region I or II of the Power to Flow Map is entered, **THEN** (Otherwise N/A)

IMMEDIATELY take actions to insert control rods to less than 95.2% loadline AND **REFER TO** 0-TI-464**,** Reactivity Control Plan Development and Implementation. ☐

[4] **RAISE** core flow to greater than 45% in accordance with 1-OI-68. ☐

[5] **INSERT** control rods to exit regions if **NOT** already exited AND **REFER TO** 0-TI-464**,** Reactivity Control Plan Development and Implementation. ☐

NOTE
The remaining subsequent action steps apply to a single Reactor Recirc Pump trip.

[6] **CLOSE** tripped Recirc Pump discharge valve. ☐

[7] **MAINTAIN** operating Recirc pump flow less than 46,600 gpm in accordance with 1-OI-68. ☐

[8] [NER/C] **WHEN** plant conditions allow, **THEN**, (Otherwise N/A)

MAINTAIN operating jet pump loop flow greater than 41×10^6 lbm/hr (1-FI-68-46 or 1-FI-68-48). [GE SIL 517] ☐

4.2 Subsequent Actions (continued)

CAUTION

The temperature of the coolant between the dome and the idle Recirc loop should be maintained within 75°F of each other. If this limit cannot be maintained, a plant cool down should be initiated. Failure to maintain this limit and **NOT** cool down could result in hangers and/or shock suppressers exceeding their maximum travel range. [GE SIL 251, 430 and 517]

[9] **IF** Recirc Pump was tripped due to dual seal failure, **THEN** (Otherwise N/A)

[9.1] **VERIFY TRIPPED**, RECIRC DRIVE 1A(1B) NORMAL FEEDER, 1-HS-57-17(14). ☐

[9.2] **VERIFY TRIPPED**, RECIRC DRIVE 1A(1B) ALTERNATE FEEDER, 1-HS-57-15(12). ☐

[9.3] **CLOSE** tripped recirc pump suction valve using, RECIRC PUMP 1A(1B) SUCTION VALVE, 1-HS-68-1(77). ☐

[9.4] **IF** it is evident that 75°F between the dome **AND** the idle Recirc loop cannot be maintained, **THEN**

COMMENCE plant shut down and cool down in accordance with 1-GOI-100-12A. ☐

[10] **NOTIFY** Reactor Engineer to perform Reactor Recirculation System Single Loop Operation, 1-SR-3.4.1(SLO) AND to refer to Station Reactor Engineer, 0-TI-248 and Tech Specs 3.4.1 as necessary. ☐

[11] [NER/C] **WHEN** the Recirc Pump discharge valve has been closed for at least five minutes (to prevent reverse rotation of the pump) [GE SIL-517], **THEN** (N/A if Recirc Pump was isolated in Step 4.2[9])

OPEN Recirc Pump discharge valve as necessary to maintain Recirc Loop in thermal equilibrium. ☐

4.2 Subsequent Actions (continued)

[12] **REFER TO** the following ICS screens to help determine the cause of recirc pump trip/core flow decrease.

- VFDPMPA(VFDPMPB) ☐
- VFDAAL(VFDBAL) ☐

[13] **CHECK** parameters associated with Recirc Drive and Recirc Pump/Motor 1A(1B) on ICS and RECIRC PMP MTR 1A & 1B WINDING & BRG TEMPS, 1-TR-68-71 to determine cause of trip. ☐

[14] **PERFORM** visual inspection of tripped Reactor Recirc Drive. ☐

[15] **PERFORM** visual inspection of Reactor Recirc Pump Drive relay boards for relay targets. ☐

[16] **IF** necessary, **THEN** (Otherwise N/A)

REFER TO 1-OI-68 for Reactor Recirc Pump trips. ☐

[17] **INITIATE** actions required to make the necessary repairs. ☐

NOTE
Restarting a Recirc Pump while in Region 1 is **NOT** allowed. Tech Spec 3.4.1.A requires that the Reactor Mode Switch be immediately placed in SHUTDOWN upon entry into Region 1

[18] **PERFORM** the following for Single Loop Operation:

[18.1] **REFER TO** 1-OI-68 for guidance on single loop operation. ☐

[18.2] **REFER TO** Tech Specs 3.4.1. ☐

[18.3] **WHEN** available, **THEN**

RETURN tripped Recirc Pump to service in accordance with 1-OI-68. ☐

40. RO 295001AK3.01 001/MEM/T1G1/RECIRC//295001AK3.01//RO/SRO/BANK

Given the following plant conditions:

- Unit 3 is operating at 100% power.
- The following alarm is received:
  - RECIRC LOOP A OUT OF SERVICE (3-XA-9-4A, Window 26)

Which ONE of the following describes the Reactor Water Level response?

A.✓ Rises initially due to swell, then returns to normal.

B. Lowers initially due to shrink, then returns to normal.

C. Rises initially due to swell and remains high due to a lower power level.

D. Lowers initially due to shrink and remains lower due to the loss of core voids.

K/A Statement:

295001 Partial or Complete Loss of Forced Core Flow Circulation / 1 & 4
AK3.01 - Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION : Reactor water level response

K/A Justification: This question satisfies the K/A statement by requiring the candidate to use specific plant conditions to determine the effect on reactor water level due to a partial loss of Recirculation flow.

References:

Level of Knowledge Justification: This question is rated as MEM due to the requirement to recall or recognize discrete bits of information.

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REFERENCE PROVIDED: None

Plausibility Analysis:

In order to answer this question correctly the candidate must determine the following:

1. The internal response of the Reactor Vessel due to a Recirc Pump trip.
2. The RPV level instrument response to the conditions determined from Item 1 above.

The RPV response occurs in three parts.

First, the trip of the recirc pump causes a sudden reduction of coolant flow up through the fuel bundles while power level remains approximately 100%. This causes a sudden increase in core void fraction. The large voiding in the active fuel region coupled with the reduction in inventory being removed from the downcomer by the tripped recirc pump results in a rapid rise in RPV level outside the core shroud while water level inside the core shroud lowers. Since RPV level is measured outside the shroud, indication rises.

Next, the large void content in the active fuel region responds quickly to insert negative reactivity, causing a large reduction in reactor power and therefore, steam (void) production. This reduction in void fraction draws water from the downcomer region outside the shroud into the active fuel region inside the shroud. Even though reactor power will drop to approximately 65% with a 100% rod pattern, core void fraction at 65% is actually greater than at 100% due to the effect of recirc flow. Therefore, RPV level indication does not immediately return to its original value.

Finally, the Feedwater Control System responds to the transient by reducing feedwater flow below steam flow to enable RPV level to slowly return to the original setpoint at a lower reactor power.

A - correct:

B - incorrect: This is plausible because level inside the core shroud initially lowers, then returns to normal. However, RPV level is NOT measured inside the core shroud.

C - incorrect: This is plausible because RPV level initially rises in response to the recirc pump trip. However, the lower power level is compensated by automatic adjustments of Feedwater Control.

D - incorrect: This is plausible because level inside the core shroud initially lowers, then returns to normal. In addition, the reduction of core voids is temporary. Final void fraction is actually higher. However, RPV level is NOT measured inside the core shroud.

41. RO 295003AA2.01 001/MEM/T1G1/0-GOI-100-4//RO 295003AA2.01//RO/SRO/NEW 10/16/07

Given the following plant conditions:

- All three units were at 100% rated power when 500KV PCB 5234 (Trinity 1 feed to Bus 1 Section 1) tripped and failed to auto close.
- The signal which caused the Power Circuit Breaker (PCB) trip CANNOT be reset.
- The Chattanooga Load Coordinator has issued a Switching Order directing BFN to open Motor Operated Disconnect (MOD) 5233 and 5235 to isolate 500KV PCB 5234 for troubleshooting.

Which ONE of the following describes your response to this Switching Order and the basis for that response?

ENSURE the PK block for PCB 5234 is______(1)______ to

prevent__________(2)__________.

	(1)	(2)
A.	removed;	to prevent electrical arcing across the MOD contacts while being opened.
B.✓	removed;	to prevent actuating the breaker failure logic and tripping the remainder of the PCBs on Bus 1.
C.	installed;	to prevent actuating the breaker failure logic and tripping the remainder of the PCBs on Bus 1.
D.	installed;	to prevent electrical arcing across the MOD contacts while being opened.

K/A Statement:

295003 Partial or Complete Loss of AC / 6
AA2.01 - Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER : Cause of partial or complete loss of A.C. power

K/A Justification: This question satisfies the K/A statement by requiring the candidate to use specific plant conditions to determine the potential cause of a partial or complete loss of AC power.

References: 0-GOI-300-4

Level of Knowledge Justification: This question is rated as MEM due to the requirement to recall or recognize discrete bits of information.

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REFERENCE PROVIDED: None

Plausibility Analysis:

In order to answer this question correctly the candidate must determine the following:

1. The function of the PK block and its relationship to the breaker failure logic.
2. Recognize the CAUTION in 0-GOI-300-4 related to PK block removal.

B - correct:

A - incorrect: This is plausible because electrical arcing across the MOD contacts is actually what causes the trip signal to be generated if the PK block is installed. The electrical arcing will occur with or without the PK block installed. However, it will not trip the 500KV breakers on the bus when it happens without the PK block installed.

C - incorrect: This is plausible since the wording is ALMOST identical to the CAUTION, however the PK block must be removed.

D - incorrect: This is plausible because electrical arcing across the MOD contacts is actually what causes the trip signal to be generated if the PK block is installed. The electrical arcing will occur with or without the PK block installed. However, it will not trip the 500KV breakers on the bus when it happens without the PK block installed.

8.2 Response to a Breaker Trip on 161kV or 500kV Breaker

CAUTION

Breaker reclosure times on opposite ends of the transmission lines leaving BFN are 15 to 17 seconds after a trip. The breakers at BFN should reclose immediately thereafter.

[1] **IF** a line trips, **THEN**

WAIT 30 seconds before resetting the disagreement to ensure adequate time for automatic reclosure. ☐

NOTE

1. 161kV breakers have high speed and standard speed reclosure.
2. For PCBs equipped with digital relays, the PCB will lockout from AUTO closure if the affected line does not reclose from the other end within approximately 1.5 seconds. Only the AUTO closure is prevented, the breaker can be manually closed with Dispatcher concurrence.

CAUTION

Induced currents in the current transformers of a 500KV PCB during cycling of the associated MOD's, in conjunction with an existing PCB trip signal, may actuate the breaker failure logic and trip all PCB's on the associated 500KV bus. Thus the MOD's associated with a tripped PCB should **NOT** be operated until the trip has been reset; or, if the trip cannot be reset, the breaker failure PK block has been removed for the associated tripped PCB during MOD operation. Contact Dispatcher for instruction or assistance to reset the tripped relay.

42. RO 295004AK1.03 001/MEM/T1G1/24VDC/VB9/295004AK1.03//RO/SRO/BANK

Given the following plant conditions:

- A reactor startup is in progress on Unit 3 and reactor power is on IRM Range 7.
- The operator observes the following indications:
  - SRM DOWNSCALE (3-XA-9-5A, Window 6)
  - IRM CH 'A', 'C', 'E', and 'G' HI-HI / INOP (3-XA-9-5A, Window 33)

Which ONE of the following power sources, IF lost, would cause these failures?

A.✓ +/-24V DC Power Distribution Panel

B. +/-48V DC Power Distribution Panel

C. 120V AC RPS Power Supply Distribution Panel

D. 120V AC Instrument and Control Power Distribution Panel

K/A Statement:

295004 Partial or Total Loss of DC Pwr / 6
AK1.03 - Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER : Electrical bus divisional separation

K/A Justification: This question satisfies the K/A statement by requiring the candidate to use specific plant conditions to determine the effect on a division of IRM instruments due to a loss of DC power.

References:

Level of Knowledge Justification: This question is rated as MEM due to the requirement to recall or recognize discrete bits of information.

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REFERENCE PROVIDED: None

Plausibility Analysis:

In order to answer this question correctly the candidate must determine the following:

1. Which of th listed power supplies input to the IRM system.
2. Which power supply, if lost, would provide only those indications listed.

A - correct:

B - incorrect: This is plausible because 48V DC supplies power to annunciator panels in the control room including the two annunciators listed in the stem. However, other annunciators would also be affected by the loss that are not included on the list.

C - incorrect: This is plausible because RPS supplies trip units associated with IRMs. However, the given annunciators are not indicative of the loads supplied by RPS.

D - incorrect: This is plausible because 120V AC I&C Buses supply power to IRM detectors and drives. A loss of that power supply would also affect the entire division. However, the indications given in the stem are not indicative of loads supplied by I&C buses.

IRM CHAN B.D.F.H HI-HI/INOP	34

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Sensor/Trip Point:

Relay K16

A. Hi-Hi
   1. 116.4 on 125 scale.
B. INOP
   1. Hi voltage low.
   2. Module unplugged.
   3. Function switch **NOT** in operate.
   4. Loss of ± 24 VDC to monitor.

Sensor Location: Control Room Panel 1-9-12.

Probable Cause:

A. Flux level at or above setpoint.
B. One or more inoperable conditions exist.
C. Testing in progress.
D. Malfunction of sensor.
E. Control rod drop accident.

Automatic Action:

A. Half-scram if one sensor actuates (except with Rx Mode Switch in RUN).
B. Reactor scram if one sensor per channel actuates (except with Rx Mode Switch in RUN).

Operator Action:

A. **STOP** any reactivity changes. ☐
B. **VERIFY** alarm by multiple indications. ☐
C. **RANGE** initiating channel or **BYPASS** initiating channel. **REFER TO** 1-OI-92A. ☐
D. With SRO permission, **RESET** Half Scram. **REFER TO** 1-OI-99
E. **IF** alarm is from a control rod drop, **THEN** **REFER TO** 1-AOI-85-1. ☐
F. [NRC/C] **IF** one or more IRM recorder reading is downscale, **THEN** **CHECK** for loss of ± 24 VDC power. ☐
G. **NOTIFY** Instrument Maintenance that functional tests of any monitors indicating an INOP condition, including a downscale reading, are required before the instrument can be considered operable. [NRC IE item 86-40-03] ☐
H. **NOTIFY** Reactor Engineer. ☐
I. **REFER TO** Tech Spec Table 3.3.1.1-1, TRM Tables 3.3.4-1 and 3.3.5-1. ☐

References:

1-45E620-6	1-730E237-6, -10	1-730E915-10
1-730E915RF-12	1-SIMI-92B	

SRM DOWNSCALE 6	Sensor/Trip Point:
(Page 1 of 1)	Relay K-19 Count rate 5 cps.

Sensor Location: Panel 1-9-12, MCR.

Probable Cause:

A. An un-bypassed SRM channel having a count rate ≤ 3 counts per second.
B. SI (or SR) in progress.
C. Malfunction of sensor.

Automatic Action: Rod block below range 3 on IRM and Rx Mode Sw. **NOT** in Run.

Operator Action:

A. **VALIDATE** SRM downscale. ☐
B. **IF** alarm valid, **THEN**
REFER TO 1-OI-92 during startup (Mode 2) operation or 0-GOI-100-3A, -3C during refuel (Mode 5) operation. ☐
C. **NOTIFY** Unit Supervisor. ☐
D. **REFER TO** Tech. Spec. Sect. 3.3.1.2, Table 3.3.1.2-1, TRM Tables 3.3.4-1 and 3.3.5-1. ☐

References: 1-45E620-6-1 1-730E237-8

2.0 SYMPTOMS (continued)

H. Loss of Main Steam Relief Valve position indication.

I. Loss of power to RCIC and HPCI Turbine Vibration circuitry and position indication for testable check valves, (Panel 9-3).

J. Loss of RHRSW and EECW Division I instrumentation, (Panel 9-3, 9-20).

K. Loss of SBGT A flow and differential pressure indication, (Panel 9-25).

L. Loss of SLC A and B amber ready lights and valve position indication, (Panel 9-5).

M. Loss of LPRM meter lights and APRM alarm lights, (Panel 9-5).

N. Loss of Condensate - Feedwater and Heater Drains instrumentation, (Panel 9-6).

O. Loss of SRM/IRM detector drive power and position indication, (Panel 9-5).

P. Loss of one-half the blue scram lights and accumulator low pressure-high level light indications (Panel 9-5, 25-04).

Q. Loss of Control Bay Emergency Ventilation System Division I.

R. Loss of AC Supply to ± 24V NEUTRON MONITORING BATT CHGR A1-1 NEG SIDE, 1-CHGD-283-0000A1-1 and ± 24V NEUTRON MONITORING BATT CHGR A2-1 POS SIDE, 1-CHGD-283-0000A2-1. (The Neutron Monitoring Battery System is rated to carry loads for 3 hours. STACK GAS CH1 RAD MON RTMR, 0-RM-090-0147B will be lost after this time period)

S. Loss of Main Steam Line B, D and Feedwater Line B flow indicators and inputs to 3 Element Control and Rod Worth Minimizer.

T. "B" Fuel Pool Demin valves:

1. 1-FCV-078-0063, FPC F/D OUTBD ISOL VLV Closes,
2. 1-FCV-078-0068, RX WELL INFL INBD VLV Closes,
3. 1-FCV-078-0066, FPC F/D 1A BYP VLV Opens.

These actions result from loss of power to A and C skimmer surge tank low-low level switches.

U. I&C BUS A VOLTAGE ABNORMAL (1-XA-55-8C, Window 21).

V. Short Cycle valves 1-FCV-002-0029A and1- FCV-002-0029B fail open due to loss of power to 1-FC-2-29.

43. RO 295005AA1.04 001/C/A/T1G1///295005AA1.04//RO/SRO/NEW 11/28/07 RMS

Given the following plant conditions:

- Unit 1 is at 100% rated power when the Desk Unit Operator notices that number 3 Main Turbine Stop Valve (MTSV) position indication is reading 0%.
- Numbers 1, 2, and 4 MTSV position indications all read 100%.
- Maintenance investigation determines that the cause of the MTSV position indication failure is due to a mechanical failure of the Linear Variable Differential Transducer (LVDT).
- The Unit 1 Main Turbine receives a trip signal.

Which ONE of the following describes the effect on Main Turbine operation and any required action?

~~Main T~~urbine operation is ____.

A. ~~affected~~. The RPS logic contact for the #3 MTSV will NOT function so a turbine trip may not initiate a scram.

B. ~~NOT affecte~~d. The RPS logic contact is already OPEN for the #3 MTSV so a turbine trip will still initiate a scram.

C✓ a~~ffecte~~d. The Generator output breaker will NOT open on a turbine trip due to a 4-out-of-4 logic arragement.

D. ~~NOT affected~~. The Generator output breaker will still open on a turbine trip due to a 2-out-of-4 logic arragement.

K/A Statement:

295005 Main Turbine Generator Trip / 3
AA1.04 - Ability to operate and/or monitor the following as they apply to MAIN TURBINE GENERATOR TRIP : Main generator controls

K/A Justification: This question satisfies the K/A statement by requiring the candidate to use specific plant conditions to determine the required action following a Main Turbine Generator trip.

References: OPL171.228

Level of Knowledge Justification: This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

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REFERENCE PROVIDED: None

Plausibility Analysis:

In order to answer this question correctly the candidate must determine the following:

1. Whether the LVDT position indicator feeds the RPS logic, the Turbine Trip logic, or both.
2. Based on the above answer, the affect of a turbine trip with the failure active.

C - correct:

A - incorrect: This is plausible because the Main Turbine operation is affected. However, the position indication supplied to RPS is a limit switch, NOT the LVDT position. Therefore, a turbine trip WILL initiate a scram signal to RPS.

B - incorrect: This is plausible because the position of #3 MTSV is supplied to RPS logic. However, the position indication supplied to RPS is a limit switch, NOT the LVDT position. Therefore, RPS logic "sees" #3 MTSV as open until the turbine trips.

D - incorrect: This is plausible based on the different logic associated with the CIVs and RPS on the main turbine. However, the logic for MTSV inputs to open the generator output breaker is a 4-out-of-4 logic. Therefore, the generator output breaker will NOT automatically open.

INSTRUCTOR NOTES

c. Consequences of Event

This caused the Bypass valves to start opening. Due to the short duration of the error signal the bypass valves did not reach full open and subsequently closed.

Operation of the bypass valves would impact Rx pressure, Rx power and Generator load.

Corrective Action - EHC logic software was modified to eliminate the possibility of this type response to a communications glitch.

2. At BFN on 1/15/2006, the Unit 3 generator breaker failed to trip as expected on a turbine trip.

PER 95370

a. Description of Event

At BFN on 1/15/2006, the Unit 3 generator breaker failed to trip as expected on a turbine trip. The logic for the generator breaker needs to see all the stop valves closed and the CIV's closed (either intercept or stop). The LVDT for S.V#1 was failed such that the generator breaker would not open on a turbine trip. Operator action was taken to manually trip the generator breaker

The metal rod moves to alter the magnetic coupling of 2 opposing transformer secondary windings to make an LVDT provide an output proportional to the position of the metal rod.

INSTRUCTOR NOTES

b. Cause of Event

The LVDT transformer coupling rod became disconnected from the valve and fell to a position which gave indication of ~ 50% valve position. The affect on the logic for tripping the generator PCB on a turbine trip was not recognized.

Work practices
Monitor all parameters during a transient and ensure automatic actions have occurred

c. Consequences of Event

Tripping of the generator breaker on a turbine trip prevents a reverse power situation where the generator and turbine could attempt to rotate backwards, causing equipment damage. The unit operator's quick recognition and response to the breaker failure to trip prevented damage.

44. RO 295006AK3.05 001/C/A/T1G1/RPS/1/295006AK3.05//RO/SRO/BANK

Given the following plant conditions:

- Power ascension is in progress on Unit 3 with the Main Turbine on line.
- Control Rods are being withdrawn to increase power.
- As reactor power approaches 35%, the Shift Technical Advisor (STA) notes that two (2) Turbine Bypass Valves are OPEN.

Which ONE of the following describes the effect on the plant?

Regarding the UFSAR Chapter 14 analyses for a turbine trip, the above condition is ______(1)______ conservative than the assumptions used in the UFSAR because ____________________(2)____________________.

	(1)	(2)
A.	more;	it lowers the actual power level at which the RPS reactor scram on turbine trip is enabled.
B.✓	less;	it raises the actual power level at which the RPS reactor scram on turbine trip is enabled.
C.	more;	it lowers the peak vessel pressure for a design basis transient in regard to transition boiling.
D.	less;	it raises the actual power level for a design basis transient in regard to peak cladding temperature.

K/A Statement:

295006 SCRAM / 1
AK3.05 - Knowledge of the reasons for the following responses as they apply to SCRAM : Direct turbine generator trip: Plant-Specific

K/A Justification: This question satisfies the K/A statement by requiring the candidate to use specific plant conditions to determine the response to a Main Turbine trip and the basis for that response related to a reactor scram.

References:

Level of Knowledge Justification: This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

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REFERENCE PROVIDED: None

Plausibility Analysis:

In order to answer this question correctly the candidate must determine the following:

1. Which assumptions are of concern regarding the UFSAR analysis for a turbine trip.
2. What affect the above conditions have on that analysis.
3. Which thermal limit is of concern regarding the analyzed transient.

B - correct:

A - incorrect: This is plausible because the initial power level prior to a scram is assumed in the analysis. However, the given conditions raise the initial power level which is less conservative.

C - incorrect: This is plausible because RPV pressure affects transition boiling. However, this is NOT the limit of concern during this analysis and the initial conditions are LESS conservative with regard to MCPR.

D - incorrect: This is plausible because the condition is less conservative based on initil power. However, the limit of concern is NOT PCT, but MCPR.

45. RO 295016AA2.04 001/C/A/T1G1/AOI-100-2//295016AA2.04//RO/SRO/NEW 12/17/2007 RMS

Given the following plant conditions:

- Unit 3 control room was abandoned due to a fire.
- Actions are being carried out in accordance with 3-AOI-100-2, Control Room Abandonment.
- RCIC is injecting with RPV level at (+)20 inches and steady.
- A cooldown has begun using Safety Relief Valves (SRVs). Reactor pressure is 850 psig and lowering.
- RHR Loop I is in Suppression Pool Cooling.

In accordance with 3-AOI-100-2, "Control Room Abandonment", a Suppression Pool Temperature limit of less than or equal to _(1)_ °F has been established. The basis for this limit is ____________(2)____________?

(1) (2)

A. 95 °F; to prevent exceeding the Technical Specification LCO before reaching Mode 4 (Cold Shutdown).

B. 110 °F; to prevent exceeding the Heat Capacity Temperature Limit before the reactor can be verified to be shutdown.

C. 120 °F; to prevent damage to the RCIC turbine from overheated lube oil, which is cooled by the Suppression Pool water.

D✓ 120 °F; to prevent exceeding the design basis maximum allowable values for primary containment temperature or pressure.

K/A Statement:

295016 Control Room Abandonment
AA2.04 - Ability to determine and/or interpret the following as they apply to CONTROL ROOM ABANDONMENT : Suppression pool temperature

K/A Justification: This question satisfies the K/A statement by requiring the candidate to use specific plant conditions to determine the limitation and basis for Suppression Pool Temperature during a Control Room Abandonment.

References: 3-AOI-100-2, Tech Spec Bases 3.6, EOIPM 0-V-B

Level of Knowledge Justification: This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

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REFERENCE PROVIDED: None

Plausibility Analysis:

In order to answer this question correctly the candidate must determine the following:

1. The Suppression Pool Temperature limit established by 3-AOI-100-2.
2. The basis for the above limit.

D - correct:

A - incorrect: This is plausible because ≤ 95°F is the normal operating limit imposed by Technical Specifications. However, this limit is NOT expected to be maintained during Control Room Abandonment.

B - incorrect: This is plausible because the basis for ≤ 110°F is correct. However, this limit is NOT expected to be maintained during Control Room Abandonment and the reactor is assumed to be shutdown.

C - incorrect: This is plausible because elevated SP temperatures can result in overheating the lube oil used for lubricating RCIC. This constitues the basis for Caution #6 in the EOIs; but, does NOT apply during Control Room Abandonment.

Date ________

4.2 Unit 3 Subsequent Actions (continued)

[15.7] **ESTABLISH** RHR system flow between 7,000 and 10,000 gpm as follows: ☐

[15.7.1] **MONITOR** RHR SYS I TOTAL FLOW, 3-FI-74-79 at Panel 3-25-32. ☐

[15.7.2] **THROTTLE OPEN** 3-HS-074-0059C, RHR SYSTEM I TEST VLV at 480V RMOV Bd 3A, Compt. 12C, ☐

[15.7.3] **WHEN** RHR SYS I TOTAL FLOW, 3-FI-74-79 indicates between 7,000 and 10,000 gpm, **THEN**

DIRECT the operator to stop throttling 3-HS-074-0059C. ☐

[15.7.4] **VERIFY CLOSED** RHR SYSTEM I MINIMUM FLOW VALVE, 3-FCV-74-7, at either of the following: ☐

- 480V RMOV Bd 3D, Compt. 4E, 3-BKR-074-0007 RHR SYSTEM I MINIMUM FLOW VLV FCV-74-7 (MO10-16A), OR (Otherwise N/A) ☐
- Rx Bldg - SW Quad - E1 541' local control switch RHR SYSTEM I MINIMUM FLOW VALVE, 3-HS-074-0007B. (Otherwise N/A) ☐

[15.8] **MONITOR** SUPPR POOL TEMPERATURE, 3-TI-64-55B, at Panel 3-25-32 **and MAINTAIN** temperature less than 120°F , ☐

BASES

ACTIONS

D.1, D.2, and D.3 (continued)

Additionally, when suppression pool temperature is > 110°F, increased monitoring of pool temperature is required to ensure that it remains ≤ 120°F. The once per 30 minute Completion Time is adequate, based on operating experience. Given the high suppression pool average temperature in this Condition, the monitoring Frequency is increased to twice that of Condition A. Furthermore, the 30 minute Completion Time is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal suppression pool average temperature condition.

E.1 and E.2

If suppression pool average temperature cannot be maintained at ≤ 120°F, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the reactor pressure must be reduced to < 200 psig within 12 hours, and the plant must be brought to at least MODE 4 within 36 hours. The allowed Completion Time is reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

Continued addition of heat to the suppression pool with suppression pool temperature > 120°F could result in exceeding the design basis maximum allowable values for primary containment temperature or pressure. Furthermore, if a blowdown were to occur when the temperature was > 120°F, the maximum allowable bulk and local temperatures could be exceeded very quickly.

(continued)

BASES

LCO (continued)

b. Average temperature ≤ 105°F when any OPERABLE IRM channel is > 70/125 divisions of full scale on Range 7 and testing that adds heat to the suppression pool is being performed. This required value ensures that the unit has testing flexibility, and was selected to provide margin below the 110°F limit at which reactor shutdown is required. When testing ends, temperature must be restored to ≤ 95°F within 24 hours according to Required Action A.2. Therefore, the time period that the temperature is > 95°F is short enough not to cause a significant increase in unit risk.

c. Average temperature ≤ 110°F when all OPERABLE IRM channels are ≤ 70/125 divisions of full scale on Range 7. This requirement ensures that the unit will be shut down at > 110°F. The pool is designed to absorb decay heat and sensible heat but could be heated beyond design limits by the steam generated if the reactor is not shut down.

Note that 70/125 divisions of full scale on IRM Range 7 is a convenient measure of when the reactor is producing power essentially equivalent to 1% RTP. At this power level, heat input is approximately equal to normal system heat losses.

APPLICABILITY

In MODES 1, 2, and 3, a DBA could cause significant heatup of the suppression pool. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining suppression pool average temperature within limits is not required in MODE 4 or 5.

(continued)

DISCUSSION: CAUTION #5 and CAUTION #6

CAUTION #5, this warns the operator of the potential plant response if injection of cold, unborated water into the core is too rapid under conditions where little or no margin to subcriticality may exist. This may result in a large increase in positive reactivity with a subsequent reactor power excursion large enough to substantially damage the core.

CAUTION #6, the HPCI and RCIC Lube Oil Coolers are cooled by routing part of the pump discharge fluid to the cooler. At elevated temperatures in the suppression pool, the turbine lube oil may get too hot to provide adequate lubrication. Only during EOI operations will the system be needed at such an extreme suppression pool temperature. Therefore, the EOIs are an appropriate location for this caution.

46. RO 295018AK2.01 001/MEM/T1G1/RBCCW/3/295018AK2.01//RO/SRO/BANK

Which ONE of the following components would lose cooling upon isolation of the RBCCW NON-ESSENTIAL LOOP ISOLATION VALVE (2-FCV-70-48)?

A. Recirculation Pump Seals.

B. Drywell Atmospheric Coolers.

C.✔ Spent Fuel Pool Cooling Heat Exchanger.

D. Drywell Equipment Drain Sump Heat Exchanger.

K/A Statement:

295018 Partial or Total Loss of CCW / 8
AK2.01 - Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER and the following: System loads

K/A Justification: This question satisfies the K/A statement by requiring the candidate to use specific system knowledge to determine the effect on RBCCW loads due to a partial loss of RBCCW.

References: 1/2/3-AOI-70-1, OPL171.047

Level of Knowledge Justification: This question is rated as MEM due to the requirement to recall or recognize discrete bits of information.

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REFERENCE PROVIDED: None

Plausibility Analysis:

In order to answer this question correctly the candidate must determine the following:

1. Which of the loads listed are part of the "Essential Loop".
2. Which of the loads listed are part of the "Non-essential Loop".

C - correct:

A - incorrect: This is an "Essential Loop" load.

B - incorrect: This is an "Essential Loop" load.

D - incorrect: This is an "Essential Loop" load.

Attachment 1
(Page 1 of 1)

Components Cooled by RBCCW During Normal Plant Operation

SYSTEM	**COMPONENTS COOLED**
Reactor Recirculation	Pump Seals Pump Motor Bearings Pump Motor Windings Pump Discharge Sample Cooler
Primary Containment	Drywell Atmosphere Cooling Coils
Reactor Water Cleanup	Non-Regenerative Heat Exchangers Pump Seals Pump Bearings
Fuel Pool Cooling and Cleanup	Fuel Pool Heat Exchangers
Equipment Drains	Reactor Building Equipment Drain Sump Heat Exchanger Drywell Equipment Drain Sump Heat Exchanger

d. Proper system flow operation is assured by monitoring the system DP (pump discharge minus pump suction). — Done Each Shift

2. RBCCW Heat Loads

a. Essential loop loads — Obj. V.B.2, Obj. V.D.2

- Drywell Blowers(10)
- Reactor recirculation pump motor coolers (2)
- Reactor recirculation pump seal coolers (2)
- Drywell equipment drain sump heat exchanger (1)

b. Non-essential loop loads — Obj. V.B.3, Obj. V.D.3

- Reactor Building equipment drain sump heat exchanger (1)
- Reactor water cleanup pump seal water coolers and bearing oil coolers (2)
- RWCU Non-regenerative heat exchangers (2)
- Fuel pool cooling heat exchangers (2)
- Reactor recirculation pump discharge sample cooler (1)

3. RBCCW Heat Exchangers

a. These provide the means for heat removal from RBCCW by RCW with Emergency Equipment Cooling Water (EECW) as a backup. — DCN 51195, replaced HX1A & 1B, HX 1C NOT replaced. OPL171.051

b. They are counter-flow type, 50% capacity each.

- RBCCW flow makes one pass through the shell side.
- RCW makes one pass through the tube side.

47. RO 295019AA2.02 001/C/A/SYS/CA//295019AA2.02//RO/SRO/BANK

Given the following plant conditions:

- Unit 2 was at 100% power when a transient occurred which resulted in a reactor scram.
- After stabilizing the unit, the scram signal is RESET.
- All eight (8) Scram Solenoid Group lights are ON.
- Approximately ten minutes later, the following conditions are present:
  - Raw Cooling Water (RCW) Low Pressure Alarm
  - CRD charging water High Pressure Alarm
  - Outboard MSIVs CLOSED, Inboard MSIVs are OPEN
  - Scram Discharge Volume (SDV) Vents and Drain Valves are CLOSED
  - Scram Solenoid Air Valves are OPEN

Which ONE of the following describes the cause for the event?

A.✓ Loss of Control Air.

B. Loss of both RPS busses.

C. Loss of Drywell Control Air.

D. Loss of 9-9 cabinet 5, Unit Non-Preferred.

K/A Statement:

295019 Partial or Total Loss of Inst. Air / 8
AA2.02 - Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR : Status of safety-related instrument air system loads (see AK2.1 - AK2.19)

K/A Justification: This question satisfies the K/A statement by requiring the candidate to use specific plant conditions to determine the effect of a loss of Control Air on safety related loads.

References: 2-AOI-32-2

Level of Knowledge Justification: This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

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REFERENCE PROVIDED: None

Plausibility Analysis:

In order to answer this question correctly the candidate must determine the following:

1. Which indications given are indicative of the possible causes listed.

A - correct:

B - incorrect: This is plausible because the scram would occur as well as scram valves open and SDV vents and drains closed, however these indications would NOT be appropriate AFTER the scram was reset. In fact, the scram could NOT be reset without RPS available.

C - incorrect: This is plausible because a loss of Drywell Control Air would cause MSIVs to close, however the INBOARD valves would close.

D - incorrect: This is plausible because the ONLY indications given that would NOT apply would be outboard MSIVs closing and SDV vents and drains failure to re-open.

1.0 PURPOSE

This Abnormal Operating Instruction provides symptoms, automatic action, operator actions and expected system responses for loss of control air.

2.0 SYMPTOMS

A. AIR COMPRESSOR ABNORMAL annunciator, (1-XA-55-20B, Window 29) is in alarm.

B. CONTROL AIR COMP G BKR ENERGIZED (0-XA-55-23B, Window 38) will reset (extinguish) when panel reset pushbutton is depressed.

C. CONTROL AIR COMP G MOTOR AMPS, 0-EI-32-2901, on Panel 1-9-20 indicates approximately zero amps.

D. Air Compressor G ICS Display shows Compressor G in an unloaded or shutdown condition.

E. Air Compressor G ICS Display shows lowering Control Air Header Pressure.

F. Control Air Compressor G breaker tripped.

G. SERVICE AIR XTIE VLV OPEN (0-FCV-33-1 Open) annunciator, 0-PA-33-1A/1(3) (Unit 1 and Unit 3) on Panel 1(3)-9-20 is in alarm at (1(3)-XA-55-20B, Window 30).

H. CONTROL AIR PRESS LOW annunciator, 0-PA-32-88 is in alarm (2-XA-55-20B, Window 32).

I. SCRAM PILOT AIR HEADER PRESS LOW annunciator, 2-PA-85-38B on Panel 9-5 is in alarm (2-XA-55-5B, Window 28).

J. MAIN STEAM LINE ISOL VLV POSN HALF SCRAM annunciator is in alarm (2-XA-55-4A, Window 30).

K. DRYWELL CONTROL AIR PRESSURE LOW 2-PA-32-70 annunciator is in alarm (2-XA-55-3E, Window 35).

L. CONDENSER A, B OR C VACUUM LOW 2-PA-47-125 annunciator is in alarm (2-XA-55-7B, Window 17).

M. OG HOLDUP LINE INLET FLOW LOW 2-FA-66-111A annunciator is in alarm (2-XA-55-53, Window 4).

N. HOTWELL A(B)(C) LEVEL ABNORMAL 2-LA-2-3(2-LA-2-6)(2-LA-2-9) is in alarm (2-XA-55-6A, Window 5(6)(7)).

2.0 SYMPTOMS (continued)

O. REACTOR WATER LEVEL ABNORMAL 2-LA-3-53 annunciator is in alarm (2-XA-55-5A, Window 8).

P. REACTOR PRESS HIGH 2-PA-3-53 annunciator is in alarm (2-XA-55-5A, Window 1).

Q. REACTOR CHANNEL A(B) AUTO SCRAM annunciator in alarm if any scram setpoint is exceeded (2-XA-55-5B, Window 1(2)).

R. MAIN STEAM RELIEF VLV AIR ACCUM PRESS LOW 2-PA-32-31 annunciator in alarm (2-XA-55-3D, Window 18).

3.0 AUTOMATIC ACTIONS

A. Unit 2 to Unit 3 Control Air Crosstie, 2-PCV-032-3901, will close when Control Air Header pressure reaches 65 psig and lowering at the valve.

B. U-1 TO U-2 CONT AIR CROSSTIE, 1-PCV-032-3901, will close to separate Units 1 and 2 when Control Air Header pressure reaches 65 psig and lowering at the valve.

C. 2-PCV-84-0654, CAD/CA FLOW SEL, will select nitrogen from CAD tank A to supply 2-FSV-64-20, 2-FSV-64-21, 2-FSV-64-221, and 2-FSV-64-222 at ≤75 psig.

D. 2-PCV-84-0033, will select nitrogen from CAD tank A to supply 2-FSV-84-19, 2-FSV-64-29, and 2-FSV-64-32.

E. 2-PCV-84-0034, will select nitrogen from CAD tank B to supply 2-FSV-84-20, 2-FSV-64-31, and 2-FSV-64-34.

4.0 OPERATOR ACTIONS

NOTE

[NER/C] Attachment 1 provides expected system responses, critical components that do not fail in intended positions should be placed in the required positions. [INPO SOER 88-001]

4.1 Immediate Actions

None

4.2 Subsequent Actions

[1] **IF** a RFP Minimum Flow Valve failed open and flow is required from the condensate/feedwater system to reactor vessel or to prevent pump overload, **THEN**

ISOLATE the associated RFP minimum flow lines in the appropriate RFPT Room as follows: (N/A any RFP valves not affected.)

- RFP 2A MIN FLOW SHUTOFF, 2-SHV-003-0508 ☐
- RFP 2B MIN FLOW SHUTOFF, 2-SHV-003-0517 ☐
- RFP 2C MIN FLOW SHUTOFF, 2-SHV-003-0526 ☐

[2] **IF** CNDS BSTR PUMPS DISCH BYPASS TO COND B, 2-FCV-2-29A

and

CNDS BSTR PUMPS DISCH BYPASS TO COND C, 2-FCV-2-29B fail CLOSED, **THEN** (Otherwise N/A)

- **VERIFY** a flow path for condensate system

  OR

- **STOP** the condensate pumps/booster pumps using 2-OI-2. ☐

[3] **IF** any outboard MSIVs fails closed, **THEN:**

PLACE associated hand-switch on Panel 2-9-3 to close position. (Otherwise N/A) ☐

4.2 Subsequent Actions (continued)

[4] **IF** RSW STRG TNK ISOLATION VALVE, 0-FCV-025-0032 FAILS CLOSED, **THEN**

START a high pressure fire pump using 0-OI-26 . ☐

[5] **OPEN** CAD SYSTEM A N2 SHUTOFF VALVE, 0-FCV-084-0005, at Panel 9-54. ☐

[6] **OPEN** CAD SYSTEM B N2 SHUTOFF VALVE, 0-FCV-084-0016, at Panel 9-55. ☐

NOTES

1) All RCW temperature control valves fail open except for 2-TCV-24-80B and 2-TCV-24-85B on 2A and 2B RBCCW heat exchangers and 2-TCV-024-0075B on the Main Turbine Oil Coolers (4" line) which fail closed.
2) The appropriate computer points may be used for monitoring for the following lube oil temperatures, or any local temperature monitoring device that may be available, as necessary

[7] **IF** RCW pump motor amps indicate that RCW System flow reduction is required, **THEN**

REDUCE RCW flows as required: (Otherwise N/A). ☐

[7.1] **CLOSE** main turbine lube oil cooler TCV isolation valve 2-SHV-024-0583 or 2-SHV-024-0584, **THEN**

ESTABLISH lube oil temperature between 80°F and 90°F using TCV BYPASS VALVE 2-BYV-024-0585 or 2-BYV-024-0586. ☐

[7.2] **CLOSE** the following RFP turbine oil cooler TCV isolation valves

- A RFP 2-24-624A or 2-24-625A ☐
- B RFP 2-24-624B or 2-24-625B ☐
- C RFP 2-24-624C or 2-24-625C ☐

1.0 PURPOSE

This abnormal operating instruction provides symptoms, automatic actions and operator actions for the loss of Drywell Control Air System for causes other than Group 6 Isolation. The loss of Drywell Control Air caused by a Group 6 Isolation is addressed in 2-AOI-64-2d.

2.0 SYMPTOMS

A. DRYWELL CONTROL AIR PRESS LOW (2-XA-55-3E, Window 35) at ≤ 87 psig.

B. MAIN STEAM RELIEF VLV AIR ACCUM PRESS LOW (2-XA-55-3D, Window 18) at ≤ 82 psig.

C. Inboard MSIV's close or start to close.

D. Drywell cooler dampers close.

3.0 AUTOMATIC ACTIONS

None

48. RO 295021G2.4.50 001/C/A/T1G1/74-1//2950212.4.50//RO/SRO/BANK

Given the following plant conditions:

- Unit 2 is aligned with RHR Loop I in Shutdown Cooling and RHR Loop II in standby readiness.
- A leak occurs in the RPV, which results in the following conditions:
  - RPV level at 0 inches and slowly lowering
  - Drywell Pressure at 3.0 psig and slowly rising
  - RHR Pumps 'A' and 'C' TRIPPED

Which ONE of the following describes the minimum actions required to align RHR Loop II for injection to the RPV?

A.✓ After FCV-74-47 OR FCV-74-48 is closed,

push the RHR SYS II SD CLG INBD INJECT ISOL RESET 2-XS-74-132.

B. After FCV-74-47 AND FCV-74-48 are closed,

start RHR Loop II pumps, reset PCIS, and open the inboard injection valve.

C. After FCV-74-47 OR FCV-74-48 is closed;

reset PCIS, push the RHR SYS II SD CLG INBD INJECT ISOL RESET 2-XS-74-132, and open the inboard injection valve.

D. After FCV-74-47 AND FCV-74-48 are closed,

reset PCIS, push the RHR SYS II SD CLG INBD INJECT ISOL RESET 2-XS-74-132, and open BOTH injection valves.

K/A Statement:

295021 Loss of Shutdown Cooling / 4
2.4.50 - Emergency Procedures / Plan Ability to verify system alarm setpoints and operate controls identified in the alarm response manual

K/A Justification: This question satisfies the K/A statement by requiring the candidate to analyze plant conditions and determine the required actions during an emergency which have resulted in a loss of shutdown cooling.

References: 2-AOI-74-1

Level of Knowledge Justification: This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

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REFERENCE PROVIDED: None

Plausibility Analysis:

In order to answer this question correctly the candidate must determine the following:

1. Current RHR Loop II status given the initial conditions.
2. Based on the RHR Loop II status, determine the minimum actions to align Loop II for injection to the RPV.

A - correct:

B - incorrect: This is plausible because the valve alignment is correct and resetting 2-XS-74-132 is correct. However, resetting PCIS, re-opening FCV 74-47 and re-starting RHR pumps are NOT required.

C - incorrect: This is plausible because the valve alignment is correct and resetting 2-XS-74-132 is correct. However, resetting PCIS and re-opening FCV 74-47 is NOT required.

D - incorrect: This is plausible because the valve alignment is correct and resetting 2-XS-74-132 is correct. However, resetting PCIS and re-opening FCV 74-47 & 48 are NOT required.

4.2 Subsequent Actions (continued)

[5] **IF** Shutdown Cooling isolates on low RPV water level or high Drywell press (GROUP 2 ISOL) AND RPV water level needs restoring using LPCI, **THEN** (Otherwise N/A)

PERFORM the following before reaching -122 inches RPV water level: ☐

NOTE
The LPCI inboard injection valve that is aligned per 2-POI-74-2 will already be in the required accident position with the breakers open and will **NOT** isolate.

[5.1] **PERFORM** the following on a group 2 isolation: ☐

[5.1.1] **IF** 2-POI-74-2 is in effect, **THEN**

VERIFY CLOSED one of the following valves: (Otherwise N/A) ☐

- RHR SHUTDOWN COOLING SUCT OUTBD ISOL VLV, 2-FCV-74-47. ☐
- RHR SHUTDOWN COOLING SUCT INBD ISOL VLV, 2-FCV-74-48.

AND ☐

- **VERIFY CLOSED** the LPCI inboard injection valve **NOT** aligned for 2-POI-74-2, (RHR SYS I LPCI INBD INJECT VALVE, 2-FCV-74-53 OR RHR SYS II LPCI INBD INJECT VALVE, 2-FCV-74-67) ☐

4.2 Subsequent Actions (continued)

[5.1.2] **IF** 2-POI-74-2 is **NOT** in effect, **THEN**

VERIFY CLOSED the following valves on a Group 2 isolation: ☐

- RHR SHUTDOWN COOLING SUCT OUTBD ISOL VLV, 2-FCV-74-47. ☐
- RHR SHUTDOWN COOLING SUCT INBD ISOL VLV, 2-FCV-74-48. ☐
- RHR SYS I LPCI INBD INJECT VALVE, 2-FCV-74-53. ☐
- RHR SYS II LPCI INBD INJECT VALVE, 2-FCV-74-67. ☐

[5.2] **DEPRESS** RHR SYS I(II) SD CLG INBD INJECT ISOL RESET, 2-XS-74-126 and 2-XS-74-132 AND **VERIFY** 2-IL-74-126 and 2-IL-74-132 extinguished. ☐

49. RO 295023AK1.02 001/C/A/T1G1/79-2/V.B.3.B/295023AK1.02//RO/SRO/MODIFIED 11/17/07

Fuel loading is in progress on Unit 1 when you notice an unexplained rise in Source Range Monitor (SRM) count rate and an indicated positive reactor period.

Which ONE of the following actions is an appropriate response?

A. Immediately EVACUATE all personnel from the refuel floor.

B. If unexpected criticality is observed following control rod withdrawal, manually SCRAM the reactor.

C.✓ If the reactor cannot be determined to be subcritical, traverse the refueling bridge and fuel assembly away from the reactor core, preferably to the area of the cattle chute.

D. If all rods are not inserted/cannot be inserted, verify the fuel grapple is latched onto the fuel assembly handle and immediately remove the fuel assembly from the reactor core.

K/A Statement:

295023 Refueling Acc Cooling Mode / 8
AK1.02 - Knowledge of the operational implications of the following concepts as they apply to REFUELING ACCIDENTS : Shutdown margin

K/A Justification: This question satisfies the K/A statement by requiring the candidate to analyze specific plant conditions to determine a reduction in Shutdown Margin has occurred and the actions required to address that condition.

References: 1-AOI-79-2

Level of Knowledge Justification: This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

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MODIFIED FROM OPL171.060 #1

REFERENCE PROVIDED: None

Plausibility Analysis:

In order to answer this question correctly the candidate must determine the following:

1. The appropriate condition and Immediate Action required by 1-AOI-79-2.

C - correct:

A - incorrect: This is plausible because the evacuation of the Refuel Floor MAY be directed, but other actions to mitigate the problem take precedence until personnel safety is compromised.

B - incorrect: This is plausible because the condition is correct, but the action to scram is incorrect. Reinserting the control rod is required.

D - incorrect: This is plausible because the required action is correct, but the condition is NOT correct. This action is based on unexplained criticality following insertion of a fuel assembly.

4.0 OPERATOR ACTIONS

4.1 Immediate Actions

[1] **IF** unexpected criticality is observed following control rod withdrawal, **THEN**

REINSERT the control rod. ☐

[2] **IF** all control rods can NOT be fully inserted, **THEN**

MANUALLY SCRAM the Reactor. ☐

[3] **IF** unexpected criticality is observed following the insertion of a fuel assembly, **THEN**

PERFORM the following:

[3.1] **VERIFY** fuel grapple latched onto the fuel assembly handle AND **IMMEDIATELY REMOVE** the fuel assembly from the Reactor core. ☐

[3.2] **IF** the Reactor can be determined to be subcritical **AND** no radiological hazard is apparent, **THEN**

PLACE the fuel assembly in a spent fuel storage pool location with the least possible number of surrounding fuel assemblies and **LEAVE** the fuel grapple latched to the fuel assembly handle. ☐

[3.3] **IF** the Reactor can NOT be determined to be subcritical **OR** adverse radiological conditions exist, **THEN**

TRAVERSE the Refueling Bridge and fuel assembly away from the Reactor core, preferably to the area of the cattle chute and **CONTINUE** at Step 4.1[4]. ☐

[4] **IF** the Reactor can NOT be determined to be subcritical **OR** adverse radiological conditions exist, **THEN**

EVACUATE the refuel floor. ☐

50. RO 295024G2.1.33 001/C/A/T1G1/CONT/PRI/B10/295024G2.1.33//RO/SRO/BANK

During operation at 100% power a gross failure of both seals on Recirculation Pump 'B' increases Drywell Pressure to 2.0 psig.

Which ONE of the following is the approximate amount and type of Reactor Coolant System (RCS) leakage resulting from this condition?

A. 30 gpm of Identified leakage

B. 30 gpm of Unidentified leakage

C. 60 gpm of Identified leakage

D.✔ 60 gpm of Unidentified leakage

K/A Statement:

295024 High Drywell Pressure / 5
2.1.33 - Conduct of Operations Ability to recognize indications for system operating parameters which are entry-level conditions for technical specifications

K/A Justification: This question satisfies the K/A statement by requiring the candidate to determine that entry into Technical Specifications is required based on conditions which have resulted in high drywell pressure.

References: U2 TSR Sections 1 & 3.4.4, 2-AOI-68-1, OPL171.007

Level of Knowledge Justification: This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

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REFERENCE PROVIDED: None

Plausibility Analysis:

In order to answer this question correctly the candidate must determine the following:

1. Whether the leakage is IDENTIFIED or UNIDENTIFIED leakage.
2. The amount of leakage associated with a gross failure of both seals on a single recirc pump.

D - correct:

A - incorrect: This is plausible because the amount of leakage is correct. However, the leakage is not IDENTIFIED because the leakage is NOT intentionally captured and directed to a sump and is NOT expected.

B - incorrect: This is plausible because the leakage is UNIDENTIFIED and equal to the Tech Spec value for total leakage, but insufficient for the conditions given.

C - incorrect: This is plausible because the leakage is equal to the Tech Spec value for total leakage, but insufficient for the conditions given. In addition, the leakage is not IDENTIFIED because the leakage is NOT intentionally captured and directed to a sump and is NOT expected.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.4 RCS Operational LEAKAGE

LCO 3.4.4 RCS operational LEAKAGE shall be limited to:

a. No pressure boundary LEAKAGE;

b. ≤ 5 gpm unidentified LEAKAGE; and

c. ≤ 30 gpm total LEAKAGE averaged over the previous 24 hour period; and

d. ≤ 2 gpm increase in unidentified LEAKAGE within the previous 24 hour period in MODE 1.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Unidentified LEAKAGE not within limit. OR Total LEAKAGE not within limit.	A.1	Reduce LEAKAGE to within limits.	4 hours
B. Unidentified LEAKAGE increase not within limit.	B.1 OR	Reduce LEAKAGE increase to within limits.	4 hours (continued)

1.1 Definitions (continued)

LEAKAGE	LEAKAGE shall be: a. <u>Identified LEAKAGE</u> 1. LEAKAGE into the drywell, such as that from pump seals or valve packing, that is captured and conducted to a sump or collecting tank; or 2. LEAKAGE into the drywell atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE; b. <u>Unidentified LEAKAGE</u> All LEAKAGE into the drywell that is not identified LEAKAGE; c. <u>Total LEAKAGE</u> Sum of the identified and unidentified LEAKAGE; d. <u>Pressure Boundary LEAKAGE</u> LEAKAGE through a nonisolable fault in a Reactor Coolant System (RCS) component body, pipe wall, or vessel wall.
LINEAR HEAT GENERATION RATE (LHGR)	The LHGR shall be the heat generation rate per unit length of fuel rod. It is the integral of the heat flux over the heat transfer area associated with the unit length.

(continued)

RECIRC PUMP A
NO. 2 SEAL
LEAKAGE HIGH
2-FA-68-55

18

(Page 1 of 2)

Sensor/Trip Point:

2-FIS-068-0055 0.1-0.2 gpm after second seal.

Sensor Location: Recirculation Pump 2A Drywell

Probable Cause:
A. Recirculation Pump 2A No. 2 (outer) seal failure.
B. Sensor malfunction.

Automatic Action: None

Operator Action:

A. **COMPARE** No. 2 cavity pressure indicator (2-PI-68-63A) to No. 1 cavity pressure indicator (2-PI-68-64A), on Panel 2-9-4 or ICS. No. 2 seal degradation is indicated if the pressure at No. 2 seal is less than 50% of the pressure at No. 1 seal. ☐

B. **IF** seal failure is indicated, **THEN** **INITIATE** seal replacement as soon as possible. Continued operation is permissible if Drywell leakage is within T.S. limits. ☐

NOTE

1) Possible indications of dual seal failure include:
- Window 25 on this panel alarming in conjunction with this window.
- Rising drywell pressure and/or temperature.
- Increased leakage into the drywell sump.
- Increased vibration of the recirc pump.

C. **IF** dual seal failure is indicated, **THEN** ☐
   1. **SHUTDOWN** Recirc Pump 2A by DEPRESSING RECIRC DRIVE 2A SHUTDOWN, 2-HS-96-19.. ☐
   2. **VERIFY TRIPPED**, RECIRC DRIVE 2A NORMAL FEEDER, 2-HS-57-17. ☐
   3. **VERIFY TRIPPED**, RECIRC DRIVE 2A ALTERNATE FEEDER, 2-HS-57-15. ☐
   4. **CLOSE** RECIRC PUMP 2A SUCTION VALVE, 2-HS-68-1. ☐

Continued on Next Page

(b) The flow keeps number 1 seal cavity clean and cool by flowing out of the seal area, along the pump shaft, and into the recirculation system.

(c) This purge flow reduces the possibility of seal damage due to foreign material entering the seal from an unclean piping system.

(8) Seal Failures

Obj. V.B.8
Obj. V.C.3
TP-6

(a) Seal failure may be assessed by the resulting changes in flows and pressures.

(b) Failure of the number 1 seal assembly would allow a higher flow to the number 2 seal cavity, forcing the number 2 seal to operate at a higher pressure (i.e., greater than 500 psig).

ARPs provide useful info/analysis.
Obj. V.D.2c
Obj. V.E.3c

(c) This failure of the number 1 seal will cause leakage through the controlled seal leak-off line to rise to approximately 1.1 gpm. A flow element in this line causes a common alarm on high flow at 0.9 gpm or on low flow at 0.5 gpm.

(d) Failure of the number 2 seal assembly would cause its seal pressure to drop (depending upon the magnitude of the failure).

(i) This failure would also cause a higher leakage through the seal leak detection line downstream from the number 2 seal.

(ii) Normally there is no flow through this line and flow switches are set to alarm at 0.1-0.2 gpm flow.

(e) Failure of both mechanical seals would result in a total seal assembly leakage of 60 gpm as limited by the seal breakdown bushings.

Would cause elevated drywell temp and pressure, and would exceed Tech Spec and EPIP limits for RCS leakage.

(f) Should the number 1 seal restricting orifice become plugged, the RECIRC PUMP A(B) NO. 1 SEAL LEAKAGE ABN annunciator will alarm on low flow (less than or equal to 0.5 gpm). Additionally, a reduction in number 2 seal pressure would be seen.

(g) Should the number 2 restricting orifice become plugged, the RECIRC PUMP A(B) NO. 1 SEAL LEAKAGE ABN annunciator would also alarm on low flow; however, number 2 seal pressure would rise to near the pressure of number 1 seal.

(9) Seal Cooling

Obj. V.B.9

(a) Cooling for the recirculation pump seals is required due to the heat generated by the friction of the sealing surfaces and the leakage of reactor water through the seal assembly.

Obj. V.C.3
Obj. V.B.19c

(b) This cooling is provided by a combination of supplied Reactor Building Closed Cooling Water (RBCCW) and the leakage of primary coolant past the seals.

51. RO 295025EK2.08 001/C/A/T1G1/EHC LOGIC//295025EK2.08//RO/SRO/BANK

Unit 2 has experienced an inadvertent Main Steam Isolation Valve (MSIV) closure and subsequent reactor scram. Consequently, RCIC was placed in level control and is also maintaining reactor pressure 900 to 1000 psig with the MSIVs still isolated.

Given these plant conditions, the digital Electro-Hydraulic Control (EHC) System is in ____(1)____ Pressure Control mode with the pressure setpoint set at ___(2)___ psig.

	(1)	(2)
A.	Reactor;	700
B.✔	Header;	700
C.	Reactor;	970
D.	Header;	970

K/A Statement:

295025 High Reactor Pressure / 3
EK2.08 - Knowledge of the interrelations between HIGH REACTOR PRESSURE and the following: Reactor/turbine pressure regulating system: Plant-Specific

K/A Justification: This question satisfies the K/A statement by requiring the candidate to use specific plant conditions to determine the response of the digital EHC system to a transient resulting in a high reactor pressure.

References: OPL171.228

Level of Knowledge Justification: This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

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REFERENCE PROVIDED: None

Plausibility Analysis:

In order to answer this question correctly the candidate must determine the following:

1. Whether the header pressure dropped sufficiently low enough to cause an automatic transfer to Header Pressure Control.
2. Whether current plant conditions have allowed EHC logic to automatically transfer back to Reactor Pressure Control.

B - correct:

A - incorrect: This is plausible because EHC automatically transfers back to Header pressure control if HEADER pressure returns above 725 psig. However, the MSIVs are still closed so the given reactor pressure is not being sensed by the header pressure instruments.

C - incorrect: This is plausible because this condition is typical for post-scram EHC conditions if the MSIVs are open.

D - incorrect: This is plausible because EHC will swap to Header pressure control, but the setpoint will drop from 970 psig to 700 psig.

INSTRUCTOR NOTES

2. During turbine start-up and for a brief time following synchronization, the bypass valve control also maintains the reactor steam pressure. Once all the bypass valves are closed, then the turbine control maintains reactor steam pressure either in Header Pressure or Reactor Pressure Control depending on which operating mode is selected.

   Monitor Plant parameters for expected response

3. Steam pressure control is selectable from either panel 9-7 or the EHC Workstation by selecting HEADER PRESSURE CONTROL or REACTOR PRESSURE CONTROL.

   Obj.V.B.9.a

4. Header Pressure Control Input Signal

   a. Two redundant pressure transmitters sense header pressure at the main steam throttle just upstream of the main turbine stop valves.

      Powered from within the EHC system

   b. Both signals are monitored for low, high, difference, and hardware failures.

   c. The higher of the two signals when no failures are detected is selected as the input.

   d. A maximum difference setpoint of 10-PSI is also established to detect a fault and/or transmitter drift from either of the inputs.

   e. In the event a fault is detected, the channel is prohibited from being used in the signal processing and the appropriate BYPASS pushbutton light will illuminate on 9-7 and on the HMI operator interface.

      Obj.V.B.9.c

   f. Once the failed signal is corrected, depressing the BYPASS pushbutton will reset the BYPASS logic and both input signals will then be processed.

INSTRUCTOR NOTES

g. This mode IS NOT single failure proof - one of the two pressure sensors failing upscale can, and generally will be selected by the logic to control. This will open the TCV's and BPV's to depressurize the header to the MSIV isolation setpoint of 852 psig in RUN Mode.

h. In the unlikely event that both inputs signals are detected as failed, the control logic will automatically switch to reactor pressure control.

i. If header pressure drops below 700-PSI, and reactor pressure control is the controlling mode of operation, the control logic will automatically transfer to header pressure control. If desired, the operator may re-select reactor pressure control after the transfer has been made even though header pressure is below 700-psi. The automatic transfer logic will re-engage if header pressure rises above 725-psi.

5. Reactor Pressure Control Input Signal — TP-3

a. Four (4) redundant pressure transmitters (PT- 204a-d) grouped in pairs with "A" and "B" constituting one pair and "C" and "D" the other pair.

b. A pressure-biasing algorithm determines the lagged high-median value of the four (4) inputs and biases the remaining three (3) input signals to that high median value.

c. The high-median signal is then averaged with the other three signals and is used as "Actual Rx Pressure". — Four biased signals are averaged.

52. RO 295026EA2.01 001/C/A/T1/G1///295026EA2.01//RO/SRO/BANK

Given the following plant conditions:

- Unit 2 is in a transient condition with current conditions as follows.

  - Suppression pool level: 13.5 feet
  - Reactor pressure: 900 psig
  - Suppression pool temperature: 105°F

Which ONE of the following describes the required action?

REFERENCE PROVIDED

A.✓ Operate ALL available Suppression Pool Cooling.

B. Rapidly depressurize via the Main Turbine Bypass Valves.

C. Emergency Depressurize the RPV by opening ALL six ADS Valves.

D. Lower Reactor Pressure to stay within the Safe Area of the Heat Capacity Temperature Limit Curve and maintain cooldown rate below 100°F/hr.

K/A Statement:

295026 Suppression Pool High Water Temp. / 5
EA2.01 - Ability to determine and/or interpret the following as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE: Suppression pool water temperature

K/A Justification: This question satisfies the K/A statement by requiring the candidate to correctly identify an adverse condition related to Suppression Pool High Temperature and then determine the action required to correct the adverse condition.

Reference: 2-EOI-2 Flowchart, EOIPM Section 0-V-D Page 85

Level of Knowledge Justification: This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to solve a problem. This requires mentally using this knowledge and its meaning to resolve the problem.

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REFERENCE PROVIDED: HCTL Curve only

Plausibility Analysis:

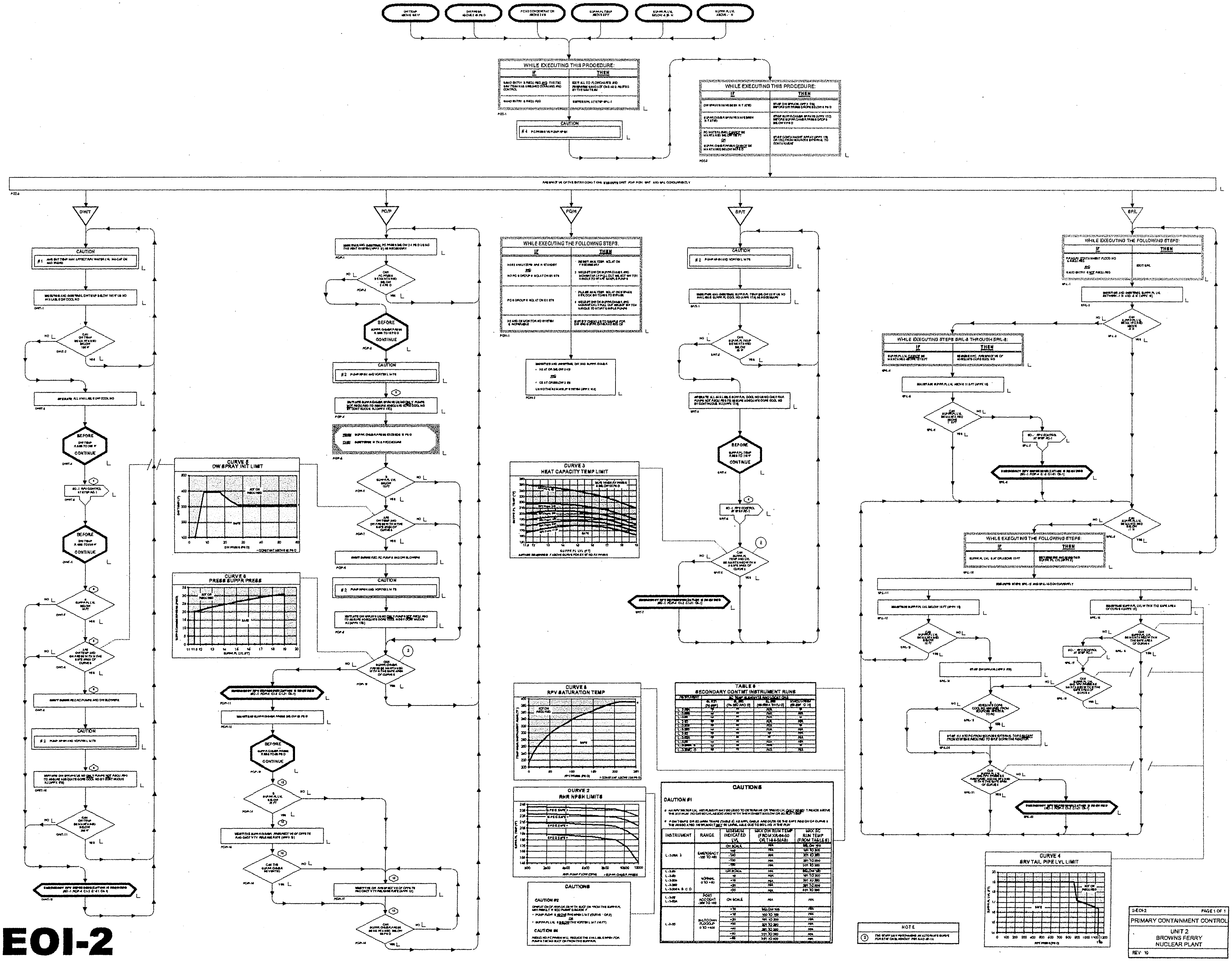
A - correct:

B - incorrect: ED would not be anticipated under these conditions unless the candidate focuses on Suppression Pool (Torus) water level which is approaching the limit of 11.5 feet for Emergency Depressurization. If this is the case, other actions on SP/L (Suppression Pool / Level leg) take priority over ED.

C - incorrect: No condition has been met requiring Emergency Depressurization at this temperature. It is plausible if the candidate focuses on the SP level, which is approaching the limit of 11.5 feet for Emergency Depressurization. If this is the case, other actions on SP/L take priority over ED.

D - incorrect: It is plausible if the candidate continues down the SP/T leg of EOI-2 and determines that exceeding the HCTL is possible. Since EOI-1 is used to lower pressure and cooldown, and no EOI-1 entry condition has been met, it is unacceptable to assume that exceeding the HCTL is possible.

2-EOI-2 PRIMARY CONTAINMENT CONTROL 2-EOI-2



2-EOI-2 PAGE 1 OF 1

PRIMARY CONTAINMENT CONTROL

UNIT 2

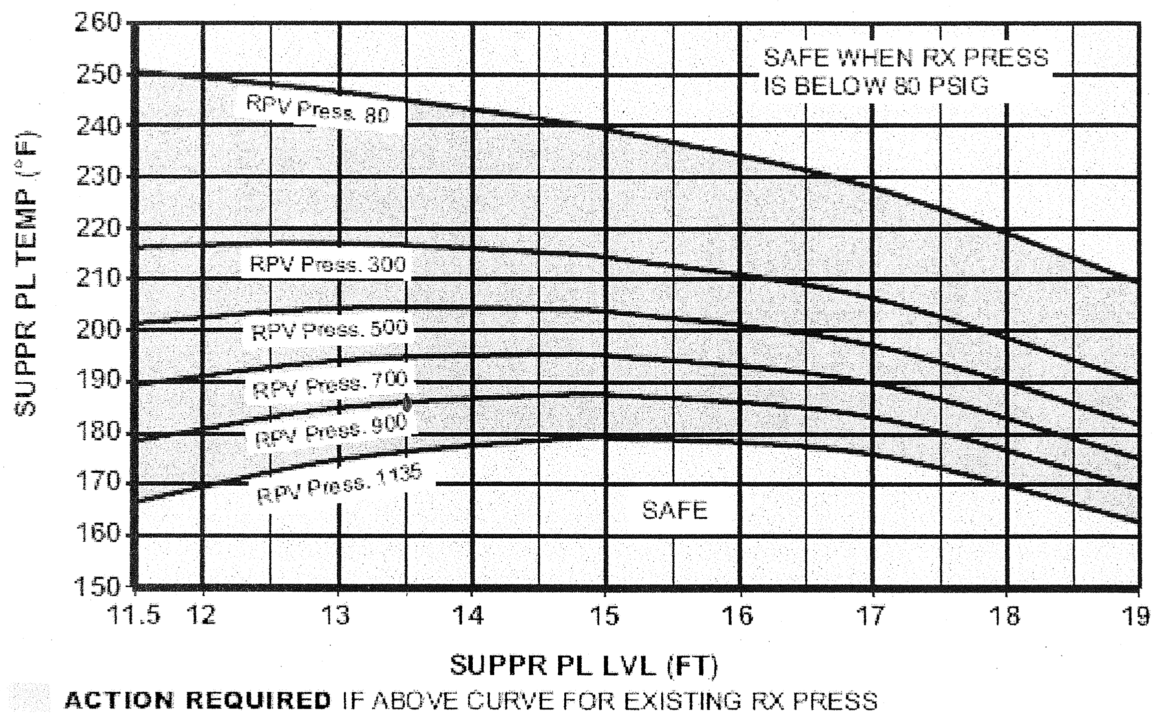
BROWNS FERRY NUCLEAR PLANT

REV: 10

EOI-2

EXAMINATION REFERENCE PROVIDED TO CANDIDATE

CURVE 3
HEAT CAPACITY TEMP LIMIT



53. RO 295028EK3.04 001/C/A/T1G1/480VLS/B5/295028EK3.04//RO/SRO/BANK

Given the following plant conditions:

- A Loss of Off-site power has occurred in conjunction with a LOCA on Unit 2.
- Plant conditions are as follows:

- Reactor Water Level	(+) 20 inches, steady
- Average Drywell Temperature	230°F, rising
- Suppression Chamber Pressure	11 psig, rising
- Emergency Diesel Generators	Tied and loaded to 4 KV Sd Bds
- Reactor pressure	Remains greater than 800 psig

Which ONE of the following describes the final status of Unit 2 Drywell cooling and Reactor Building Closed Cooling Water (RBCCW)?

A.✓ Drywell coolers are operating and RBCCW is available.

B. Drywell coolers are operating; but, RBCCW is NOTavailable.

C. Drywell coolers must be manually restarted and RBCCW is available.

D. Drywell coolers must be manually restarted; but, RBCCW is NOT available.

K/A Statement:

295028 High Drywell Temperature / 5
EK3.04 - Knowledge of the reasons for the following responses as they apply to HIGH DRYWELL TEMPERATURE : Increased drywell cooling

K/A Justification: This question satisfies the K/A statement by requiring the candidate to use specific plant conditions to determine the status of drywell cooling following a transient which results in high drywell temperature.

References:

Level of Knowledge Justification: This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

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REFERENCE PROVIDED: None

Plausibility Analysis:

In order to answer this question correctly the candidate must determine the following:

1. Whether a Common Accident Signal (CAS) 480V Load Shed has been initiated based on the given conditions.
2. The status of RBCCW and DW Blowers based on the answer to Item #1 above.

A - correct:

B - incorrect: This is plausible because the Drywell Blowers would be operating. However, RBCCW does not receive a trip signal because a 480V Load Shed signal has NOT yet been initiated. RPV level and pressure are too high.

C - incorrect: This is plausible because following a 480V Load Shed, the Drywell Blowers, on the accident unit, MUST be manually started and RBCCW would be available. However, a 480V Load Shed signal has NOT yet been initiated. RPV level and pressure are too high.

D - incorrect: This is plausible because following a 480V Load Shed, the Drywell Blowers, on the accident unit, MUST be manually started. However, RBCCW does NOT receive a trip signal because a 480V Load Shed signal has NOT yet been initiated.

X. Lesson Body

A. The 480V Load Shedding Logic System removes selected loads from 480V boards which are powered from the 4kV Shutdown Boards — Obj. V.B.1/V.D.1; TP-1, 2

1. The load shedding is initiated by an accident signal on Unit 1 or 2 with **a diesel generator supplying one 4kV Shutdown Board as its only source of power** — Obj. V.B.3/ V.D.3; Obj. V.C.2

AND

2. The accident signal is generated in the Core Spray System logic — TP-3; Obj. V.B.2./V.D.2; Obj. V.C.1

   a. Low-low-low reactor water level (-122"/Level 1)

   OR — CASA signal

   b. High drywell pressure (2.45 psig) with low reactor pressure (450 psig)

   c. For load shed signal on U1 or U2, the accident is for either unit

   d. Unit 3 accident signal won't cause Unit 1 or 2 load shed or vice versa

3. The signal representing "diesel generator supplying a 4KV shutdown board" is called "DGVA" — TP-4

4. For DGVA logic to be satisfied, both conditions must be present:

   a. The DG output breaker or the U2 tie breaker to U3 being closed

   b. The normal and alternate feeder breaker must be open

5. All Unit 1-2 DGVA contacts are in parallel

6. Any D/G tied to its Shutdown Board with an accident signal present will initiate U1-2 load shed logic

3.0 AUTOMATIC ACTIONS (continued)

V. Unit 1/2 480V Load Shed occurs on a loss of offsite power in conjunction with a LOCA signal:

1. One RBCCW pump auto restarts (after 40 seconds on U1 and U2).

2. Drywell Blowers auto restart on non-accident unit (after 40 seconds). Drywell Blowers with their respective auto restart inhibit switches in the INHIBIT position will not auto restart.

3. Drywell coolers are manually restarted on the accident unit. A Drywell Blower with its auto restart inhibit switch in the INHIBIT position can be manually restarted after a ten minute time delay.

4. SGT TRAINS A & B trip, but will AUTO RESTART in 40 seconds when an initiation signal is present.

5. Loss of Control Bay Chilled Water Pumps A & B. (may be restarted after 10 minutes with use of bypass switch).

W. Unit 3 480V load shedding occurs as follows:

1. Division I 480V load shedding will occur when an accident signal is present and diesel generator voltage is available on the 4160V shutdown board supplying the 480V shutdown board 3A as follows:

   a. RBCCW pump 3A trips

   b. Drywell blowers 3A1 & 3A2 trip

   c. After a 40 second time delay, with the control switch in Normal After Start, RBCCW pump 3A restarts

   d. After a 40 second time delay, Drywell blowers 3A1 and 3A2 can be manually restarted

   e. Drywell blowers 3A3, 3A4 and 3A5 cannot be restarted until the load shed signal is corrected

54. RO 295030EA1.06 001/C/A/T1G1/3.5/3.5//295030EA1.06//RO/SRO/BANK

Given the following plant conditions:

- A LOCA has caused gross fuel failure on Unit 3.
- The Site Emergency Director (SED) / SRO has approved implementation of EOI Appendix 18, "Suppression Pool Water Inventory Removal and Makeup."
- The control room crew has just CLOSED the RHR RADWASTE SYS FLUSH VALVE, 3-FCV-74-63.
- Suppression Pool level is (-)3.5 inches and steady.

Which ONE of the following describes the next appropriate action(s)?

A. Verify OPEN 3-FCV-73-40, HPCI CST SUCTION VALVE, and OPEN 3-FCV-73-30, HPCI PUMP MIN FLOW VALVE.

B. OPEN 3-FCV-74-63, RHR RADWASTE SYS FLUSH VALVE, and direct Suppression Pool water to Radwaste ONLY.

C. OPEN 3-FCV-74-62, RHR MAIN CNDR FLUSH VALVE, and direct Suppression Pool water to the Main Condenser ONLY.

D✓ Inform the SRO that Appendix 18, "Suppression Pool Water Inventory Removal and Makeup" is COMPLETE and that Suppression Pool level is acceptable.

K/A Statement:

295030 Low Suppression Pool Water Level / 5
EA1.06 - Ability to operate and/or monitor the following as they apply to LOW SUPPRESSION POOL WATER LEVEL: Condensate storage and transfer (make-up to the suppression pool): Plant-Specific

K/A Justification: This question satisfies the K/A statement by requiring the candidate to use specific plant conditions to determine the effectiveness of actions to control Suppression Pool level using the Condensate storage and transfer system.

References: 2-EOI Appendix 18

Level of Knowledge Justification: This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

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REFERENCE PROVIDED: None

Plausibility Analysis:

In order to answer this question correctly the candidate must determine the following:

1. Which actions are required based on the given conditions.

NOTE: Each distractor is plausible because they are all actions directed by 2-EOI Appendix 18 "Suppression Pool Water Inventory Removal and Makeup" to control Suppression Pool (Torus) water level.

D - correct:

A - incorrect: The given Suppression Pool (Torus) water level is sufficiently high enough that additional inventory makeup is NOT necessary.

B - incorrect: The given Suppression Pool (Torus) water level is sufficiently low enough that additional inventory removal is NOT necessary. However, following a gross fuel failure, rejecting water to Radwaste is more appropriate than to the main condenser.

C - incorrect: The given Suppression Pool (Torus) water level is sufficiently low enough that additional inventory removal is NOTnecessary. In addition, following a gross fuel failure, rejecting water to the main condenser is also NOT appropriate.

3-EOI APPENDIX-18

SUPPRESSION POOL WATER INVENTORY REMOVAL AND MAKEUP

LOCATION: Unit 3 Control Room

ATTACHMENTS: None (√)

****

CAUTION

[NRC/C] Suppression Pool water will be highly radioactive after a LOCA. Chemical Engineering recommendations are used to determine location to pump contaminated water. [NRC Inspection Report 89-16]

****

NOTE: All panel operations performed at Control Room Panel 3-9-3 unless otherwise stated.

1. IF ..... Suppression Pool Water makeup is required,
   THEN ... **CONTINUE** in this procedure at Step 5. ____

2. IF ..... Gross fuel failure is suspected,
   THEN ... **OBTAIN** SED/SRO permission to pump down Suppression Pool BEFORE continuing in this procedure. ____

3. IF ..... Directed by SRO,
   THEN ... **REMOVE** water from Suppression Pool as follows:

   a. **DISPATCH** personnel to perform the following (Unit 3 RB, El 519 ft, Torus Area):

      1) **VERIFY OPEN** 3-SHV-074-0786A(B), RHR DR PUMP A(B) DISCH SHUTOFF VALVE. ____

      2) **OPEN** the following valves:

         - 3-SHV-074-0564A(B), RHR DR PUMP A(B) SEAL WTR SPLY ____
         - 3-SHV-074-0529A(B), RHR DR PUMP A(B) SHUTOFF VLV. ____

      3) **UNLOCK** and **OPEN** 3-SHV-074-0765A(B), RHR DR PUMP A(B) DISCH. ____

      4) **NOTIFY** Unit Operator that RHR Drain Pump 3A(3B) is lined up to remove water from Suppression Pool. ____

      5) **REMAIN** at torus area UNTIL Unit 3 Operator directs starting of RHR Drain Pump 3A(3B). ____

3. (continued from previous page)

   b. IF..... Main Condenser is desired drain path,
      THEN ... **OPEN** 3-FCV-74-62, RHR MAIN CNDR FLUSH VALVE. ____

   c. IF..... Radwaste is desired drain path,
      THEN ... **PERFORM** the following:

      1) **ESTABLISH** communications with Radwaste. ____

      2) **OPEN** 3-FCV-74-63, RHR RADWASTE SYS FLUSH VALVE. ____

   d. **NOTIFY** personnel in Unit 3 RB, El 519 ft, Torus Area to start RHR Drain Pump 3A(3B). ____

   e. **THROTTLE** 3-FCV-74-108, RHR DR PUMP 3A/B DISCH HDR VALVE, as necessary. ____

4. WHEN ... Suppression Pool level reaches -5.5 in.,
   THEN ... **SECURE** RHR Drain System as follows:

   a. **DISPATCH** personnel to **STOP** the Drain System as follows (Unit 3 RB, El 519 ft, Torus Area):

      1) **STOP** RHR Drain Pump 3A(3B). ____

      2) **CLOSE** the following valves:

         - 3-SHV-074-0564A(B), RHR DR PUMP A(B) SEAL WTR SPLY ____
         - 3-SHV-074-0529A(B), RHR DR PUMP A(B) SHUTOFF VLV. ____

      3) **CLOSE** and **LOCK** 3-SHV-074-0765A(B), RHR DR PUMP A(B) DISCH. ____

   b. **CLOSE** 3-FCV-74-108, RHR DR PUMP 3A/B DISCH HDR VALVE. ____

   c. **VERIFY CLOSED** 3-FCV-74-62, RHR MAIN CNDR FLUSH VALVE. ____

   d. **VERIFY CLOSED** 3-FCV-74-63, RHR RADWASTE SYS FLUSH VALVE. ____

   e. WHEN ... Suppression Pool level can be maintained between -1 in. and -5.5 in.,
      THEN ... **EXIT** this procedure. ____

5. IF ..... Directed by SRO to Emergency Makeup to the Suppression Pool from Standby Coolant,
THEN ... **CONTINUE** in this procedure at Step 9. ____

6. IF ..... Directed by SRO to add water to suppression pool,
THEN ... **MAKEUP** water to Suppression Pool as follows:

   a. **VERIFY OPEN** 3-FCV-73-40, HPCI CST SUCTION VALVE. ____

   b. **OPEN** 3-FCV-73-30, HPCI PUMP MIN FLOW VALVE. ____

   c. IF ..... HPCI is NOT available for Suppression Pool makeup,
   THEN ... **MAKEUP** water to Suppression Pool using RCIC as follows:

      1) **VERIFY OPEN** 3-FCV-71-19, RCIC CST SUCTION VALVE. ____

      2) **OPEN** 3-FCV-71-34, RCIC PUMP MIN FLOW VALVE. ____

   d. IF ..... 3-FCV-71-34, RCIC PUMP MIN FLOW VALVE, CANNOT be opened from control room,
   THEN ... **DISPATCH** personnel to 250V DC RMOV Board 3B, Compartment 5D, to perform the following:

      1) **PLACE** 3-XS-071-0034, RCIC PUMP MIN FLOW VALVE EMER TRANS SWITCH, to EMERG. ____

      2) **OPEN** 3-FCV-71-34, RCIC PUMP MIN FLOW VALVE. ____

7. WHEN ... Suppression Pool level reaches -5.5 in.,
THEN ... **VERIFY CLOSED** the following valves:

   - 3-FCV-73-30, HPCI PUMP MIN FLOW VALVE ____
   - 3-FCV-71-34, RCIC PUMP MIN FLOW VALVE. ____

8. **DISPATCH** personnel to 250V DC RMOV Board 3B, Compartment 5D, to **VERIFY** 3-XS-071-0034, RCIC PUMP MIN FLOW VALVE EMER TRANS SWITCH, in NORMAL. ____

55. RO 295031G2.4.6 001/C/A/T1G1/C1//295031G2.4.6//RO/SRO/BANK

Given the following plant conditions:

- Unit 2 was operating at 98% power when an automatic scram occurs due to a Group I Isolation.
- ALL control rods fully insert as reactor water level immediately drops below Level 2.
- BOTH Recirc Pumps trip.
- HPCI automatically initiates but immediately isolates due to a ruptured inner turbine exhaust rupture diaphragm.
- RCIC is currently tagged out of service to repair an oil leak.
- ALL other systems are operable.
- EOI-1, RPV Control, is entered.
- Pressure control was established with Safety Relief Valves (SRVs).

The remaining high pressure injection systems are unable to maintain reactor water level which is currently at (-)150 inches and lowering.

Which ONE of the following contingency procedures would be appropriate to execute?

A.✔ C1, "Alternate Level Control."

B. C2, "Emergency RPV Depressurization."

C. C4, "RPV Flooding."

D. C5, "Level/Power Control."

K/A Statement:

295031 Reactor Low Water Level / 2
2.4.6 - Emergency Procedures / Plan Knowledge symptom based EOP mitigation strategies

K/A Justification: This question satisfies the K/A statement by requiring the candidate to use specific plant conditions to determine the appropriate Emergency Procedure used to mitigate a low reactor water level condition.

References: 2-EOI-1, EOIPM Sections 0-V-C and 0-V-G

Level of Knowledge Justification: This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

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REFERENCE PROVIDED: None

Plausibility Analysis:

In order to answer this question correctly the candidate must determine the following:

1. Whether the given conditions are indicative of a loss of HP injection.
2. Based on Item #1 above, which EOI Contingency is appropriate to mitigate that condition.

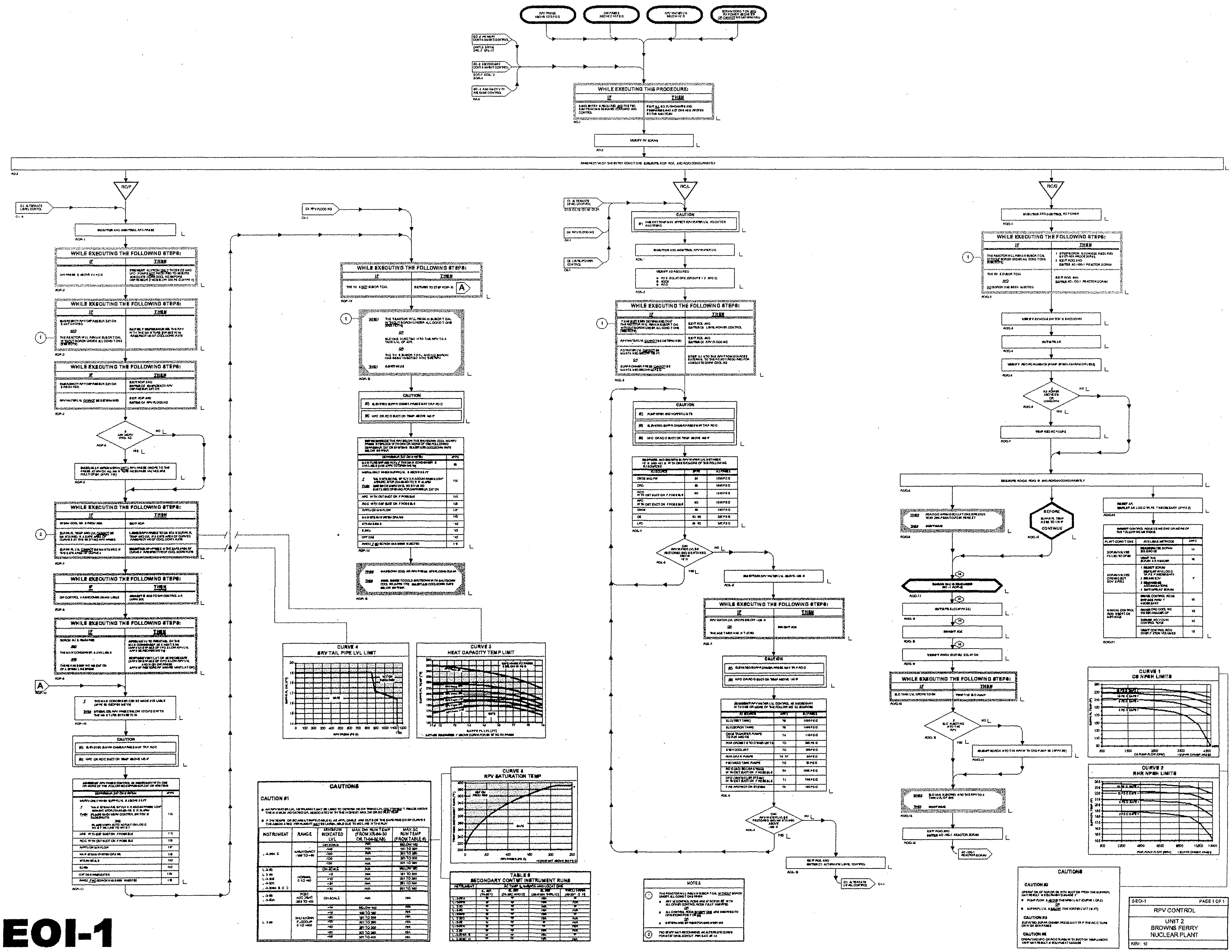
A - correct:

B - incorrect: This is plausible since Emergency Depressurization (ED) will eventually become necessary following the initial actions of EOI-C1. However, additional actions are required before EOI-C2 is appropriate.

C - incorrect: This is plausible since Drywell temperature may be high enough, following Emergency Depressurization, to create environmental conditions where RPV level instruments become inaccurate / unavailable. However, additional actions are required before EOI-C4 is appropriate.

D - incorrect: This is plausible since the ONLY given condition which contradicts the use of EOI-C5 is the current rod pattern. However, with ALL rods inserted, EOI-C5 is NOT appropriate.

2-EOI-1 RPV CONTROL 2-EOI-1



EOI-1

56. RO 295037EK2.11 001/C/A/T1G1/RMCS//295037EK2.11//RO/SRO/BANK

A hydraulic ATWS has occurred on Unit 2 and the Operator At-The-Controls is inserting control rods in accordance with the following:

- EOI Appendix 1D, "Insert Control Rods Using Reactor Manual Control System,"
- EOI Appendix 1F, "Manual Scram," and
- EOI Appendix 2, "Defeating ARI Logic Trips."

During execution of these appendices, ____________________________________.

A.✔ ALL potential Control Rod Insert Block signals are bypassed.

B. Rod Drift indication is received as soon as rod motion begins.

C. Stabilizing Valves are OPEN to provide increased drive water pressure.

D. ALL Reactor Manual Control System (RMCS) timer functions are bypassed except for the Settle Bus Timer.

K/A Statement:

295037 SCRAM Condition Present and Power Above APRM Downscale or Unknown/1
EK2.11 - Knowledge of the interrelations between SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN and the following: RMCS: Plant-Specific

K/A Justification: This question satisfies the K/A statement by requiring the candidate to use specific plant conditions to determine the status of the RMCS while executing procedures to mitigate an ATWS condition.

References: 2-EOI Appendicies 1D, 1F, and 2

Level of Knowledge Justification: This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

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REFERENCE PROVIDED: None

Plausibility Analysis:

In order to answer this question correctly the candidate must determine the following:

1. What affect the actions performed by EOI Appendix implementation have on the RMCS system.
2. What affect the RMCS manipulations required by implementation of the EOI Appendices have on plant indications.

A - correct:

B - incorrect: This is plausible since a Rod Drift indication will occur for each inserted control rod. However, the indication does NOT occur until the rod is fully inserted and the CRD NOTCH OVERRIDE switch is released.

C - incorrect: This is plausible because CRD Stabilizing Valves do have an effect on drive water pressure, but the efffect is to prevent oscillations while moving control rods, NOT increase pressure.

D - incorrect: This is plausible since RMCS timers are bypassed by using the CRD NOTCH OVERRIDE switch in accordance with EOI Appendix 1D. However, the Settle Bus Timer is also bypassed.

2-EOI APPENDIX-1D

INSERT CONTROL RODS USING REACTOR MANUAL CONTROL SYSTEM

LOCATION: Unit 2 Control Room, Panel 9-5

ATTACHMENTS: 1. Tools and Equipment
2. Core Position Map

(√)

NOTE: This EOI Appendix may be executed concurrently with EOI Appendix 1A or 1B at SRO's discretion when time and manpower permit.

1. **VERIFY** at least one CRD pump in service. ____

NOTE: Closing 2-85-586, CHARGING WATER ISOL, valve may reduce the effectiveness of EOI Appendix 1A or 1B.

2. IF ..... Reactor Scram or ARI CANNOT be reset,
THEN ... **DISPATCH** personnel to close 2-SHV-85-586, CHARGING WATER SHUTOFF (RB NE, El 565 ft). ____

3. **VERIFY** REACTOR MODE SWITCH in SHUTDOWN. ____

4. **BYPASS** Rod Worth Minimizer. ____

5. **REFER TO** Attachment 2 and **INSERT** control rods in the area of highest power as follows:

   a. **SELECT** control rod. ____

   b. **PLACE** CRD NOTCH OVERRIDE switch in EMERG ROD IN position UNTIL control rod is NOT moving inward. ____

   c. **REPEAT** Steps 5.a and 5.b for each control rod to be inserted. ____

NOTE: A ladder may be required to perform the following step. REFER TO Tools and Equipment, Attachment 1.

IF necessary, an alternate ladder is available at the HCU Modules, EAST and West banks. It is stored by the CRD Charging Cart.

6. WHEN ... NO further control rod movement is possible or desired,
THEN ... **DISPATCH** personnel to verify open 2-SHV-85-586, CHARGING WATER SHUTOFF (RB NE, El 565 ft). ____

END OF TEXT

2-EOI APPENDIX-1F

MANUAL SCRAM

LOCATION: Unit 2 Control Room

ATTACHMENTS: 1. Tools and Equipment
2. Panel 2-9-15, Rear
3. Panel 2-9-17, Rear

(√)

1. **VERIFY** Reactor Scram and ARI reset. ____

   a. IF ..... ARI CANNOT be reset,
   THEN ... **EXECUTE** EOI Appendix 2 concurrently with Step 1.b of this procedure. ____

   b. IF ..... Reactor Scram CANNOT be reset,
   THEN ... **DISPATCH** personnel to Unit 2 Auxiliary Instrument Room to defeat ALL RPS logic trips as follows:

      1) **REFER** to Attachment 1 and **OBTAIN** four 3-ft banana jack jumpers from EOI Equipment Storage Box. ____

      2) **REFER** to Attachment 2 and **JUMPER** the following relay terminals in Panel 2-9-15, Rear:

         a) Relay 5A-K10A (DQ) Terminal 2 to Relay 5A-K12E (ED) Terminal 4, Bay 1. ____

         b) Relay 5A-K10C (AT) Terminal 2 to Relay 5A-K12G (BH) Terminal 4, Bay 3. ____

      3) **REFER** to Attachment 3 and **JUMPER** the following relay terminals in Panel 2-9-17, Rear:

         a) Relay 5A-K10B (DQ) Terminal 2 to Relay 5A-K12F (ED) Terminal 4, Bay 1. ____

         b) Relay 5A-K10D (AT) Terminal 2 to Relay 5A-K12H (BH) Terminal 4, Bay 3. ____

2. WHEN ... RPS Logic has been defeated,
THEN ... **RESET** Reactor Scram. ____

3. **VERIFY OPEN** Scram Discharge Volume vent and drain valves. ____

57. RO 295038EK1.01 001/MEM/T1G1/NEW//295038EK1.01//RO/SRO/MODIFIED 11/17/07

Given the following plant conditions:

- Unit 2 has experienced a LOCA with a loss of Primary Containment.
- You have volunteered for a team, dispatched from the Operations Support Center (OSC), that will enter the Reactor Building and attempt to energize '2D' 480V RMOV board.
- Due to environmental and radiological conditions present in the Reactor Building, Radiation Protection personnel provide each team member with a Sodium Chloride (NaCl) and Potassium Iodide (KI) tablet during the briefing.

Which ONE of the following describes the benefit of ingesting Potassium Iodide prior to the Reactor Building entry?

Potassium Iodide will reduce the ______________________________________.

A. risk of dehydration and heat stress.

B. absorption of radioactive Iodine in the lungs.

C✓ absorption of radioactive Iodine by the thyroid.

D. absorption of radioactive Potassium in the blood stream.

K/A Statement:

295038 High Off-site Release Rate / 9
EK1.01 - Knowledge of the operational implications of the following concepts as they apply to HIGH OFF-SITE RELEASE RATE : Biological effects of radioisotope ingestion

K/A Justification: This question satisfies the K/A statement by requiring the candidate to correctly identify the pathway and adverse effect of iodine ingestion.

Reference: EPIP -14 Revision 18, page 4

Level of Knowledge Justification: This question is rated as MEM due to the requirement to recall or recognize discrete bits of information.

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MODIFIED FROM 295038EK1.01 #1

REFERENCE PROVIDED: None

Plausibility Analysis:

C - correct:

A - incorrect. The sodium chloride tablets would be used for this purpose. It is plausible if the candidate is unsure of the purpose of KI tablets.

B - incorrect: Only the thyroid is the organ at risk, but it is plausible if the candidate assumes that airborne ingestion is limited to absorption by the lungs.

D - incorrect: Iodine is the element that is absorbed. Potassium becomes a plausible answer due to recent media coverage regarding health risks related to low potassium levels in the blood stream.

3.6 Issuing Potassium Iodide (KI)

3.6.1 If the TSC RP Manager has reason to believe that a person's projected cumulative dose to the thyroid from inhalation of radioactive iodine might exceed 10 rems (see Appendix A), the exposed person should be started immediately on a dose regimen of KI. This decision shall be immediately communicated to the SED.

3.6.1.1 If the TSC is not staffed or the RP Manager position has not been filled, then the senior onsite RP Supervisor has the authority to issue KI utilizing the bases described in step 3.6.1.

3.6.1.2 The initial dose of KI should be not delayed since thyroid blockage requires 30 to 60 minutes. Anyone authorized to initiate KI shall be familiar with the Food and Drug Administration (FDA) patient package insert and be sure that each recipient is similarly informed.

3.6.1.3 Prior to issuing KI to an individual, the person should be asked if he/she is allergic to iodine. If the person indicates a possible sensitivity to iodine they should not be issued KI.

3.6.2 KI is stored in the plant RP supply cage and the REP Van instrument kits.

3.6.3 RP normally will not dispense a container or package of KI to TVA Personnel involved in activities to support a radiological emergency. RP will however dispense a single individual dose of KI to team members dispatched from the OSC.

3.6.4 Follow the dosage outlined on the FDA patient package insert (Appendix B). A copy of the FDA approved patient package insert shall accompany the issuance of KI. If KI is distributed in individual doses then verbal instructions of the significant information on the patient package insert by a knowledgeable individual is sufficient.

3.6.5 Complete the KI Issue Report (Appendix C) or document on an RWP time sheet as appropriate for issuance of KI. If the RWP time sheet is used to document distribution of the KI, note the time of KI distribution on the back of the time sheet.

58. RO 600000AA1.08 001/MEM/T1G1/RSW//600000AA1.08//RO/SRO/NEW 11/20/07 RMS

Which ONE of the following describes the appropriate fire extinguishing agent for the specific class of fire?

A. Water used on Class 'B' fires.

B.✓ Low pressure CO_2 used on Class 'C' fires.

C. Dry Chemical (PKP) used on Class 'C' fires.

D. Aqueous Film Forming Foam (AFFF) used on Class 'A' fires.

K/A Statement:

600000 Plant Fire On-site / 8
AA1.08 - Ability to operate and / or monitor the following as they apply to PLANT FIRE ON SITE: Fire fighting equipment used on each class of fire

K/A Justification: This question satisfies the K/A statement by requiring the candidate to identify the correct fire fighting agent for a specific class of fire.

References: TVA Safety Manual

Level of Knowledge Justification: This question is rated as MEM due to the requirement to recall or recognize discrete bits of information.

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REFERENCE PROVIDED: None

Plausibility Analysis:

In order to answer this question correctly the candidate must determine the following:

1. Which flammable material is of concern based on Fire Class A, B and C.
2. Which extinguishing agent is appropriate for each class of fire.
3. Which extinguishing agent is inappropriate for a given class of fire.

B - correct:

A - incorrect: Class "B" fires are flammable liquids. Using water could cause serious damage by allowing the liquid to splatter and spread.

C - incorrect: Dry chemical agents are extremely corrosive to electrical components and insulation typical of Class "B" electrical fires.

D - incorrect: AFFF is designed as a flooding and diluting agent for Class "B" flammable liquid fires. Application on a Class "A" fire is not effective in extinguishing flammable materials such as wood and paper.

59. RO 295009AK2.01 001/C/A/T1G2/PR.INSTR/13/295009AK2.01/9619/RO/SRO/NEW 10/16/07

Given the following Unit 1 plant conditions:

- Due to multiple high pressure injection system failures, 1-EOI-C1, "Alternate Level Control" has been entered.
- RHR Pump '1A' is running and lined up for LPCI injection.
- Core Spray Pumps '1B' and '1D' are running and lined up for injection.
- Drywell Temperature is 240 °F and rising slowly.

Which ONE of the following conditions describes the appropriate point where Emergency Depressurization may be performed in accordance with 1-EOI-C1, "Alternate Level Control?"

Post Accident Flooding Range Level Instrument, 3-LI-3-52, is reading __(1)__ with reactor pressure at ___(2)___.

REFERENCES PROVIDED

	(1)	(2)
A.	(-)205 inches;	350 psig
B.	(-)210 inches;	500 psig
C.	(-)220 inches;	900 psig
D.✔	(-)225 inches;	800 psig

K/A Statement:

295009 Low Reactor Water Level / 2
AK2.01 - Knowledge of the interrelations between LOW REACTOR WATER LEVEL and the following: Reactor water level indication

K/A Justification: This question satisfies the K/A statement by requiring the candidate to use specific plant conditions to determine actual reactor water level under conditions of low reactor water level.

References: 1-EOI-C1 Flowchart, PIP-95-64 Rev 12

Level of Knowledge Justification: This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

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REFERENCE PROVIDED: 1-EOI-C1 Flowchart, PIP-95-64 Rev. 12

Plausibility Analysis:

In order to answer this question correctly the candidate must determine the following:

1. Recognize the requirement that RPV level must be less than -162 inches before Emergency Depressurization is appropriate.
2. Recognize that the indicated RPV level must be corrected for pressure using PIP-95-64.
3. Recognize that two or more injection systems must be lined up with pumps running to meet the requirement to Emergency Depressurize.
4. Recognize that only one RHR pump is required to qualify as an injection subsystem since each RHR pump is rated for 100% capacity.

NOTE: Each distractor is plausible because the conditions specified are possible given the current plant conditions.

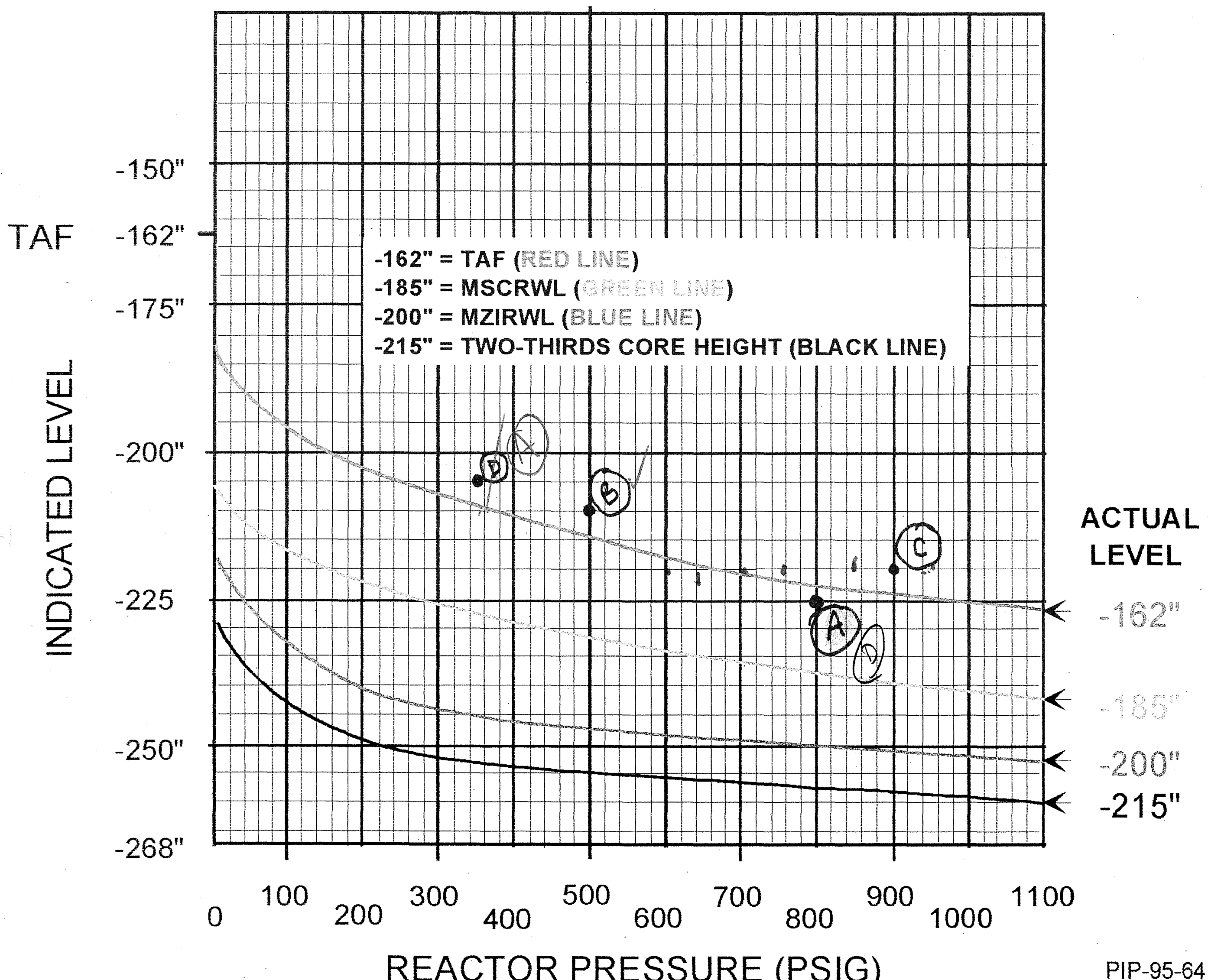
D - correct:

A - incorrect: Level is 5 inches too high or pressure is 100 psig too high.

B - incorrect: Level is ~4 inches too high or pressure is 240 psig too high.

C - incorrect: Level is ~4 inches too high or pressure is 100 psig too high.

3-LI-3-52 & 62 CORRECTION CURVES



PIP-95-64
REV. 12

INSTRUCTOR NOTES

Since no trips or alarms are associated with this range, this level signal is not directed through the Analog Trip System.

(d) One MCR indicator on Panel 9-3 monitors this range of level indication.

(4) Post-accident Flood Range

(a) -268" to +32" range covering active core area and overlapping the lower portion of the Normal Control Range.

(b) Referenced to instrument zero

(c) Intended for use only under accident conditions with reactor at 0 psig and recirculation pumps tripped.

(d) Variable leg tap is from diffuser of jet pumps 1 and 6 (or 11 and 16).

(e) Per Safety Analysis on water level instruments the conclusion is that the accident range instruments adequately indicate water level--provided they are corrected for off-calibration conditions of RPV pressure utilizing the operator aid on Panel 9-3 for level correction.

Injecting with RHR LI-3-52 and 62 (Accident Range) Technical Support letter dated 9/13/95 (See LP Folder) Use Conservative Decision Making Obj. V.B.15. Obj. V.B.11.

(f) An interlock associated with this range will prevent using the RHR System for containment de pressurization when it is needed to flood the core region.

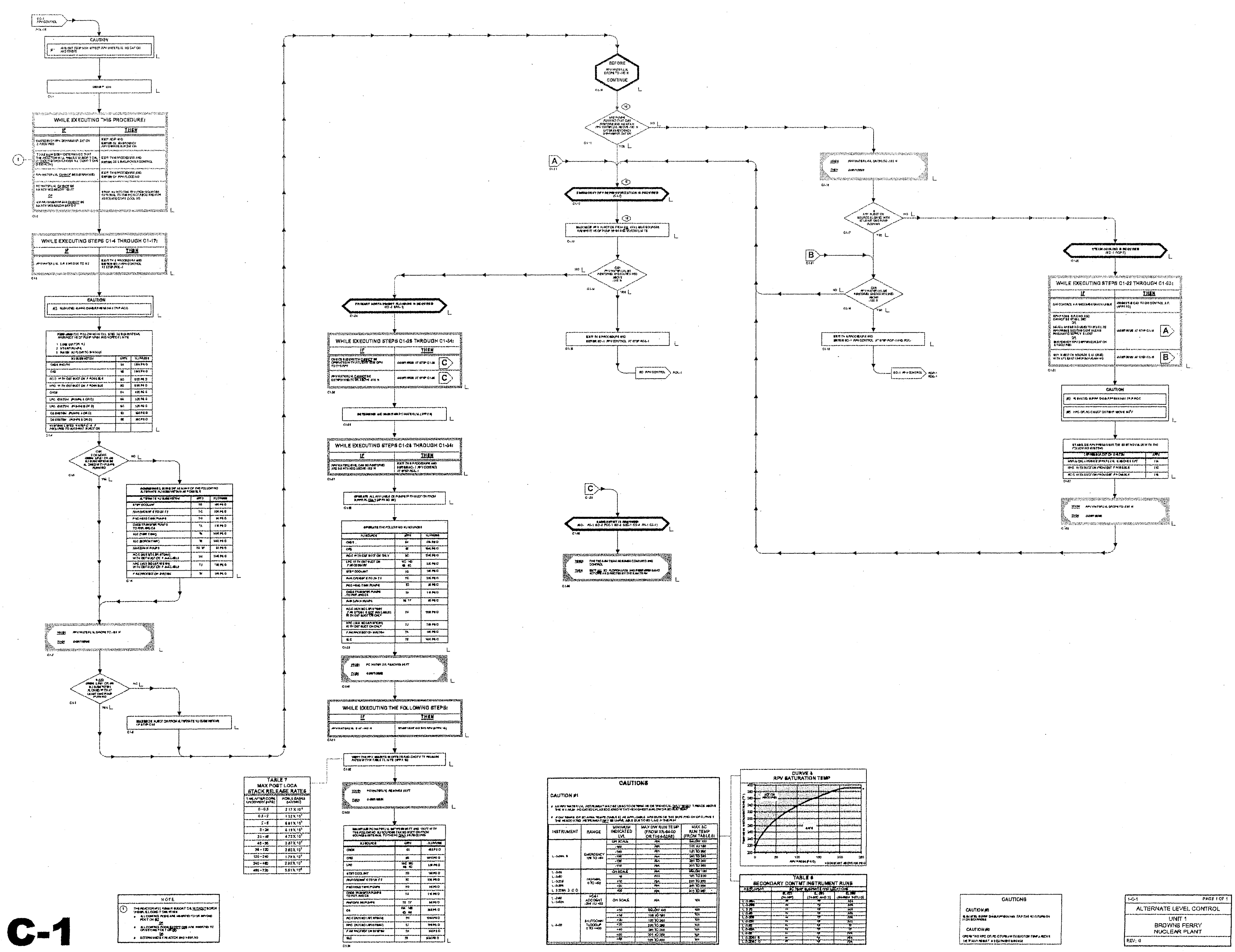
(g) The -68" to -168" portion of this range is recorded in the MCR on 2-LI-3-62 Recorder and two indicators monitor the full range of these instruments.

Unit 3 Recorder displays a scale of +32" to -268"

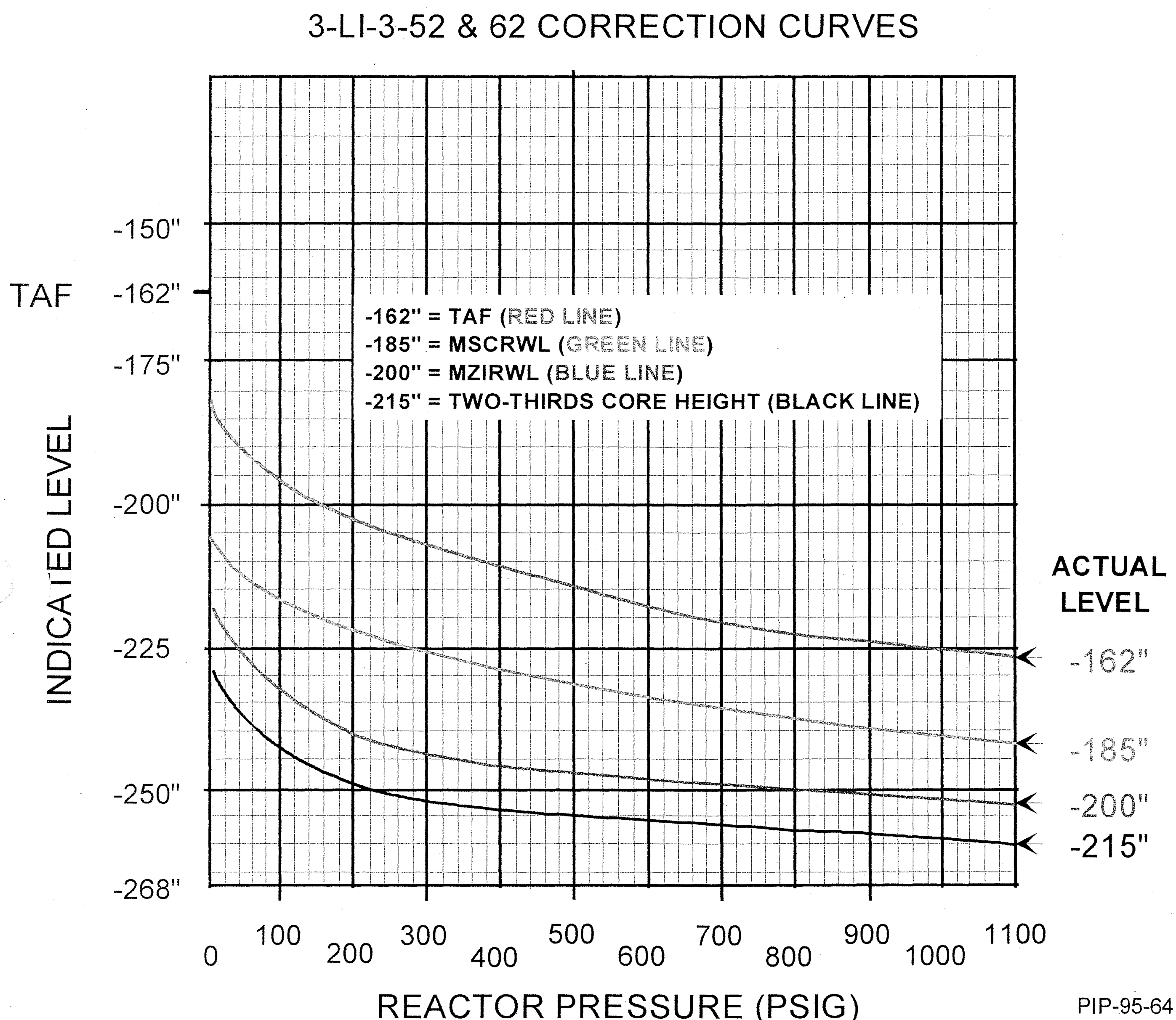
EXAMINATION REFERENCE PROVIDED TO CANDIDATE

ALTERNATE LEVEL CONTROL

1-C-1



1-C-1	PAGE 1 OF 1
ALTERNATE LEVEL CONTROL	
UNIT 1 BROWNS FERRY NUCLEAR PLANT	
REV: 0	



PIP-95-64
REV. 12

60. RO 295012G2.2.22 001/C/A/T1G2/64/12/295012G2.2.22//RO/SRO/BANK

Given the following plant conditions:

- You are the oncoming Unit 3 Unit Supervisor.
- During turnover, the onshift Unit Supervisor informs you that two (2) Drywell Coolers had been secured during his shift while performing ground isolation on '3C' 480V RMOV board.
- Drywell Average Temperature is 152°F and stable.

Which ONE of the following describes the appropriate condition and required action?

A.✓ Exceeded 3-SR-2, "Instrument Checks and Observations", Drywell temperature limit. Address Tech Spec section 3.6.

B. Exceeded the normal operating Drywell temperature limit. Drywell temperature must be logged hourly until below the limit.

C. Exceeded the normal operating Drywell temperature limit. Restore Drywell average air temperature below the limit in 24 hours.

D. Exceeded 3-EOI-2, "Primary Containment Control" entry condition. Enter and execute 3-EOI-2, Primary Containment Control.

K/A Statement:

295012 High Drywell Temperature / 5
2.2.22 - Equipment Control - Knowledge of limiting conditions for operations and safety limits.

K/A Justification: This question satisfies the K/A statement by requiring the candidate to use specific plant conditions to determine that Technical Specification limits have been exceeded.

References: Unit 3 Tech Specs Section 3.6.1.4

Level of Knowledge Justification: This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

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REFERENCE PROVIDED: 1-EOI-C1 Flowchart, PIP-95-64 Rev 12.

Plausibility Analysis:

In order to answer this question correctly the candidate must determine the following:

1. The appropriate entry condition for U3 Tech Spec Section 3.6.1.4.
2. The appropriate entry condition for 3-EOI-2, Primary Containment Control.
3. The appropriate action based on the given condition.

A - correct:

B - incorrect: This is plausible because the Tech Spec limt was exceeded. However, the required action is to restore the Drywell Temperature within the limit in 8 hours. There is NO requirement for hourly logging of DW temperature.

C - incorrect: This is plausible because the Tech Spec limt was exceeded. However, the required action is to restore the Drywell Temperature within the limit in 8 hours. The 24 hour limit is based on performing the surveillance on Drywell Temperature.

D - incorrect: This is plausible because the entry condition for 3-EOI-2 is only 8 °F above the given temperature. However, the entry condition has NOT been met and DW temperature was reported as "stable".

3.6 CONTAINMENT SYSTEMS

3.6.1.4 Drywell Air Temperature

LCO 3.6.1.4 Drywell average air temperature shall be ≤ 150°F.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Drywell average air temperature not within limit.	A.1 Restore drywell average air temperature to within limit.	8 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.4.1	Verify drywell average air temperature is within limit.	24 hours

61. RO 295015AK1.02 001/MEM/T1G2/BASIS//295015AK1.02///BANK

Which ONE of the following describes why the Automatic Depressurization System (ADS) is inhibited once Standby Liquid Control injection has begun in accordance with EOI-1, "RPV Control" path RC/Q?

A. The operator can control pressure better than an automatic system like ADS.

B. ADS actuation would impose a severe pressure and temperature transient on the reactor vessel.

C.✔ Severe core damage from a large power excursion could result, if ~~low pressure systems automatically injected on depressurization~~.

D. If only steam driven high pressure injection systems are available an ADS actuation could lead to a loss of adequate core cooling.

K/A Statement:

295015 Incomplete SCRAM / 1

AK1.02 - Knowledge of the operational implications of the following concepts as they apply to INCOMPLETE SCRAM : (CFR 41.8 to 41.10) Cooldown effects on reactor power

K/A Justification: This question satisfies the K/A statement by requiring the candidate to use specific plant conditions to determine the effect of a significant cooldown when an incomplete scram has occurred.

References:

Level of Knowledge Justification: This question is rated as MEM due to the requirement to recall or recognize discrete bits of information.

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REFERENCE PROVIDED: None

Plausibility Analysis:

In order to answer this question correctly the candidate must determine the following:

1. The basis for inhibiting ADS under the specific conditions of boron injection.

NOTE: Each of the three distractors are plausible based on their relationship to the bases for inhibiting ADS under circumstances OTHER than boron injection. Specifically, Alternate RPV Level Control actions. Refer to the attached excerpt from EOIPM Section 0-V-G.

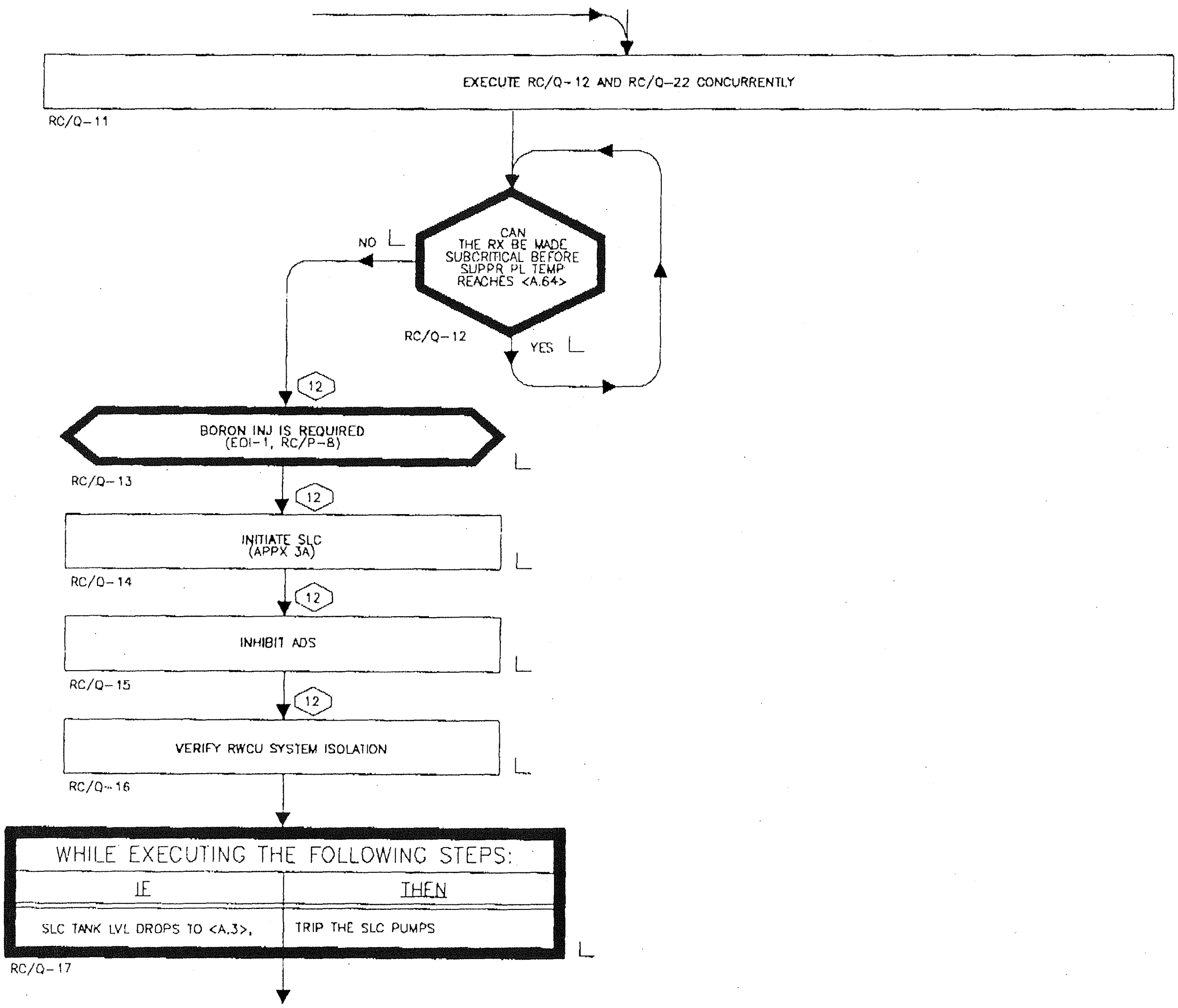
C - correct:

A - incorrect: This statement is true, but is NOT addressed in the basis for boron injection.

B - incorrect: This applies whenever ADS actuates, but is ONLY the **precursor** to the issue related to boron injection.

D - incorrect: This statement applies particularly to a low RPV level condition.

STEP: RC/Q-14 and RC/Q-15



034

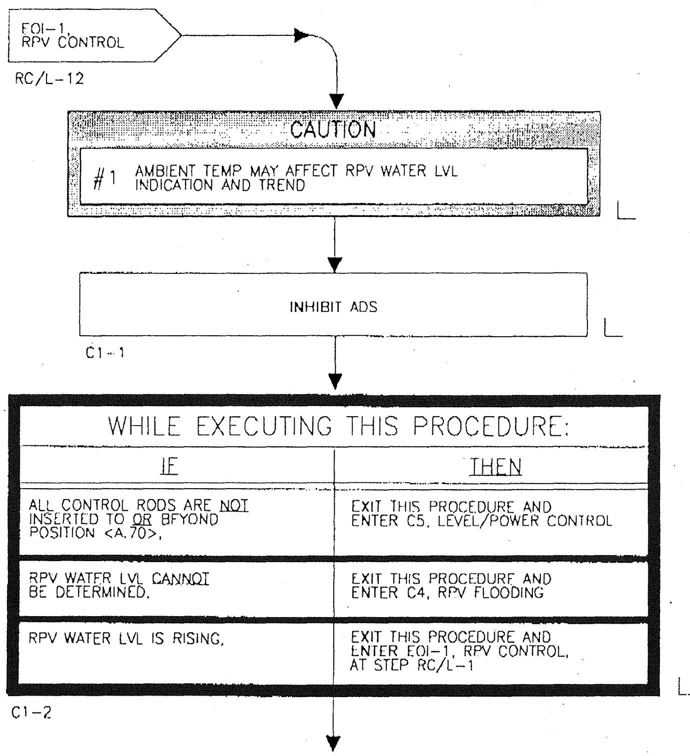
DISCUSSION: STEP RC/Q-14 and RC/Q-15

The **RC/Q-14** action step directs the operator to manually initiate the SLC System. Because this step is prioritized with the miniature before decision step RC/Q-12 symbol, this action should be performed before suppression pool temperature reaches <A.64>, Boron Injection Initiation Temperature. EOI Appendix 3A provides step-by-step guidance for manual initiation of the SLC System. Boron in solution absorbs neutrons, providing negative reactivity to achieve reactor subcriticality, since the reactor is not yet subcritical on control rod insertion alone.

The **RC/Q-15** action step directs the operator to defeat automatic ADS function by placing the ADS inhibit switches in the inhibit position. Because this step is prioritized with the miniature before decision step RC/Q-12 symbol, this action should be performed before suppression pool temperature reaches <A.64>, Boron Injection Initiation Temperature.

ADS initiation may result in the injection of large amounts of relatively cold, unborated water from low pressure injection systems. With the reactor still critical or subcritical on boron, the positive reactivity addition due to boron dilution and temperature reduction from injection of cold water may result in a reactor power excursion large enough to cause substantial core damage. Defeating ADS is, therefore, appropriate whenever boron injection is required. If emergency depressurization of the RPV is subsequently required, explicit direction is provided in the appropriate EOI. Therefore, the ability to maintain automatic initiation capability of ADS is not required.

STEP: C1-1


EOI-1, RPV CONTROL
RC/L-12
CAUTION
#1 AMBIENT TEMP MAY AFFECT RPV WATER LVL INDICATION AND TREND
INHIBIT ADS
C1-1
WHILE EXECUTING THIS PROCEDURE:
IF
THEN
ALL CONTROL RODS ARE NOT INSERTED TO OR BEYOND POSITION <A.70>,
EXIT THIS PROCEDURE AND ENTER C5, LEVEL/POWER CONTROL
RPV WATER LVL CANNOT BE DETERMINED,
EXIT THIS PROCEDURE AND ENTER C4, RPV FLOODING
RPV WATER LVL IS RISING,
EXIT THIS PROCEDURE AND ENTER EOI-1, RPV CONTROL, AT STEP RC/L-1
C1-2

DISCUSSION: STEP C1-1

This action step directs the operator to defeat automatic ADS function. An ADS actuation with the RPV at pressure imposes a severe thermal transient on the RPV and may significantly complicate efforts to restore and maintain RPV water level as specified in this procedure.

Because ADS initiation logic receives limited input signals, a variety of plant conditions may exist where automatic depressurization of the RPV is not appropriate. In certain cases (e.g., RCIC available but LPCI/CS injection valves closed and control power for their operation not available) ADS actuation may directly lead to loss of adequate core cooling and core damage, conditions that might otherwise have been avoided. Further, conditions assumed in the design of ADS actuation logic (e.g., no operator action for ten minutes) do not exist when actions specified in this procedure are being carried out.

Finally, an operator can draw on much more plant information than is available to ADS logic (e.g., equipment out of service for maintenance, operating experience with certain systems, probability of restoration of offsite power, etc.) and thus can better judge, based on logic specified in this procedure, when and how to depressurize the RPV. For all of these reasons, it is appropriate to prevent automatic initiation of ADS as specified.

62. RO 295020AK3.08 001/MEM/EOI/BASIS//295020AK3.08///BANK

Unit 2 was at 100% rated power when a spurious Group I Isolation occurred. The pressure transient caused a small-break LOCA to occur inside the Drywell.

Which ONE of the following describes the basis for actions before AND after reaching 12 psig Suppression Chamber Pressure?

A. Drywell sprays must be initiated prior to this pressure to prevent opening the Suppression Chamber to Reactor Building vacuum breakers and de-inerting the containment.

B.✔ Above this pressure indicates that almost all of the nitrogen and other non-condensible gases in the drywell have been transferred to the torus and chugging is possible.

C. Above this pressure indicates that almost all of the nitrogen and other non-condensible gases in the torus have been transferred to the drywell air space and Suppression Chamber Sprays will be ineffective.

D. Above this pressure indicates that almost all of the nitrogen and other non-condensible gases in the drywell have been transferred to the torus so initiating Drywell Sprays will not result in containment failure.

K/A Statement:

295020 Inadvertent Cont. Isolation / 5 & 7
AK3.08 - Knowledge of the reasons for the following responses as they apply to INADVERTENT CONTAINMENT ISOLATION: Suppression chamber pressure response

K/A Justification: This question satisfies the K/A statement by requiring the candidate to use specific plant conditions to determine the effect on Suppression Chamber pressure due to an inadvertent containment isolation and the basis for that response.

References: EOIPM Section 0-V-D

Level of Knowledge Justification: This question is rated as MEM due to the requirement to recall or recognize discrete bits of information.

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REFERENCE PROVIDED: None

Plausibility Analysis:

In order to answer this question correctly the candidate must determine the following:

1. The basis for the Pressure Supression Pressure Limit of 12 psig Suppression Chamber pressure.

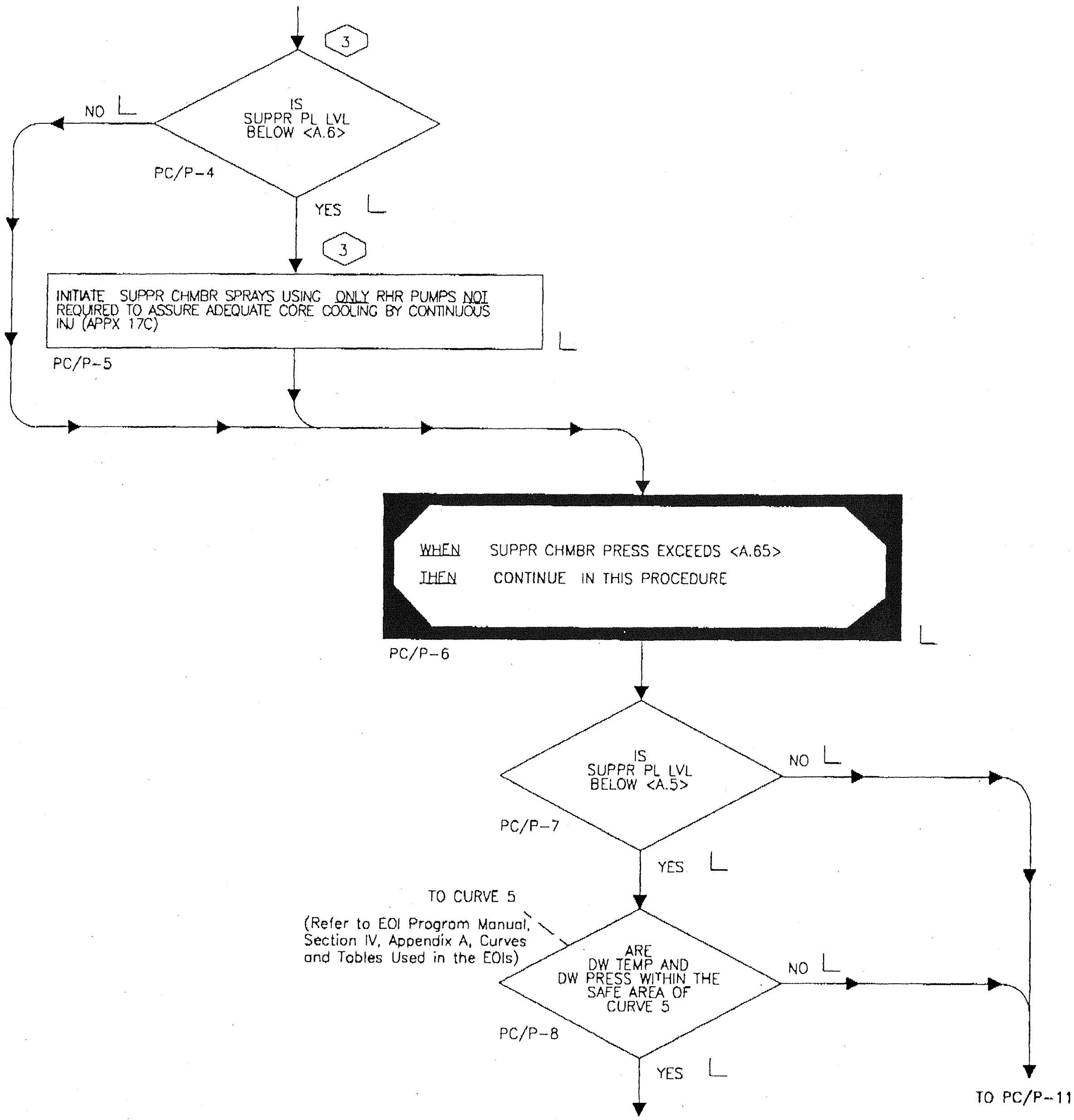
B - correct:

A - incorrect: This is plausible because initiation of DW sprays at high SC pressure could reduce pressure low enough to open the Suppression Chamber to Reactor Building Vacuum Breakers. However, this is part of the bases for the Drywell Spray Initiation Pressure Limit Curve #5.

C - incorrect: This is plausible if the LOCA occurred inside the Suppression Chamber and NOT the Drywell as given in the stem.

D - incorrect: This is plausible because initiating SC sprays with high temperature non-condensible gases in the SC will result in evaporative cooling and a rapid pressure drop. However, the SC to DW vacuum relief system is capable of compensating for this pressure drop. This is also part of the bases for the Drywell Spray Initiation Pressure Limit Curve #5.

STEP: PC/P-6


3
IS SUPPR PL LVL BELOW <A.6>
NO
YES
PC/P-4
3
INITIATE SUPPR CHMBR SPRAYS USING ONLY RHR PUMPS NOT REQUIRED TO ASSURE ADEQUATE CORE COOLING BY CONTINUOUS INJ (APPX 17C)
PC/P-5
WHEN SUPPR CHMBR PRESS EXCEEDS <A.65>
THEN CONTINUE IN THIS PROCEDURE
PC/P-6
IS SUPPR PL LVL BELOW <A.5>
NO
YES
PC/P-7
TO CURVE 5
(Refer to EOI Program Manual, Section IV, Appendix A, Curves and Tables Used in the EOIs)
ARE DW TEMP AND DW PRESS WITHIN THE SAFE AREA OF CURVE 5
NO
YES
PC/P-8
TO PC/P-11

DISCUSSION: STEP PC/P-6

This contingent action step requires the operator to wait until the stated condition has been met before continuing in EOI-2. Performance of subsequent actions in this section of EOI-2 will not be performed until suppression chamber pressure exceeds Suppression Chamber Spray Initiation Pressure.

Engineering calculations have determined that if suppression chamber pressure exceeds <A.65>, Suppression Chamber Spray Initiation Pressure, there is no assurance that chugging will be prevented at downcomer openings of the drywell vents. This value is rounded off in the EOI to use the closest, most conservative value that can be accurately determined on available instrumentation.

Suppression Chamber Spray Initiation Pressure is defined to be the lowest suppression chamber pressure that can occur when 95% of noncondensables in the drywell have been transferred to airspace of the suppression chamber. Scale model tests have demonstrated that chugging will not occur so long as the drywell atmosphere contains at least 1% noncondensables. To prevent the occurrence of conditions under which chugging may happen, Suppression Chamber Spray Initiation Pressure is conservatively defined by specifying 5% noncondensables.

Chugging is the cyclic condensation of steam at downcomer openings of the drywell vents. Chugging occurs when steam bubbles collapse at the exit of downcomers. The rush of water that fills the void (some of which is drawn up into the downcomer pipe) induces a severe stress at the junction of the downcomer and vent header. Repeated application of this stress can cause these joints to experience fatigue failure (cracks), thereby creating a pathway that bypasses the pressure suppression function of primary containment. Subsequent steam that discharges through downcomers would then exit through the fatigued cracks, and directly pressurize suppression chamber air space, rather than discharging to and condensing in the suppression pool.

Although operation of suppression chamber sprays by itself will not prevent chugging, the requirement to wait to initiate drywell sprays until reaching Suppression Chamber Spray Initiation Pressure assures that suppression chamber spray operation is attempted before operation of drywell sprays. Therefore, actions to initiate drywell sprays need to be directed only if suppression chamber sprays were unable to reduce primary containment pressure or they could not be initiated.

DISCUSSION: STEP PC/P-8

This decision step has the operator evaluate the present status of drywell pressure and drywell temperature to determine if conditions are favorable for drywell spray operation.

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

Evaporative cooling occurs when water is sprayed into a superheated atmosphere. Water at the surface of each droplet is heated and flashes to steam, absorbing heat energy from the drywell atmosphere until the atmosphere reaches saturated conditions. In the drywell, with a typical drywell spray flowrate, the evaporative cooling process results in an immediate, rapid, large reduction in pressure. This pressure reduction occurs at a rate much faster than can be compensated for by the primary containment vacuum relief system. Unrestricted operation of drywell sprays could cause an excessive negative differential pressure to occur between the drywell and suppression chamber, large enough to cause a loss of primary containment integrity.

Convective cooling occurs when water is sprayed into a saturated atmosphere. Sprayed water droplets absorb heat from the surrounding atmosphere through convective heat transfer (sensible heat from the atmosphere is transferred to the water droplets). This effect reduces drywell ambient temperature and pressure until equilibrium conditions are established. The convective cooling process occurs at a rate much slower than the evaporative cooling process. An operator can effectively control the magnitude of a containment temperature/pressure reduction from convective cooling by terminating operation of drywell sprays.

Considering the pressure drop concerns described above, engineering calculations have determined that primary containment integrity is assured when drywell sprays are operated in the safe area of Drywell Spray Initiation Limit Curve (Curve 5). Drywell Spray Initiation Limit is defined to be the highest drywell temperature at which initiation of drywell sprays will not result in an evaporative cooling pressure drop to below either: 1) drywell-below-suppression chamber differential pressure capability, or 2) high drywell pressure scram setpoint.

If drywell temperature and pressure are within the safe area of Curve 5, the operator continues at Step PC/P-9.

If drywell temperature and pressure are not within the safe area of Curve 5, then drywell spray operation is not permitted, and the operator is directed to Step PC/P-11.

63. RO 295032EA1.01 001/C/A/T1G1/EOI-3//295032EA1.01//RO/SRO/MODIFIED 11/17/07

Given the following plant conditions:

- Unit 2 experiences a Main Steam Line break at full power.
- Both Inboard and Outboard Main Steam Isolation Valves (MSIVs) on the 'B' steam line fail to isolate. However, the reactor scrams and ALL rods fully insert.
- Steam Leak Detection Panel 9-21 indications are as follows:
  - 2-TI-1-60A 320 °F
  - 2-TI-1-60B 323 °F
  - 2-TI-1-60C 337 °F
  - 2-TI-1-60D 318 °F
- NO other temperature indications are alarming at this time.

Which ONE of the following describes the appropriate operator actions and the reason for those actions?

REFERENCE PROVIDED

A. Emergency depressurize the reactor due to two (2) areas being above Max Safe in EOI-3, "Secondary Containment Control."

B.✓ Enter 2-EOI-1, "RPV Control" and initiate a Reactor Scram due to one (1) area being above Max Safe in EOI-3, "Secondary Containment Control."

C. Rapidly depressurize the reactor due to one (1) area above Max Safe and one (1) area approaching Max Safe in EOI-3, "Secondary Containment Control."

D. Enter 2-GOI-100-12A, "Unit Shutdown" and commence a normal shutdown / cooldown due to a primary system discharging outside Primary Containment.

K/A Statement:

295032 High Secondary Containment Area Temperature / 5
EA1.01 - Ability to operate and/or monitor the following as they apply to HIGH SECONDARY CONTAINMENT AREA TEMPERATURE : Area temperature monitoring system

K/A Justification: This question satisfies the K/A statement by requiring the candidate to use specific plant conditions to determine the required actions which result from high secondary containment temperatures as indicated by Area Temperature Monitoring instrumentation.

References: 2-EOI-3 Flowchart, EOIPM Section 0-V-E

Level of Knowledge Justification: This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

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MODIFIED FROM OPL171.204 #20
REFERENCE PROVIDED: 2-EOI-3 Flowchart

Plausibility Analysis:

In order to answer this question correctly the candidate must determine the following:

1. Which area(s) are above or approaching Max Safe
2. Based on Item #1 above, determine the appropriate action and the basis for that action.

B - correct:

A - incorrect: This is plausible because ALL four temperatures provided are greater than 315°F as indicated on Table 3. However, ONLY one indicator applies to an EOI-3 area, therefore ONLY one area is above Max Safe.

C - incorrect: This is plausible because one area is above Max Safe and given conditions indicate an unisolable leak exists; which implies conditions are degrading. However, with NO other temerature indications in alarm, anticipating the requirement to Emergency Depressurize is NOT ppropriate.

D - incorrect: This is plausible because ALL four temperatures provided are greater than 315°F as indicated on Table 3. However, ONLY one indicator applies to an EOI-3 area, therefore ONLY one area is above Max Safe. In addition, this step is ONLY addressed if Emergency Depressurization will NOT reduce the discharge into Secondary Containment. In this case, it would reduce the discharge.

DISCUSSION: ENTRY CONDITIONS: EOI-3

Entry conditions for this procedure are symptomatic of conditions which, if not corrected, could degrade into an emergency. Adverse affects on equipment operability and conditions that directly challenge secondary containment integrity were specifically considered in the selection of these entry conditions. Following is a description of each entry condition:

Area temperature above the maximum normal operating value of Table 3

A secondary containment area temperature above the maximum normal operating value of Table 3, Secondary Containment Area Temperature, is an indication that steam from a primary system may be discharging into secondary containment. As temperatures continue to increase, continued operability of equipment needed to carry out EOI actions may be compromised. High area temperatures also present a danger to personnel since access to secondary containment may be required by actions specified by EOIs.

Maximum normal operating temperature is defined to be the highest value of a secondary containment area temperature expected to occur during normal plant operating conditions with all directly associated support and control systems functioning properly.

Differential pressure at or above <A.38> inches of water

High secondary containment differential pressure is indicative of a potential loss of secondary containment structural integrity, and could result in uncontrolled release of radioactivity to the environment.

Reactor Zone Ventilation exhaust radiation level above <A.39>

High Reactor Zone Ventilation exhaust radiation levels may indicate that radioactivity is being released to the environment when the system should have automatically isolated.

Refuel Zone Ventilation exhaust radiation level above <A.40>

High Refuel Zone Ventilation exhaust radiation levels may indicate radioactivity is being released to the environment when the system should have automatically isolated.

Floor drain sump water level above <A.41>

A secondary containment floor drain sump water level above maximum normal operating level is an indication that steam or water may be discharging into secondary containment.

Maximum normal operating floor drain sump water level is defined to be the highest value of secondary containment floor drain sump water level expected to occur during normal plant operating conditions with all directly associated support and control systems functioning properly.

Area water level above <A.42>

Secondary containment area water level above maximum normal operating level is an indication that steam or water may be discharging into secondary containment.

Maximum normal operating secondary containment area water level is defined to be the highest value of secondary containment area water level expected to occur during normal plant operating conditions with all directly associated support and control systems functioning properly.

DISCUSSION: SC/T-6 and SC/T-7

Step SC/T-6 is a before decision step that has the operator evaluate current and future efforts to lower secondary containment area temperatures, in relation to the current value and trend of secondary containment area temperatures, to determine if a reactor scram is necessary. The before decision step requires that this determination and subsequent actions be performed before any secondary containment area temperature reaches its respective maximum safe operating temperature value provided in Table 3.

Maximum safe operating temperature is defined to be the highest temperature at which neither: 1) equipment necessary for the safe shutdown of the plant will fail, nor 2) personnel access necessary for safe shutdown of the plant will be prevented. The maximum safe operating temperature value for all secondary containment areas is provided in Table 3, Secondary Containment Area Temperature.

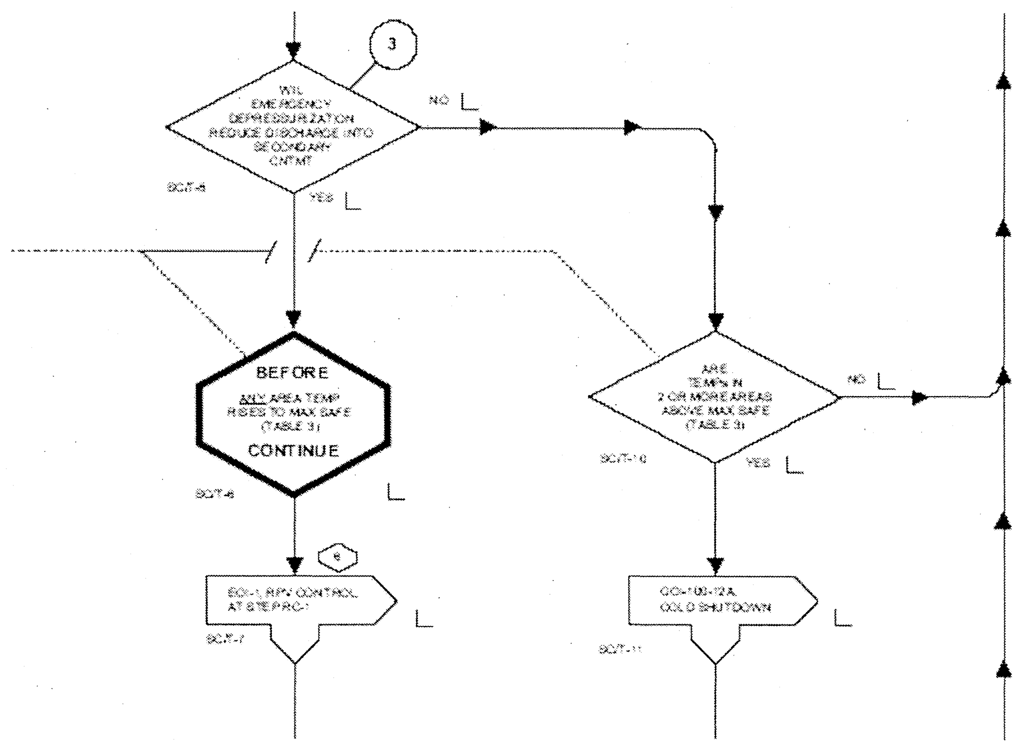
This step is reached only when additional actions have been required to reverse an increasing secondary containment area temperature trend. If all secondary containment area temperatures can be maintained below their respective maximum safe operating values, the operator returns to Step SC/T-1. If it is determined that all secondary containment area temperatures cannot be maintained below their respective maximum safe operating values, the operator continues at Step SC/T-7.

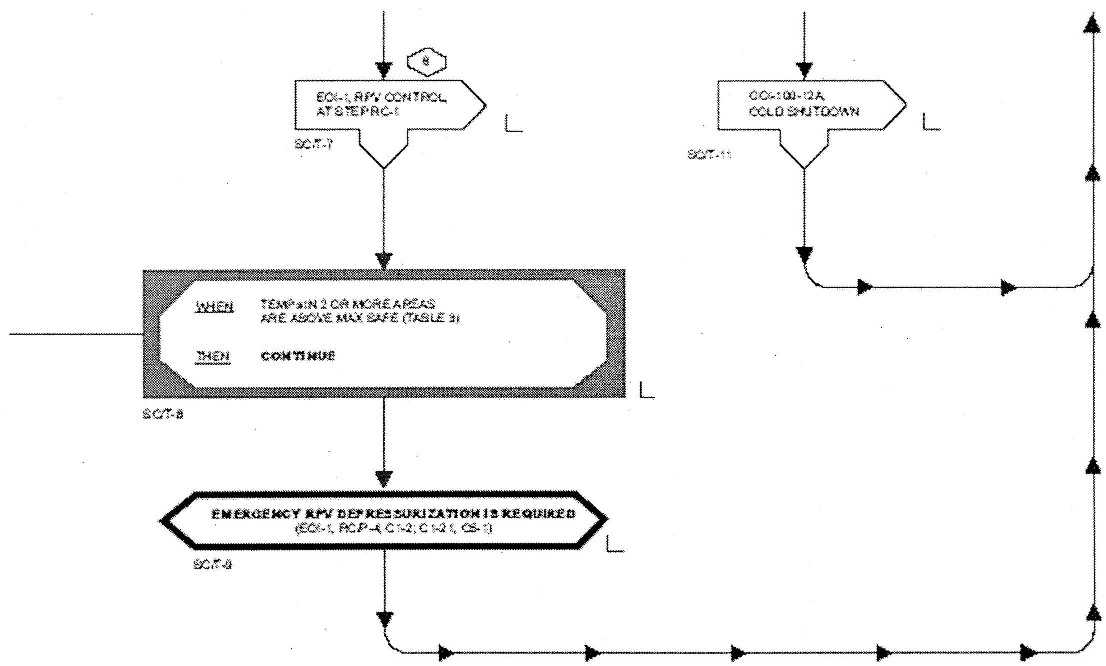
Step SC/T-7 is an enter and execute concurrently step that requires the operator to enter EOI-1, RPV Control, at Step RC-1, and to perform the actions concurrently with this procedure. Because this step is prioritized with the miniature before decision step symbol relating to SC/T-6, this action should be performed before any secondary containment area temperature reaches its respective maximum safe operating value.

Initiation of reactor scram (Step RC-1) before any secondary containment area temperature reaches its respective maximum safe operating value may halt the increase in secondary containment area temperature(s), since the RPV is the only significant source of heat, other than a fire, that could cause secondary containment area temperatures to exceed their respective maximum safe operating values.

TABLE 3
SECONDARY CNTMT AREA TEMP

AREA	PANEL 9-3 ALARM WINDOW (UNLESS NOTED)	PANEL 9-21 TEMP ELEMENT (UNLESS NOTED)	MAX NORMAL VALUE °F	MAX SAFE VALUE °F	POTENTIAL ISOLATION SOURCES
RHR SYS I PUMPS	XA-55-3E-4	74-95A	ALARMED	150	FCV-74-47, 48
RHR SYS II PUMPS	XA-55-3E-4	74-95B	ALARMED	210	FCV-74-47, 48
HPCI ROOM	XA-55-3F-10	73-55A	ALARMED	270	FCV-73-2, 3, 44, 81
CS SYS I PUMPS RCIC ROOM	XA-55-3D-10	71-41A	ALARMED	190	FCV-71-2, 3, 39
CS SYS II PUMPS	XA-55-3E-29	75-69B (PANEL 9-3)	ALARMED	150	NONE
TOP OF TORUS	XA-55-3D-10 XA-55-3F-10 XA-55-3E-4	71-41B, C, D 73-55B, C, D 74-95G, H	ALARMED ALARMED ALARMED	200 240 240	FCV-71-2, 3 FCV-73-2, 3, 81 FCV-74-47, 48
STEAM TUNNEL (RB)	XA-55-3D-24	1-60A (PANEL 9-3)	ALARMED	315	MSIVs FCV-71-2, 3, FCV-69-1, 2, 12
DW ACCESS	XA-55-3E-4	74-95E	ALARMED	170	FCV-74-47, 48
RB EL 565 W (RWCU PIPE TRENCH)	XA-55-5B-32 (PANEL 9-5) XA-55-5B-33 (PANEL 9-5)	69-835A, B, C, D (AUX INST ROOM)	ALARMED ALARMED	170	FCV-69-1, 2, 12
RWCU H. X. ROOM	XA-55-3D-17	69-29F, G, H	ALARMED	220	FCV-69-1, 2, 12
RWCU PUMP A	XA-55-3D-17	69-29D	ALARMED	215	FCV-69-1, 2, 12
RWCU PUMP B	XA-55-3D-17	69-29E	ALARMED	215	FCV-69-1, 2, 12
RB EL 593	XA-55-3E-4	74-95C, D	ALARMED	195	FCV-74-47, 48
RB EL 621	XA-55-3E-4	74-95F	ALARMED	155	FCV-43-13, 14


3
WILL EMERGENCY DEPRESSURIZATION REDUCE DISCHARGE INTO SECONDARY CNTMT
SC/T-5
NO
YES
BEFORE
ANY AREA TEMP RISES TO MAX SAFE (TABLE 3)
CONTINUE
SC/T-6
6
EOI-1, RPV CONTROL AT STEP RC-1
SC/T-7
ARE TEMPS IN 2 OR MORE AREAS ABOVE MAX SAFE (TABLE 3)
SC/T-10
NO
YES
GOI-100-12A, COLD SHUTDOWN
SC/T-11



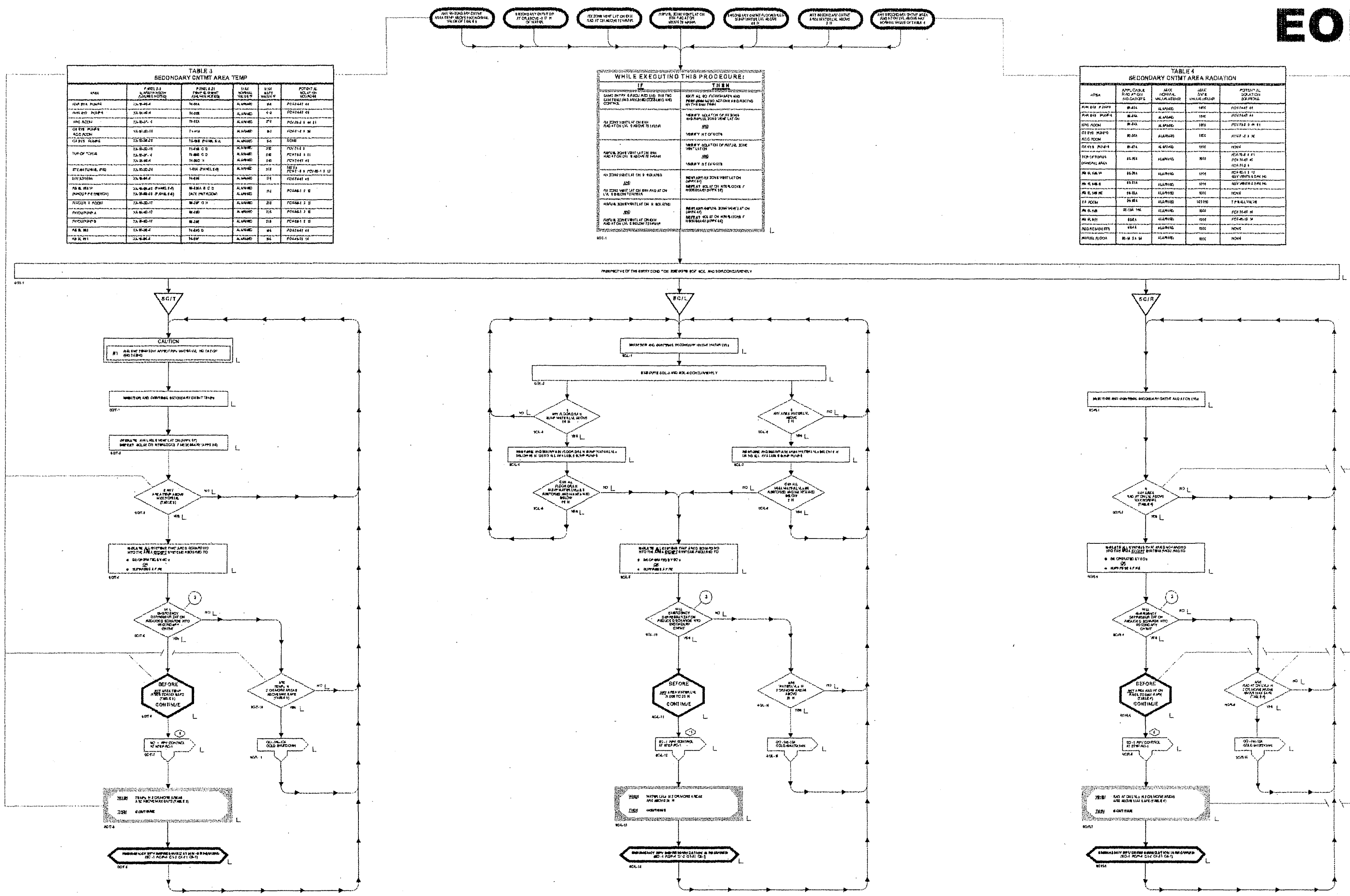
EOI-1, RPV CONTROL, AT STEP RC-1
SC/T-7
EOI-100-12A, COLD SHUTDOWN
SC/T-11
WHEN TEMP ≥ IN 2 OR MORE AREAS ARE ABOVE MAX SAFE (TABLE 3)
THEN CONTINUE
SC/T-8
EMERGENCY RPV DEPRESSURIZATION IS REQUIRED
(EOI-1, RC/P-4, C1-2, C1-2.1, CS-1)
SC/T-9

WHILE EXECUTING THE FOLLOWING STEPS:	
IF	**THEN**
EMERGENCY RPV DEPRESSURIZATION IS ANTICIPATED AND THE REACTOR WILL REMAIN SUBCRITICAL WITHOUT BORON UNDER ALL CONDITIONS (SEE NOTE)	**RAPIDLY DEPRESSURIZE** THE RPV WITH THE MAIN TURB BYPASS VLVs IRRESPECTIVE OF COOLDOWN RATE

RC/P-3

EXAMINATION REFERENCE PROVIDED TO CANDIDATE

EOI-3



CAUTIONS

CAUTION #1

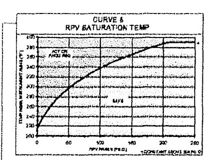


TABLE 6
SECONDARY CNTMT INSTRUMENT RUNS

NOTES

2-EOI-3 PAGE 1 OF 1
SECONDARY CONTAINMENT CONTROL
UNIT 2
BROWNS FERRY
NUCLEAR PLANT
REV 11

EOI-3

64. RO 295033EA2.01 001/C/A/T1G2/SC/R//295033EA2.01//RO/SRO/BANK

Given the following plant conditions:

- Unit 2 is at 100% rated power.
- A Reactor Water Cleanup (RWCU) drain line cracks and is spilling into the Reactor Building.
- Area Radiation Monitors in the Reactor Building read as follows:

Reactor Building Elevation 593	1100 mR/hr
Reactor Building Elevation 565 West	800 mR/hr
Reactor Building Elevation 565 East	850 mR/hr
Reactor Building Elevation 565 Northeast	1050 mR/hr
All other Reactor Building areas	NOT ALARMED

- Approximately one (1) minute later, RWCU is successfully isolated.

Which ONE of the following describes the required action that MUST be directed by the Unit Supervisor and/or Shift Manager?

REFERENCE PROVIDED

A. Enter 2-EOI-1, "RPV Control" and initiate a Reactor Scram due to two (2) areas being above Max Safe in EOI-3, "Secondary Containment Control."

B. Scram the reactor and Emergency Depressurize the reactor due to two (2) areas being above Max Safe in EOI-3, "Secondary Containment Control."

C. Rapidly depressurize the reactor, to the Main Condenser, with the Main Turbine Bypass Valves due to an Emergency Depressurization being anticipated.

D.✓ Enter 2-GOI-100-12A, "Unit Shutdown" and commence a normal shutdown / cooldown due to two (2) areas above Max Safe with the source of the leak isolated.

K/A Statement:

295033 High Secondary Containment Area Radiation Levels / 9
EA2.01 - Ability to determine and/or interpret the following as they apply to HIGH SECONDARY CONTAINMENT AREA RADIATION LEVELS : Area radiation levels

K/A Justification: This question satisfies the K/A statement by requiring the candidate to use specific plant conditions to determine the required actions which result from high secondary containment radiation levels as indicated by Area Radiation Monitoring instrumentation.

References: 2-EOI-3 Flowchart

Level of Knowledge Justification: This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

0610 NRC Exam

REFERENCE PROVIDED: 2-EOI-3 Flowchart

Plausibility Analysis:

In order to answer this question correctly the candidate must determine the following:

1. Which area(s) are above or approaching Max Safe
2. Based on Item #1 above, determine the appropriate action.

D - correct:

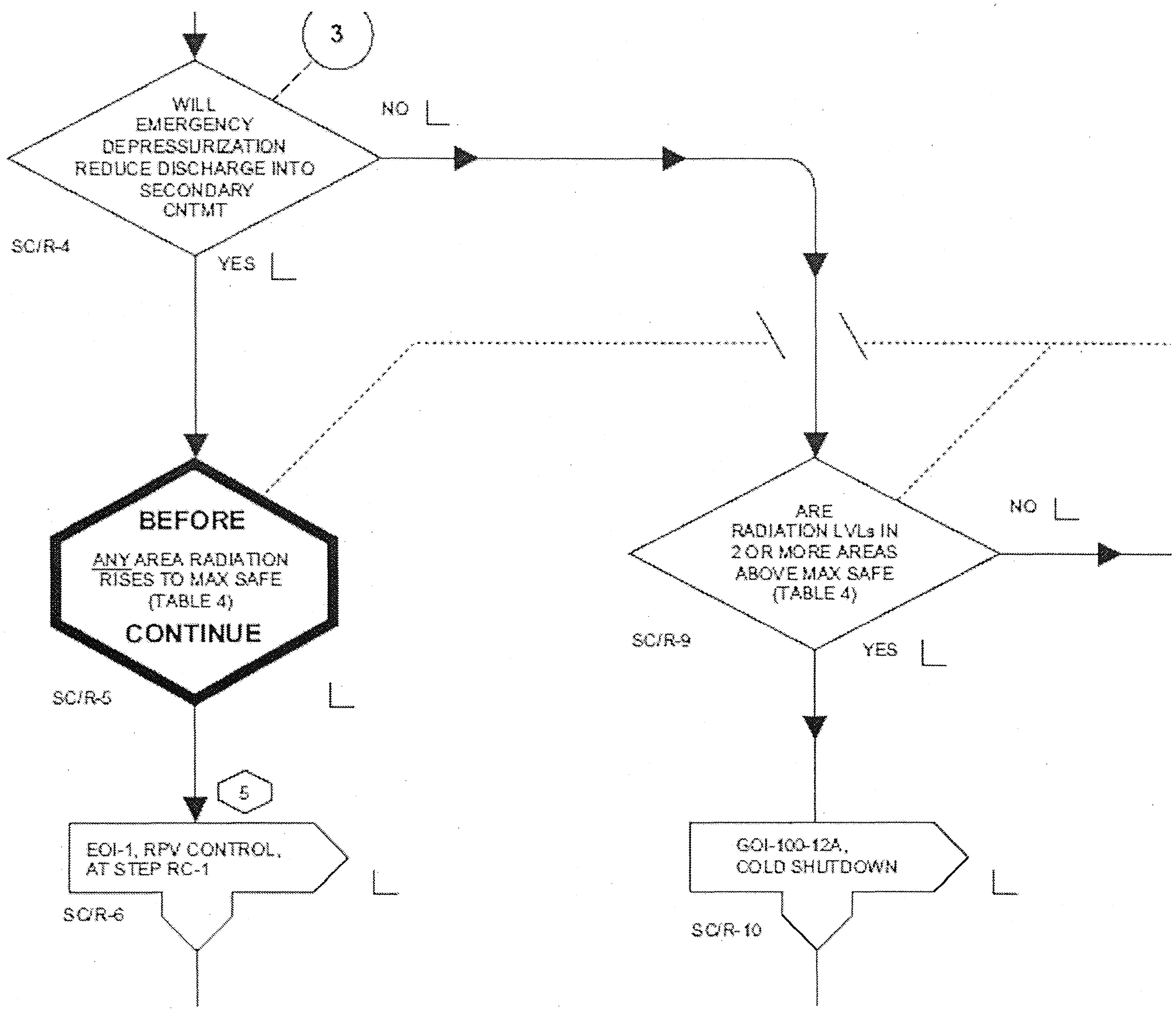
A - incorrect: This is plausible because this action requires at least one area greater than Max Safe. However, this is not appropriate since the source of the leak is isolated.

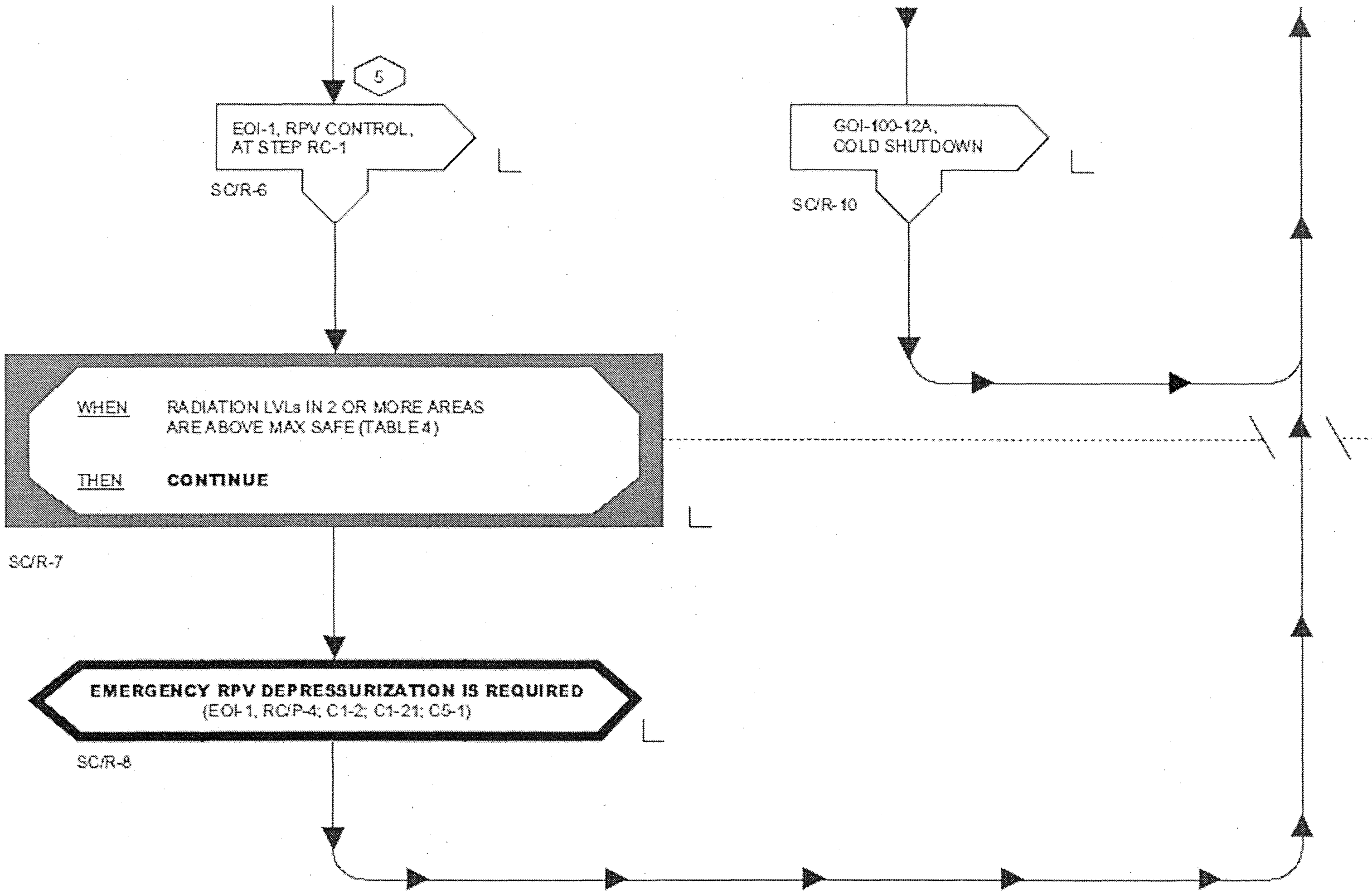
B - incorrect: This is plausible because this action requires two areas greater than Max Safe. However, this is not appropriate since the source of the leak is isolated.

C - incorrect: This is plausible because this action requires at least one area greater than Max Safe and another area approaching Max Safe. However, this is not appropriate since the source of the leak is isolated.

TABLE 4
SECONDARY CNTMT AREA RADIATION

AREA	APPLICABLE RADIATION INDICATORS	MAX NORMAL VALUE MR/HR	MAX SAFE VALUE MR/HR	POTENTIAL ISOLATION SOURCES
RHR SYS I PUMPS	90-25A	ALARMED	1000	FCV-74-47, 48
RHR SYS II PUMPS	90-28A	ALARMED	1000	FCV-74-47, 48
HPCI ROOM	90-24A	ALARMED	1000	FCV-73-2, 3, 44, 81
CS SYS I PUMPS RCIC ROOM	90-26A	ALARMED	1000	FCV-71-2, 3, 39
CS SYS II PUMPS	90-27A	ALARMED	1000	NONE
TOP OF TORUS GENERAL AREA	90-29A	ALARMED	1000	FCV-73-2, 3, 81 FCV-74-47, 48 FCV-71-2, 3
RB EL 565 W	90-20A	ALARMED	1000	FCV-69-1, 2, 12 SDV VENTS & DRAINS
RB EL 565 E	90-21A	ALARMED	1000	SDV VENTS & DRAINS
RB EL 565 NE	90-23A	ALARMED	1000	NONE
TIP ROOM	90-22A	ALARMED	100,000	TIP BALL VALVE
RB EL 593	90-13A, 14A	ALARMED	1000	FCV-74-47, 48
RB EL 621	90-9A	ALARMED	1000	FCV-43-13, 14
RECIRC MG SETS	90-4A	ALARMED	1000	NONE
REFUEL FLOOR	90-1A, 2A, 3A	ALARMED	1000	NONE


3
WILL EMERGENCY DEPRESSURIZATION REDUCE DISCHARGE INTO SECONDARY CNTMT
SC/R-4
NO
YES
BEFORE
ANY AREA RADIATION RISES TO MAX SAFE (TABLE 4)
CONTINUE
SC/R-5
5
EOI-1, RPV CONTROL, AT STEP RC-1
SC/R-6
ARE RADIATION LVLs IN 2 OR MORE AREAS ABOVE MAX SAFE (TABLE 4)
SC/R-9
NO
YES
GOI-100-12A, COLD SHUTDOWN
SC/R-10


5
EOI-1, RPV CONTROL,
AT STEP RC-1
SC/R-6
GOI-100-12A,
COLD SHUTDOWN
SC/R-10
WHEN RADIATION LVLs IN 2 OR MORE AREAS ARE ABOVE MAX SAFE (TABLE 4)
THEN CONTINUE
SC/R-7
EMERGENCY RPV DEPRESSURIZATION IS REQUIRED
(EOI-1, RC/P-4; C1-2; C1-21; C5-1)
SC/R-8

WHILE EXECUTING THE FOLLOWING STEPS:

IF	THEN
EMERGENCY RPV DEPRESSURIZATION IS ANTICIPATED AND THE REACTOR WILL REMAIN SUBCRITICAL WITHOUT BORON UNDER ALL CONDITIONS (SEE NOTE)	**RAPIDLY DEPRESSURIZE** THE RPV WITH THE MAIN TURB BYPASS VLVs IRRESPECTIVE OF COOLDOWN RATE

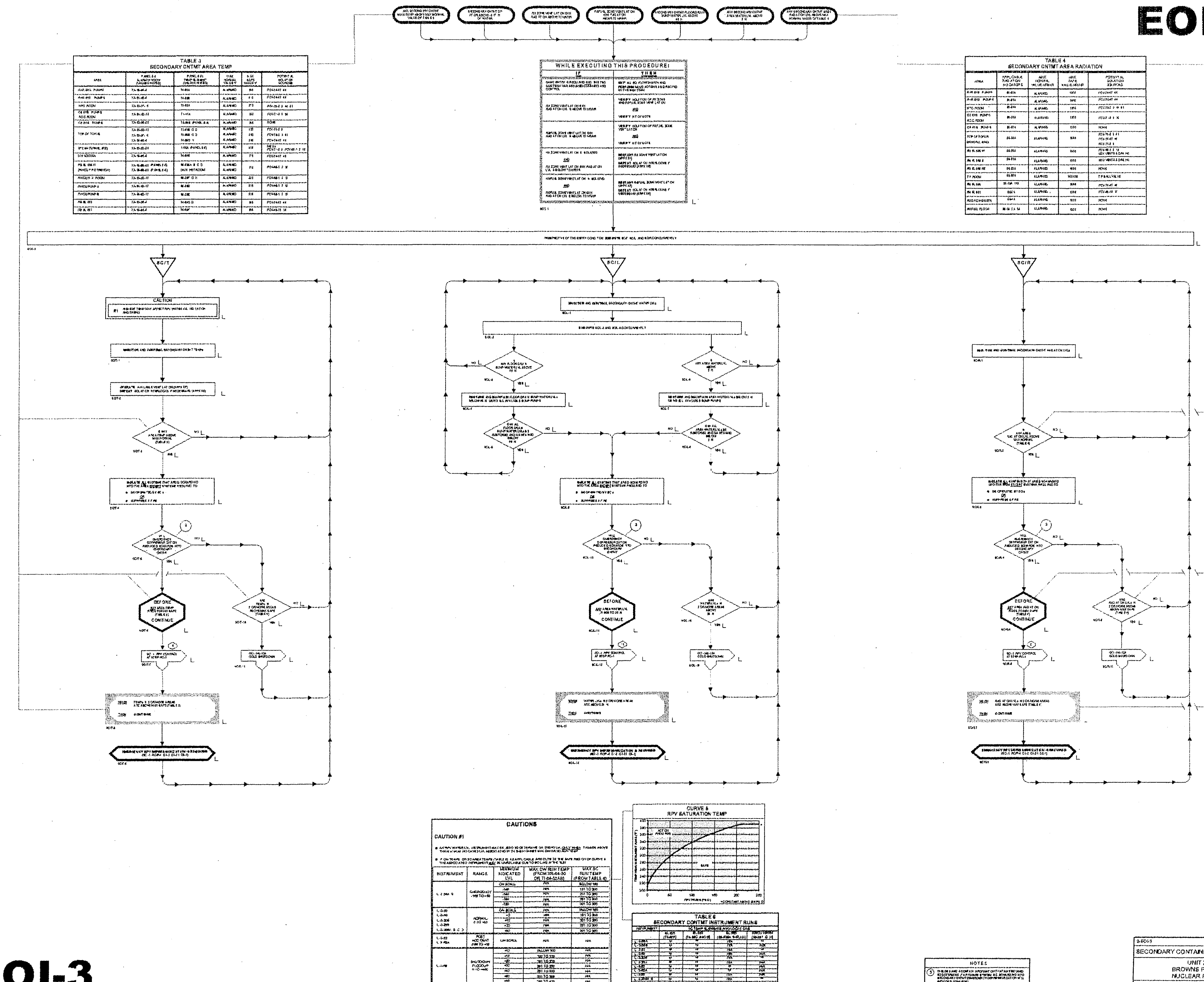
L

RC/P-3

EXAMINATION REFERENCE PROVIDED TO CANDIDATE

EOI-3

EOI-3
UNIT 2



EOI-3

2-EOI-3	PAGE 1 OF 1
SECONDARY CONTAINMENT CONTROL	
UNIT 2 BROWNS FERRY NUCLEAR PLANT	
REV 11	

65. RO 295035EA2.02 001/C/A/EOI/EOI-3/S85/295035EA2.02//RO/SRO/BANK

Given the following plant conditions:

- Unit 2 is at 100% power.
- During the backwash of a Reactor Water Cleanup (RWCU) Demineralizer, the Backwash Receiving Tank ruptured.
- The RWCU system has been isolated.
- Secondary Containment conditions are as follows:

  - ALL Reactor and Refuel Zone radiation monitors trip on high radiation.

  - ONLY Standby Gas Treatment (SGT) train 'C' can be started. It is operating at 10,000 scfm and taking suction on the Refuel and Reactor Zones.

- Refuel zone pressure:	(-)0.12 inches of water
- Reactor zone pressure:	(+)0.02 inches of water

- AREA RADIATION LEVELS

RB EL 565 W, 565 E, 565 NE:	250 mr/hr
RB EL 593	upscale
RB EL 621	upscale

Which ONE of the following describes the required action and the type of radioactive release in progress?

REFERENCE PROVIDED

A. Initiate a shutdown per 2-GOI-100-12A, "Unit Shutdown." An Elevated radiation release is in progress.

B✓ Initiate a shutdown per 2-GOI-100-12A, "Unit Shutdown." A Ground-level radiation release is in progress.

C. Scram the reactor and Emergency Depressurize the RPV. An Elevated radiation release is in progress.

D. Scram the reactor and Emergency Depressurize the RPV. A Ground-level radiation release is in progress.

K/A Statement:

295035 Secondary Containment High Differential Pressure
EA2.02 - Ability to determine and/or interpret the following as they apply to SECONDARY CONTAINMENT HIGH DIFFERENTIAL PRESSURE: Off-site release rate: Plant-Specific

K/A Justification: This question satisfies the K/A statement by requiring the candidate to correctly identify the type of off-site release and required actions due to high differential pressure in the secondary containment.

References: 2-EOI-3 Flowchart

Level of Knowledge Justification: This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to solve a problem. This requires mentally using this knowledge and its meaning to resolve the problem.

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REFERENCE PROVIDED: 2-EOI-3 flowchart

Plausibility Analysis:

In order to answer this question correctly the candidate must determine the following:

1. Which area(s) are above or approaching Max Safe.
2. Based on Item #1 above, determine the appropriate action.
3. Whether plant conditions indicate an elevated or ground-level release.

NOTE: EOI-3 steps SC/R-8 and SC/R-9 apply, requiring shutdown per 2-GOI-100-12A because two (2) or more areas are above max safe rad levels; but, a primary system is NOT discharging to the Reactor Building. Insufficient Reactor Building-to-atmosphere dp (greater than -0.25 inches of water) indicates loss of secondary containment integrity. The positive Reactor Zone pressure is causing an unmonitored and uncontrolled ground-level release of radioactive contaminants.

B - correct:

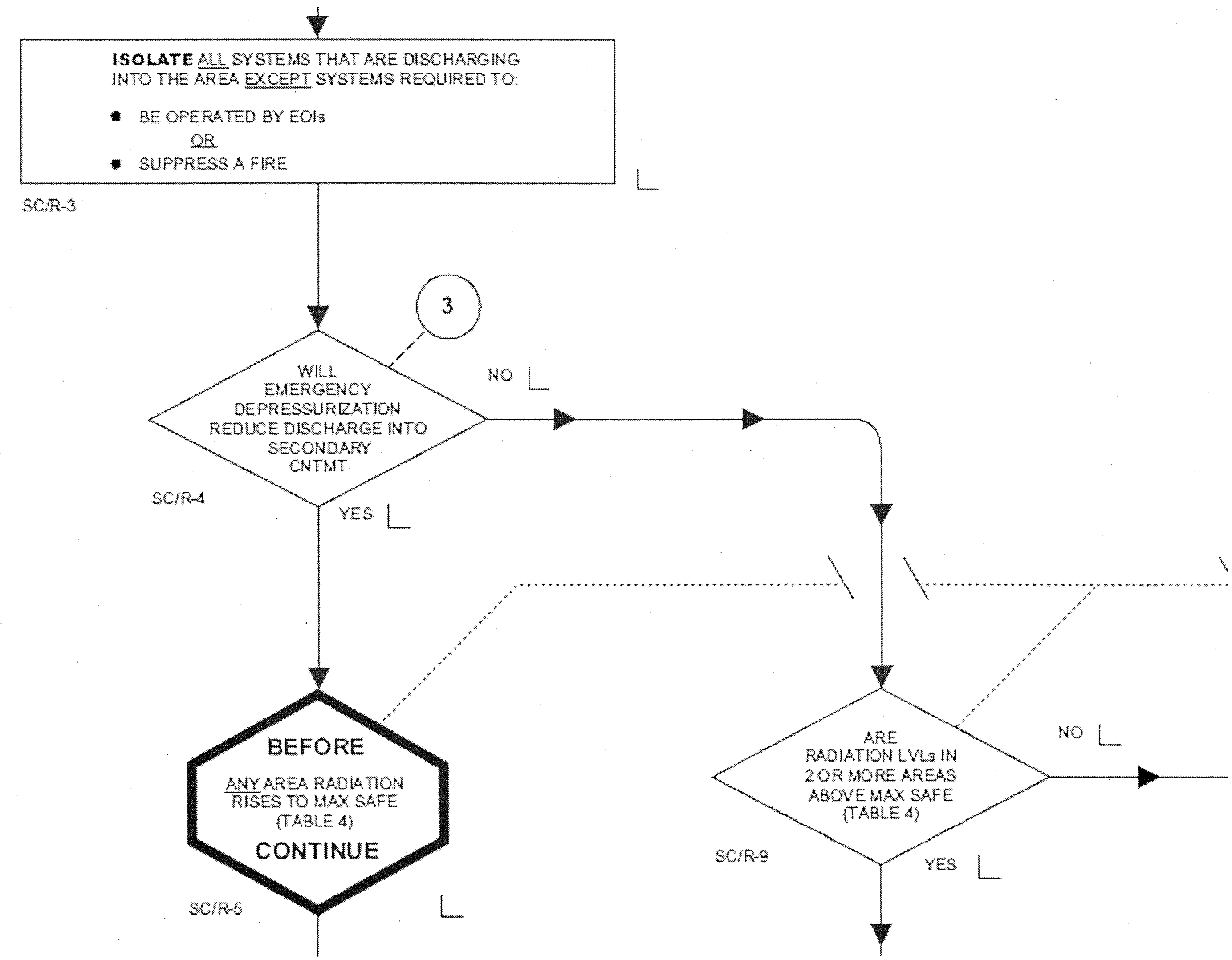
A - incorrect: The release from the Reactor Building is NOT elevated. This is plausible because the required actions are correct except the differential pressure results in a ground-level release.

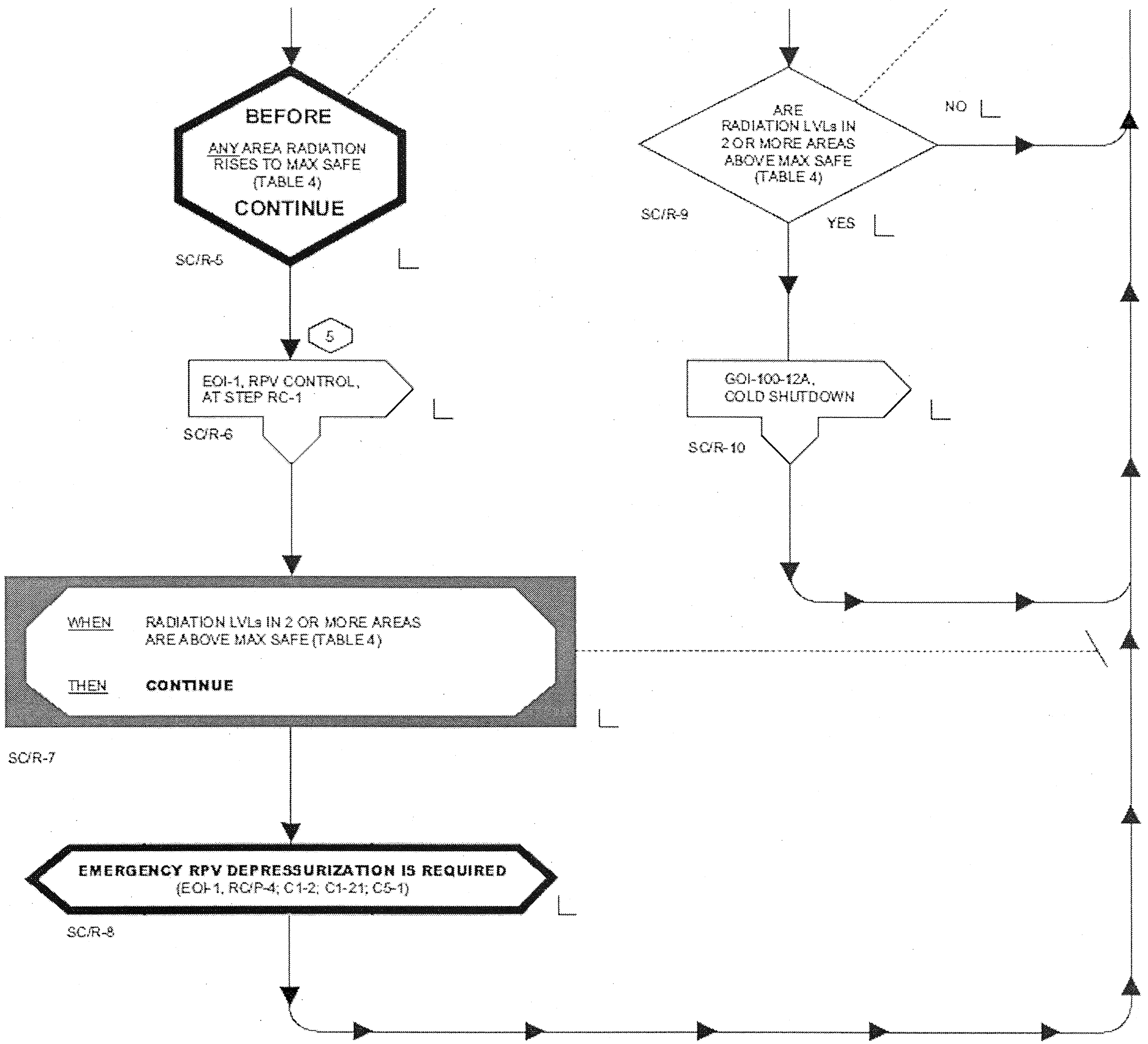
C - incorrect: Conditions DO NOT warrant a scram at this point. In addition, the release from the Reactor Building is NOT elevated. This is plausible if the candidate fails to recognize that a primary system is NOT discharging to the Reactor Building.

D - incorrect: Conditions DO NOT warrant a scram at this point. This is plausible if the candidate fails to recognize that a primary system is NOT discharging to the Reactor Building.

TABLE 4
SECONDARY CNTMT AREA RADIATION

AREA	APPLICABLE RADIATION INDICATORS	MAX NORMAL VALUE MR/HR	MAX SAFE VALUE MR/HR	POTENTIAL ISOLATION SOURCES
RHR SYS I PUMPS	90-25A	ALARMED	1000	FCV-74-47, 48
RHR SYS II PUMPS	90-28A	ALARMED	1000	FCV-74-47, 48
HPCI ROOM	90-24A	ALARMED	1000	FCV-73-2, 3, 44, 81
CS SYS I PUMPS RCIC ROOM	90-26A	ALARMED	1000	FCV-71-2, 3, 39
CS SYS II PUMPS	90-27A	ALARMED	1000	NONE
TOP OF TORUS GENERAL AREA	90-29A	ALARMED	1000	FCV-73-2, 3, 81 FCV-74-47, 48 FCV-71-2, 3
RB EL 565 W	90-20A	ALARMED	1000	FCV-69-1, 2, 12 SDV VENTS & DRAINS
RB EL 565 E	90-21A	ALARMED	1000	SDV VENTS & DRAINS
RB EL 565 NE	90-23A	ALARMED	1000	NONE
TIP ROOM	90-22A	ALARMED	100,000	TIP BALL VALVE
RB EL 593	90-13A, 14A	ALARMED	1000	FCV-74-47, 48
RB EL 621	90-9A	ALARMED	1000	FCV-43-13, 14
RECIRC MG SETS	90-4A	ALARMED	1000	NONE
REFUEL FLOOR	90-1A, 2A, 3A	ALARMED	1000	NONE


ISOLATE ALL SYSTEMS THAT ARE DISCHARGING INTO THE AREA EXCEPT SYSTEMS REQUIRED TO:
BE OPERATED BY EOIs
OR
SUPPRESS A FIRE
SC/R-3
3
WILL EMERGENCY DEPRESSURIZATION REDUCE DISCHARGE INTO SECONDARY CNTMT
SC/R-4
NO
YES
BEFORE
ANY AREA RADIATION RISES TO MAX SAFE (TABLE 4)
CONTINUE
SC/R-5
ARE RADIATION LVLs IN 2 OR MORE AREAS ABOVE MAX SAFE (TABLE 4)
SC/R-9
NO
YES

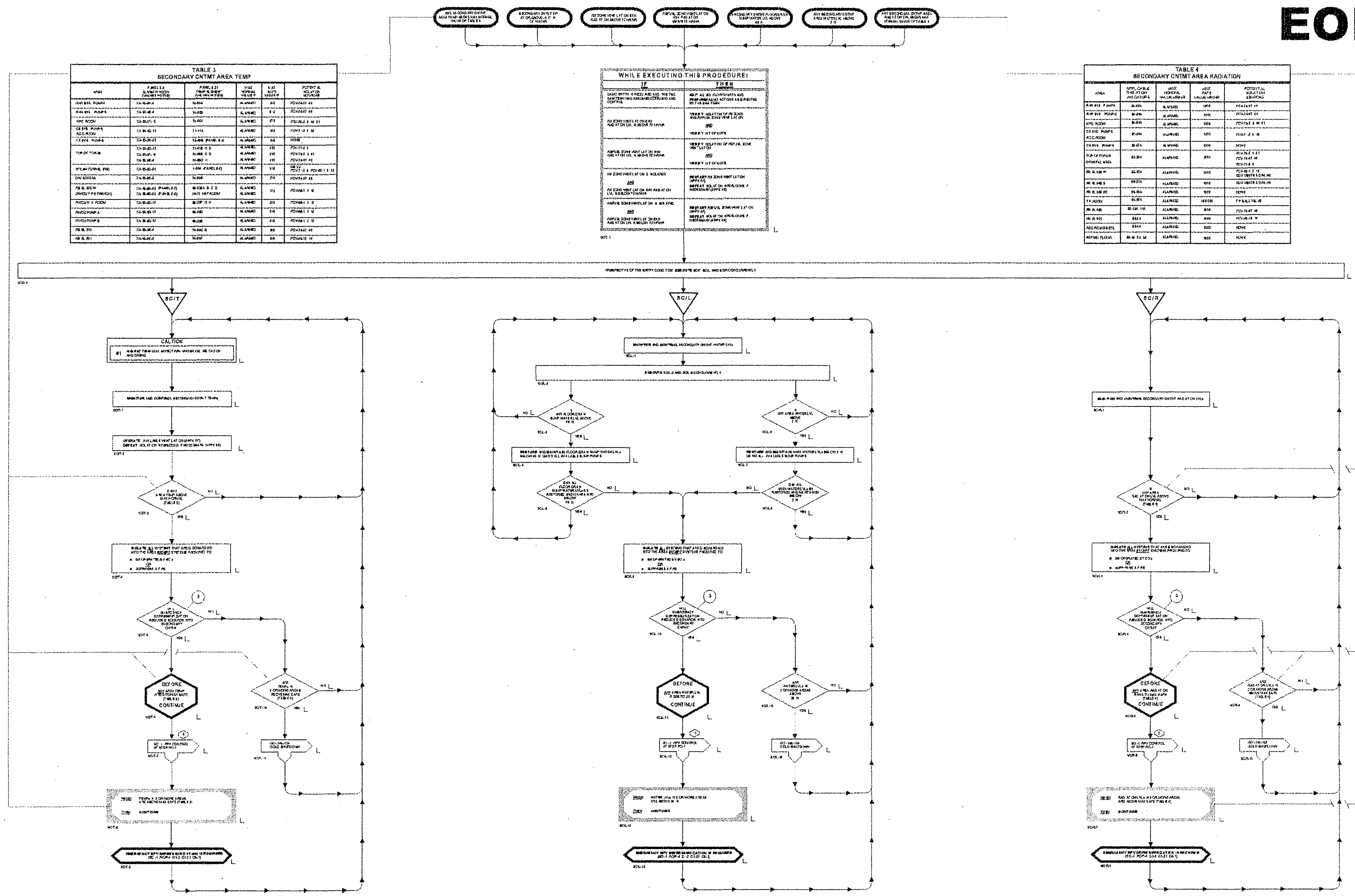

BEFORE
ANY AREA RADIATION RISES TO MAX SAFE (TABLE 4)
CONTINUE
SC/R-5
5
EOI-1, RPV CONTROL, AT STEP RC-1
SC/R-6
WHEN
RADIATION LVLs IN 2 OR MORE AREAS ARE ABOVE MAX SAFE (TABLE 4)
THEN
CONTINUE
SC/R-7
EMERGENCY RPV DEPRESSURIZATION IS REQUIRED
(EOI-1, RC/P-4; C1-2; C1-21; C5-1)
SC/R-8
ARE RADIATION LVLs IN 2 OR MORE AREAS ABOVE MAX SAFE (TABLE 4)
NO
YES
SC/R-9
GOI-100-12A, COLD SHUTDOWN
SC/R-10

EXAMINATION REFERENCE PROVIDED TO CANDIDATE

EOI-3

UNIT 2

EOI-3



CAUTIONS

CAUTION #1

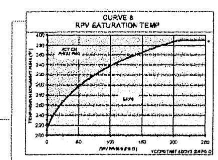


TABLE 6
SECONDARY CNTMT INSTRUMENT RUNS

NOTES

2-EOI-3	PAGE 1 OF 1
SECONDARY CONTAINMENT CONTROL	
UNIT 2 BROWNS FERRY NUCLEAR PLANT	
REV. 11	

EOI-3

REFERENCE PROVIDED: None

Plausibility Analysis:

In order to answer this question correctly the candidate must determine the following:

1. Tech Spec applicability for the listed systems with the given plant condition.

NOTE: The distractors are all plausibe since only one system or function is incorrect in each distractor.

C - correct:

A - incorrect: The RBM is NOT required until >27% rated power.

B - incorrect: The APRM Hi (120%) is NOT required until Mode 1.

D - incorrect: The OPRMs are NOT required until Mode 1 and >25% rated power.

Table 3.3.2.1-1 (page 1 of 1)
Control Rod Block Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Rod Block Monitor				
a. Low Power Range - Upscale	(a)	2	SR 3.3.2.1.1 SR 3.3.2.1.4 SR 3.3.2.1.8	(e)
b. Intermediate Power Range - Upscale	(b)	2	SR 3.3.2.1.1 SR 3.3.2.1.4 SR 3.3.2.1.8	(e)
c. High Power Range - Upscale	(f),(g)	2	SR 3.3.2.1.1 SR 3.3.2.1.4 SR 3.3.2.1.8	(e)
d. Inop	(g),(h)	2	SR 3.3.2.1.1	NA
e. Downscale	(g),(h)	2	SR 3.3.2.1.1 SR 3.3.2.1.4	(i)
2. Rod Worth Minimizer	$1^{(c)}, 2^{(c)}$	1	SR 3.3.2.1.2 SR 3.3.2.1.3 SR 3.3.2.1.5 SR 3.3.2.1.7	NA
3. Reactor Mode Switch — Shutdown Position	(d)	2	SR 3.3.2.1.6	NA

(a) THERMAL POWER ≥ 27% and ≤ 62% RTP and MCPR less than the value specified in the COLR.

(b) THERMAL POWER > 62% and ≤ 82% RTP and MCPR less than the value specified in the COLR.

(c) With THERMAL POWER ≤ 10% RTP.

(d) Reactor mode switch in the shutdown position.

(e) Less than or equal to the Allowable Value specified in the COLR.

(f) THERMAL POWER > 82% and < 90% RTP and MCPR less than the value specified in the COLR.

(g) THERMAL POWER ≥ 90% RTP and MCPR less than the value specified in the COLR.

(h) THERMAL POWER ≥ 27% and < 90% RTP and MCPR less than the value specified in the COLR.

(i) Greater than or equal to the Allowable Value specified in the COLR.

Table 3.3.1.1-1 (page 1 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Intermediate Range Monitors					
a. Neutron Flux - High	2	3	G	SR 3.3.1.1.1 SR 3.3.1.1.3 SR 3.3.1.1.5 SR 3.3.1.1.6 SR 3.3.1.1.9 SR 3.3.1.1.14	≤ 120/125 divisions of full scale
	5(a)	3	H	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.9 SR 3.3.1.1.14	≤ 120/125 divisions of full scale
b. Inop	2	3	G	SR 3.3.1.1.3 SR 3.3.1.1.14	NA
	5(a)	3	H	SR 3.3.1.1.4 SR 3.3.1.1.14	NA
2. Average Power Range Monitors					
a. Neutron Flux - High, Setdown	2	3(b)	G	SR 3.3.1.1.1 SR 3.3.1.1.6 SR 3.3.1.1.7 SR 3.3.1.1.13 SR 3.3.1.1.16	≤ 15% RTP
b. Flow Biased Simulated Thermal Power - High	1	3(b)	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.7 SR 3.3.1.1.13 SR 3.3.1.1.16	≤ 0.66 W + 66% RTP and ≤ 120% RTP(c)
c. Neutron Flux - High	1	3(b)	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.7 SR 3.3.1.1.13 SR 3.3.1.1.16	≤ 120% RTP

(continued)

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

(b) Each APRM channel provides inputs to both trip systems.

(c) [0.66 W + 66% - 0.66 Δ W] RTP when reset for single loop operation per LCO 3.4.1, "Recirculation Loops Operating."

Table 3.3.1.1-1 (page 2 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. Average Power Range Monitors (continued)					
d. Inop	1,2	3(b)	G	SR 3.3.1.1.16	NA
e. 2-Out-Of-4 Voter	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.14 SR 3.3.1.1.16	NA
f. OPRM Upscale	1	3(b)	I	SR 3.3.1.1.1 SR 3.3.1.1.7 SR 3.3.1.1.13 SR 3.3.1.1.16 SR 3.3.1.1.17	NA
3. Reactor Vessel Steam Dome Pressure - High(d)	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.14	≤ 1090 psig
4. Reactor Vessel Water Level - Low, Level 3(d)	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≥ 528 inches above vessel zero
5. Main Steam Isolation Valve - Closure	1	8	F	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 10% closed
6. Drywell Pressure - High	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 2.5 psig
7. Scram Discharge Volume Water Level - High					
a. Resistance Temperature Detector	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 50 gallons
	5(a)	2	H	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 50 gallons

(continued)

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

(b) Each APRM channel provides inputs to both trip systems.

(d) During instrument calibrations, if the As Found channel setpoint is conservative with respect to the Allowable Value but outside its acceptable As Found band as defined by its associated Surveillance Requirement procedure, then there shall be an initial determination to ensure confidence that the channel can perform as required before returning the channel to service in accordance with the Surveillance. If the As Found instrument channel setpoint is not conservative with respect to the Allowable Value, the channel shall be declared inoperable.

Prior to returning a channel to service, the instrument channel setpoint shall be calibrated to a value that is within the acceptable As Left tolerance of the setpoint; otherwise, the channel shall be declared inoperable.

The nominal Trip Setpoint shall be specified on design output documentation which is incorporated by reference in the Updated Final Safety Analysis Report. The methodology used to determine the nominal Trip Setpoint, the predefined As Found Tolerance, and the As Left Tolerance band, and a listing of the setpoint design output documentation shall be specified in Chapter 7 of the Updated Final Safety Analysis Report.

67. RO GENERIC 2.1.16 001/MEM/T3//B17/G2.1.16//RO/SRO/NEW 10/16/07

Which ONE of the following announcements is an INAPPROPRIATE use of the Plant Paging System in accordance with OPDP-1, "Conduct of Operations?"

A. "Shift Manager dial 2391. Shift Manager dial 2391."

B. "Operations will be starting the 2 Alpha RHR pump."

C.✓ "All personnel evacuate the Unit 2 Reactor Building due to high radiation. This is a drill."

D. "There is a fire in the Unit 2 Shutdown Board Room. I repeat. There is a fire in the Unit 2 Shutdown Board Room."

K/A Statement:

Conduct of Operations
2.1.16 Ability to operate plant phone, paging system, and two-way radio

K/A Justification: This question satisfies the K/A statement by requiring the candidate to demonstrate specific knowledge of the use of the Plant Paging System while communicating with plant personnel.

References: OPDP-1

Level of Knowledge Justification: This question is rated as MEM due to the requirement to recall or recognize discrete bits of information.

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REFERENCE PROVIDED: None

Plausibility Analysis:

In order to answer this question correctly the candidate must determine the following:

1. The requirements associated with making Page Announcements per OPDP-1.
2. Whether the announcement meets those requirements

C - correct: The line "This is a drill" is required at the beginning AND end of each communication during drills / exercises. In addition, an announcement of such urgency should be repeated.

A - incorrect: This is plausible since the page is repeated. However, there may NOT be a requirement to repeat the announcement, but it is NOT an inappropriate action.

B - incorrect: This is plausible since the page is NOT repeated. However, repeating pages for a normal operation is NOT required.

D - incorrect: This is plausible because it is an expected announcement during a fire.

Appendix I
(Page 3 of 5)

Communications

b. Use equipment noun names and/or identification (ID) numbers to describe a component.

c. The use of sign language is undesired but maybe used when verbal communications is not practical.

d. Take time when reporting abnormal conditions. Speak deliberately, distinctly and calmly. Identify yourself and watch station or your location. Describe the nature and severity of the problem. State the location of the problem if appropriate. Keep the communication line open if possible or until directed otherwise.

e. The completion of directed actions should be reported to the governing station, normally the control room.

f. Require other plant personnel (including contractors) conducting operational communication to do so in accordance with this procedure.

g. If there is any doubt concerning any portion of the communication or task assigned, resolve it before taking any action.

h. When making announcements for drills or exercises begin and end the announcement with "This is a Drill."

4. Emergency Communications Systems

When personnel are working in areas where the public address (PA) system or emergency signals cannot be heard, alternate methods for alerting these persons should be devised. Flashing lights, personal pagers that vibrate and can be felt, and persons dedicated to notifications are examples of alternate methods.

5. PA System

a. Use of the plant PA system shall be limited to ensure it retains its effectiveness in contacting plant personnel. Excessive use of the PA system should be avoided. Plant telephones and other point-to-point communications channels should be used in lieu of the PA system whenever practical.

b. The announcement of planned starting or stopping large equipment should be made to alert personnel working in that area.

c. The plant PA system may be used in abnormal or emergency conditions, to announce change of plant status, or give notification of major plant events either in progress or anticipated.

Appendix I
(Page 4 of 5)

Communications

d. When using the plant PA system:

(1) Speak slowly and deliberately in a normal tone of voice.

(2) When announcements of abnormal or emergency conditions are made, they shall be made at least twice.

(3) When making announcements for drills or exercises begin and end the announcement with "This is a Drill."

6. Plant Telephones

When using Plant telephones:

a. Identify yourself and watch station.

b. When trying to make contact with the main Control Room, if the message is of a routine nature, the sender should hang up when the main Control Room fails to answer after the fifth ring to avoid unnecessary Control Room noise. The phone shall be allowed to ring until answered if the information is important to Operations.

c. During times when the DO NOT DISTRUB (DND) function has been used by MCR personnel, follow the directions on the recording as appropriate.

d. When making announcements for drills or exercises begin and end the announcement with "This is a Drill."

7. Radio/phone Communication

Radio/phone usage shall not be allowed in areas where electronic interference with plant equipment may result.

a. When making announcements for drills or exercises, begin and end the announcement with "This is a Drill."

b. Sender should identify themselves by watch station.

c. Three way communications should be used.

d. Clear concise language should be used since radio/phone contact does not have the advantage of face to face communication.

68. RO GENERIC 2.1.18 001/MEM/T3/12.1//GENERIC 2.1.18//RO/SRO/NEW 10/16/07

Which ONE of the following is an INEFFECTIVE use of the phonetic alphabet in accordance with OPDP-1, "Conduct of Operations?"

A. "Start 2 Alpha RHR pump per 3-OI-74."

B. "Place Golf IRM in Bypass per 1-OI-92-Bravo."

C. "Transfer 2 Alpha 480 Volt Shutdown Board to Alternate."

D.✓ "Place Romeo Papa Sierra 2 Alpha on Alternate per 2-Oscar-India-99."

K/A Statement:

Conduct of Operations
2.1.18 Ability to make accurate, clear and concise logs, records, status boards, and reports.

K/A Justification: This question satisfies the K/A statement by requiring the candidate to demonstrate knowledge of the requirements related to verbal communications or reports during shift operations.

References: OPDP-1

Level of Knowledge Justification: This question is rated as MEM due to the requirement to recall or recognize discrete bits of information.

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REFERENCE PROVIDED: None

Plausibility Analysis:

In order to answer this question correctly the candidate must determine the requirements for use of the phonetic alphabet and apply that knowledge to the given communications.

NOTE: Each distractor is plausible because they all contain at least one use of the phonetic alphabet.

D - correct: The use of the phonetic alphabet for common acronyms, such as RPS, is not required and could reduce the effectiveness of the communication.

A - incorrect: This communication is appropriate.

B - incorrect: This communication is appropriate.

C - incorrect: This communication is appropriate.

Appendix I
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Communications

b. The receiver repeats back the message to the sender. The repeat back can be verbatim or functional. In many cases a functional repeat back best communicates the receivers understanding of the message. This can be done in several ways to accomplish the desired goals. For example the sender might say, "Bob, report RCS pressure and trend." The receiver could respond in either of two ways.

(1) The receiver could respond with, "Report RCS pressure and trend. RCS pressure is 2250 psig and stable."

Or

(2) The receiver could respond with, "RCS pressure is 2250 psig and stable."

c. The sender verbally acknowledges that the receiver correctly understood the message. The verbal acknowledgement can be simple such as, "That is correct". If the sender has requested and received information then the sender shall provide either verbatim or functional repeat back to demonstrate his understanding of the receiver's message. For the example above the sender could respond with, "I understand 2250 and stable."

2. Phonetic Alphabet

The phonetic alphabet is a tool to improve communications. In general, operations communication should use the phonetic alphabet except when well established acronyms describe the subject. If use of phonetic alphabet will reduce effectiveness of communications then it should not be used. The following are examples of when the phonetic alphabet should not be used:

a. It is not desirable to use Romeo-Charlie-Sierra to describe the RCS (Reactor Coolant System).

b. If a procedural step is written using acronyms, it may be read and ordered as such.

c. If a component tag or label is written using acronyms then the acronyms may be used.

3. General Standards

a. All communications shall be clear, concise, and precise. All operational communications shall be conducted in a formal and professional manner. In all communications, the sender and intended receiver should be readily identifiable.

69. RO GENERIC 2.2.13 001/MEM/T3/10.2/7/18/GENERIC 2.2.13/3.6/3.8/RO/SRO/NEW 10/16/07

Which ONE of the following describes the requirements when placing a clearance on Air-Operated Valves (AOVs)?

An Air-Operated Valve that fails ____.

A. "OPEN" on loss of air SHALL NOT be used for blocking purposes.

B.✔ "OPEN" on loss of air, SHALL be held closed with a gagging device that is tagged as a clearance boundary.

C. "AS-IS" on loss of air SHALL NOT be used for blocking purposes until it is verified closed and a gagging device installed.

D. "CLOSED" on loss of air SHALL NOT be considered closed for blocking purposes unless it is held closed with a gagging device.

K/A Statement:

Equipment Control
2.2.13 Knowledge of tagging and clearance procedures

K/A Justification: This question satisfies the K/A statement by requiring the candidate to demonstrate knowledge of the Clearance and Tagging requirements.

References: SPP 10.2

Level of Knowledge Justification: This question is rated as MEM due to the requirement to recall or recognize discrete bits of information.

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REFERENCE PROVIDED: None

Plausibility Analysis:

In order to answer this question correctly, the candidate must determine the requirements for Clearance and Tagging procedure, SPP-10.2 and apply that knowledge to the given conditions.

B - correct:

A - incorrect: This is plausible since using a "Fail-Open" valve presents a difficult problem. However, SPP-10.2 provides specific additional guidance to support the allowance of their use as a clearance boundary.

C - incorrect: This is plausible because, in most cases, it is true. However, SPP-10.2 provides specific guidance and controls to allow using them as a clearance boundary under a specific condition that the clearance be considered "working on energized equipment".

D - incorrect: This is plausible since a locking device would ensure the valve does NOT open. However, SPP-10.2 requires the air supply that actuates the valve be mechanically or electrically isolated.

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Special Requirements for Mechanical Clearances

1.0 REQUIREMENTS

A. An air-operated valve that fails open on a loss of air is not be considered closed for blocking purposes unless it is held closed with an installed jacking device or device used to secure the valve in the required position. A clearance tag will be issued and attached to the jacking or other device.

B. An air-operated valve that fails closed must have its air supply electrically or mechanically isolated, depressurized, and the valve visually checked-to-be-closed by local or remote indication. The air supply energy-isolating devices must be tagged.

C. An air-operated valve that fails "as is" shall be closed and mechanically restrained. Its air supply should be electrically or mechanically isolated, depressurized, and the valve visually checked to be closed by local or remote indication. The air supply energy-isolating devices and mechanical restraint must be tagged.

D. In cases where it is not possible to physically secure an air operated valve that fails "as-is" in the closed position, the valve will be tagged closed by applying closing air to the valve diaphragm by the use of the solenoid valve air overrides and tagging both the hand-switch in the closed position and the solenoid valve air overrides. Prior to allowing work to begin, the equipment will be drained and de-pressurized to ensure the boundary valves are holding. This condition will be noted in the remarks section of the clearance sheet to inform PAE/Authorized Employee(s) that pressurized air is required to ensure the valve remains closed. This work is considered "working on energized equipment" and must be approved by the management official in charge.

E. Pressure controlled valves, relief valves, and check valves will not be used as isolation boundary valves under normal conditions. Where such a valve does not have an external means of physical restraint, the work is considered "working on energized equipment" and must be approved by the management official in charge.

F. The following instructions govern the use of freeze plugs

1. The clearance should be in place, but not issued, before establishing the freeze plug.

2. The need for the freeze plug should be identified on the Remarks Section of the clearance sheet. The freeze plug should not be listed as a device held on the clearance sheet. The establishment and maintenance of the freeze plug shall be in accordance with approved procedures or work documents.

3. The freeze plug must be attended by qualified personnel to ensure that it is maintained intact until all work is complete and the proper Post Maintenance Tests (PMTs) are performed.

4. If the clearance must be released to allow performance of a PMT, the equipment must be retagged before allowing the freeze plug to thaw. This will prevent migration of a portion of the plug.

70. RO GENERIC 2.2.33 001/C/A/SYS/RWM//G2.2.33/RO 2.5/RO/SRO/BANK

Given the following plant conditions:

- A reactor startup is in progress with power currently at 3%
- Rod Worth Minimizer (RWM)is latched into Group 8 (12 control rods)
- Group 9 rods are the same rods as Group 8.
- Sequence Control: ON
- Group 8 Limits: 08-12
- Group 9 Limits: 12-16

Which ONE of following describes when the RWM will automatically latch up to Group 9?

A. ALL rods EXCEPT 3 in group 8 are withdrawn to the group Withdraw Limit, and a rod in group 9 is selected.

B.✓ ALL rods in group 8 have been withdrawn to the group 8 Withdraw Limit and a rod in group 9 has been selected.

C. ALL rods EXCEPT 1 in group 8 are withdrawn to the group Withdraw Limit and a rod in group 9 has to be selected and moved.

D. The last rod in group 8 is withdrawn to the group 8 Withdraw Limit and the in-sequence rod in group 9 has NOT been selected.

K/A Statement:

Equipment Control
2.2.33 Knowledge of control rod programming.

K/A Justification: This question satisfies the K/A statement by requiring the candidate to recognize and apply limitations on control rod programming enforced by the Rod Worth Minimizer program.

References: Lesson Plan OPL171.024 Rev. 13 pages 13 - 15

Level of Knowledge Justification: This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to solve a problem. This requires mentally using this knowledge and its meaning to resolve the problem.

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(b) After returning the RWM to service:

(1) Group 7 will be the latched group

(2) Rod 30-03 will be displayed as a withdraw error.

(3) The withdraw block status indicators will indicate a withdrawal block condition on the RWM system displays and RWM switch panel.

(4) No other control rod may be inserted or withdrawn until the withdraw error rod from Group 8 (30-03) is corrected. It can only be inserted.

(c) The proper way to correct the out of sequence condition is to insert the withdraw error rod (30-03) to position "04".

NOTE: Upon select of rod 30-03, an RWM system message will be generated indicating a target position of notch "04" for this control rod

This removes the withdraw error; leaves group 7 as the latched group, and removes the withdraw block indications on the RWM system displays and RWM switch panel.

11. Automatic Latching Up/Down

Obj. V.B.10

a. The automatic latching process depends on whether or not RWM Sequence Control is ON or OFF. Sequence Control is normally selected (ON) and enforces a specific order to pull rods within a latched group.

b. When operating below the LPSP with sequence control "ON", latching to the next higher or next lower rod group is done internally by the RWM program only after a rod in the next group is selected.

NOTE: Latching within Transistion Zone will be discussed later.

REFERENCE PROVIDED: None

Plausibility Analysis:

B - correct:

A - incorrect: This is plausible because the RWM normally allows three insert errors without generating a rod block, however it will NOT latch up to a higher group under this condition because the three rods are more than one notch from the withdraw limit.

C - incorrect: The selected control rod does NOT have to be moved to latch to Group 9. This is plausible because the RWM will latch to the highest group with one rod past the insert limit if the RWM is latching to a group from an unknown condition. Since this is a known condition, the RWM will latch to Group 9 without moving the selected rod.

D - incorrect: The RWM will NOT latch to the next group until the correct rod is selected in Group 9; because Sequence Control is ON. This is plausible if Sequence Control is OFF. Under that condition, the RWM only looks for rods within the Group and NOT within a specific sequence. With Sequence Control in OFF, the RWM will latch up to Group 9 as soon as the last rod reaches the withdraw limit.

INSTRUCTOR NOTES

(8) Upon demand by the operator via the Scan/Latch request function.

(9) Following correction of Insert or Withdraw Errors.

d. The latched group is the highest group which can be achieved without producing an active insert block condition.

(1) The RWM system will latch to the highest group in the sequence with:

(a) At least one rod withdrawn past the group insert limit and

(b) No other groups below have three insert errors

(2) Example: Relatch at an intermediate power level

(a) Assume that RWM has been out of service and rods have been moved out of sequence. The following rod distribution exists:

(1) All rods in Group 1 thru 7 are at their withdraw limit, except rods 30-35, 38-43 and 38-27 (GP. 7) which are at position 02.

These 3 rods would cause an insert block if GP 8 were latched.

(2) All rods in Groups 8 and above are at their insert limit (04) except for rod 30-03 (GP 8) which is at position 06.

(3) No rod is selected

INSTRUCTOR NOTES

(1) The program will latch down (latch the next lower group) when all the rods in the presently latched group have been inserted to the group insert limit and a rod in the next group is selected.

NOTE: Will latch down if insert errors in GP is lower than latch GP.

(2) The program will latch up (latch the next higher group) upon selection of a rod within the next higher group provided that only 2 insert errors or less result from within the current latched group and/or any lower groups.

Will latch up provided that the number of insert errors produced will not give an insert block.

c. When sequence control is NOT selected, (OFF), latching automatically occurs based on rod movement within repeating BPWS banked groups (ex: 2/3/4/5/6 and 7/8/9/10/11/12).

(1) For example, if the rods in a group (GP. 4) are the same rods as in the next higher group (GP. 5), the RWM will NOT latch up based solely upon control rod selection. Latch up to Group 5 will automatically occur when any of the rods Group 4 are moved to a position defined for Group 5 provided that <3 insert errors would result.

(2) If the rods in a group (GP. 5) are the same rods as the next lower group (GP. 4), the RWM will not latch down based solely upon control rod selection. Latch down to the next lower RWM group will generally occur in this case based upon movement of any of the rods within the group to a position defined for the next lower RWM group.

With rods at both a GP 4 and GP 5 defined position, the latched GP after a movement will be the GP moved into.

(3) If the next rod group is NOT repeating, then latching occurs when the next rod is selected.

71. RO GENERIC 2.3.10 001/C/A/T3/GENERIC/C/A/G2.3.10//RO/SRO/BANK

Given the following conditions at a work site.

Airborne activity:	3 DAC
Radiation level:	40 mr/hr
Radiation level with shielding:	10 mr/hr
Time to place shielding:	15 minutes
Time to conduct task with respirator:	1 hour
Time to conduct task without respirator:	30 minutes

Assume the following:

- the airborne dose with a respirator will be zero (0).
- a dose rate of 40 mr/hr will be received while placing the shielding.
- all tasks will be performed by one worker.
- shielding can be placed in 15 minutes with or without a respirator.

Which ONE of the following would result in the lowest whole body dose?

A. Conduct the task WITHOUT a respirator or shielding.

B. Conduct the task with a respirator and WITHOUT shielding.

C. Place the shielding while wearing a respirator and conduct the task with a respirator.

D.✓ Place the shielding while wearing a respirator and conduct the task WITHOUT a respirator.

K/A Statement:

Radiation Control

2.3.10 Ability to perform procedures to reduce excessive levels of radiation and guard against personnel exposure.

K/A Justification: This question satisfies the K/A statement by requiring the candidate to calculate the expected exposure for a job and determine the correct precautions and radiological controls required to minimize exposure.

Level of Knowledge Justification: This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

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REFERENCE PROVIDED: None

Plausibility Analysis:

This question requires the candidate to calculate the exposure received for each of the four options in the distractors. Plausibility of distractors is based upon either calculation errors or misapplication of ALARA principles; and is based entirely on the type of decision which must be made while performing duties as a Licensed Operator. Using the calculation below, the candidate must correctly perform the analysis and apply ALARA principles to select the correct answer.

Calculations required:

3 DAC x 2.5 mr/DAC X 0.5 hours = 3.75 mr

a. 10 mr placing shielding, 10 mr conducting task, zero airborne = **20 mr**

b. 10 mr placing shielding, 5 mr conducting task, 3.75 mr airborne = **18.75 mr (lowest dose = Correct)**

c. 40 mr conducting task, zero airborne = **40 mr**

d. 20 mr conducting task, 3.75 mr airborne = **23.75 mr**

D - correct:

A - incorrect: Would NOT result in dose being maintained ALARA, based upon the calculations above.

B - incorrect: Would NOT result in dose being maintained ALARA, based upon the calculations above.

C - incorrect: Would NOT result in dose being maintained ALARA, based upon the calculations above.

72. RO GENERIC 2.3.9 001/MEM/T3/PR.CMPTR//RO GENERIC 2.3.9//RO/SRO/BANK 11/27/07 RMS

Unit 2 reactor shutdown is in progress and primary containment de-inerting has been authorized.

Which ONE of the following is the basis for preventing the simultaneous opening of BOTH the SUPPR CHBR ATM SPLY INBD ISOLATION VLV (2-FCV-64-19) and the DRYWELL ATM SUPPLY INBD ISOLATION VLV (2-FCV-64-18) during the performance of this evolution?

A. To prevent the high flow rate from damaging the non-hardened ventilation ducts.

B.✓ To prevent the possibility of overpressurizing the primary containment during a LOCA.

C. To prevent release of the drywell atmosphere through an unmonitored ventilation flow path.

D. To prevent creating a high differential pressure between the primary containment and the Reactor Building.

K/A Statement:

Radiation Control
2.3.9 Knowledge of the process for performing a containment purge.

K/A Justification: This question satisfies the K/A statement by requiring the candidate to use specific plant conditions and times to correctly determine the process for performing a containment purge.

References: 2-OI-64, Rev.106, section 8.1

Level of Knowledge Justification: This question is rated as MEM due to the requirement to recall or recognize discrete bits of information.

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REFERENCE PROVIDED: None

Plausibility Analysis:

In order to answer this question correctly the candidate must determine the requirements for de-inerting the Primary Containment and their bases.

B - correct:

A - incorrect: This is plausible becaue high flowrates would result from both valves being open. However, the vent ducts are designed to accomodate such flowrates.

C - incorrect: This is plausible since the vent path is unmonitored. However, having both valves open simultaneously provides NO additional path for a release.

D - incorrect: This is plausible because the de-inerting lineup raises the differential pressure between the Drywell and Reactor Building. However, the rise is relatively insignificant and well within the design limits.

8.0 INFREQUENT OPERATIONS

8.1 Purging the Drywell and Suppression Chamber with Primary Containment Purge Filter Fan

NOTES

1) TOE 970823 identified a potential for a bypass flow path to exist between the Drywell and Suppression Chamber when purging the Drywell and Suppression Chamber at the same time (both FCV-64-18 and 64-19 opened concurrently). Should a design basis LOCA occur with these two valves opened at the same time with the Reactor **NOT** in Cold Shutdown (Mode 4 or 5), a potential exists for overpressurizing primary containment due to the pressure suppression function being bypassed. Therefore, when Primary Containment purging is required with the Reactor **NOT** in Cold Shutdown (Mode 4 or 5), the Suppression Chamber and the Drywell are purged separately.

2) This section is used when purging both the Drywell and Suppression Chamber concurrently with the Reactor in Cold Shutdown (Mode 4 or 5).

3) When the Reactor is **NOT** in Cold Shutdown (Mode 4 or 5), the Suppression Chamber and the Drywell are purged separately.

[1] **REVIEW** all Precautions and Limitations in Section 3.0. ☐

[2] **VERIFY** all Prestartup/Standby Readiness requirements in Section 4.0 are satisfied. ☐

[3] **VERIFY** the following initial conditions are satisfied:

- Drywell vented to less than 0.25 psig. **REFER TO** Section 6.1. ☐
- H_2O_2 analyzers are in service **REFER TO** 2-OI-76 ☐
- Suppression Chamber vented to less than 0.25 psig. **REFER TO** Section 6.2. ☐
- Reactor Zone Fans in operation with Reactor Zone Supply and Exhaust Fan in fast speed. **REFER TO** 2-OI-30B. ☐

[4] **REQUEST** Chemistry to obtain a Drywell sample. **REFER TO** 2-SI-4.8.B.2-6. ☐

[5] **IF** sample is within limits of 2-SI-4.8.B.2-6, **THEN**

NOTIFY Shift Manager. ☐

73. RO GENERIC 2.4.47 001/C/A/T3/C4/6/G2.4.47//RO/SRO/BANK

Given the following plant conditions:

- Reactor pressure is being maintained at 50 psig.
- Temperature near the water level instrument run in the Drywell is 220 °F.
- The Shutdown Vessel Flooding Range Instrument (LI-3-55) is reading (+)35".

Which ONE of the following describes the highest Drywell Run Temperature at which the LI-3-55 reading (+)35 inches is considered valid?

REFERENCE PROVIDED

A. 200 °F

B.✓ 250 °F

C. 270 °F

D. 300 °F

K/A Statement:

Emergency Procedures /Plan
2.4.47 Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material.

K/A Justification: This question satisfies the K/A statement by requiring the candidate to use specific plant conditions to determine the correct reactor water level under emergency conditions.

References: 2-EOI-1 Flowchart

Level of Knowledge Justification: This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

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REFERENCE PROVIDED: 2-EOI-1 flowchart

Plausibility Analysis:

In order to answer this question correctly, the candidate must use EOI Caution #1 to determine operable RPV water level instruments.

B - correct:

A - incorrect: This is plausible since 200°F is a valid indication, however the question calls for the HIGHEST temperature.

C - incorrect: This is plausible if the candidate interpolates the Caution #1 table, however this is NOT permissible.

D - incorrect: This is plausible if the candidate interpolates the Caution #1 table, however this is NOT permissible.

CAUTIONS

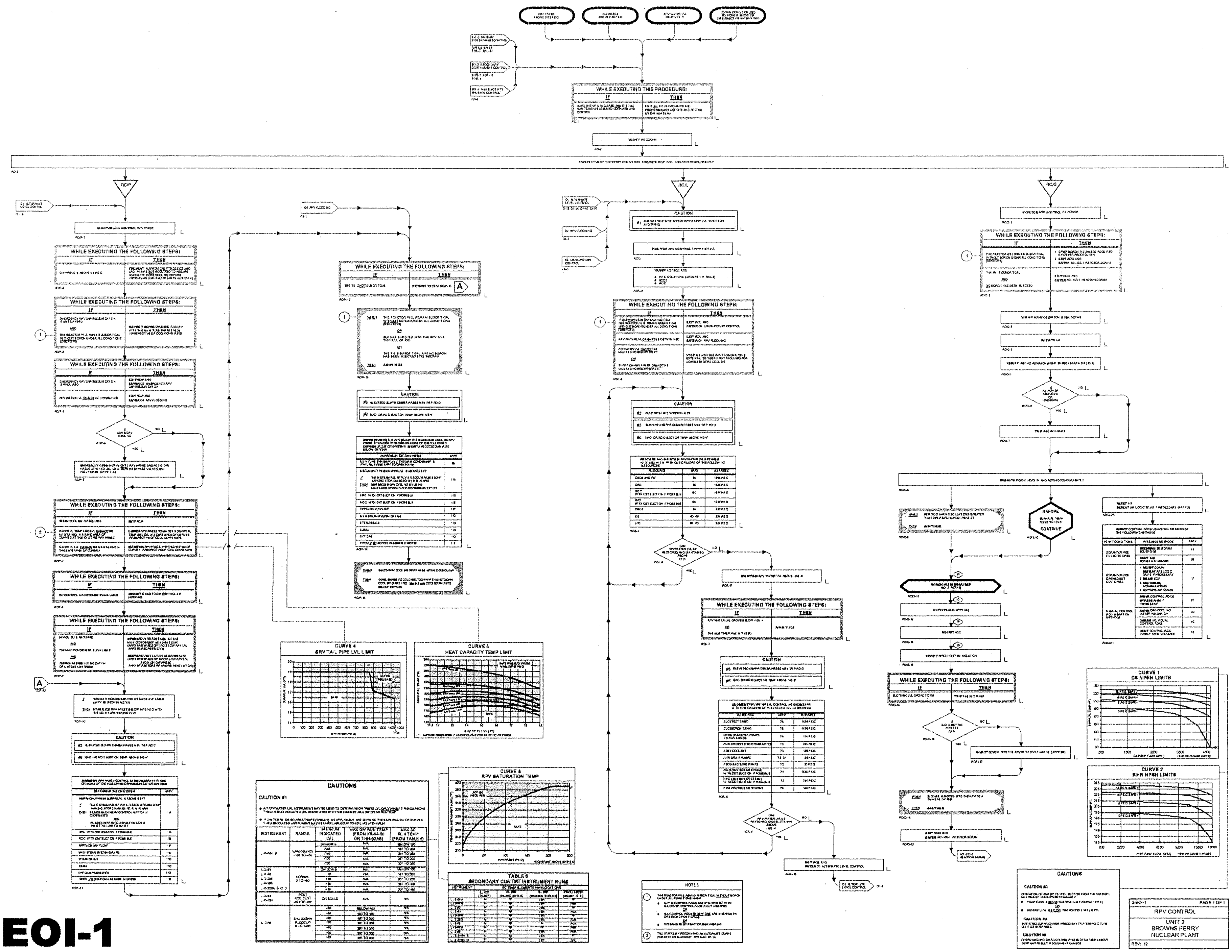
CAUTION #1

- AN RPV WATER LVL INSTRUMENT MAY BE USED TO DETERMINE OR TREND LVL ONLY WHEN IT READS ABOVE THE MINIMUM INDICATED LVL ASSOCIATED WITH THE HIGHEST MAX DW OR SC RUN TEMP.
- IF DW TEMPS, OR SC AREA TEMPS (TABLE 6), AS APPLICABLE, ARE OUTSIDE THE SAFE REGION OF CURVE 8, THE ASSOCIATED INSTRUMENT MAY BE UNRELIABLE DUE TO BOILING IN THE RUN.

INSTRUMENT	RANGE	MINIMUM INDICATED LVL	MAX DW RUN TEMP (FROM XR-64-50 OR TI-64-52AB)	MAX SC RUN TEMP (FROM TABLE 6)
LI-3-58A, B	EMERGENCY -155 TO +60	ON SCALE	N/A	BELOW 150
		-145	N/A	151 TO 200
		-140	N/A	201 TO 250
		-130	N/A	251 TO 300
		-120	N/A	301 TO 350
LI-3-53 LI-3-60 LI-3-206 LI-3-253 LI-3-208A, B, C, D	NORMAL 0 TO +60	ON SCALE	N/A	BELOW 150
		+5	N/A	151 TO 200
		+15	N/A	201 TO 250
		+20	N/A	251 TO 300
		+30	N/A	301 TO 350
LI-3-52 LI-3-62A	POST ACCIDENT -268 TO +32	ON SCALE	N/A	N/A
LI-3-55	SHUTDOWN FLOODUP 0 TO +400	+10	BELOW 100	N/A
		+15	100 TO 150	N/A
		+20	151 TO 200	N/A
		+30	201 TO 250	N/A
		+40	251 TO 300	N/A
		+50	301 TO 350	N/A
		+65	351 TO 400	N/A

EXAMINATION REFERENCE PROVIDED TO CANDIDATE

2-EOI-1 RPV CONTROL 2-EOI-1



EOI-1

74. RO GENERIC 2.4.15 001/MEM/T3///GENERIC 2.4.15//RO/SRO/NEW 10/16/07

Given the following plant conditions:

- Unit 2 has scrammed and multiple control rods failed to insert.
- The Unit Supervisor has entered EOI-1, "RPV Control", and C-5, "Level/Power Control."
- You have been designated to assist the crew by performing EOI Appendicies as they are assigned.

Which ONE of the following precludes the use of a handheld radio to communicate with Control Room personnel?

A. EOI Appendix 2 in the 2A Electrical Board Room.

B. EOI Appendix 16H at the '2C' 250V RMOV Board.

C.✓ EOI Appendix 1C in the Auxiliary Instrument Room.

D. EOI Appendix 1B on the Reactor Building 565' Elevation.

K/A Statement:

Emergency Procedures /Plan
2.4.15 Knowledge of communications procedures associated with EOP implementation.

K/A Justification: This question satisfies the K/A statement by requiring the candidate to demonstrate knowledge of communication requirements that apply during execution of Emergency Operating Instructions.

References: OPDP-1

Level of Knowledge Justification: This question is rated as MEM due to the requirement to recall or recognize discrete bits of information.

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REFERENCE PROVIDED: None

Plausibility Analysis:

In order to answer this question correctly the candidate must determine which of the given locations violates the requirements of OPDP-1, Conduct of Operation.

C - correct:

A - incorrect: This is plausible bacuse of the safety related equiment powered from 2A Electric Board Room; however, radio communication is authorized.

B - incorrect: This is plausible because of the safety related equipment fed from '2C' 250V RMOV Board; however, radio communication is authorized.

D - incorrect: This is plausible because of the proximity of the RPV level instrumentation; however, radio communication is authorized.

L. Exiting the EOIs

The operators remain in the EOIs until either directed out by the EOI or when the SM/US concludes that an emergency condition no longer exists. Exit from EOI-1 and associated contingency procedures always requires SM/US determination, since these procedures have no explicit exit to other plant procedures except from RC/Q to AOI-100-1. Appendix 100-1 should be reviewed prior to EOI exit to determine, restore, and document abnormal alterations that were established during EOI execution.

After exiting the EOIs the operator surveys the present plant conditions to ensure no reason for re-entry to the EOIs exist.

During EOI execution, a SAMG ENTRY IS REQUIRED condition may arise. Entry into and execution of Severe Accident Management Guidelines (SAMGs) are the responsibility of the SED in the TSC. Significant time may be required to man the TSC with the appropriate SAM Team members and turn over plant conditions between the control room and the TSC. The control room staff terminate execution of ALL EOI flowcharts ONLY when the SED declares that the SAM Team has assumed command and control. EOI appendices may continue in use as directed by the SAMGs.

During the time between the development of the SAMG ENTRY IS REQUIRED condition and the time of assumption of command and control by the TSC, the control room staff shall continue use of available EOI guidance to mitigate the event. Development of a SAMG ENTRY IS REQUIRED condition always requires entry into the SAMGs when the TSC SAM Team assumes command and control, even if plant conditions subsequently develop which seem to no longer satisfy a requirement to enter SAMGs.

3.5 Duties of the Control Room Team Members While Executing EOIs

The specific duties of the Control Room Team Members are outlined in Conduct of Operations.

3.6 Shift Communications During Execution of EOIs

The methodology associated with communications during execution of the EOIs is outlined in Conduct of Operations

3.7 Use of Instrumentation and ICS/Safety Parameter Display System (SPDS)

Various instruments in the control room are qualified for Post Accident Monitoring. These instruments are identified with black labels. During the performance of the EOIs, these instruments are required to be utilized as much as practical. For parameters that have multiple readouts in the control room, the operator should observe as many of the multiple readouts as practical for a verification of the values being observed.

Most instruments in the control room are provided with what may be considered standard scale divisions (increments of 1, 5, 10, etc.), although there are some that may be considered off-normal (increments of 2, 3, 4, etc.). Some pressure instruments may read out in PSIA rather than the more common value of PSIG.

The operator is required to remain aware of these possible differences when reading the values from the instruments. For pressure instruments, the pressure should be called out in values of PSIA or PSIG, as applicable. When the operator reading the flowchart asks for the value of a pressure parameter, it should be assumed that the value be given as PSIG unless he/she solicits the value in PSIA.

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Communications

d. When using the plant PA system:

(1) Speak slowly and deliberately in a normal tone of voice.

(2) When announcements of abnormal or emergency conditions are made, they shall be made at least twice.

(3) When making announcements for drills or exercises begin and end the announcement with "This is a Drill."

6. Plant Telephones

When using Plant telephones:

a. Identify yourself and watch station.

b. When trying to make contact with the main Control Room, if the message is of a routine nature, the sender should hang up when the main Control Room fails to answer after the fifth ring to avoid unnecessary Control Room noise. The phone shall be allowed to ring until answered if the information is important to Operations.

c. During times when the DO NOT DISTRUB (DND) function has been used by MCR personnel, follow the directions on the recording as appropriate.

d. When making announcements for drills or exercises begin and end the announcement with "This is a Drill."

7. Radio/phone Communication

Radio/phone usage shall not be allowed in areas where electronic interference with plant equipment may result.

a. When making announcements for drills or exercises, begin and end the announcement with "This is a Drill."

b. Sender should identify themselves by watch station.

c. Three way communications should be used.

d. Clear concise language should be used since radio/phone contact does not have the advantage of face to face communication.

75. RO GENERIC 2.4.8 001/MEM/T3///GENERIC 2.4.8//RO/SRO/NEW 10/16/07

Which ONE of the following describes the use of Event-Based procedures during Symptom-Based Emergency Operating Instructions (EOI) execution?

Event-Based procedures are ____.

A. NOT used during Symptom-Based EOI execution.

B. ONLY used if specifically directed by an EOI flowchart step.

C✓ ONLY used if they do not interfere with EOI implementation.

D. ALWAYS used if equipment or plant status require their implementation.

K/A Statement:

Emergency Procedures /Plan
2.4.8 Knowledge of how the event-based emergency/abnormal operating procedures are used in conjunction with the symptom-based EOPs.

K/A Justification: This question satisfies the K/A statement by requiring the candidate to demonstrate knowledge of procedure hiearchy during execution of Emergency Operating Instructions.

References: EOIPM Section 0-VIII-A

Level of Knowledge Justification: This question is rated as MEM due to the requirement to recall or recognize discrete bits of information.

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REFERENCE PROVIDED: None

Plausibility Analysis:

In order to answer this question correctly, the candidate must deterine the rules for using Event-Based procedures during EOI execution.

C - correct:

A - incorrect: This is plausible based on the contradiction often found between Event-Based and Symptom-Based guidance. However, their use is permitted under controls circumstances.

B - incorrect: This is plausible because several EOI steps direct actions in accordance with Event-Based procedures. However, it is NOT a prerequisite to their use.

D - incorrect: This is plausible because no specific Event-Based procedure is expressly prohibited from use. However, if a conflict exists between the Event-Based procedure and the EOI, the EOI takes precedence.

You have completed the test!

I. EOI Flowchart Use With Other Plant Procedures

The EOIs are entered, based upon specific conditions symptomatic of emergencies, or conditions that could degrade into emergencies. Therefore the operator actions, provided within the EOIs, allow the operator to mitigate the consequences of a broad range of accidents and multiple equipment failures.

Other procedures, such as AOIs, ARPs, EPIPs, etc., have event specific entry conditions and may be used to supplement EOIs. In some instances the EOIs will direct the operators to the unit operating procedures (OIs, GOIs, and AOIs) for completion of specific tasks. Usually, the EOIs direct the operators to specific EOI Appendices. The Appendices are specific task related procedures written to satisfy directives given within the EOIs.

Actions that contradict any direction given by the EOIs, or reduce the effectiveness of any directions given by the EOIs, WILL NOT be implemented for any reason.

The exception to this rule are the SSIs and AOI-100-2. The conditions which cause entry into the SSIs are such that the reliability of the information systems required to execute the EOIs are no longer at a confidence level that would make the EOIs effective. Any time that the operators must leave the control room, as directed by AOI-100-2, the EOIs shall be exited and AOI-100-2 shall be used to shut down and cool down the reactor. The EOIs are not designed, or written, to support their use outside of the main control room.

Conditions may arise under Station Blackout (SBO) conditions in which the rate of RPV cooldown is reduced, or alternate Heat Capacity Temperature Limit or Pressure Suppression Pressure curves are appropriate to avoid an unnecessary emergency depressurization, in order to maintain RCIC injection capability. The TSC staff or an associated abnormal operating instruction may recommend use of these alternate curves, which have been calculated as part of EOIPM section 2- or 3-VI-F and -H. These alternate curves meet the assumptions used within the EOIs.

It is recognized that during execution of the EOIs the control room will receive assistance from various support groups. This is especially the case under conditions in the EPIPs that result in the Technical Support Center (TSC) being staffed. For example, the TSC may make recommendations regarding when it is best to vent primary containment, based upon present or predicted meteorological conditions. This would not contradict the directions provided by the EOIs, but help to meet the intent of minimizing radiological releases to the general public.

J. Execution of EOI Appendixes

The EOIs rely heavily upon the EOI Appendices to implement EPG and PSTG actions and tasks that are too involved to outline on the flowchart procedure. These tasks include the defeating of various interlocks and logic systems. The steps within the Appendices involve the removing of fuses, placing jumpers across terminals, and placing boots on relay contacts, as well as some of the more common functions such as opening and closing valves and operation of systems to support the EOI flowchart procedure steps.