1. Unit 3 is operating at 100% Rated Thermal Power (RTP) when a Loss of Offsite Power occurs.

   Which **ONE** of the following completes the statement below in accordance with 0-AOI-57-1A, Loss of Offsite Power (161 and 500 KV)/Station Blackout?

   Diesel Generators (EDGs) are required to automatically start and tie on to their respective 4KV Shutdown Boards within __________ seconds.

   A. 5

   B. 6

   C. 10

   D. 14

2. An event occurs on Unit 3 that results in 480V Load Shed. In accordance with 3-AOI-57-1D, 480V Load Shed, if the load shed logic cannot be reset, Battery Charger 3 may be returned to service by placing the charger select switch in ___**(1)**___.

If Battery Charger 3 will not reset, Battery Charger ___**(2)**___ may be used as a spare.

A. **(1)** EMERG
(2) 2A

B. **(1)** EMERG
(2) 2B

C. **(1)** OFF and then back ON
(2) 2A

D. **(1)** OFF and then back ON
(2) 2B

3. Which **ONE** of the following completes the statement below relative to the frequency of panel walkdowns in accordance with OPDP-1, Conduct of Operations?

   The Unit Operator is to perform a panel walk down a minimum of once every __________.

   A. 1 hour

   B. 2 hours

   C. 4 hours

   D. 6 hours

4. Which **ONE** of the following completes the statement below?

   Regarding 4KV Start Bus transfer schemes, the _______ device generates a bus lockout and **MUST** be cleared prior to transfer.

   A. 27

   B. 51

   C. 52

   D. 86

5. All three Units are operating at 100% RTP when the following conditions occur:

- 0-RM-90-259A, U1/U2 Control Room Ventilation Radiation Monitor indicates 120 cpm
- 0-RM-90-259B, U3 Control Room Ventilation Radiation Monitor indicates 230 cpm

Given the conditions above, which **ONE** of the following completes the statements below?

The Control Room Ventilation System will automatically isolate on ___**(1)**____.

Control Room Emergency Ventilation (CREV) will automatically respond with ___**(2)**_____ CREV fan(s) starting.

A. **(1)** Unit 3 **ONLY**
(2) ONLY the selected

B. **(1)** Unit 3 **ONLY**
(2) BOTH

C. **(1) ALL** three (3) Units
(2) ONLY the selected

D. **(1) ALL** three (3) Units
(2) BOTH

6. Which **ONE** of the following completes the statement below in regards to Simulated Thermal Power (STP)?

STP consists of an average of all Local Power Range Monitor (LPRM) signals from the Average Power Range Monitor (APRM) Channels with a _________ second filter applied.

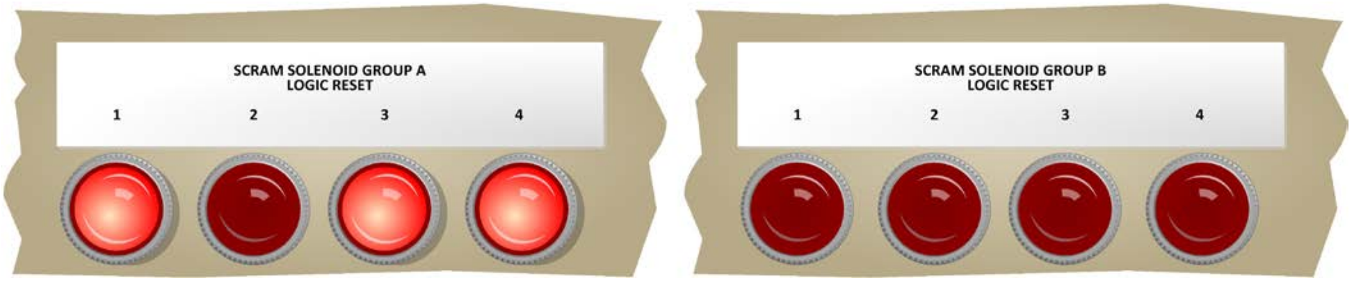
A. 2

B. 6

C. 10

D. 37

7.



Given that Reactor Protection System (RPS) Bus 'B' power has just been lost in conjunction with a **PREVIOUSLY** blown fuse on Group A, which **ONE** of the following completes the statements below?

In this condition, the loss of RPS Bus 'B' will **DIRECTLY** cause ___**(1)**___ of the Control Rods to SCRAM.

This condition will cause ___**(2)**___ Scram Discharge Volume(s) (SDV) to fill.

A. **(1)** one quarter (1/4)
   (2) BOTH

B. **(1)** one quarter (1/4)
   (2) ONLY the WEST

C. **(1)** one half (1/2)
   (2) BOTH

D. **(1)** one half (1/2)
   (2) ONLY the WEST

8. Which **ONE** of the following completes the statement below?

The **NORMAL** electrical power supply to Unit 1's Condensate Pump 1A and Condensate Booster Pump 1A is ___________ via a 4KV Unit Board.

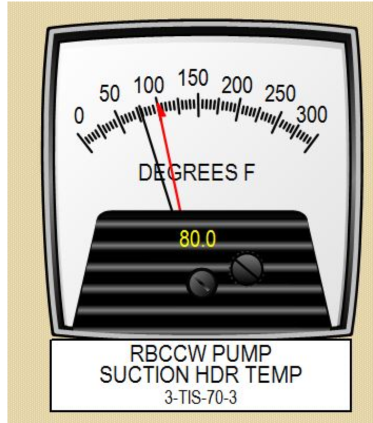
A. 4KV Start Bus 1A

B. 4KV Start Bus 1B

C. Unit Station Service Transformer 1A

D. Unit Station Service Transformer 1B

9. Which **ONE** of the following completes the statements below?



The 3-TIS-70-3, RBCCW PUMP SUCTION HEADER TEMP indicator is located on Panel ___**(1)**___.

In relation to the Reactor Building Closed Cooling Water (RBCCW) System, Fuel Pool Cooling Heat Exchangers are considered ___**(2)**___ loop loads.

A. **(1)** 3-9-4
(2) essential

B. **(1)** 3-9-4
(2) non-essential

C. **(1)** 3-9-21
(2) essential

D. **(1)** 3-9-21
(2) non-essential

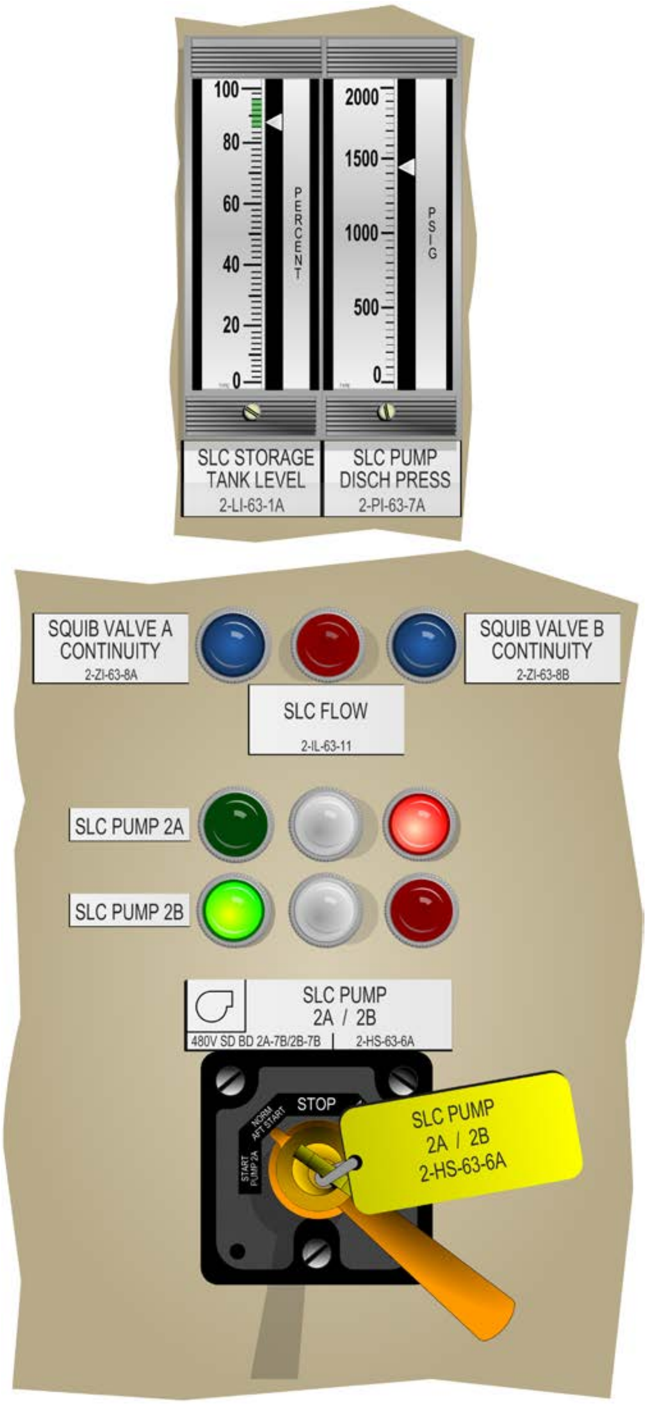
10. Given the following conditions and indications on Unit 2:

- An ATWS required the execution of 2-EOI Appendix-3A, SLC Injection
- Reactor Pressure is 995 psig

Which **ONE** of the following completes the statements below?

SLC ____**(1)**____ currently injecting to the Reactor.

The analog level and pressure indicators, included here, are located on Panel ____**(2)**____.



A. **(1)** is
(2) 2-9-5

B. **(1)** is
(2) 2-9-4

C. **(1)** is NOT
(2) 2-9-5

D. **(1)** is NOT
(2) 2-9-4

11. A **Unit 1** Operator is walking down Panel 1-9-3 in preparation for shift turnover.

In accordance with BFN-ODM-4.5, Operator Aids and Operator Information Systems, which **ONE** of the following types of hand switch tags would indicate to the **Unit 1** Operator that Panel 1-9-3 components are aligned to support Shutdown Cooling on **Unit 2**?

A. Blue Tag

B. Green Tag

C. Hot Pink Tag

D. Orange Tag

12. Unit 2 has experienced a loss of Drywell Control Air. Which **ONE** of the following completes the statements below in regards to Main Steam Relief Valves?

To assure that the valves can be held open following a failure of the air supply, ___**(1)**___ are equipped with accumulators.

In accordance with 2-AOI-32A-1, Loss of Drywell Control Air, accumulators are designed to hold sufficient air to allow a **MINIMUM** of five (5) ____**(2)**____.

Note: Automatic Depressurization System (ADS)
Main Steam Relief Valve (SRV)

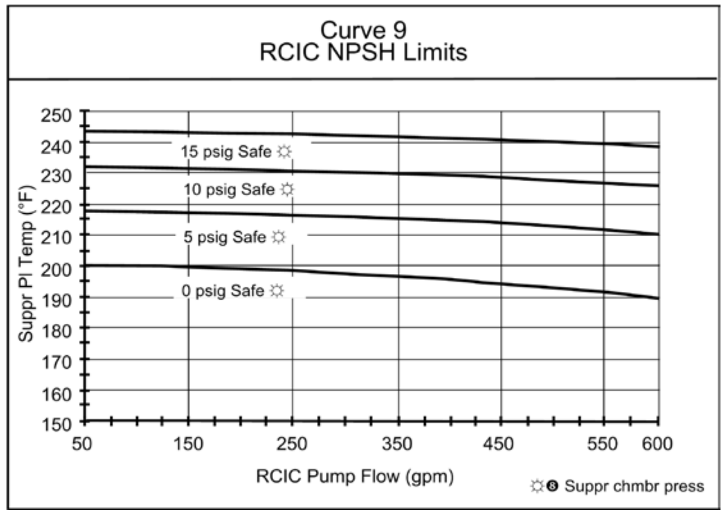
A. **(1) ONLY** the ADS SRVs
(2) hours of operations

B. **(1) ONLY** the ADS SRVs
(2) valve operations

C. **(1) ALL** SRVs
(2) hours of operations

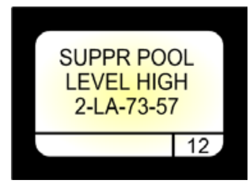
D. **(1) ALL** SRVs
(2) valve operations

13. Unit 2 suffered a small break Loss of Coolant Accident (LOCA), with the following conditions:



- Suppression Pool Bulk Temperature is 210°F
- Suppression Chamber Pressure is 5 psig
- RCIC is currently taking suction from the Condensate Storage Tank (CST) at exactly **ONE-HALF** of normal rated flow

Subsequently, SUPPRESSION POOL LEVEL HIGH (2-9-3F, Window 12) alarms.



Which **ONE** of the following completes the statement below in accordance with the EOI-5 Cautions?

Assume **NO** Operator action has been taken.

Operating RCIC under these current conditions, __________.

A. **NO** equipment damage would be anticipated

B. equipment damage from cavitation **ONLY** would be anticipated

C. equipment damage from inadequate lube oil cooling **ONLY** would be anticipated

D. equipment damage from **BOTH** cavitation and inadequate lube oil cooling would be anticipated

14. The following conditions exist on Unit 3:

- A Reactor SCRAM occurred and Control Rods failed to insert
- Initial Reactor Power following the SCRAM was 23%
- Initial SLC Storage Tank Level was 87%

Current conditions:

- SLC is injecting into the Reactor and SLC Storage Tank Level is 74%
- Reactor Water Level is (-) 90 inches
- All APRMs are DOWNSCALE
- SRM Period indication is (-) 40 seconds
- IRMs are inserted, indicating on Ranges 4 and 5

Which **ONE** of the following completes the statements below?

At this time, the Reactor ___**(1)**___ subcritical.

If, at the current SLC Tank Level, Reactor Water Level is restored to (+) 2 to (+) 51 inches, the Reactor will be ___**(2)**___.

A. **(1)** is
(2) critical

B. **(1)** is
(2) subcritical

C. **(1)** is NOT
(2) critical

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

15. The Shift Manager has directed entering 3-AOI-100-2, Control Room Abandonment, due to heavy smoke in the Unit 3 Main Control Room (MCR).

Which **ONE** of the following completes the statements below in accordance with 3-AOI-100-2, Control Room Abandonment?

The Immediate Actions are taken to SCRAM the Reactor and to place the Unit in the most stable configuration possible to ___**(1)**___.

If the Reactor fails to SCRAM during the performance of Immediate Actions due to an Electrical Anticipated Transient Without SCRAM (ATWS), continue in 3-AOI-100-2 since the **PRIORITY** in the subsequent actions will be to ___**(2)**___.

A. (1) allow time to prepare for plant cooldown
(2) vent the overpiston area

B. (1) allow time to prepare for plant cooldown
(2) pull RPS Scram Solenoid Fuses

C. (1) limit the heat load on RBCCW
(2) vent the overpiston area

D. (1) limit the heat load on RBCCW
(2) pull RPS Scram Solenoid Fuses

16. Unit 2 is operating at 25% RTP with a Reactor Shutdown in progress.

Which **ONE** of the following completes the statements below?

Given that the Main Turbine trips, Extraction Non-Return Valves automatically close in order to prevent Main Turbine ___**(1)**___.

Feedwater Temperature will ___**(2)**___ as a result of the Main Turbine trip.

A. **(1)** overspeed
(2) lower

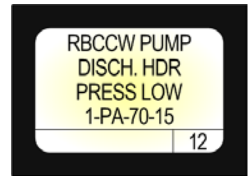
B. **(1)** overspeed
(2) remain the same

C. **(1)** overheating
(2) lower

D. **(1)** overheating
(2) remain the same

17. Unit 1 is operating at 100% RTP, when RBCCW Pump 1A trips, resulting in the following:

- RBCCW PUMP DISCHARGE HEADER PRESSURE LOW, (1-9-4C, Window 12), in alarm



Given the conditions above, which **ONE** of the following system loads is still being cooled by RBCCW?

Note: Reactor Water Cleanup (RWCU)

A. RWCU Non-Regenerative Heat Exchangers

B. Reactor Recirculation Pump Motor Coolers

C. RWCU Pump Seal Water and Bearing Oil Coolers

D. Reactor Building Equipment Drain Sump Heat Exchanger

18. Which **ONE** of the following completes the statement below pertaining to High Drywell Pressure Primary Containment Isolation System (PCIS) isolations?

PCIS Group 3, RWCU (Reactor Water Cleanup System) ___**(1)**___ isolate
and PCIS Group 8, TIP (Traverse Incore Probe System) ___**(2)**___ isolate.

A. **(1)** will
(2) will

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

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

D. **(1)** will NOT
(2) will NOT

19. Unit 1 is operating at 100% Rated Thermal Power (RTP). Upon completion of 1-SR-3.4.2.1, Jet Pump Mismatch And Operability, it was determined that Jet Pump No. 7 is INOPERABLE.

An engineering evaluation has determined that Jet Pump #7 has failed mechanically.

Given the conditions above, which **ONE** of the following correctly describes the reason behind the required Tech Spec action to be in MODE 3 in 12 hours?

A. Raises the probability of thermal hydraulic instability events at lower Power levels during low flow conditions.

B. Changes core neutron flux distribution due to the change in Core Flow, making the APRM indications unreliable.

C. Reduces the capability of re-flooding to two thirds (2/3) Core Height following a Loss of Coolant Accident (LOCA).

D. Causes the APRM Flow Biased SCRAM and Rod Block setpoints to drift due to the impact of flow changes in the affected loop.

20. Unit 2 is being shut down for an outage with the following conditions:

- 2-HS-99-5A-S1, REACTOR MODE SWITCH is in SHUTDOWN
- Reactor Vessel Head Closure Bolts are still fully tensioned
- Residual Heat Removal (RHR) Loop I is in Shutdown Cooling

Subsequently, a complete Loss of Shutdown Cooling occurs and results in the following Reactor Coolant Temperature response:

TIME	Reactor Coolant Temperature (°F)
0800	113 °F
0802	116 °F
0804	119 °F

Given the conditions above, which **ONE** of the following completes the statements below?

The current Heatup Rate is ___**(1)**___the limit specified in 2-SR-3.4.9.1(1), Reactor Heatup and Cooldown Rate Monitoring.

Given that all other associated MODE requirements remain unchanged and based upon the constant trend, ___**(2)**___ is the **EARLIEST** time that Unit 2 will enter MODE 3 due to the rising Reactor Coolant Temperature.

A. **(1)** below
(2) 0858

B. **(1)** below
(2) 0906

C. **(1)** above
(2) 0858

D. **(1)** above
(2) 0906

21. Unit 2 SCRAMs with the following current conditions:

- Main Turbine Bypass Valves are **NOT** available
- Reactor Pressure is 990 psig and rising
- Reactor Water Level is 30 inches and stable using Reactor Core Isolation Cooling (RCIC)

Which **ONE** of the following completes the statement below in accordance with the EOI Program Manual?

Given the conditions above, the advantage of using High Pressure Coolant Injection (HPCI) in Pressure Control Mode **STRICTLY**, versus the SRVs for Reactor Pressure Control, is

__________.

A. that HPCI can lower Reactor Pressure **MORE** quickly than a single SRV being opened

B. the injection of relatively cold water into the Reactor, **IN THIS MODE,** aids in lowering Reactor Pressure

C. **MORE** than one SRV would be required to be opened to initiate a cooldown thus adding more heat to the Suppression Pool

D. the Suppression Pool would heat up **LESS,** for an equivalent amount of steam, due to the steam transferring some of its energy into HPCI Turbine work

22. Unit 1 was operating at 100% RTP when an event occurred resulting in the following conditions:

- Drywell Pressure is 5 psig and rising
- Drywell Temperature is 188 °F and rising
- Suppression Chamber Pressure is 6 psig and rising
- Suppression Chamber Temperature is 216 °F and rising
- SRVs are being cycled for Reactor Pressure Control
- Suppression Pool Water Level is 13 feet and lowering

Which **ONE** of the following completes the statements below?

Given the conditions above, the ____**(1)**____ will be uncovered **FIRST**.

____**(2)**____ Sprays will be more effective at mitigating the **CURRENT** Primary Containment conditions.

A. **(1)** HPCI Turbine exhaust
(2) Drywell

B. **(1)** HPCI Turbine exhaust
(2) Suppression Chamber

C. **(1)** Downcomer opening
(2) Drywell

D. **(1)** Downcomer opening
(2) Suppression Chamber

23. Which **ONE** of the following completes the statement below in accordance with the **BASES** for Technical Specification 3.6.2.2, Suppression Pool Water Level?

The essential accident / transient mitigating feature associated with Suppression Pool Water Level is providing **___(1)___**.

Additionally, Suppression Pool Water Level below **___(2)___WITH DIFFERENTIAL PRESSURE CONTROL ESTABLISHED** will invalidate the Safety Analysis Initial Conditions.

A. **(1)** adequate steam quenching
(2) (-) 6.25 inches

B. **(1)** adequate steam quenching
(2) (-) 7.25 inches

C. **(1)** an emergency water supply to ECCS
(2) (-) 6.25 inches

D. **(1)** an emergency water supply to ECCS
(2) (-) 7.25 inches

24. An ATWS has occurred on Unit 2.

- ATWS actions are in progress
- Reactor Water Level currently indicates (+) 40 inches
- Reactor Power is 46%
- Standby Liquid Control (SLC) is injecting

Given the conditions above, which **ONE** of the following completes the statement below?

2-EOI-1A, ATWS RPV CONTROL, requires Operators to STOP and PREVENT **ALL** injection into the Reactor **EXCEPT** **___(1)___** to mitigate the consequences of the failure to SCRAM by **___(2)___,** which adds negative reactivity.

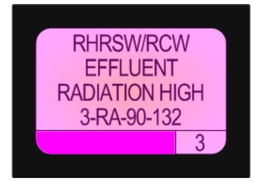
A. **(1)** CRD, and SLC **ONLY**
(2) increasing natural circulation to mix the injected boron

B. **(1)** CRD, and SLC **ONLY**
(2) reducing natural circulation resulting in increased void fraction in the core

C. **(1)** RCIC, CRD, and SLC **ONLY**
(2) increasing natural circulation to mix the injected boron

D. **(1)** RCIC, CRD, and SLC **ONLY**
(2) reducing natural circulation resulting in increased void fraction in the core

25. Unit 3 is at 100% RTP when the following alarm is received:



- RHRSW/RCW EFFLUENT RADIATION HIGH (3-9-3A, Window 3)

Which **ONE** of the following completes the statement below?

A probable cause for this alarm is __________.

A. tube leaks in the RBCCW Heat Exchanger

B. tube leaks in the RHR Pump Seal Heat Exchanger

C. tripped RHR Service Water (RHRSW) Sample Pump

D. tube leaks in the Reactor Water Cleanup (RWCU) Heat Exchanger

26. A fire has occurred in the Unit 3 Reactor Building.

Which **ONE** of the following completes the statement below?

In accordance with the **NOTES and CAUTIONS** of 0-AOI-26-1, Fire Response, the reason Assistant Unit Operators (AUOs) report to their assigned Control Room is to __________.

Note: Self-Contained Breathing Apparatus (SCBA)
Fire Safe Shutdown (FSS)

A. retrieve necessary SCBA kits

B. perform FSS Recovery Actions

C. perform Personnel Accountability

D. retrieve the Hard Hat head lamps

27. Unit 3 is operating at 100% RTP when the following conditions occur:

- The Transmission System Operator (TOp) notifies the Shift Manager (SM) that the 500KV System voltage is 508 KV
- GENERATOR LOAD is 1240 MWe
- GENERATOR MVAR is (+) 200 MVAR
- GENERATOR HYDROGEN PRESSURE is 60 psig

Given the conditions above, which **ONE** of the following completes the statements below?

The current System voltage ___**(1)**___ within **NORMAL** limits in accordance with 0-AOI-57-1E, Grid Instability.

The generator ___**(2)**___ operating within the limitations of the Generator Capability Curve.

[REFERENCE PROVIDED]

A. **(1)** is
(2) is

B. **(1)** is
(2) is NOT

C. **(1)** is NOT
(2) is

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

28. Which **ONE** of the following completes the statements below?

In accordance with EOI-1A, ATWS RPV Control, Reactor Water Level is lowered to (-) 50 inches in order to maintain Reactor Water Level ___**(1)**___ the Feedwater Spargers.

In accordance with AOI-100-1, Reactor SCRAM Hard Card during the execution of ATWS actions, when Reactor Water Level reaches (-) 50 inches, the Operator is required to **REPORT** ___**(2)**___ to the Unit Supervisor.

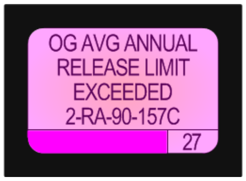
A. **(1)** below
(2) Reactor Water Level **ONLY**

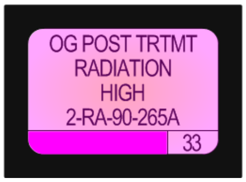
B. **(1)** below
(2) Reactor Water Level **AND** Reactor Power

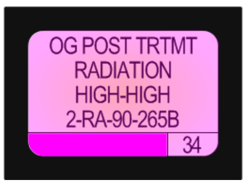
C. **(1)** above
(2) Reactor Water Level **ONLY**

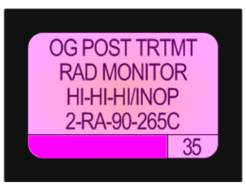
D. **(1)** above
(2) Reactor Water Level **AND** Reactor Power

29. In accordance with the Unit 2 Alarm Response Procedures, which **ONE** of the following, when alarming, requires that Operators ensure 2-FCV-66-28, OFFGAS SYSTEM ISOLATION VALVE, is CLOSED?

A. 

B. 

C. 

D. 

30. The following plant conditions exist on Unit 2:

- The Reactor is in MODE 4
- RHR Loop I is in Shutdown Cooling
- RHR Loop II is in Suppression Pool Cooling

During the performance of a Reactor Water Level Surveillance, the Instrument Mechanics (IMs) inadvertently cause a Primary Containment Isolation System (PCIS) Group 2 Isolation.

Which **ONE** of the following completes the statements below?

RHR Loop I ___**(1)**___ remain in Shutdown Cooling.

RHR Loop II ___**(2)**___ remain in Suppression Pool Cooling.

A. **(1)** will
(2) will

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

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

D. **(1)** will NOT
(2) will NOT

31. Unit 3 has suffered a small break LOCA and HPCI is injecting.

Suppression Pool Water Level rises to (+) 5.3 inches.

Given the conditions above, which **ONE** of the following completes the statement below in accordance with 3-OI-73, High Pressure Coolant Injection System?

The HPCI suction path will **___(1)___** transfer to the Suppression Pool **AND** the Condensate Storage Tank (CST) suction path will **___(2)___** after the new suction path is established.

A. **(1)** automatically
(2) automatically close

B. **(1)** automatically
(2) require manual closure

C. **(1)** require manual
(2) automatically close

D. **(1)** require manual
(2) require manual closure

32. Which **ONE** of the following completes the statement below?

___(1)____ of the **EMERGENCY** Range Level Instrument indication(s) is/are affected by High RWCU Heat Exchanger Room Temperatures.

EMERGENCY Range Level Instruments **___(2)____** located on Panel 3-9-5?

[REFERENCE PROVIDED]

A. **(1)** One
(2) are

B. **(1)** One
(2) are **NOT**

C. **(1)** None
(2) are

D. **(1)** None
(2) are **NOT**

33. The following conditions are observed on Unit 1:

<u>1-RM-90-140/142</u>

- Reactor Zone 1-RM-90-142A indicates 65 mR/hr
- Reactor Zone 1-RM-90-142B indicates 67 mR/hr
- Refuel Zone 1-RM-90-140A indicates 75 mR/hr
- Refuel Zone 1-RM-90-140B indicates 78 mR/hr

<u>1-RM-90-141/143</u>

- Reactor Zone 1-RM-90-143A indicates 68 mR/hr
- Reactor Zone 1-RM-90-143B indicates downscale
- Refuel Zone 1-RM-90-141A indicates 70 mR/hr
- Refuel Zone 1-RM-90-141B indicates 69 mR/hr

Which **ONE** of the following identifies the Ventilation System response?

A. Refuel Zone isolation **ONLY**

B. Reactor **AND** Refuel Zone isolation

C. Refuel Zone isolation **AND** CREV auto initiation

D. Reactor Zone isolation **AND** CREV auto initiation

34. Which **ONE** of the following completes the statements below?

Entry into EOI-3, Secondary Containment Control, is required when **ANY** Secondary Containment Area Water Level is above ___**(1)**___.

In accordance with EOI-3, ___**(2)**___ is required when a Primary System is discharging into Secondary Containment and Secondary Containment Water Level exceeds Max Safe in two or more areas.

A. **(1)** 2 inches
(2) a normal Reactor Shutdown

B. **(1)** 2 inches
(2) Emergency Depressurization

C. **(1)** 66 inches
(2) a normal Reactor Shutdown

D. **(1)** 66 inches
(2) Emergency Depressurization

35. Unit 2 was operating at 100% RTP when plant events resulted in the following:

- Reactor Pressure is 405 psig
- Reactor Water Level is (-) 140 inches
- 480V RMOV Board 2D is de-energized
- Assume **NO** Operator actions has been taken

Given the conditions above, which **ONE** of the following completes the statement below?

2-FCV-74-52, RHR SYS I LPCI **OUTBOARD** INJECTION VALVE is ___**(1)**___ and
2-FCV-74-53; RHR SYS I LPCI **INBOARD** INJECTION VALVE is ___**(2)**___.

A. **(1)** CLOSED
(2) CLOSED

B. **(1)** CLOSED
(2) OPEN

C. **(1)** OPEN
(2) CLOSED

D. **(1)** OPEN
(2) OPEN

36. Unit 1 is preparing for a Refueling outage with the following conditions:

- RHR SYS I FLOW is 7500 gpm for Shutdown Cooling (SDC)
- **NO** Recirc Pumps are in service
- Reactor Coolant Temperature is 175 °F
- **NO** other testing or evolutions are in progress

Which **ONE** of the following completes the statements below?

Given the conditions above, in accordance with 1-OI-74, Residual Heat Removal System, Reactor Water Level should be maintained ____**(1)**____.

The purpose of maintaining this SDC Reactor Water Level is to ____**(2)**____.

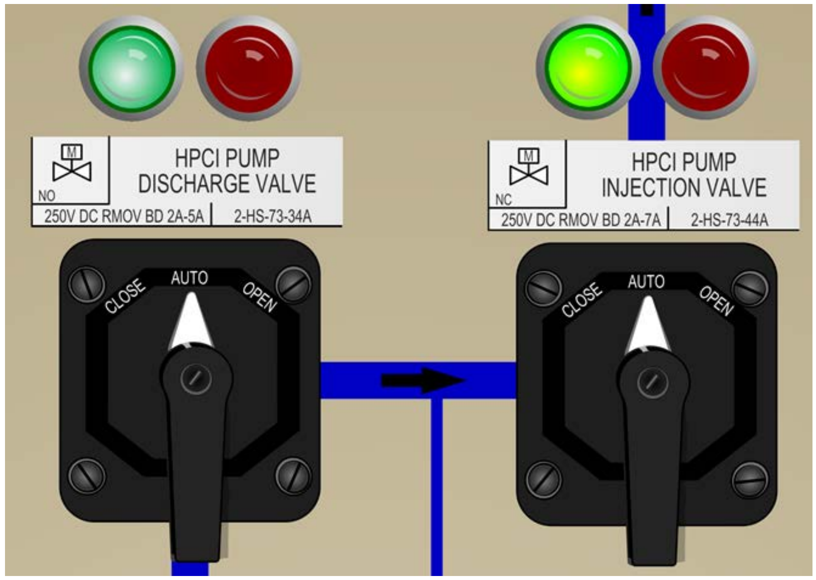
A. **(1)** < 70 inches
(2) prevent thermal stratification

B. **(1)** < 70 inches
(2) prevent jet pump cavitation

C. **(1)** 70 to 90 inches
(2) prevent thermal stratification

D. **(1)** 70 to 90 inches
(2) prevent jet pump cavitation

37. Given that an Operator error has resulted in the Unit 2 HPCI System lineup detailed as indicated **AND** the "VERIFY HPCI is in Standby Readiness" step has been circle/slashed as completed, which **ONE** of the following completes the statements below?



If the HPCI System were to be **MANUALLY** started (for Reactor Pressure Vessel (RPV) injection) with the given lineup using 2-OI-73, HPCI Injection System Lineup Hard Card, HPCI ___**(1)**___ inject to the RPV.

If the HPCI System were to receive an **AUTOMATIC** start signal, HPCI ___**(2)**___ inject to the RPV.

A. **(1)** would
(2) would NOT

B. **(1)** would
(2) would

C. **(1)** would NOT
(2) would NOT

D. **(1)** would NOT
(2) would

38. A LOCA occurred on Unit 3 resulting in the following conditions:

- Reactor Water Level reaches (-) 122 inches and is slowly lowering
- ‘3B’ EDG fails to start

Given the conditions above, which **ONE** of the following completes both statements below?

Core Spray Pump 3A ___**(1)**___.

Core Spray Pump 3D starts in ___**(2)**___.

A. **(1)** starts automatically
(2) 7 seconds

B. **(1)** starts automatically
(2) 21 seconds

C. **(1)** can be manually started **ONLY**
(2) 7 seconds

D. **(1)** can be manually started **ONLY**
(2) 21 seconds

39. Which **ONE** of the following completes the statement below?

Unit 1 is at 100% RTP performing 1-SR-3.1.7.2, Continuity Verification of Explosive Charges in the SLC Injection Valves. This surveillance is performed every ___**(1)**___ and Squib Valve Continuity Amperage can be observed at ___**(2)**___.

A. (1) 24 hours
(2) the back of Panel 1-9-5

B. (1) 24 hours
(2) Panel 1-PNLA-925-0057 in the Unit 1 Auxiliary Instrument Room

C. (1) 31 days
(2) the back of Panel 1-9-5

D. (1) 31 days
(2) Panel 1-PNLA-925-0057 in the Unit 1 Auxiliary Instrument Room

40. Which **ONE** of the following completes the statement in regards to the SLC System?

During **NORMAL** operation, Sodium Pentaborate is maintained in solution by the use of **________**.

A. mechanical tank agitation **ONLY**

B. tank heaters **ONLY**

C. piping heat tracing **ONLY**

D. mechanical tank agitation **AND** tank heaters

41. Which **ONE** of the following completes the statement below pertaining to the Unit 2 ADS automatic initiation logic requirements?

The **MINIMUM** required ECCS pump(s) permissive will be met when __________ is/are running.

A. **ANY** RHR Pump

B. **ANY** Core Spray Pump

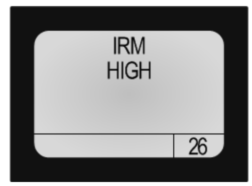
C. RHR Pumps 2C **AND** 2D

D. Core Spray Pumps 2C **AND** 2D

42. **UNIT 2** is conducting a Reactor Startup and is currently in MODE 2.

Intermediate Range Monitor (IRM) 'B' is currently on Range 7 and reading as indicated.

Given the conditions above, which **ONE** of the following completes the statements below?



If the Range Switch for IRM 'B' is turned counterclockwise to the 'Range 6' position, the resultant reading on IRM 'B' will be larger by a factor of ___**(1)**___.

As a result of the switch being 'ranged down', IRM HIGH (2-9-5A, Window 26) annunciator ___**(2)**___ illuminate.

A. **(1)** 2.3
(2) will

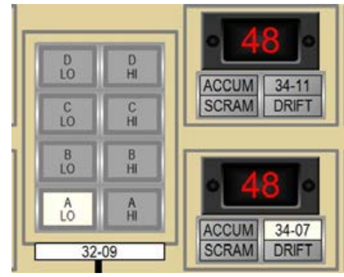
B. **(1)** 2.3
(2) will NOT

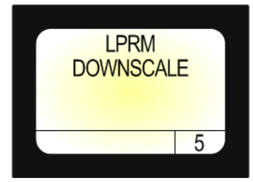
C. **(1)** 3.16
(2) will

D. **(1)** 3.16
(2) will NOT

43. Unit 2 is operating at 96% RTP and returning Control Rod 34-07 to service in accordance with 2-OI-85, Control Rod Drive System when the following conditions occur:

- LPRM 32-09 Detector A indicates 'LO'
- LPRM DOWNSCALE (2-9-5A, Window 5) is in alarm





Which **ONE** of the following completes the statements below?

The above condition is generated from the LPRM reaching ___(1)___.

Relative to Control Rod coupling integrity, if uncoupled, the **FOUR ROD** display digital read-out for Control Rod 34-07 would ___(2)___.

A. **(1)** 3%
   (2) remain illuminated

B. **(1)** 3%
   (2) extinguish

C. **(1)** 5%
   (2) remain illuminated

D. **(1)** 5%
   (2) extinguish

44. Which **ONE** of the following completes the statement below?

The LPRM design feature that is utilized to offset the effects of detector aging is that the __________.

A. flux amplifier gain can be adjusted

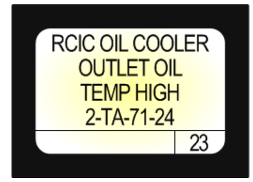
B. detector chamber is coated with enriched U-235

C. detector chamber is filled with high pressure Argon gas

D. ion chamber high voltage power supply can be lowered

45. Unit 2 is operating at 100% RTP with the following plant conditions:

- RCIC is running CST to CST for a flow test following repairs
- RCIC OIL COOLER OUTLET OIL TEMPERATURE HIGH (2-9-3C, Window 23) has just alarmed



Given the conditions above, which **ONE** of the following completes the statement below?

If RCIC lube oil cooling is in fact compromised or lost, troubleshooting and response will be conducted using procedures associated with the ________ System.

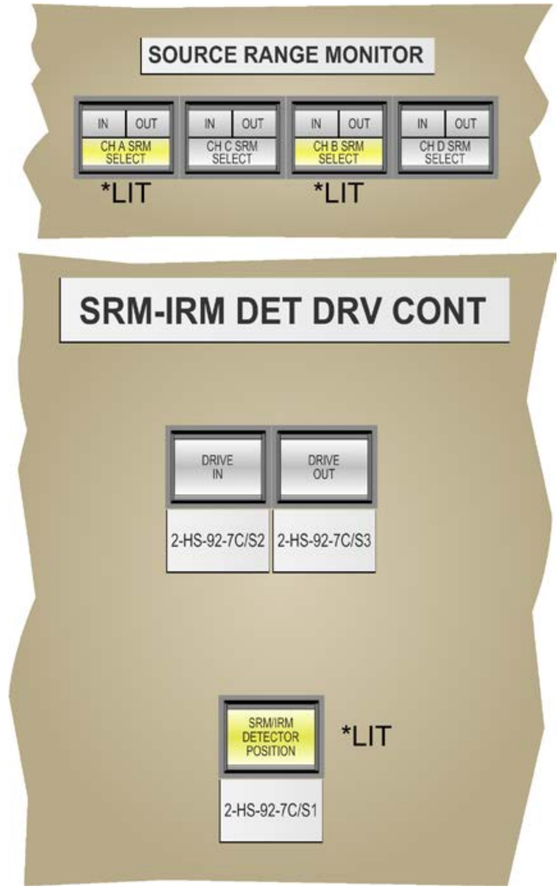
A. Raw Cooling Water

B. Reactor Core Isolation Cooling

C. Reactor Building Closed Cooling Water

D. Emergency Equipment Closed Cooling Water

46. **UNIT 2** is conducting a Reactor Startup and is currently in MODE 2. Source Range Monitors (SRMs) are in the depicted configuration.



Given the conditions above, which **ONE** of the following completes the statements below?

(CONSIDER EACH STATEMENT INDEPENDENTLY)

If the **DRIVE IN** Pushbutton is pressed **AND** released after one second, SRMs ‘A’ and ‘B’ will travel into the core ___**(1)**___.

If the **DRIVE OUT** Pushbutton is pressed **AND** released after one second, SRMs ‘A’ and ‘B’ will travel outwards from the core ___**(2)**___.

(Assume **NO** further operator actions)

A. **(1)** until the button is released
(2) until the button is released

B. **(1)** until the button is released
(2) until the full-out electrical stop is reached

C. **(1)** until the full-in electrical stop is reached
(2) until the button is released

D. **(1)** until the full-in electrical stop is reached
(2) until the full-out electrical stop is reached

47. Unit 2 was operating at 100% RTP when a LOCA occurred resulting in the following conditions:

- Drywell Pressure is 10 psig, rising slowly
- Reactor Water Level is (-) 180 inches, steady
- RHR Pumps 2A and 2C started
- RHR Pumps 2B and 2D failed to start
- Core Spray Pump 2A and 2B started

Subsequently, Operators noticed ADS initiated, but the Unit Supervisor directed the crew to secure ADS.

Given the conditions above, which **ONE** of the following manual actions would cause the ADS valves to **CLOSE**?

A. Secure **BOTH** RHR Pumps.

B. Secure **EITHER** Core Spray Pump.

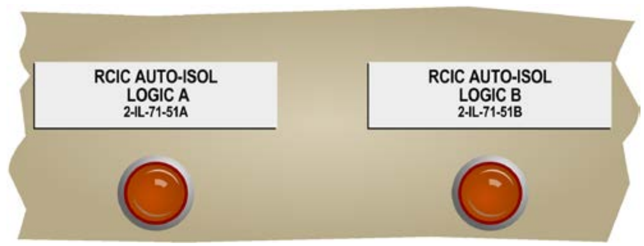
C. **RAISE** Reactor Water Level to (-) 162 inches.

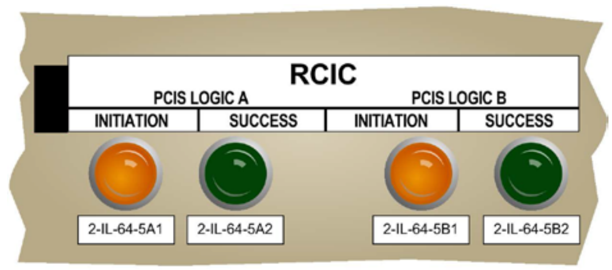
D. Depress **BOTH** 2-XS-1-159 and -161, Timer Reset buttons.

48. Given the following conditions for **UNIT 2**:

- Accident conditions have resulted in an EOI directed Emergency Depressurization
- Reactor Pressure is currently 59 psig
- **ALL** systems functioned as designed

Which **ONE** of the following completes the statements below?





As a result of the above conditions, the amber RCIC AUTO-**ISOLATION** LOGIC 'A' / 'B' lights on Panel 9-3, are ___**(1)**___.

As a result of the above conditions, the amber RCIC PCIS LOGIC 'A' / 'B' **INITIATION** lights on CISS Panel 9-4, are ___**(2)**___.

A. **(1)** lit
   (2) lit

B. **(1)** lit
   (2) **NOT** lit

C. **(1)** **NOT** lit
   (2) lit

D. **(1)** **NOT** lit
   (2) **NOT** lit

49. Unit 2 is operating at 100% RTP when the following occurs:

- 250V DC RMOV Board **2A** lost power

Subsequently, a LOCA occurred resulting in Reactor Water Level lowering to (-) 51 inches.

Given the conditions above, which **ONE** of the following completes the statements below?

HPCI ___**(1)**___ auto initiate.

HPCI ___**(2)**___ be manually initiated.

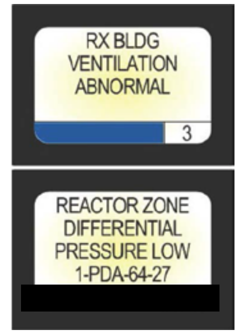
A. **(1)** will
(2) can

B. **(1)** will
(2) can NOT

C. **(1)** will NOT
(2) can

D. **(1)** will NOT
(2) can NOT

50. Unit 1 is operating at 100% RTP when the following conditions occur:



- REACTOR BUILDING VENTILATION ABNORMAL (1-9-3D, Window 3) alarms
- REACTOR ZONE DIFFERENTIAL PRESSURE LOW (1-9-3D, Window 32) alarms
- AUO reports REACTOR ZONE DIFFERENTIAL PRESSURE is (+) .5 inches of water locally
- Panel 1-9-25, amber light illuminates for REACTOR ZONE ISOLATION
- Assume **NO** Operator action has been taken

Which **ONE** of the following completes the statements below?

Given the conditions above, Standby Gas Treatment (SGT) ___**(1)**___ automatically started.

1-EOI-3, Secondary Containment Control, entry ___**(2)**___ required.

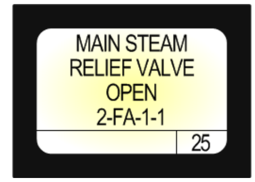
A. **(1)** has
   (2) is

B. **(1)** has
   (2) is NOT

C. **(1)** has NOT
   (2) is

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

51. Unit 2 was operating at 100% RTP when the following occurs:



- MAIN STEAM RELIEF VALVE OPEN, (2-9-3C, Window 25) alarms
- SRV 1-31 indicates full OPEN on the SRV TAILPIPE FLOW MONITOR
- 2-AOI-1-1, Relief Valve Stuck Open actions are in progress

Given the conditions above, which **ONE** of the following completes the statements below?

The given annunciator is **DIRECTLY** actuated by the tailpipe ____**(1)**____.

In accordance with 2-AOI-1-1, the Operator is allowed to cycle the control switch for SRV 1-31 ____**(2)**____ from CLOSE to OPEN to CLOSE positions.

A. **(1)** acoustic monitor
(2) UP TO three (3) times

B. **(1)** acoustic monitor
(2) ONLY one (1) time

C. **(1)** discharge temperature
(2) UP TO three (3) times

D. **(1)** discharge temperature
(2) ONLY one (1) time

52. The Unit 1 Unit Preferred Inverter is operating in a normal lineup, when a Loss of Offsite Power occurs **AND** ‘A’ EDG fails to start.

Which **ONE** of the following completes the statement below?

Given the conditions above, the Unit Preferred Inverter is **CURRENTLY** powered from the

__________.

A. 480V RMOV Board 1A

B. 250 VDC Battery Board 4

C. 250 VDC Battery Board 5

D. Unit Preferred Transformer

53. Which **ONE** of the following is correct with regards to the **NORMAL** and **ALTERNATE** power supplies to 250VDC RMOV Board 3C?

The **NORMAL** power supply is Battery Board ___**(1)**___ and the **ALTERNATE** power supply is Battery Board ___**(2)**___.

A. **(1)** 3
(2) 2

B. **(1)** 3
(2) 1

C. **(1)** 2
(2) 1

D. **(1)** 2
(2) 3

54. All three Units are operating at 100% RTP with all battery chargers in normal operation.

Which **ONE** of the following completes both statements below in accordance with 0-OI-31, Control Bay and Off-Gas Treatment Building Air Conditioning System?

Battery Room ventilation is required to be in operation to prevent ___**(1)**___.

Obtain ___**(2)**___ permission prior to shutting down the Battery Room exhaust fan.

A. **(1)** excessive temperatures
(2) Electrical Maintenance

B. **(1)** excessive temperatures
(2) Unit Supervisor

C. **(1)** the buildup of hydrogen
(2) Electrical Maintenance

D. **(1)** the buildup of hydrogen
(2) Unit Supervisor

55. ‘3A’ EDG has been started for the 3-SR-3.8.1.1(3A), Monthly Operability Test.

Which **ONE** of the following will occur if the ‘3A’ EDG Output Breaker is closed with 3-HS-82-3A/5A, DG MODE SELECTOR SWITCH in the **SINGLE UNIT** position?

A. The ‘3A’ EDG Output Breaker trips on overspeed.

B. The ‘3A’ EDG Output Breaker trips on undervoltage.

C. The 4KV Shutdown Board 3EA Normal Feeder Breaker trips on overload.

D. The 4KV Shutdown Board 3EA Normal Feeder Breaker trips on reverse power.

56. Given the following conditions:

- All three Units were operating at 100% RTP
- A Loss of Offsite Power occurred
- An Accident Signal is received on Unit 1

Subsequently,

- One (1) minute later, 'C' EDG trips on overspeed

Which **ONE** of the following completes the statement below?

Given the conditions above, the current status of RHR Pump 1B is ___**(1)**___ and RHR Pump 1C is ___**(2)**___.

A. **(1)** running
(2) running

B. **(1)** running
(2) NOT running

C. **(1)** NOT running
(2) running

D. **(1)** NOT running
(2) NOT running

57. Unit 2 is operating at 100% RTP when a Control Air leak develops, resulting in the following indication:

Given the indication, which **ONE** of the following identifies the correct plant status in accordance with 0-AOI-32-1, Loss of Control and Service Air Compressors?

A. 0-FCV-33-1, SERVICE AIR CROSSTIE VALVE, is CLOSED

B. 2-PCV-032-3901, CONTROL AIR CROSSTIE, is CLOSED

C. 2-FCV-2-130, CONDENSATE DEMIN BYP VALVE, is OPEN

D. OUTBOARD MAIN STEAM ISOLATION VALVES are CLOSED

58. Unit 1 is in MODE 4 with the following plant conditions:

- RHR Pump 1A and RHRSW Pump A2 is aligned for Shutdown Cooling (SDC)
- RHRSW Pump C2 is being utilized for dilution flow at 1500 gpm with RHR Heat Exchanger (HX) 1C
- The Unit 1 Operator observes Panel 1-9-21, 1-TR-74-80, RHR HX A/C COMBINED DISCHARGE Temperature is 142ºF and rising

Given the conditions above, which **ONE** of the following completes the statements below?

In accordance with 1-OI-74, Residual Heat Removal System, the Unit Operator is required to ___**(1)**___.

Additionally, do **NOT** exceed the **RATED** RHRSW flow of ___**(2)**___ gpm through RHR HX 1A or 1C.

Note: 1-FCV-23-34, RHR HX 1A RHRSW OUTLET VALVE
1-FCV-23-40, RHR HX 1C RHRSW OUTLET VALVE

A. **(1)** throttle open 1-FCV-23-34
(2) 4000

B. **(1)** throttle open 1-FCV-23-34
(2) 4500

C. **(1)** throttle open 1-FCV-23-40
(2) 4000

D. **(1)** throttle open 1-FCV-23-40
(2) 4500

59. Unit 1 is operating at 100% RTP when Recirc Pump 1A trips. The following conditions exist:

- Crew entered 1-AOI-68-1A, Recirc Pump Trip/Core Flow Decrease OPRMs Operable
- Core Flow is indicating 53% on the APRMs
- **NO** Operator actions have been taken

Given the conditions above, which **ONE** of the following identifies the **CURRENT** APRM Flow Biased SCRAM Setpoint in accordance with 1-OI-92B, Average Power Range Monitoring?

A. 87.38%

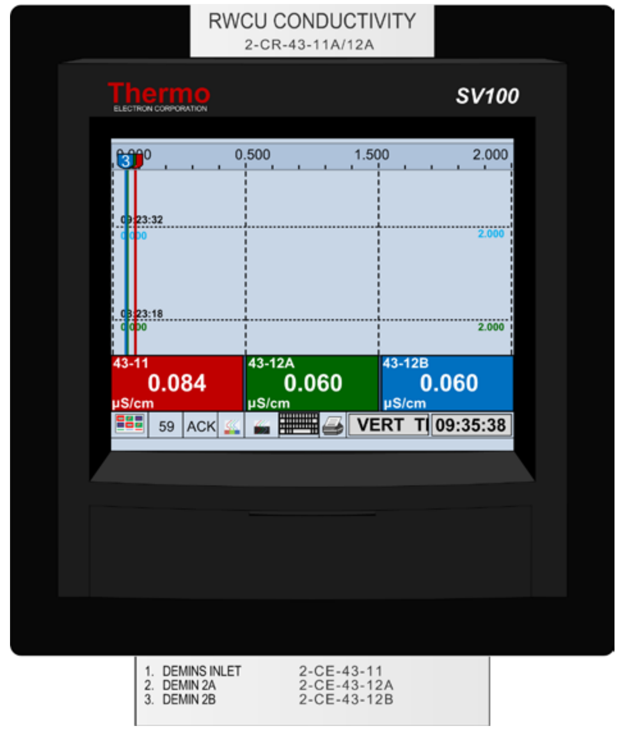
B. 93.38%

C. 93.65%

D. 93.98%

60. Unit 2 is operating at 100% RTP when the following occurs:

- RWCU Conductivity is reading as indicated



Subsequently,

- 480V RMOV BD 2A suddenly loses power

Given the conditions above, which **ONE** of the following completes the statements below?

RWCU ___**(1)**___ isolated.

After the power loss, Chemistry ___**(2)**___ required to be notified.

A. **(1)** is
(2) is

B. **(1)** is
(2) is NOT

C. **(1)** is NOT
(2) is

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

61. Unit 3 was operating at 100% RTP when the following plant conditions occur:

- Control Rod 30-31 lost its **ONES** digit position indication
- 3-AOI-85-4, Loss of RPIS is entered

Given the conditions above, which **ONE** of the following completes the statements below in accordance with 3-AOI-85-4, Loss of RPIS?

Control Rod 30-31 ___**(1)**___ be moved to an Operable Position Indication as a means of position verification.

If it is determined that Operators must SCRAM Control Rod 30-31, this will be conducted from the ___**(2)**___ .

A. (1) can
(2) Aux Instrument Room

B. (1) can
(2) Battery Board Room 3

C. (1) can NOT
(2) Aux Instrument Room

D. (1) can NOT
(2) Battery Board Room 3

62. Unit 3 was operating at 100% RTP when an event occurred requiring the insertion of a manual SCRAM, resulting in the following conditions:

- Reactor Power is 3%
- 3-EOI-1A, ATWS RPV Control entered
- Reactor Water Level is (+) 33 inches
- The SRO has directed 3-EOI Appendix-1F, Manual SCRAM

Given the conditions above, which **ONE** of the following completes the statements below?

Upon completion of Appendix-1F, the Operator __**(1)**__ be able to reset the SCRAM. The performance of the outside portions of Appendix-1F requires the use of __**(2)**__.

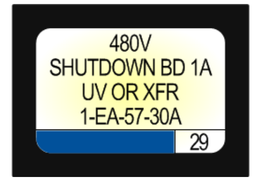
A. **(1)** will
(2) jumpers

B. **(1)** will
(2) keylocks

C. **(1)** will NOT
(2) jumpers

D. **(1)** will NOT
(2) keylocks

63. Unit 1 is operating at 100% RTP when the following occurs:



- 480V SHUTDOWN BOARD 1A UNDERVOLTAGE OR TRANSFER (1-9-8B, Window 29) alarms due to a fault

Given the condition above, which **ONE** of the following completes the statement below?

RHR Loop I ____**(1)**____ available for the Suppression Pool Cooling Mode and **AUTOMATIC** board transfer to the alternate power supply ____**(2)**____ occur.

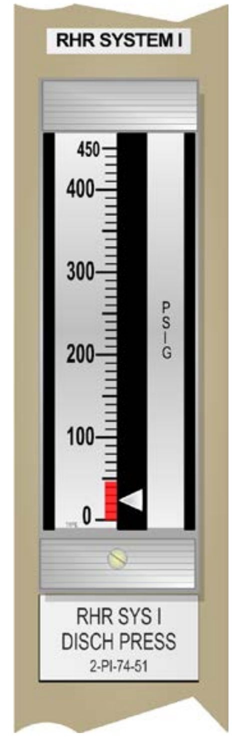
A. **(1)** is
(2) will

B. **(1)** is
(2) will NOT

C. **(1)** is NOT
(2) will

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

64. Unit 2 was operating at 100% RTP when a SCRAM and a small break LOCA occurred inside Containment resulting in the following conditions:



- Reactor Water Level is currently (+) 25 inches and stable
- Reactor Pressure is 850 psig and lowering
- Drywell Pressure 5 psig and rising
- Suppression Chamber Pressure 4 psig and rising

The SRO directs 2-EOI Appendix-17C, RHR System Operation Suppression Chamber Sprays be placed in service. The provided indication is then noted on Panel 2-9-3.

Given the conditions above, which **ONE** of the following completes the statements below?

RHR System piping is **NORMALLY** maintained in a filled condition by the ___**(1)**___.

If Suppression Chamber Sprays are placed in service, the concern is ___**(2)**___.

A. (1) PSC Head Tank
(2) water hammer

B. (1) PSC Head Tank
(2) cavitation

C. (1) Condensate Storage and Supply System
(2) water hammer

D. (1) Condensate Storage and Supply System
(2) cavitation

65. Unit 1 is operating at 24% RTP following a Startup with the following conditions:

- Reactor Feedwater Pump (RFPT) 1A is in service
- 1-LI-3-208D, RX WATER LEVEL NORMAL RANGE, failed **DOWNSCALE**

Subsequently, the Unit Operator observes 1-LI-3-208A, RX WATER LEVEL NORMAL RANGE drifting **DOWNSCALE**.

Given the conditions above, which **ONE** of the following completes the statements below?

If actual Reactor Water Level **RISES** to (+) 55 inches, the Main Turbine __**(1)**__ trip.

In accordance with OPDP-1, Conduct of Operations, a manual Reactor SCRAM __**(2)**__ required.

A. **(1)** will
(2) is

B. **(1)** will
(2) is NOT

C. **(1)** will NOT
(2) is

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

66. Which **ONE** of the following completes the statements below?

In accordance with OI-66, Off-Gas System, above ___**(1)**___ power operation, the discharge of the Steam Jet Air Ejectors (SJAEs) is required to be routed through the charcoal adsorber.

Excess moisture ___**(2)**___ affect the charcoal bed adsorber efficiency for the removal of iodine.

A. **(1)** 15%
(2) will

B. **(1)** 15%
(2) will NOT

C. **(1)** 25%
(2) will

D. **(1)** 25%
(2) will NOT

67. Which **ONE** of the following completes the statements below?

Unit 2 Technical Specification 3.4.9, RCS Pressure and Temperature (P/T) Limits, ___**(1)**___ applicable **AT ALL TIMES**.

When starting a Reactor Recirculation Pump, the difference between Bottom Head Temperature and RPV Coolant Temperature must be verified **NO MORE THAN** ___**(2)**___ minutes prior to starting each Recirculation Pump.

A. **(1)** is
(2) 15

B. **(1)** is
(2) 30

C. **(1)** is NOT
(2) 15

D. **(1)** is NOT
(2) 30

68. Given the following drawing excerpt of Unit 2 RCIC Initiation Logic, which **ONE** of the following completes the statements below in accordance with 2-45E626-1, Wiring Diagram, RCIC System Schematic Diagram?

The four primary contacts in the Reactor Vessel Low Water Level portion of the circuit are actuated directly by ___**(1)**___ System Relays.

The **LAST** RCIC Relay that energizes to cause the RCIC AUTO-INITIATE light, shown here, to illuminate on an initiation signal ___**(2)**___ a seal-in relay.

[SEE THE ATTACHED RCIC DRAWING, 2-45E626-1]

A. **(1)** RHR
   (2) is

B. **(1)** RHR
   (2) is NOT

C. **(1)** RCIC
   (2) is

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

69. Which **ONE** of the following meets the requirements to be considered an "Infrequently Performed Test or Evolution" (IPTE) per NPG-SPP-10.6, Infrequently Performed Test or Evolution?

A. 1-SR-3.5.1.7(COMP), HPCI Comprehensive Pump Test

B. 2-SR-3.5.1.6(RHR I), Quarterly RHR System Rated Flow Test Loop I

C. 0-SR-3.8.1.9(A), Diesel Generator ‘A’ Emergency Unit 1 Load Acceptance Test

D. 0-GOI-300-4, Switchyard Manual, Switching Order to remove the West Point 500KV line

70. Which **ONE** of the following completes the statement below in accordance with Unit 3 Tech Spec LCO 3.9.6 **RPV WATER LEVEL** for Refueling Operations?

RPV WATER LEVEL shall be **__________** above the top of the **RPV FLANGE** during movement of irradiated fuel assemblies in the RPV.

A. ≥ 21.5 feet

B. ≥ 22.0 feet

C. ≥ 23.5 feet

D. ≥ 25.0 feet

71. Which **ONE** of the following completes the statement below?

The Area Radiation Monitors (ARMs) are individual detectors that provide indications and alarms in the Main Control Room of ___**(1)**___ radiation levels from selected plant locations and the amber 'HIGH' light will **FIRST** illuminate when the MAX ___**(2)**___ radiation value has been reached.

A. **(1)** neutron
(2) SAFE

B. **(1)** neutron
(2) NORMAL

C. **(1)** gamma
(2) SAFE

D. **(1)** gamma
(2) NORMAL

72. Unit 1 is in a Refueling Outage with the following conditions:

- 1-HS-99-5A-S1, REACTOR MODE SWITCH is in REFUEL
- Fuel movements are in progress

Which **ONE** of the following completes the statements below?

Three direction movements are allowed **ONLY** in the **___(1)___**.

In the event gas bubbles are visible in the Spent Fuel Pool, the **IMMEDIATE ACTION** in accordance with 1-AOI-79-1, Fuel Damage During Refueling is to **___(2)___**.

A. **(1)** Reactor Vessel area
(2) evacuate the Refueling Floor

B. **(1)** Reactor Vessel area
(2) evaluate Radiation Levels

C. **(1)** Spent Fuel Pool
(2) evacuate the Refueling Floor

D. **(1)** Spent Fuel Pool
(2) evaluate Radiation Levels

73. Unit 2 is operating at 100% RTP with a steam leak in the 2A SJAE room. An Operator has been assigned to investigate. Radiation Protection reports that general area radiation levels are 120 mR/hr.

Given the above, which **ONE** of the following completes the statement below in accordance with NPG-SPP-5.18, Radiation Work Permit (RWP) requirements?

A ____**(1)**____ RWP will be used to enter the SJAE room and a documented RWP briefing will be conducted by ____**(2)**____.

A. **(1)** General
(2) the Shift Manager

B. **(1)** General
(2) Radiation Protection

C. **(1)** Specific
(2) the Shift Manager

D. **(1)** Specific
(2) Radiation Protection

74. Unit 1 is operating at 100% RTP.

Which **ONE** of the following completes the statement below?

When assessing the EOI Exclusion Plot Status Boxes on the Safety Parameter Display System (SPDS) while using Integrated Computer System (ICS), ___**(1)**___ is expected to be colored RED.

In accordance with 0-OI-48, Integrated Computer System, the SPDS component of ICS ___**(2)**___ qualified as independent decision making instrumentation for operating the plant.

Note: Curve 5 – Drywell Spray Initiation Limit
Curve 6 – Pressure Suppression Pressure

A. **(1)** Curve 5
(2) is

B. **(1)** Curve 5
(2) is NOT

C. **(1)** Curve 6
(2) is

D. **(1)** Curve 6
(2) is NOT

75. Unit 1 is operating at 100% RTP when an event with the following plant conditions occurs:

- HPCI and RCIC automatically initiate
- HPCI automatically isolates due to a steam supply line break

Which **ONE** of the following completes the statement below?

To respond to this event, the Unit Supervisor will enter __________.

Note: 1-EOI-1, RPV Control

1-EOI-2, Primary Containment Control

1-EOI-3, Secondary Containment Control

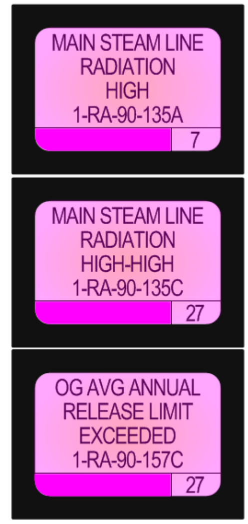
A. 1-EOI-1 **ONLY**

B. 1-EOI-2 **ONLY**

C. 1-EOI-2 **AND** 1-EOI-3

D. 1-EOI-1 **AND** 1-EOI-3

76. Unit 1 has been operating with rising amounts of fuel bundle leaks. Suppression efforts have been unsuccessful, and Unit 1 is approaching an Operations Decision Making Instruction (ODMI) trigger value for shutting down the Reactor on excessive Stack Release Rates when the following alarms are received:



- MAIN STEAM LINE RADIATION HIGH (1-9-3A, Window 7)
- MAIN STEAM LINE RADIATION HIGH-HIGH (1-9-3A, Window 27)
- OFFGAS AVERAGE ANNUAL RELEASE RATE EXCEEDED (1-9-4C, Window 27)

Which **ONE** of the following completes the statement below?

Given the conditions above, the Unit Supervisor will direct a ___**(1)**___.

Once in MODE 3, the Main Steam Isolation Valves (MSIVs) ___**(2)**___ required to be **CLOSED**.

Note: 1-AOI-100-1, Reactor SCRAM
1-GOI-100-12A, Unit Shutdown from Power Operations to Cold Shutdown and Reductions in Power During Power Operations

A. **(1)** manual Reactor SCRAM in accordance with 1-AOI-100-1
(2) are

B. **(1)** manual Reactor SCRAM in accordance with 1-AOI-100-1
(2) are NOT

C. **(1)** normal Reactor Shutdown in accordance with 1-GOI-100-12A
(2) are

D. **(1)** normal Reactor Shutdown in accordance with 1-GOI-100-12A
(2) are NOT

77. Unit 1 is in MODE 4 in preparation for a Refueling Outage with the following conditions:

- RHR Pump 1B is in Shutdown Cooling (SDC)
- RWCU is in service for Reactor Water Level Control in conjunction with CRD

Subsequently, the following has occurred:

- A leak developed on the discharge of RHR Pump 1B
- Reactor Water Level is (-) 5 inches and steady
- Reactor Coolant Temperature is 150 °F and rising slowly

Given the conditions above, which **ONE** of the following completes the statements below?

With Loop II RHR aligned for SDC, Reactor Coolant is taken from the Recirc Pump ___**(1)**___ suction flowpath.

The Unit Supervisor will direct ___**(2)**___ to mitigate this event.

Note: 1-AOI-64-2A, Group 3 Reactor Water Cleanup Isolation
1-AOI-74-1, Loss of Shutdown Cooling

A. **(1)** 1B
(2) 1-AOI-74-1 **ONLY**

B. **(1)** 1B
(2) 1-AOI-64-2A **AND** 1-AOI-74-1

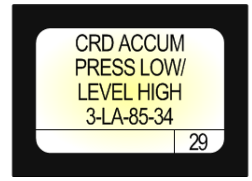
C. **(1)** 1A
(2) 1-AOI-74-1 **ONLY**

D. **(1)** 1A
(2) 1-AOI-64-2A **AND** 1-AOI-74-1

78. Unit 3 is operating at 100% RTP at the End of Core Life with **ALL** Control Rods fully withdrawn.

CRD Pump 3A has been tagged out for repairs when the following plant conditions occur:

- At 05:15 - 4kV Shutdown Board 3EA de-energizes and locks out due to a fault
- At 05:16 - Charging Water Header Pressure is 930 psig and slowly lowering
- At 05:19 - CRD ACCUMULATOR PRESS LOW/LEVEL HIGH (3-9-5A, Window 29) alarms
- At 05:23 - AUO reports Control Rod 30-31 Accumulator Pressure is 900 psig
- At 05:24 - AUO reports Control Rod 22-19 Accumulator Pressure is 920 psig



Given the conditions above, which **ONE** of the following completes the statements below?

In accordance with Tech Specs, the **ACCUMULATORS** for Control Rods 30-31 and 22-19 are **___(1)___**.

In accordance with 3-AOI-85-3, CRD System Failure, the Unit Supervisor is required to direct a manual SCRAM **NO LATER THAN ___(2)___**.

A. **(1)** OPERABLE
(2) 0544

B. **(1)** OPERABLE
(2) 0536

C. **(1)** INOPERABLE
(2) 0544

D. **(1)** INOPERABLE
(2) 0536

79. Which **ONE** of the following completes the statements below in accordance with the EOI Program Manual?

The Unit Supervisor __**(1)**__ direct exceeding the Tech Spec Cooldown Limit in order to remain within the **SAFE** region of Curve 3, Heat Capacity Temperature Limit (HCTL).

If the **UNSAFE** region of Curve 3, HCTL is entered, the Unit Supervisor __**(2)**__ allowed to lower Reactor Pressure in an attempt to return to the **SAFE** region to avoid an unnecessary Emergency Depressurization.

A. **(1)** can
(2) is

B. **(1)** can
(2) is NOT

C. **(1)** can NOT
(2) is

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

80. Which **ONE** of the following completes the statements below concerning High Drywell Temperature (DW/T)?

In accordance with Tech Spec Bases, the highest initial Drywell Average Air Temperature which ensures that the peak Drywell Temperature will **NOT** be reached following a Design Basis Accident (DBA) LOCA is ___**(1)**___.

In accordance with EOI-2, DW/T leg, Emergency Depressurization is required when DW/T cannot be restored and maintained below ___**(2)**___.

A. (1) 160 °F
(2) 280 °F

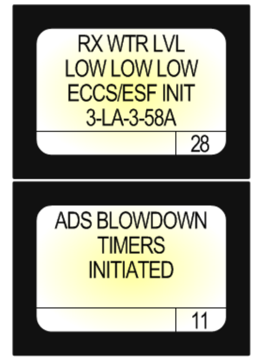
B. (1) 160 °F
(2) 350 °F

C. (1) 150 °F
(2) 280 °F

D. (1) 150 °F
(2) 350 °F

81. Unit 3 was operating at 100% RTP when a LOCA occurred with the following conditions:

- At 1000, a Loss of Offsite Power occurs
- At 1002, Drywell Pressure was 2.5 psig
- At 1010, REACTOR WATER LEVEL LOW LOW LOW ECCS/ESF INITIATION, (3-9-3C, Window 28) alarms
- At 1010, ADS BLOWDOWN TIMERS INITIATED, (3-9-3C, Window 11) alarms



Subsequently, the Operator reports the following:

- Reactor Pressure is 300 psig
- Reactor Water Level **INDICATED** (-) 210 inches and **RISING** on 3-LI-3-52, REACTOR WATER LEVEL ACCIDENT RANGE
- All RHR/Core Spray Pumps have started

Given the conditions above, which **ONE** of the following completes the statements below?

With **NO** Operator action, the **EARLIEST** time ADS will automatically initiate, is ___**(1)**___.

As the Unit Supervisor, the **HIGHEST** required Emergency Classification Action Level (EAL) to report is (a)an ___**(2)**___.

Note: Site Emergency Director Judgement shall **NOT** be used as a basis for classification

[REFERENCE PROVIDED]

A. **(1)** 95 seconds
(2) ALERT

B. **(1)** 95 seconds
(2) SITE AREA EMERGENCY

C. **(1)** 265 seconds
(2) ALERT

D. **(1)** 265 seconds
(2) SITE AREA EMERGENCY

82. On 10/17 at 1000, all Units are operating at 100% RTP with preparations underway to conduct the Fire Protection Surveillance Requirement (FPSR) Functional run of each electric motor driven High Pressure Fire Pump.

At 1015, it was determined that the weekly performance was **NOT** performed as required and should have, in fact, been started on 10/16 at 1100.

Given the conditions above, which **ONE** of the following completes the statements below?

In accordance with the FPSR Requirements, declaring the Fire Protection Limiting Condition for Operation (FPLCO) not met may be delayed, up to 24 hours or up to the limit of the specified Frequency, whichever is ___**(1)**___.

The High-Pressure Fire Protection System is considered FUNCTIONAL with ___**(2)**___ electric and one diesel pump in accordance with FPLCO 9.3.11.B.1.

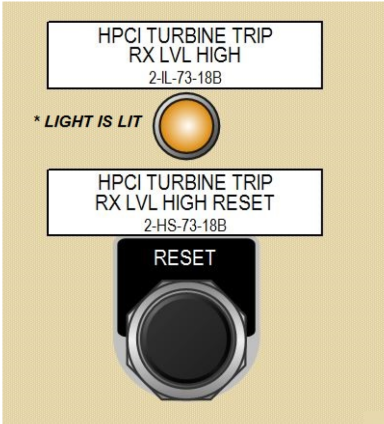
A. **(1)** less
(2) one

B. **(1)** less
(2) two

C. **(1)** greater
(2) one

D. **(1)** greater
(2) two

83. Unit 2 is operating at 100% RTP when an event results in the following plant indication:



Subsequent to the condition above, Reactor Water Level **LOWERS** to (-) 50 inches.

Assume **NO** Operator action has been taken.

Given the condition above, which **ONE** of the following completes the statement below?

HPCI ___**(1)**___ running and the Unit Supervisor will direct Reactor Water Level Control in accordance with ___**(2)**___.

Note: 2-AOI-3-1, Loss of Reactor Feedwater or Reactor Water Level High/Low
2-EOI Appendix-5A, Injection Systems Lineup Condensate/Feedwater

A. **(1)** is
(2) 2-AOI-3-1

B. **(1)** is
(2) 2-EOI Appendix-5A

C. **(1)** is NOT
(2) 2-AOI-3-1

D. **(1)** is NOT
(2) 2-EOI Appendix-5A

84. Unit 1 is operating at 100% RTP when the following plant conditions occur:

- A Control Rod drifts from position 00 to 48 and **CANNOT** be re-inserted
- Reactor Power reaches a **MAXIMUM** of 102% during the transient
- Operators lowered Recirc Flow to restore Reactor Power to 100%
- MCPR reached a **LOW** of 1.05 during the transient
- MCPR is **CURRENTLY** 1.35

Which **ONE** of the following completes the statements below?

Given the conditions above, a Reactor Shutdown ___**(1)**___ required in accordance with Technical Specifications. The significance level for this event is determined using the criteria stated in ___**(2)**___.

Note: OPDP-1, Conduct of Operations
NPG-SPP-10.4, Reactivity Management Program

A. **(1)** is
(2) NPG-SPP-10.4

B. **(1)** is
(2) OPDP-1

C. **(1)** is NOT
(2) NPG-SPP-10.4

D. **(1)** is NOT
(2) OPDP-1

85. Unit 3 has experienced an event requiring RHR Loop I to be used for Containment Sprays.

Given the conditions above, which **ONE** of the following completes the statements below?

In accordance with the EOI Program Manual Bases, ___**(1)**___ is the **HIGHEST** Suppression Pool Water Level **ALLOWED** prior to directing the termination of Drywell Sprays.

This ensures that ___**(2)**___.

A. **(1)** 19 feet
(2) maximum indicated Suppression Pool Water Level is **NOT** reached

B. **(1)** 19 feet
(2) Suppression Chamber-to-Drywell vacuum breaker openings are **NOT** submerged

C. **(1)** 20 feet
(2) maximum indicated Suppression Pool Water Level is **NOT** reached

D. **(1)** 20 feet
(2) Suppression Chamber-to-Drywell vacuum breaker openings are **NOT** submerged

86. Units 1, 2, and 3 are all operating at 100% RTP in their normal electrical lineups when:

- 250V DC Battery Board 4 is lost

Given the conditions above, which **ONE** of the following completes the statements below?

As a result, if AC power is **SUBSEQUENTLY** lost to the affected Unit's 9-9 Cabinet 5, automatic transfer of Cabinet 5 **___(1)___** occur.

As the Unit Supervisor, the mitigative actions for the given abnormal conditions would be directed in accordance with **___(2)___**.

Note: 0-AOI-57-6, Loss of Plant Non Preferred
2-AOI-57-4, Loss of Unit Preferred

A. **(1)** will
(2) 2-AOI-57-4

B. **(1)** will
(2) 0-AOI-57-6

C. **(1)** will NOT
(2) 2-AOI-57-4

D. **(1)** will NOT
(2) 0-AOI-57-6

87. Unit 2 is operating at 8% RTP with the following conditions:

- A Reactor Startup is in progress
- AUO reports that RCIC sight glass lube oil level is **NOT** visible

Given the conditions above, which **ONE** of the following completes the statements below in accordance with Tech Specs?

HPCI **OPERABILITY** is required to be verified by performing ____**(1)**____.

Under these conditions, the REACTOR MODE SWITCH ____**(2)**____ be placed in RUN.

A. (1) administrative checks
(2) can NOT

B. (1) administrative checks
(2) can

C. (1) HPCI flow rate surveillance
(2) can NOT

D. (1) HPCI flow rate surveillance
(2) can

88. At **1500**, an event occurred on Unit 2 involving fuel damage.

The following plant conditions exist:

- Stack Noble Gas (WRGERMS): 6.0×10^{8} µCi/sec
- 2-RM-90-250, Reactor Building/Turbine Building Exhaust on Panel 1-9-2 indicates 3.3×10^{5} µCi/sec

Given the above conditions, which **ONE** of the following completes the statements below?

At **1515**, the **HIGHEST REQUIRED** Emergency Classification Action Level (EAL) is ___**(1)**___.

Entry into 0-EOI-4, Radioactivity Release Control, ___**(2)**___ required.

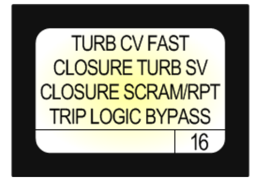
[REFERENCE PROVIDED]

A. **(1)** UNUSUAL EVENT
(2) is NOT

B. **(1)** UNUSUAL EVENT
(2) is

C. **(1)** ALERT
(2) is NOT

D. **(1)** ALERT
(2) is



89. A Unit 3 Reactor Startup is in progress with the following:

- TURBINE CONTROL VALVE FAST CLOSURE TURBINE STOP VALVE CLOSURE SCRAM/RECIRC PUMP TRIP LOGIC BYPASS, (3-9-5B, Window 16) is illuminated

Which **ONE** of the following completes the statements below?

The above alarm will **FIRST** clear when Thermal Power is **≥ ___(1)____** during a Reactor Startup.

If the alarm does **NOT** clear as expected, the Unit Supervisor will direct actions in accordance with **___(2)____** to comply with Tech Specs.

Note: 3-OI-47, Turbine-Generator System
3-OI-99, Reactor Protection System

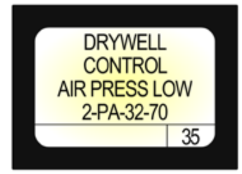
A. **(1)** 26%
(2) 3-OI-47

B. **(1)** 26%
(2) 3-OI-99

C. **(1)** 30%
(2) 3-OI-47

D. **(1)** 30%
(2) 3-OI-99

90. Following a Reactor SCRAM on Unit 2, the following plant conditions exist:



- DRYWELL CONTROL AIR PRESSURE LOW, (2-9-3E, Window 35), is in alarm
- Reactor Water Level lowered to (-) 10 inches, currently at (+) 30 inches
- Assume **NO** Operator action has been taken

Given the conditions above, which **ONE** of the following completes the statements below?

Currently, the **INBOARD** MSIVs are ____**(1)**____.

The Unit Supervisor will direct Pressure to be restored in accordance with ____**(2)**____ .

Note: 2-AOI-32-2, Loss of Control Air
2-EOI Appendix-8G, Crosstie CAD to Drywell Control Air

A. **(1)** OPEN
(2) 2-EOI Appendix-8G

B. **(1)** OPEN
(2) 2-AOI-32-2

C. **(1)** CLOSED
(2) 2-EOI Appendix-8G

D. **(1)** CLOSED
(2) 2-AOI-32-2

91. Which **ONE** of the following completes the statements below in accordance with Technical Specifications?

The Limiting Condition for Operation (LCO) statement for Unit 1 Safety Relief Valves (SRVs), LCO 3.4.3, requires that the safety function of ___**(1)**___ SRVs **SHALL** be OPERABLE in MODES 1, 2, and 3.

In accordance with Tech Spec Bases, this is to ensure the design SRV capacity can maintain Reactor Pressure below the American Society of Mechanical Engineers (ASME) Code Limit of ___**(2)**___ during the Design Basis Event from the closure of **ALL** MSIVs, followed by a SCRAM at 100% RTP.

A. **(1)** 12
(2) 1250 psig

B. **(1)** 12
(2) 1375 psig

C. **(1)** 13
(2) 1250 psig

D. **(1)** 13
(2) 1375 psig

92. Unit 2 was operating at 100% RTP when multiple events occurred resulting in the following conditions:

- MSIVs are CLOSED
- SRV 1-31 is stuck OPEN
- 2-AOI-1-1, Relief Valve Stuck Open actions are in progress

Three (3) minutes later, **WITHOUT** any Operator action

- Drywell Pressure is 7.0 psig and rising
- Drywell Temperature is 150°F and rising
- Suppression Chamber Pressure is 7.2 psig and rising
- Suppression Chamber Air Temperature is 220 °F and rising
- Suppression Pool Level is 11.5 feet and lowering

Given the conditions above, which **ONE** of the following completes the statements below?

These conditions indicate that ____**(1)**____.

In accordance with EOI-2, Primary Containment Control, the Unit Supervisor will direct ____**(2)**____.

A. **(1)** SRV discharges are uncovered
(2) Spraying the Drywell

B. **(1)** SRV discharges are uncovered
(2) Emergency Depressurization

C. **(1)** an SRV tail pipe has ruptured
(2) Spraying the Drywell

D. **(1)** an SRV tail pipe has ruptured
(2) Emergency Depressurization

93. Unit 2 was operating at 100% RTP when a Design Basis Accident (DBA) LOCA occurred. The following conditions exist:

- Drywell Pressure is 25 psig and steady
- Reactor Pressure is 400 psig and lowering
- Actual Reactor Water Level is (-) 162 inches and lowering
- RPV injection is maximized using Appendix 5B, Injection System Lineup CRD
- **NO** other RPV injection sources are available

Which **ONE** of the following completes the statements below?

The analysis of a DBA-LOCA is based on the double-ended break of one of the Recirculation Loop's ___**(1)**___ piping.

Given the conditions above, Steam Cooling ___**(2)**___ required in accordance with 2-EOI-1, RPV Control.

A. **(1)** discharge
(2) is NOT

B. **(1)** discharge
(2) is

C. **(1)** suction
(2) is NOT

D. **(1)** suction
(2) is

94. All three Units are operating at 100% RTP.

Operations is scheduled to transfer Diesel Aux Board A to the **ALTERNATE** supply breaker in accordance with 0-OI-57B, Section 8.12, Transfer of 480V RMOV Boards A, B, C and Diesel Aux Boards A&B, 3EA, & 3EB.

Which **ONE** of the following completes the statements below?

The Unit 1 and 2 Diesel Aux Board A is considered ___**(1)**___ when being supplied by its alternate power supply.

If Unit 1 and 2 Diesel Aux Board A subsequently fails to transfer and remains de-energized, the **MOST LIMITING** Tech Spec required action is to restore ___**(2)**___.

[REFERENCE PROVIDED]

A. **(1)** OPERABLE
(2) Unit 1 and 2 Diesel Aux Board A to OPERABLE status within 5 days

B. **(1)** OPERABLE
(2) all but one Unit 1 and 2 EDGs to OPERABLE status within 2 hours

C. **(1)** INOPERABLE
(2) Unit 1 and 2 Diesel Aux Board A to OPERABLE status within 5 days

D. **(1)** INOPERABLE
(2) all but one Unit 1 and 2 EDGs to OPERABLE status within 2 hours

95. Unit 2 is in MODE 5 during a scheduled refueling outage. A **NON-SPIRAL** core reload is in progress with the following conditions:

- SRM B is **INOPERABLE**
- The next fuel bundle to be moved is currently located in the Spent Fuel Pool and is designated for Reactor Cavity position 09-42
- As the respective fuel bundle is grappled, SRM D fails downscale and is declared **INOPERABLE**
- **ALL** other SRMs are **OPERABLE**

Which **ONE** of the following completes the statement below?

Given the conditions above, fuel moves ________ in accordance with Tech Specs and 0-GOI-100-3C, Fuel Movement Operations During Refueling.

[SEE THE ATTACHED CORE QUADRANT ILLUSTRATION]

A. **CAN** continue since the SRM in the **AFFECTED** core quadrant is **OPERABLE**

B. **CAN** continue since the SRM in the **ADJACENT** core quadrant is **OPERABLE**

C. **CANNOT** continue since the SRM in the **AFFECTED** core quadrant is **INOPERABLE**

D. **CANNOT** continue since the SRM in the **ADJACENT** core quadrant is **INOPERABLE**

96. Unit 3 was operating at 100% RTP and performing a HPCI calibration surveillance when the IMs report that relays 14A-K5A and 14A-K5B have failed to close during the testing.

Given the conditions above, which **ONE** of the following is correct in accordance with Tech Spec Required Actions.

[REFERENCE PROVIDED]

A. Declare HPCI INOPERABLE within one (1) hour **ONLY**

B. Place affected channel(s) in trip in 24 hours **ONLY**

C. Declare supported ECCS features INOPERABLE within one (1) hour **ONLY**

D. Declare HPCI INOPERABLE within one (1) hour **AND** place channel in trip in 24 hours

97. Given the generic Tech Spec LCO ACTIONS Table example, and the following information:

- Both Pumps are required by the LCO Statement
- Pump 'A' becomes **INOPERABLE** at 0600 on November 19
- Pump 'B' becomes **INOPERABLE** at 0100 on November 20

EXAMPLE 1.3-2

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One pump inoperable.	A.1	Restore pump to OPERABLE status.	7 days
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 3.	12 hours
	AND		
	B.2	Be in MODE 4.	36 hours

Considering Tech Spec Rules and Applications, which **ONE** of the following completes the statements below?

At 0100 on November 20, ___(1)___. If Pump 'A' is restored to **OPERABLE** status at 0800 on November 20, the **LATEST TIME** that Pump 'B' can be restored to **OPERABLE** status without entering CONDITION B is ___(2)___.

A. **(1)** LCO 3.0.3 is entered
(2) 0600 on November 26

B. **(1)** LCO 3.0.3 is entered
(2) 0100 on November 27

C. **(1)** a new CONDITION A time clock is started for Pump 'B'
(2) 0600 on November 26

D. **(1)** a new CONDITION A time clock is started for Pump 'B'
(2) 0100 on November 27

98. Which **ONE** of the following completes the statements below in accordance with the EOI Program Manual Bases?

The preferred Primary Containment vent path is from the **(1)** .

The Unit Supervisor may exceed Release Rate Limits while directing Primary Containment Pressure reduction to reduce total offsite radiation dose in accordance with **(2)** .

Note: EOI Appendix-12, Primary Containment Venting
EOI Appendix-13, Emergency Venting Primary Containment

A. **(1)** Drywell
(2) EOI Appendix-13

B. **(1)** Drywell
(2) EOI Appendix-12

C. **(1)** Suppression Chamber
(2) EOI Appendix-13

D. **(1)** Suppression Chamber
(2) EOI Appendix-12

99. Due to multiple common cause failures in the Reactor Pressure Control Systems, Unit 3 experienced a Reactor Pressure spike of 1225 psig before **AUTOMATICALLY** SCRAMMING.

Which **ONE** of the following completes the statement below in accordance with NPG-SPP-03.5, Regulatory Reporting Requirements?

As a result of these events, the **EARLIEST** notification to the NRC is a/an __________hour report.

[REFERENCE PROVIDED]

A. one (1)

B. four (4)

C. eight (8)

D. twenty-four (24)

100 Unit 1 was operating at 100% RTP when an event occurred with the following conditions:

- A manual Reactor SCRAM was inserted
- **ALL** 8 red lights for RPS Group A (4) and B (4), respectively are illuminated on Panel 9-5
- 1-AOI-100-1, Reactor SCRAM, ATWS actions are complete

Subsequently,

- Reactor Power is 23%
- 1-EOI Appendix 2, Defeating ARI Logic Trips, is complete

Given the conditions above, which **ONE** of the following completes the statements below?

In accordance with the EOIs, the Unit Supervisor will direct the AUO to perform ___**(1)**___ in an attempt to insert the Control Rods.

As a result, Control Rods will insert due to the ___**(2)**___ being **DE-ENERGIZED**.

Note: 1-EOI Appendix 1F, Manual SCRAM
1-EOI Appendix 1A, Removal and Replacement of RPS SCRAM Solenoid Fuses

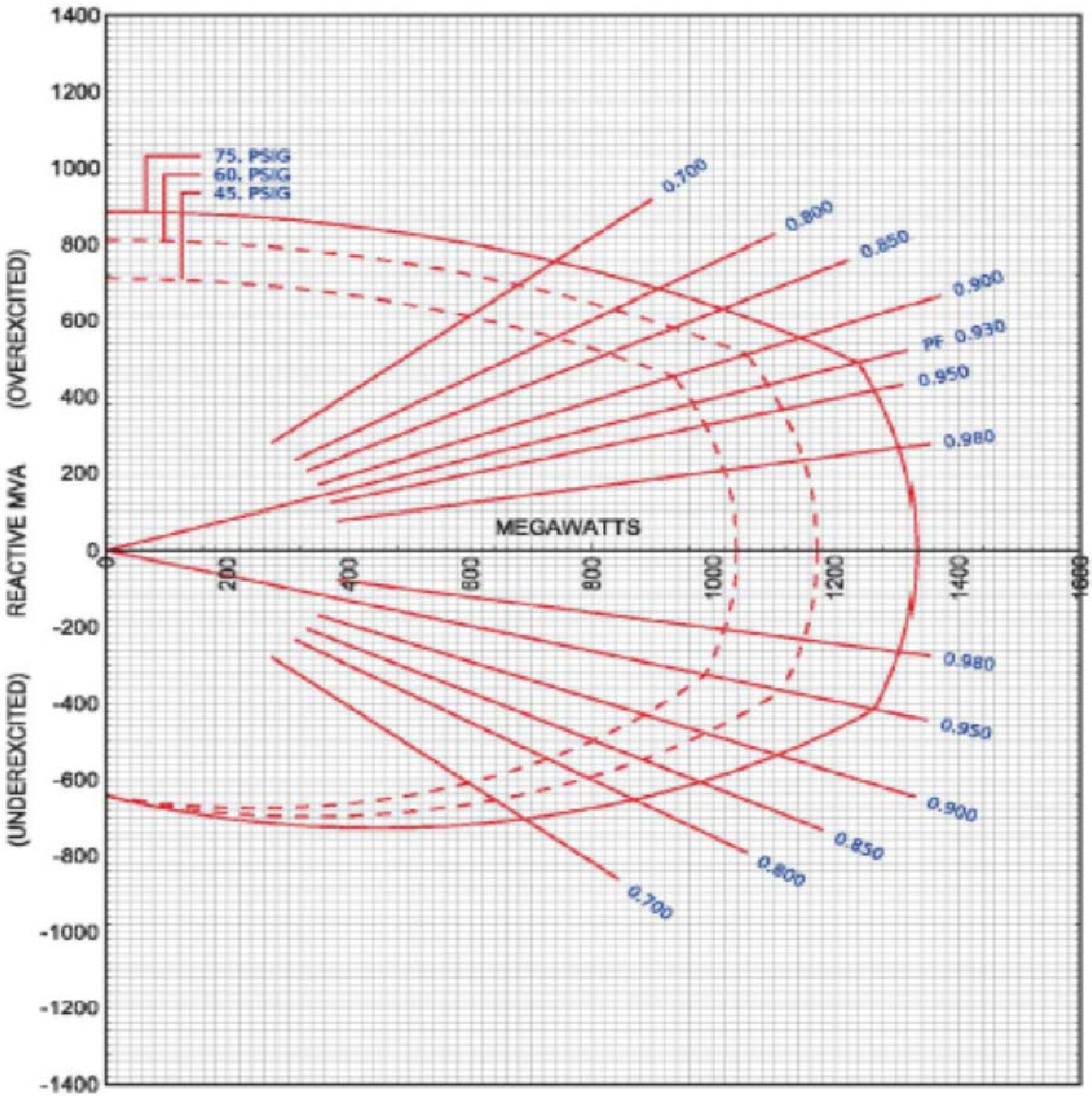
A. **(1)** EOI Appendix 1F
(2) SCRAM solenoids

B. **(1)** EOI Appendix 1F
(2) Backup SCRAM valves

C. **(1)** EOI Appendix 1A
(2) SCRAM solenoids

D. **(1)** EOI Appendix 1A
(2) Backup SCRAM valves

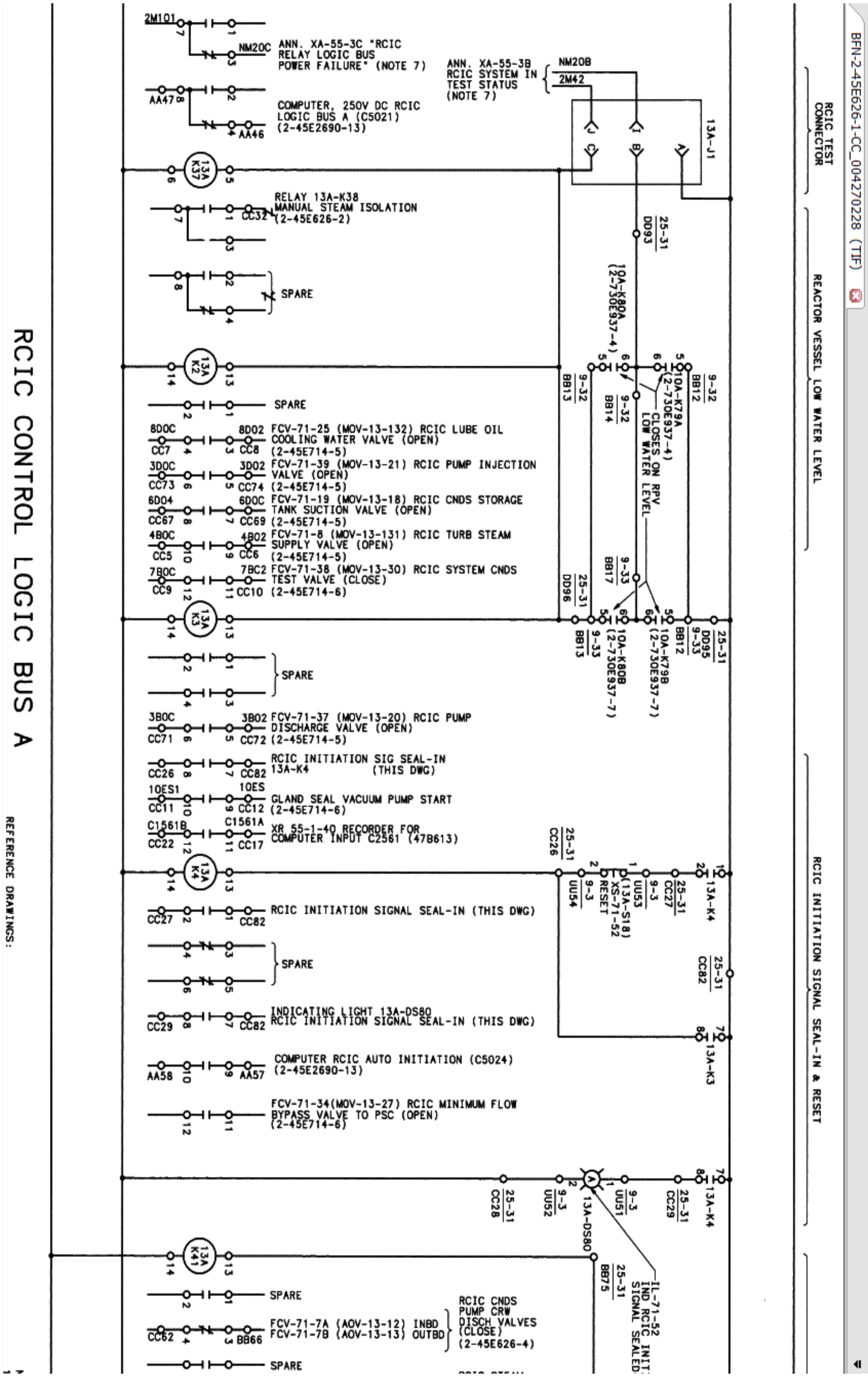
#27 REFERENCE PROVIDED



#32 - REFERENCE PROVIDED

Table 6 Secondary Cntmt Instrument Runs				
INSTRUMENT	SC TEMP ELEMENTS AND LOCATIONS			
	EI 621 (74-95F)	EI 593 (74-95C and D)	EI 565 (69-835A thru D)	RWCU HXRM (69-29F, G, H)
LI-3-58A	°F	°F	N/A	°F
LI-3-58B	°F	°F	N/A	N/A
LI-3-53	°F	°F	N/A	°F
LI-3-60	°F	°F	N/A	N/A
LI-3-206	°F	°F	N/A	°F
LI-3-253	°F	°F	N/A	N/A
LI-3-52	°F	°F	°F	N/A
LI-3-62A	°F	°F	°F	N/A
LI-3-55	°F	°F	N/A	N/A
LI-3-208A, B	°F	°F	N/A	°F
LI-3-208C, D	°F	°F	N/A	N/A

#68 PROVIDED ATTACHED RCIC DRAWING, 2-45E626-1



Attachment 1
(Page 2 of 4)

HOT INITIATING CONDITIONS – MODES 1 - 3

BROWNS FERRY
HOT INITIATING CONDITIONS – MODES 1 – 2 – 3

FISSION PRODUCT BARRIERS

FG1 - Loss of ANY Two Barriers AND Loss or Potential Loss of Third Barrier

FS1 - Loss or Potential Loss of ANY Two Barriers

FA1 - ANY Loss or ANY Potential Loss of EITHER Fuel Clad OR RCS Barrier

NOTE: *Multiple events could occur which result in the conclusion that exceeding the loss or potential loss thresholds is IMMINENT. In IMMINENT loss situations use judgment and classify as if the thresholds are exceeded.*

Fuel Clad Barrier		RCS Barrier		Containment Barrier	
Loss	Potential Loss	Loss	Potential Loss	Loss	Potential Loss

1. RCS Activity/Primary Containment/Primary Containment Conditions

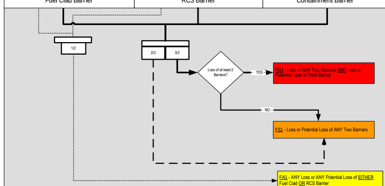
2. RPV Water Level

3. RCS Leak/Primary Containment Isolation

4. Primary Containment Radiation

5. Other Indications (Not Applicable)

6. SED Judgement



FA1 - ANY Loss or ANY Potential Loss of EITHER Fuel Clad OR RCS Barrier

Definitions

RADIOLOGICAL	SYSTEM MALFUNCTIONS			HAZARDS				
EFFLUENTS	ELECTRICAL	CONTROL ROOM INDICATION/ COMMUNICATIONS	ATWS/RCS PARAMETERS/ SAFETY SYSTEMS	SECURITY	NATURAL / DESTRUCTIVE	FIRE/ CR EVACUATION	TOXIC / FLAMMABLE	SED DISCRETION

GENERAL EMERGENCY

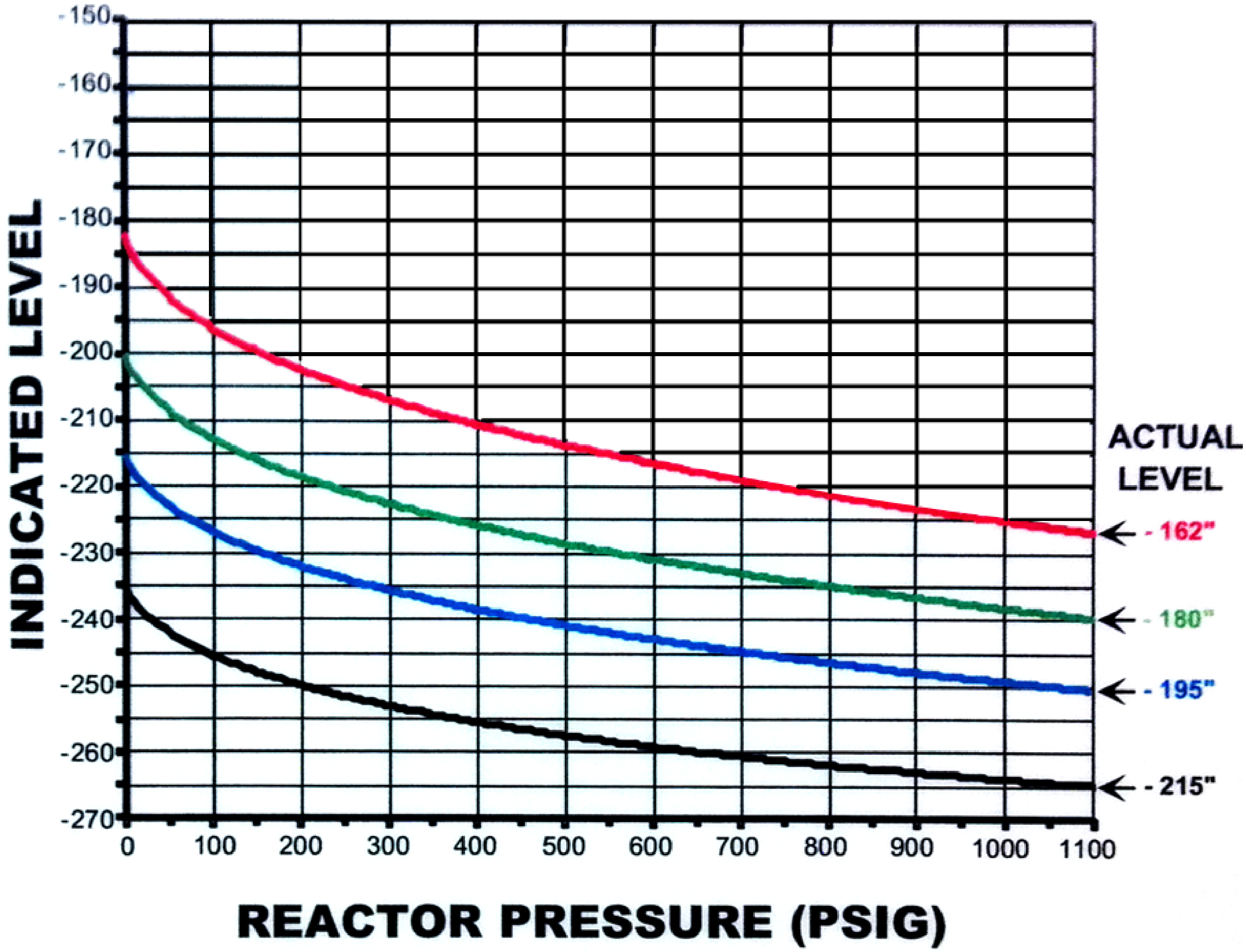
SITE AREA EMERGENCY

ALERT

NOTE:
Refer to SA9 under the "ATWS/RCS Parameter/Safety Systems" Column Under "System Malfunctions" to evaluate System Damage.

UNUSUAL EVENT

LI-3-52 & 62 CORRECTION CURVES



Attachment 1
(Page 2 of 4)

HOT INITIATING CONDITIONS – MODES 1 - 3

BROWNS FERRY
HOT INITIATING CONDITIONS – MODES 1 – 2 – 3

FISSION PRODUCT BARRIERS

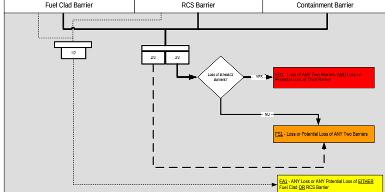
FG1 - Loss of ANY Two Barriers AND Loss or Potential Loss of Third Barrier

FS1 - Loss or Potential Loss of ANY Two Barriers

FA1 - ANY Loss or ANY Potential Loss of EITHER Fuel Clad OR RCS Barrier

NOTE: Multiple events could occur which result in the conclusion that exceeding the loss or potential loss thresholds is IMMINENT. In IMMINENT loss situations use judgment and classify as if the thresholds are exceeded.

Fuel Clad Barrier		RCS Barrier		Containment Barrier	
Loss	Potential Loss	Loss	Potential Loss	Loss	Potential Loss
1. RCS Activity/Primary Containment/Primary Containment Conditions					
A. Primary coolant activity greater than 300 µCi/gm dose equivalent Iodine-131.	Not Applicable	A. Drywell pressure at or above 2.45 psig due to RPV leakage.	Not Applicable	A. UNPLANNED rapid drop in primary containment pressure following primary containment pressure rise OR B. Primary containment pressure response not consistent with LOCA conditions.	A. Drywell pressure greater than 50 psig. OR B. H_2 greater than 6% AND O_2 greater than 5% exists inside primary containment OR C. HCTL exceeded.
2. RPV Water Level					
A. SAMG entry is required.	A. RPV water level cannot be restored and maintained above -162 inches. OR cannot be determined.	A. RPV water level cannot be restored and maintained above -162 inches. OR cannot be determined.	Not Applicable	Not Applicable	A. SAMG entry is required.
3. RCS Leak/Primary Containment Isolation					
Not Applicable	Not Applicable	A. UNISOLABLE Main Steamline OR Feedwater break resulting in exceeding the MSIV temperature isolation setpoint. OR B. Emergency RPV Depressurization.	A. UNISOLABLE primary system leakage that results in exceeding EITHER of the following: 1. Max Normal Operating Temperature OR 2. Max Normal Operating Area Radiation Level.	A. UNISOLABLE direct downstream pathway to the environment exists after primary containment isolation signal OR B. Intentional primary containment venting per EOIs OR C. UNISOLABLE primary system leakage that results in exceeding EITHER of the following: 1. Max Safe Operating Temperature. OR 2. Max Safe Operating Area Radiation Level.	Not Applicable
4. Primary Containment Radiation					
A. ANY primary containment radiation monitor reading greater than: UNIT 1 1-RM-90-272A >3.368E+03 R/HR 1-RM-90-273A >3.335E+03 R/HR UNIT 2 2-RM-90-272A >2.552E+03 R/HR 2-RM-90-273A >3.335E+03 R/HR UNIT 3 3-RM-90-272A >3.368E+03 R/HR 3-RM-90-273A >3.335E+03 R/HR	Not Applicable	A. ANY primary containment radiation monitor reading greater than: UNIT 1 1-RM-90-272A >3.582E+01 R/HR 1-RM-90-273A >3.557E+01 R/HR UNIT 2 2-RM-90-272A >2.722E+01 R/HR 2-RM-90-273A >3.557E+01 R/HR UNIT 3 3-RM-90-272A >3.582E+01 R/HR 3-RM-90-273A >3.557E+01 R/HR	Not Applicable	Not Applicable	A. ANY primary containment radiation monitor reading greater than: UNIT 1 1-RM-90-272A >1.119E+05 R/HR 1-RM-90-273A >1.111E+05 R/HR UNIT 2 2-RM-90-272A >8.502E+04 R/HR 2-RM-90-273A >1.111E+05 R/HR UNIT 3 3-RM-90-272A >1.119E+05 R/HR 3-RM-90-273A >1.111E+05 R/HR
5. Other Indications (Not Applicable)					
6. SED Judgement					
A. ANY condition in the opinion of the SED that indicates Loss of the Fuel Clad Barrier.	A. ANY condition in the opinion of the SED that indicates Potential Loss of the Fuel Clad Barrier.	A. ANY condition in the opinion of the SED that indicates Loss of the RCS Barrier.	A. ANY condition in the opinion of the SED that indicates Potential Loss of the RCS Barrier.	A. ANY condition in the opinion of the SED that indicates Loss of the Containment Barrier.	A. ANY condition in the opinion of the SED that indicates Potential Loss of the Containment Barrier.
Loss	Potential Loss	Loss	Potential Loss	Loss	Potential Loss
Fuel Clad Barrier		RCS Barrier		Containment Barrier	



Definitions

CONFINEMENT BOUNDARY: Spent Fuel Storage Canister consisting of Multi-Purpose Canister (MPC) shell, bottom base plate, MPC lid (including the vent and drain port cover plates), MPC closure ring, and associated welds.

CONTAINMENT CLOSURE: Containment condition where at least one integral barrier to the release of radioactive material is provided. ...

EXPLOSION: A rapid, violent and catastrophic failure of a piece of equipment due to combustion, chemical reaction or overpressurization. ...

FIRE: Combustion characterized by heat and light. ...

HOSTILE ACTION: An act toward a NPP or its personnel that includes the use of violent force to destroy equipment, take HOSTAGES, and/or intimidate the licensee to achieve an end. ...

HOSTILE FORCE: One or more individuals who are engaged in a determined assault, overtly or by stealth and deception, equipped with suitable weapons capable of killing, maiming, or causing destruction.

IMMINENT: The trajectory of events or conditions is such that an EAL will be met within a relatively short period of time regardless of mitigation or corrective actions.

OWNER CONTROLLED AREA: The TVA-owned site property under the control of Site Security.

PROTECTED AREA: The area which is encompassed by the security fence and to which access is controlled.

REFUELING PATHWAY: The reactor refueling cavity, spent fuel pool, or fuel transfer canal.

SAFETY SYSTEM: A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. ...

SECURITY CONDITION: Any Security Event as listed in the approved security contingency plan that constitutes a threat/compromise to site security, threat/risk to site personnel, or a potential degradation to the level of safety of the plant. A SECURITY CONDITION does not involve a HOSTILE ACTION.

UNISOLABLE: An open or breached system line that cannot be isolated, remotely or locally.

UNPLANNED: A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant transient. ...

VISIBLE DAMAGE: Damage to a SAFETY SYSTEM train that is readily observable without measurements, testing, or analysis. ...

Recognition categories (matrix columns)

RADIOLOGICAL	SYSTEM MALFUNCTIONS			HAZARDS				
EFFLUENTS	ELECTRICAL	CONTROL ROOM INDICATION/ COMMUNICATIONS	ATWS/RCS PARAMETERS/ SAFETY SYSTEMS	SECURITY	NATURAL / DESTRUCTIVE	FIRE/ CR EVACUATION	TOXIC / FLAMMABLE	SED DISCRETION

Rows: GENERAL EMERGENCY; SITE AREA EMERGENCY; ALERT; UNUSUAL EVENT.

NOTE: Refer to SA9 under the "ATWS/RCS Parameter/Safety Systems" Column Under "System Malfunctions" to evaluate System Damage.

[illegible]

3.8 ELECTRICAL POWER SYSTEMS

3.8.1 AC Sources - Operating

LCO 3.8.1 The following AC electrical power sources shall be OPERABLE:

a. Two qualified circuits between the offsite transmission network and the onsite Class 1E AC Electrical Power Distribution System;

b. Unit 1 and 2 diesel generators (DGs) with two divisions of 480 V load shed logic and common accident signal logic OPERABLE; and

c. Unit 3 DG(s) capable of supplying the Unit 3 4.16 kV shutdown board(s) required by LCO 3.8.7, "Distribution Systems - Operating."

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

---NOTE---
LCO 3.0.4.b is not applicable to DGs.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required offsite circuit inoperable.	A.1 Verify power availability from the remaining OPERABLE offsite transmission network. AND	[redacted] AND Once per 8 hours thereafter
		(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2 Declare required feature(s) with no offsite power available inoperable when the redundant required feature(s) are inoperable. AND	24 hours from discovery of no offsite power to one shutdown board concurrent with inoperability of redundant required feature(s)
	A.3 Restore required offsite circuit to OPERABLE status.	7 days AND 21 days from discovery of failure to meet LCO
B. One required Unit 1 and 2 DG inoperable.	B.1 Verify power availability from the offsite transmission network. AND	[redacted] AND Once per 8 hours thereafter

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. (continued)	B.2	Evaluate availability of both temporary diesel generators (TDGs).	[redacted] AND Once per 12 hours thereafter
	B.3.	Declare required feature(s), supported by the inoperable Unit 1 and 2 DG, inoperable when the redundant required feature(s) are inoperable.	4 hours from discovery of Condition B concurrent with inoperability of redundant required feature(s)
	AND		
	B.4.1	Determine OPERABLE Unit 1 and 2 DG(s) are not inoperable due to common cause failure.	24 hours
	OR		
	B.4.2	Perform SR 3.8.1.1 for OPERABLE Unit 1 and 2 DG(s).	24 hours
	AND		

(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. (continued)	B.5	Restore Unit 1 and 2 DG to OPERABLE status.	7 days from discovery of unavailability of TDG(s) <u>AND</u> 24 hours from discovery of Condition B entry $\geq$ 6 days concurrent with unavailability of TDG(s) <u>AND</u> 14 days <u>AND</u> 21 days from discovery of failure to meet LCO

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. One division of 480 V load shed logic inoperable.	C.1	Restore required division of 480 V load shed logic to OPERABLE status.	7 days
D. One division of common accident signal logic inoperable.	D.1	Restore required division of common accident signal logic to OPERABLE status.	7 days
E. Two required offsite circuits inoperable.	E.1	Declare required feature(s) inoperable when the redundant required feature(s) are inoperable.	12 hours from discovery of Condition E concurrent with inoperability of redundant required feature(s)
	<u>AND</u>		
	E.2	Restore one required offsite circuit to OPERABLE status.	24 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
-----------------NOTE------------- Only applicable when more than one 4.16 kV shutdown board is affected. ------------------------------------- F. One required offsite circuit inoperable. AND One Unit 1 and 2 DG inoperable.	-------------------NOTE---------------- Enter applicable Conditions and Required Actions of LCO 3.8.7, "Distribution Systems - Operating," when Condition F is entered with no AC power source to any 4.16 kV shutdown board. --- F.1 Restore required offsite circuit to OPERABLE status. OR F.2 Restore Unit 1 and 2 DG to OPERABLE status.	 12 hours 12 hours
-----------------NOTE------------- Applicable when only one 4.16 kV shutdown board is affected. ------------------------------------- G. One required offsite circuit inoperable. AND One Unit 1 and 2 DG inoperable.	G.1 Declare the affected 4.16 kV shutdown board inoperable.	[redacted]

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
H. Two or more Unit 1 and 2 DGs inoperable.	H.1 Restore all but one Unit 1 and 2 DG to OPERABLE status.	2 hours
I. Required Action and Associated Completion Time of Condition A, B, C, D, E, F, or H not met.	I.1 Be in MODE 3. AND I.2 Be in MODE 4.	12 hours 36 hours
J. One or more required offsite circuits and two or more Unit 1 and 2 DGs inoperable. OR Two required offsite circuits and one or more Unit 1 and 2 DGs inoperable. OR Two divisions of 480 V load shed logic inoperable. OR Two divisions of common accident signal logic inoperable.	J.1 Enter LCO 3.0.3.	[redacted]

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
K. One or more required Unit 3 DGs inoperable.	K.1	Declare required feature(s) supported by the inoperable Unit 3 DG inoperable when the redundant required feature(s) are inoperable.	4 hours from discovery of Condition K concurrent with inoperability of redundant required feature(s)
	<u>AND</u>		
	K.2	Declare affected SGT and CREVs subsystem(s) inoperable.	30 days

3.8 ELECTRICAL POWER SYSTEMS

3.8.7 Distribution Systems - Operating

LCO 3.8.7 The following AC and DC electrical power distribution subsystems shall be OPERABLE:

a. Unit 1 and 2 4.16 kV Shutdown Boards;

b. Unit 2 480 V Shutdown Boards;

c. Unit 2 480 V RMOV Boards 2A, 2B, 2D, and 2E;

d. Unit 1 and 2 DG Auxiliary Boards;

e. Unit DC Boards and 250 V DC RMOV Boards 2A, 2B, and 2C;

f. Unit 1 and 2 Shutdown Board DC Distribution Panels; and

g. Unit 1 and 3 AC and DC Boards needed to support equipment required to be OPERABLE by LCO 3.6.4.3, "Standby Gas Treatment (SGT) System," and LCO 3.7.3, "Control Room Emergency Ventilation (CREV) System."

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One Unit 1 and 2 4.16 kV Shutdown Board inoperable.	-------------------NOTE---------------- Enter applicable Conditions and Required Actions of Condition B, C, D, and G when Condition A results in no power source to a required 480 volt board. --	
	A.1 Restore the Unit 1 and 2 4.16 kV Shutdown Board to OPERABLE status.	5 days <u>AND</u> 12 days from discovery of failure to meet LCO
	<u>AND</u>	
	A.2 Declare associated diesel generator inoperable.	[redacted]

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One Unit 2 480 V Shutdown Board inoperable. <u>OR</u> 480 V RMOV Board 2A inoperable. <u>OR</u> 480 V RMOV Board 2B inoperable.	------------------NOTE--------------- Enter Condition C when Condition B results in no power source to 480 volt RMOV board 2D or 2E. --- B.1 Restore Board to OPERABLE status.	8 hours <u>AND</u> 12 days from discovery of failure to meet LCO
C. Unit 2 480 V RMOV Board 2D inoperable. <u>OR</u> Unit 2 480 V RMOV Board 2E inoperable.	C.1 Declare the affected RHR subsystem inoperable.	[redacted]
D. One Unit 1 and 2 DG Auxiliary Board inoperable.	D.1 Restore Unit 1 and 2 DG Auxiliary Board to OPERABLE status.	5 days <u>AND</u> 12 days from discovery of failure to meet LCO

(continued)

ACTIONS (continued)

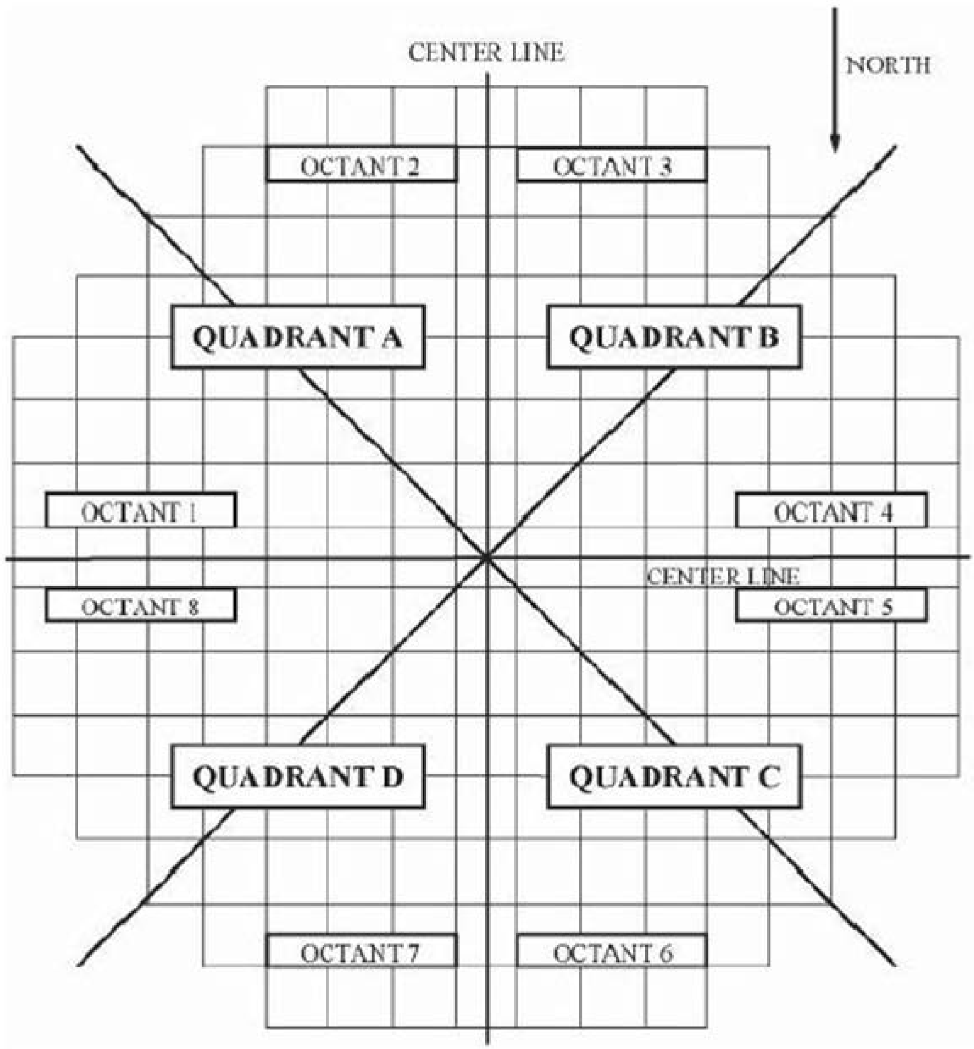
CONDITION	REQUIRED ACTION		COMPLETION TIME
E. One Unit DC Board inoperable. OR One Unit 1 and 2 Shutdown Board DC Distribution Panel inoperable. OR 250 V DC RMOV Board 2A inoperable. OR 250 V DC RMOV Board 2B inoperable. OR 250 V DC RMOV Board 2C inoperable.	E.1	Restore required Board or Shutdown Board DC Distribution Panel to OPERABLE status.	7 days AND 12 days from discovery of failure to meet LCO

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. Unit 1 and 2 4.16 kV Shutdown Board A and B inoperable. OR Unit 1 and 2 4.16 kV Shutdown Board C and D inoperable.	-------------------NOTE---------------- Enter applicable conditions and required actions of Condition B, C, D, and G when Condition F results in no power source to a required 480 volt board. -- F.1 Restore one 4.16 kV Shutdown Board to OPERABLE status.	 8 hours AND 12 days from discovery of failure to meet LCO
G. One or more required Unit 1 or 3 AC or DC Boards inoperable.	G.1 Declare the affected SGT or CREV subsystem inoperable.	[redacted]
H. Required Action and associated Completion Time of Condition A, B, D, E, or F not met.	H.1 Be in MODE 3. AND H.2 Be in MODE 4.	12 hours 36 hours
I. Two or more electrical power distribution subsystems inoperable that result in a loss of function.	I.1 Enter LCO 3.0.3.	[redacted]

#95 REFERENCE PROVIDED



3.3 INSTRUMENTATION

3.3.5.1 Emergency Core Cooling System (ECCS) Instrumentation

LCO 3.3.5.1 The ECCS instrumentation for each Function in Table 3.3.5.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.5.1-1.

ACTIONS

--NOTE--
Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more channels inoperable.	A.1 Enter the Condition referenced in Table 3.3.5.1-1 for the channel.	[redacted]

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	B.1 -------------NOTES----------- 1. Only applicable in MODES 1, 2, and 3. 2. Only applicable for Functions 1.a, 1.b, 2.a, and 2.b. --------------------------------- Declare supported ECCS feature(s) inoperable when its redundant feature ECCS initiation capability is inoperable.	[redacted] from discovery of loss of initiation capability for features in both divisions
	AND	
	B.2 --------------NOTE------------ Only applicable for Functions 3.a and 3.b. --------------------------------- Declare High Pressure Coolant Injection (HPCI) System inoperable.	[redacted] from discovery of loss of HPCI initiation capability
	AND	
	B.3 Place channel in trip.	24 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	C.1 -------------NOTES----------- 1. Only applicable in MODES 1, 2, and 3. 2. Only applicable for Functions 1.c, 1.e, 2.c, 2.d, and 2.f. --------------------------------- Declare supported ECCS feature(s) inoperable when its redundant feature ECCS initiation capability is inoperable.	 [redacted] from discovery of loss of initiation capability for features in both divisions
	<u>AND</u>	
	C.2 Restore channel to OPERABLE status.	24 hours
D. As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	D.1 --------------NOTE------------ Only applicable if HPCI pump suction is not aligned to the suppression pool. --------------------------------- Declare HPCI System inoperable.	 [redacted]

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
E. As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	E.1	-------------NOTES----------- 1. Only applicable in MODES 1, 2, and 3. 2. Only applicable for Function 1.d. ---------------------------------- Declare supported ECCS feature(s) inoperable when its redundant feature ECCS initiation capability is inoperable.	[redacted] from discovery of loss of initiation capability for subsystems in both divisions
	<u>AND</u>		
	E.2	Restore channel to OPERABLE status.	7 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
F. As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	F.1	Declare Automatic Depressurization System (ADS) valves inoperable.	[redacted] from discovery of loss of ADS initiation capability in both trip systems
	AND		
	F.2	Place channel in trip.	96 hours from discovery of inoperable channel concurrent with HPCI or reactor core isolation cooling (RCIC) inoperable AND 8 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
G. As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	G.1 Declare ADS valves inoperable.	[redacted] from discovery of loss of ADS initiation capability in both trip systems
	AND	
	G.2 Restore channel to OPERABLE status.	96 hours from discovery of inoperable channel concurrent with HPCI or RCIC inoperable AND 8 days
H. Required Action and associated Completion Time of Condition B, C, D, E, F, or G not met.	H.1 Declare associated supported ECCS feature(s) inoperable.	[redacted]

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

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Core Spray System					
a. Reactor Vessel Water Level — Low Low Low, Level 1(f)	1,2,3, 4(a), 5(a)	4(b)	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	[redacted] above vessel zero
b. Drywell Pressure — High(f)	1,2,3	4(b)	B	SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	[redacted] psig
c. Reactor Steam Dome Pressure — Low (Injection Permissive and ECCS Initiation)(f)	1,2,3	4(b) 2 per trip system	C	SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.6	[redacted] psig and [redacted] psig
	4(a), 5(a)	4 2 per trip system	B	SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.6	[redacted] psig and [redacted] psig
d. Core Spray Pump Discharge Flow — Low (Bypass)	1,2,3, 4(a), 5(a)	2 1 per subsystem	E	SR 3.3.5.1.2 SR 3.3.5.1.5	[redacted] gpm and [redacted] gpm
e. Core Spray Pump Start — Time Delay Relay					
Pumps A,B,C,D (with diesel power)	1,2,3, 4(a), 5(a)	4 1 per pump	C	SR 3.3.5.1.5 SR 3.3.5.1.6	[redacted] seconds and [redacted] seconds
Pump A (with normal power)	1,2,3, 4(a), 5(a)	1	C	SR 3.3.5.1.5 SR 3.3.5.1.6	[redacted] seconds and [redacted] second
Pump B (with normal power)	1,2,3, 4(a), 5(a)	1	C	SR 3.3.5.1.5 SR 3.3.5.1.6	[redacted] seconds and [redacted] seconds

(continued)

(a) When associated subsystem(s) are required to be OPERABLE.

(b) Channels affect Common Accident Signal Logic. Refer to LCO 3.8.1, "AC Sources - Operating."

(f) 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.

Table 3.3.5.1-1 (page 2 of 6)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Core Spray System (continued)					
e. Core Spray Pump Start — Time Delay Relay (continued)					
Pump C (with normal power)	1,2,3, 4[(a)], 5[(a)]	1	C	SR 3.3.5.1.5 SR 3.3.5.1.6	[redacted] seconds and [redacted] seconds
Pump D (with normal power)	1,2,3, 4[(a)], 5[(a)]	1	C	SR 3.3.5.1.5 SR 3.3.5.1.6	[redacted] seconds and [redacted] seconds
2. Low Pressure Coolant Injection (LPCI) System					
a. Reactor Vessel Water Level — Low Low Low, Level 1[(f)]	1,2,3, 4[(a)], 5[(a)]	4	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	[redacted] above vessel zero
b. Drywell Pressure — High[(f)]	1,2,3	4	B	SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	[redacted] psig
c. Reactor Steam Dome Pressure — Low (Injection Permissive and ECCS Initiation)[(f)]	1,2,3	4	C	SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.6	[redacted] psig and psig
	4[(a)], 5[(a)]	4	B	SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.6	[redacted] psig and psig

(continued)

(a) When associated subsystem(s) are required to be OPERABLE.

(b) Deleted.

(f) 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.

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

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. LPCI System (continued)					
d. Reactor Steam Dome Pressure — Low (Recirculation Discharge Valve Permissive)(f)	1(c),2(c), 3(c)	4	C	SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.6	[redacted] psig and [redacted] psig
e. Reactor Vessel Water Level — Level 0	1,2,3	2 1 per subsystem	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	[redacted] above vessel zero
f. Low Pressure Coolant Injection Pump Start — Time Delay Relay					
Pump A,B,C,D (with diesel power)	1,2,3, 4(a), 5(a)	4	C	SR 3.3.5.1.5 SR 3.3.5.1.6	[redacted] seconds and [redacted] second
Pump A (with normal power)	1,2,3, 4(a), 5(a)	1	C	SR 3.3.5.1.5 SR 3.3.5.1.6	[redacted] seconds and [redacted] second
Pump B (with normal power)	1,2,3, 4(a), 5(a)	1	C	SR 3.3.5.1.5 SR 3.3.5.1.6	[redacted] seconds and [redacted] seconds
Pump C (with normal power)	1,2,3, 4(a), 5(a)	1	C	SR 3.3.5.1.5 SR 3.3.5.1.6	[redacted] seconds and [redacted] seconds
Pump D (with normal power)	1,2,3, 4(a), 5(a)	1	C	SR 3.3.5.1.5 SR 3.3.5.1.6	[redacted] seconds and [redacted] seconds

(continued)

(a) When the associated subsystem(s) are required to be OPERABLE.

(c) With associated recirculation pump discharge valve open.

(f) 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.

Table 3.3.5.1-1 (page 4 of 6)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
3. High Pressure Coolant Injection (HPCI) System					
a. Reactor Vessel Water Level — Low Low, Level 2(f)	1, 2(d), 3(d)	4	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	[redacted] inches above vessel zero
b. Drywell Pressure — High(f)	1, 2(d),3(d)	4	B	SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	[redacted] psig
c. Reactor Vessel Water Level — High, Level 8	1, 2(d), 3(d)	2	C	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	[redacted] inches above vessel zero
d. Condensate Header Level — Low	1, 2(d), 3(d)	1	D	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.6	[redacted] feet
e. Suppression Pool Water Level — High	1, 2(d), 3(d)	1	D	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.6	[redacted] inches above instrument zero
f. High Pressure Coolant Injection Pump Discharge Flow—Low (Bypass)	1, 2(d), 3(d)	1	E	SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	[redacted] gpm
4. Automatic Depressurization System (ADS) Trip System A					
a. Reactor Vessel Water Level — Low Low Low, Level 1(f)	1, 2(d), 3(d)	2	F	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	[redacted] inches above vessel zero

(continued)

(d) With reactor steam dome pressure [redacted] psig.

(f) 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.

Table 3.3.5.1-1 (page 5 of 6)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
4. ADS Trip System A (continued)					
b. Drywell Pressure — High(f)	1, 2(d), 3(d)	2	F	SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	[redacted] psig
c. Automatic Depressurization System Initiation Timer	1, 2(d), 3(d)	1	G	SR 3.3.5.1.5 SR 3.3.5.1.6	[redacted] seconds
d. Reactor Vessel Water Level — Low, Level 3 (Confirmatory)(f)	1, 2(d), 3(d)	1	F	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	[redacted] inches above vessel zero
e. Core Spray Pump Discharge Pressure — High	1, 2(d), 3(d)	4	G	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.6	[redacted] psig and [redacted] psig
f. Low Pressure Coolant Injection Pump Discharge Pressure — High	1, 2(d), 3(d)	8	G	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.6	[redacted] psig and [redacted] psig
g. Automatic Depressurization System High Drywell Pressure Bypass Timer	1, 2(d), 3(d)	2	G	SR 3.3.5.1.5 SR 3.3.5.1.6	[redacted] seconds
5. ADS Trip System B					
a. Reactor Vessel Water Level — Low Low Low, Level 1(f)	1, 2(d), 3(d)	2	F	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	[redacted] inches above vessel zero

(continued)

(d) With reactor steam dome pressure [redacted] psig.

(f) 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.

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

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
5. ADS Trip System B (continued)					
b. Drywell Pressure — High[f]	1, 2[d], 3[d]	2	F	SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	[redacted] psig
c. Automatic Depressurization System Initiation Timer	1, 2[d], 3[d]	1	G	SR 3.3.5.1.5 SR 3.3.5.1.6	[redacted] seconds
d. Reactor Vessel Water Level — Low, Level 3 (Confirmatory)[f]	1, 2[d], 3[d]	1	F	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	[redacted] inches above vessel zero
e. Core Spray Pump Discharge Pressure — High	1, 2[d], 3[d]	4	G	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.6	[redacted] psig and [redacted] psig
f. Low Pressure Coolant Injection Pump Discharge Pressure — High	1, 2[d], 3[d]	8	G	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.6	[redacted] psig and [redacted] psig
g. Automatic Depressurization System High Drywell Pressure Bypass Timer	1, 2[d], 3[d]	2	G	SR 3.3.5.1.5 SR 3.3.5.1.6	[redacted] seconds

(d) With reactor steam dome pressure [redacted] psig.

(f) 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.

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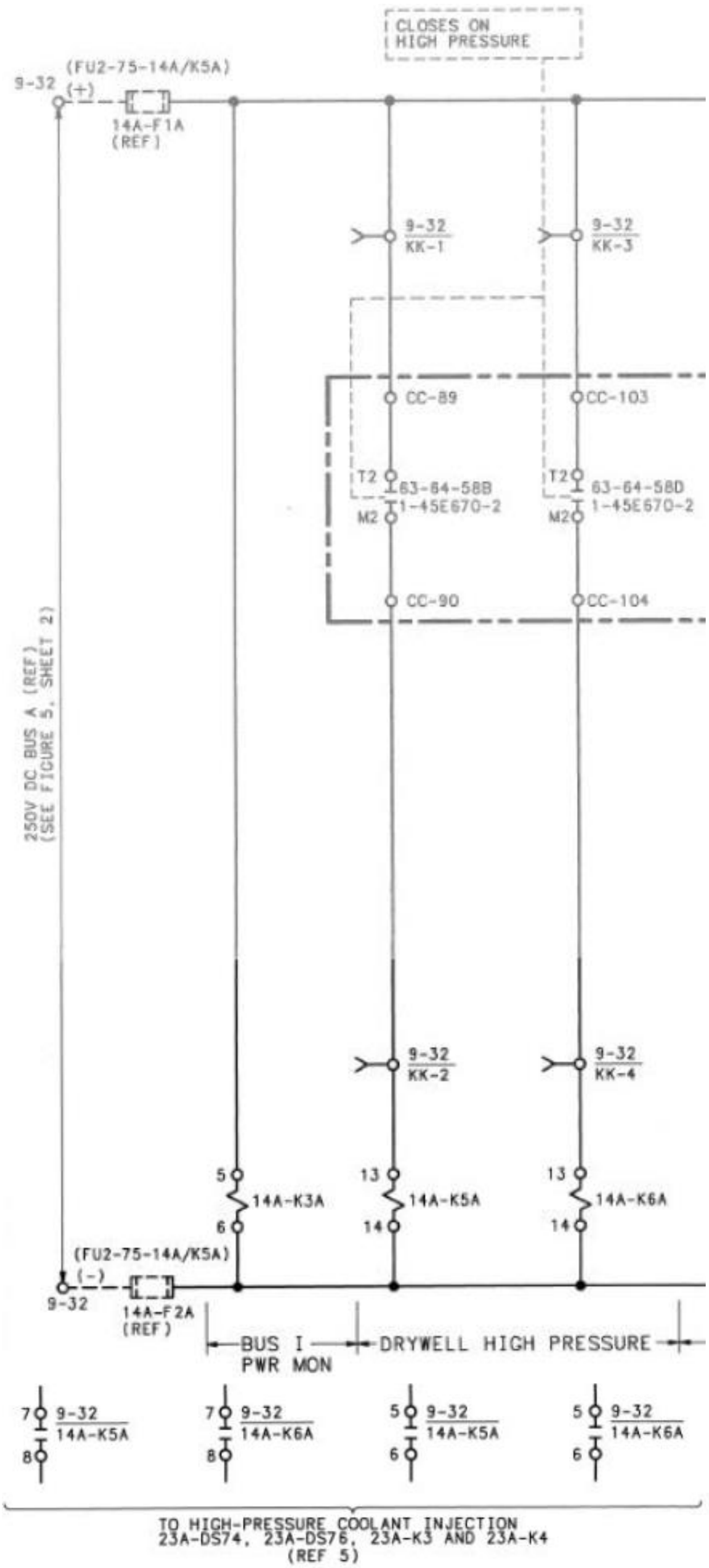

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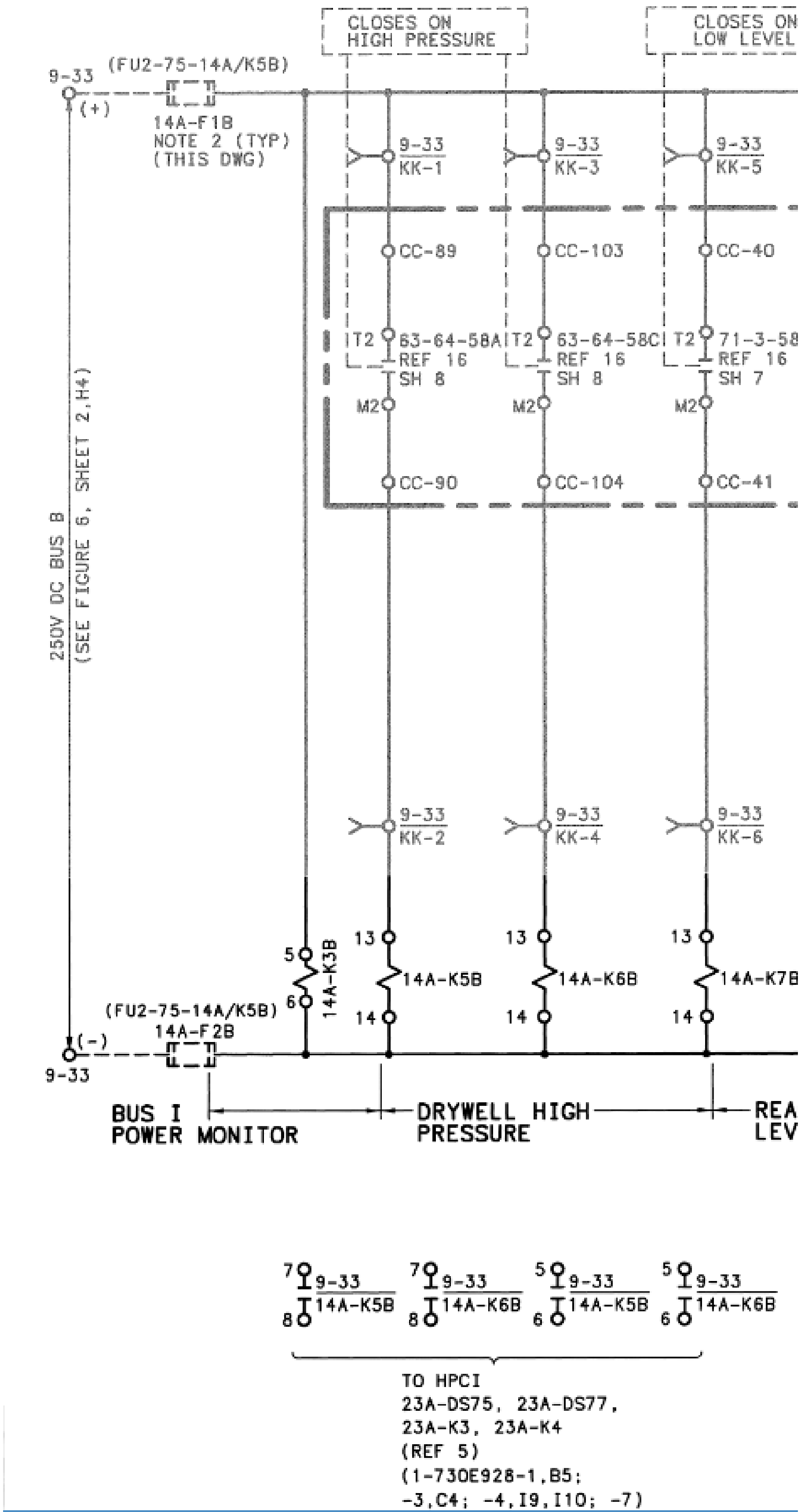

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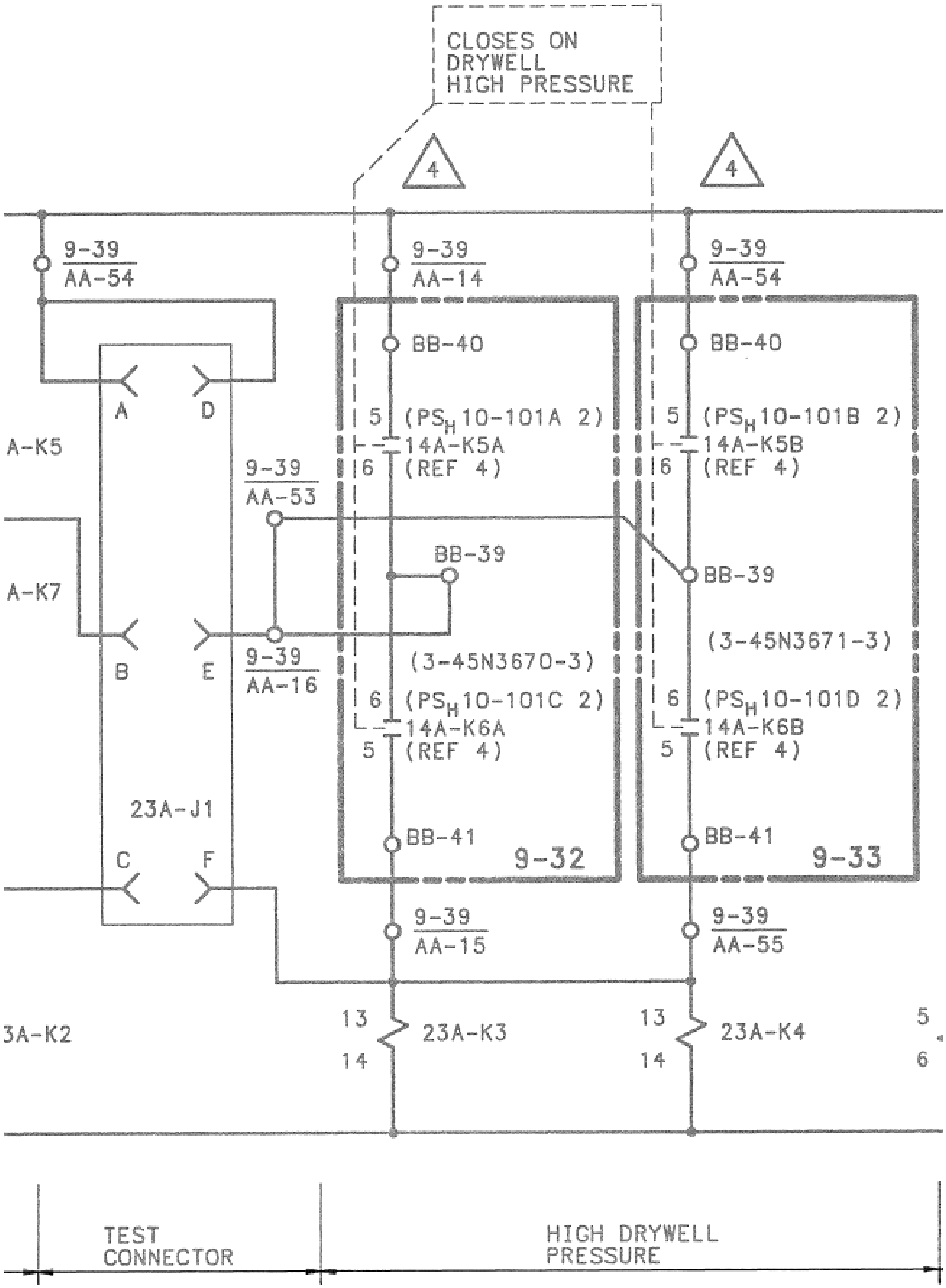

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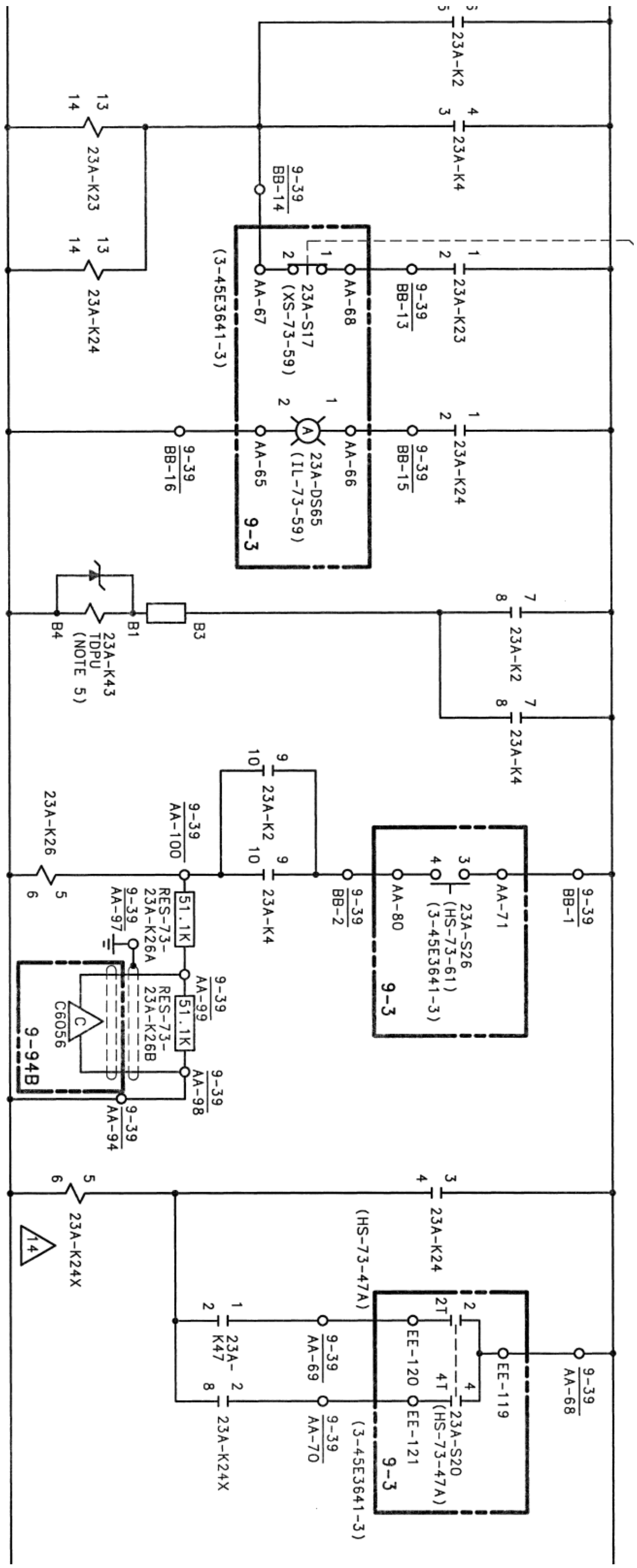

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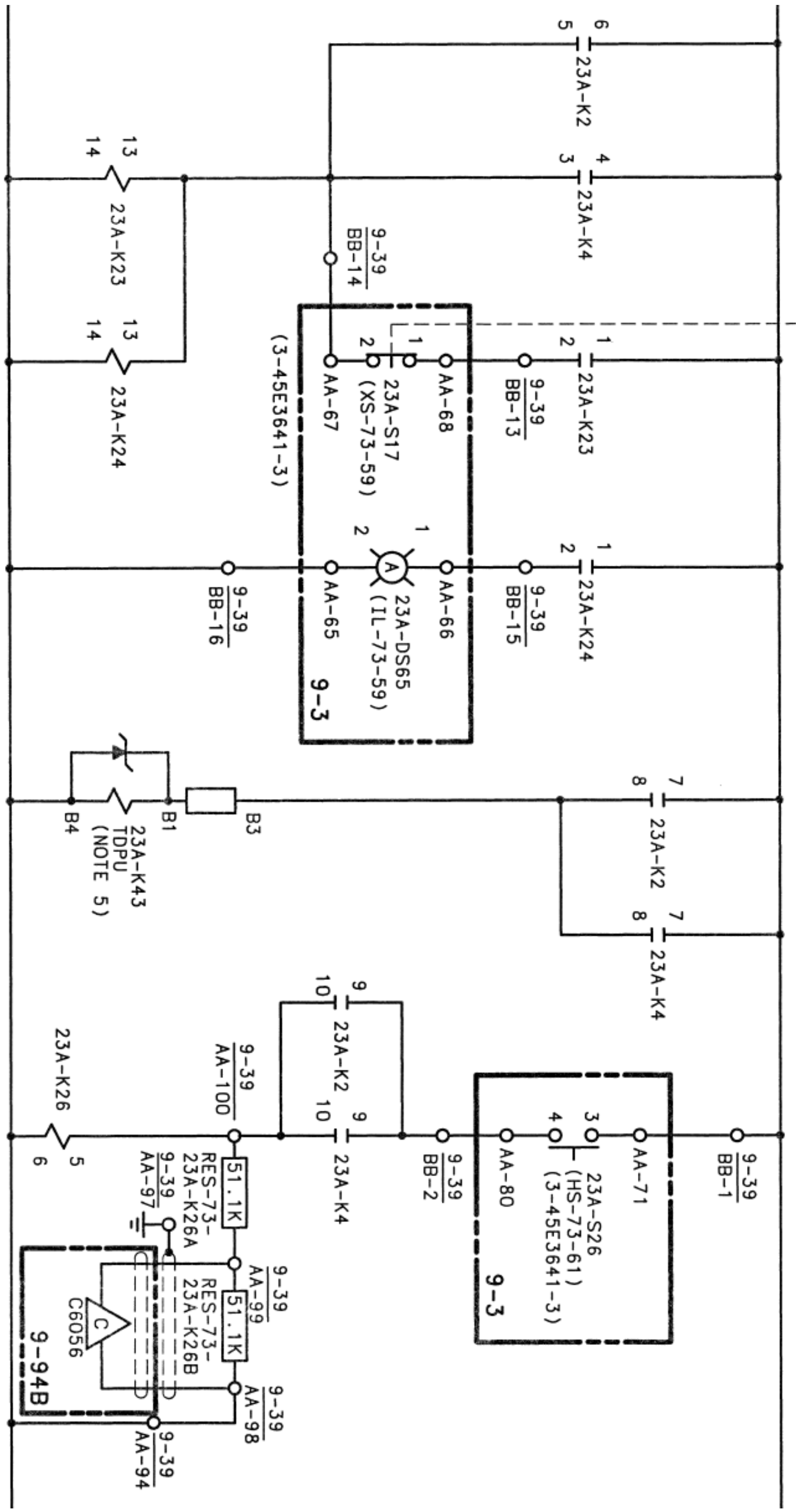

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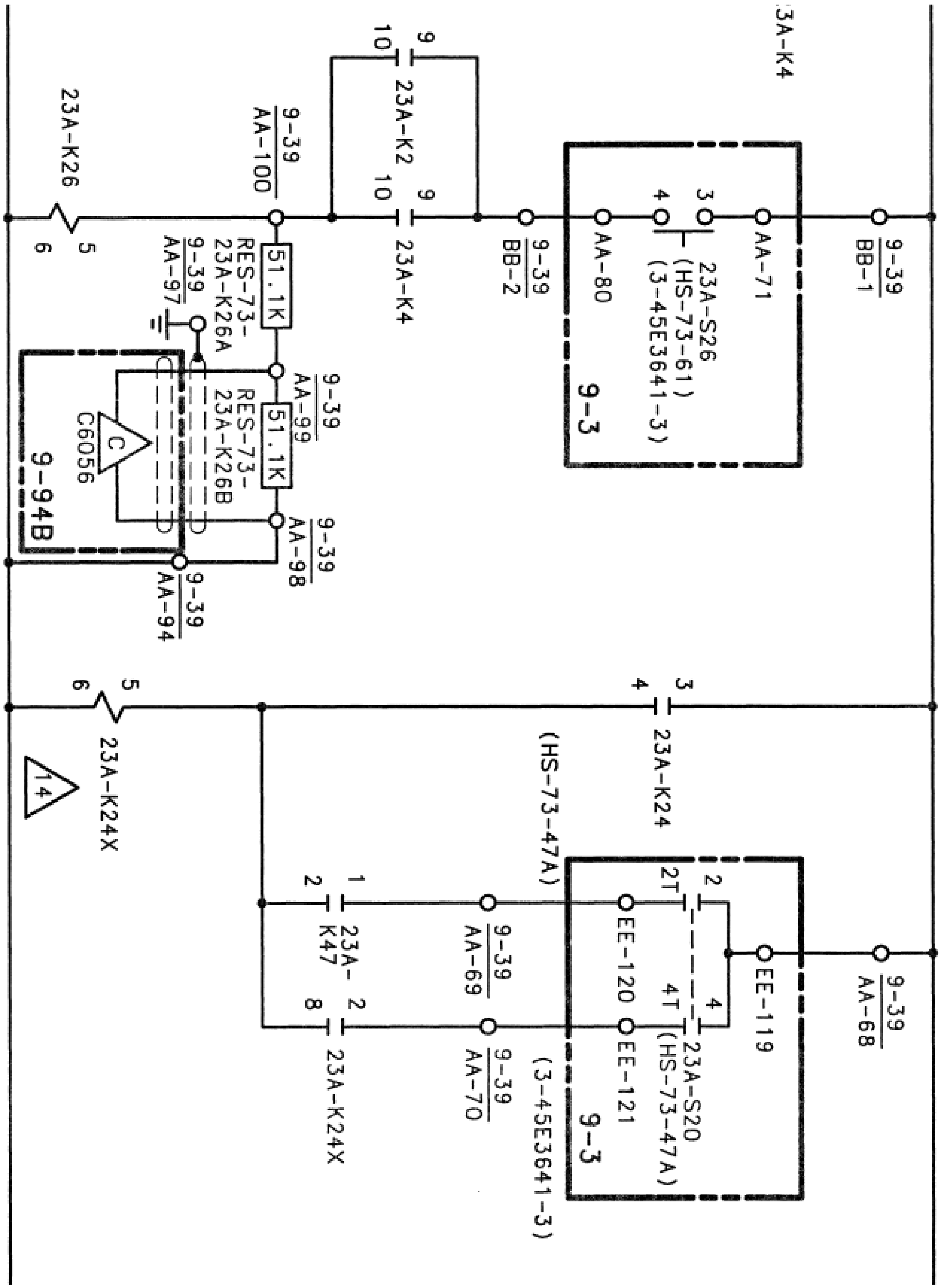

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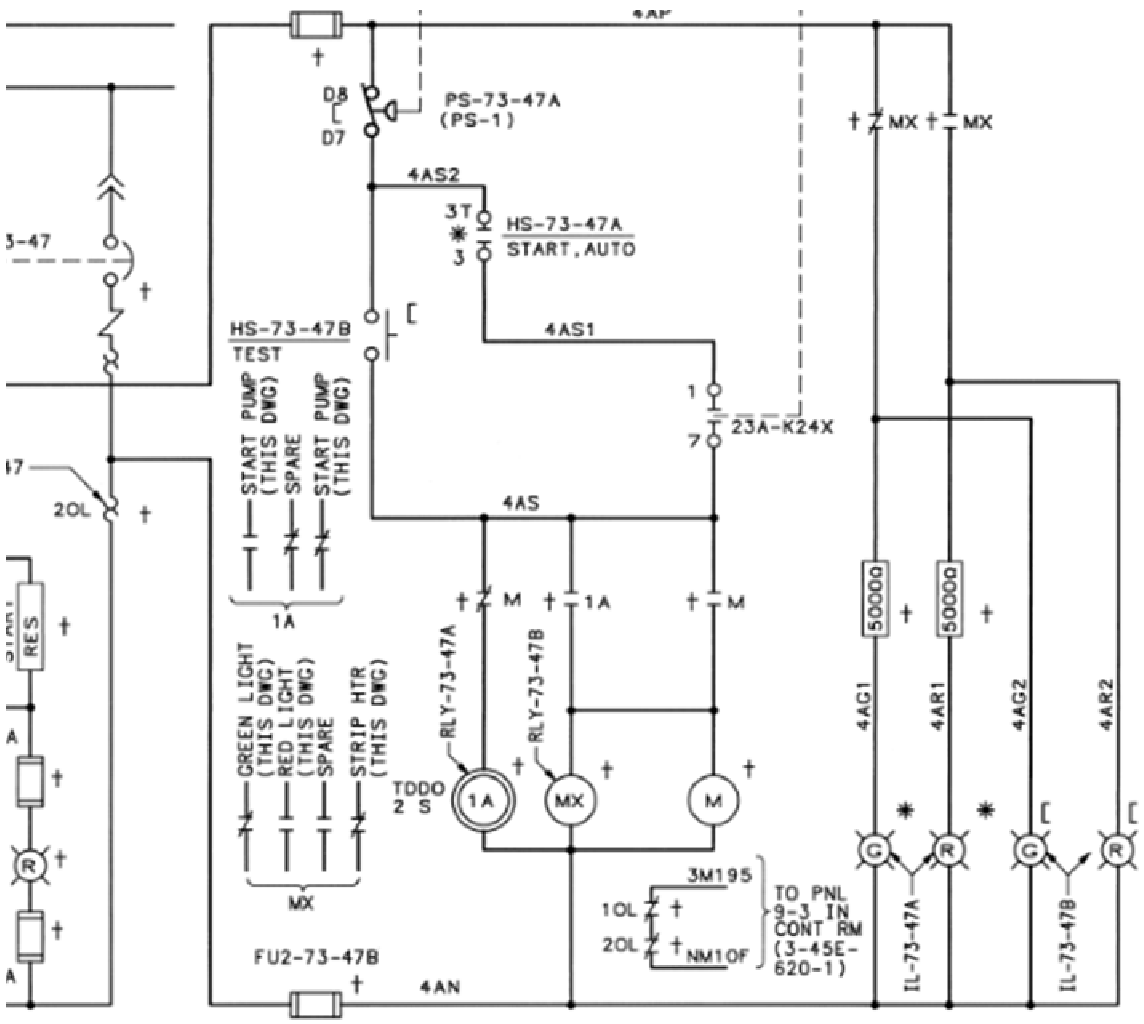

HPCI TURBINE AUX OIL PUMP

Attachment 1
(Page 1 of 18)

Reporting of Events or Conditions Affecting Licensed Nuclear Power Plants

1.0 PURPOSE

This Attachment identifies reporting requirements; and instructions for determining reportability, preparation, and transmittal of LERs; and notification to NRC for events occurring at TVA's licensed nuclear plants.

2.0 SCOPE

TVA is required by §50.72 and §50.73 to promptly report various types of conditions or events and provide written follow-up reports, as appropriate. This Attachment provides reporting guidance applicable to licensed power reactors.

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(Page 2 of 18)

Reporting of Events or Conditions Affecting Licensed Nuclear Power Plants

2.0 SCOPE (continued)

NOTES

1) Attachment 2, provides additional reporting criteria found in §Part 20, 30, 40, and 70 that may be applicable to events involving byproduct, source or special nuclear material possessed by the licensed nuclear plant. Site Licensing and Site Radiation Protection are responsible for making the reportability determinations for §Part 20, 30, 40, or 70 events associated with their site. Corporate Nuclear Regulatory Affairs and Corporate Chemistry are responsible for making the reportability determinations for §Part 20, 30, 40, or 70 events associated with all other TVA licensed activities. Depending on the location of the licensed activity, either Site or Corporate Nuclear Regulatory Affairs is responsible for developing (with input from affected organizations) and submitting the immediate notification and written reports to NRC in accordance with §Part 20, 30, 40, or 70 requirements. Reporting requirements for personnel exposure required by §Part 20 are contained in RCTP-105, Personnel Inprocessing and Dosimetry Administrative Processes.

2) Attachment 3 contains the criteria for reporting if events or conditions affecting ISFSI. TVA, as the general licensee of the ISFSI, is required by §72.216 to make initial and written reports in accordance with §72.74 and §72.75. Operations is responsible for making the reportability determinations for §72.74 and §72.75 reports. For any event, condition, or issue having the potential for being reportable, contact Site Licensing for consultation and concurrence on the reportability determination. In no event will the lack of licensing concurrence result in a failure to meet specified reporting timeframes. Operations is responsible for making the immediate notification to NRC in accordance with §72.74. Operations is responsible for making the immediate, 4-hour, and 24-hour notifications to NRC in accordance with §72.75. Site Licensing is responsible for developing (with input from affected organizations) and submitting the written reports required by §72.75.

3) Reporting requirements for events or conditions affecting the physical protection of the licensed nuclear plant specified in §73.71 are contained in NSDP-1, Safeguards Event Reporting Guidelines. Responsibilities for determining reportability and immediate notification requirements are assigned to Site Nuclear Security and Corporate Nuclear Security. The Site Operations Shift Manager is responsible for making the immediate notification to the NRC. Site Licensing is responsible for developing (with input from affected organizations) and submitting the written reports required by §73.71.

4) Reporting requirements for events or conditions affecting cyber security specified in §73.77 are contained in NPG-SPP-12.8.8, Cyber Security Incident Response. Responsibilities for determining reportability and immediate notification requirements are assigned to Nuclear Cyber Security (Corporate Computer Engineering Group). The Site Operations Shift Manager is responsible for making the immediate notification to the NRC. Site Licensing is responsible for developing (with input from affected organizations) and submitting the written reports required by §73.77.

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(Page 3 of 18)

Reporting of Events or Conditions Affecting Licensed Nuclear Power Plants

3.0 REQUIREMENTS

NOTES

1) Internal management notification requirements for reportable plant events and conditions are found in Procedure NPG-SPP-01.12, TVA Nuclear Event Response Process. The Operations Shift Manager is responsible for notifying Site Operations Management and the Site Duty Plant Manager. The Site Duty Plant Manager is responsible for making the remaining internal management notifications to the NDO who communicates to the fleet executives, in accordance with NPG-SPP-01.6, Nuclear Duty Officer. Internal management notification of emergent issues is described in Procedure NPG-SPP-01.12, TVA Nuclear Event Response Process.

2) NRC NUREG-1022, Revision 3 and subsequent supplements and revisions should be used, in its entirety, as guidance for determining reportability of plant events pursuant to §50.72 and §50.73. A text searchable copy of NUREG-1022 is maintained on the TVA Nuclear Regulatory Affairs SharePoint.

3) In addition to reviewing the clarifying discussion and examples associated with specific reporting criteria, the discussion of utilization of engineering judgment when evaluating Unanalyzed Conditions in NUREG -1022, Section 3.2.4(B), NUREG-1022, Section 2, Reporting Areas Warranting Special Mention, should also be reviewed. [R.1]

3.1 Immediate Notification - NRC

TVA is required by §50.72 and §73.71 to notify NRC immediately if certain types of events occur. This Attachment contains the types of events and the allotted time in which NRC must be notified. (Refer to NRC Form 361 at www.nrc.gov). Operations is responsible for making the reportability determinations for §50.72 and §50.73 reports. Site Nuclear Security and Corporate Nuclear Security are responsible for making the reportability determinations for §73.71 reports. For any §50.72, §50.73, or §73.71 event, condition, or issue having the potential for being reportable, contact Site Licensing for consultation and concurrence on the reportability determination. In no event will the lack of licensing concurrence result in a failure to meet specified reporting timeframes. Operations is responsible for making the immediate notification to NRC in accordance with §50.72. The Site Security Manager will request the Plant Shift Manager to call the NRC Operations Center, when appropriate.

Notification is via the Emergency Notification System. If the Emergency Notification System is not operative, use a telephone, telegraph, mailgram, or facsimile.

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(Page 4 of 18)

Reporting of Events or Conditions Affecting Licensed Nuclear Power Plants

3.1 Immediate Notification - NRC (continued)

NOTE

The NRC Event Notification Worksheet may be used in preparing for notifying the NRC. This Worksheet may be obtained directly from the NRC website (www.nrc.gov) by performing a "Form 361" search. Attachment 12 provides guidance for completing NRC Form 361.

A. The Immediate Notification Criteria of §50.72 is divided into 1-hour, 4-hour, and 8- hour phone calls. Notify the NRC Operations Center within the applicable time limit for any item which is identified in the Immediate Notification Criteria.

B. The following criteria require 1-hour notification:

1. 10 CFR 50.36(c)(1)(i)(A), (Technical Specifications) - Safety Limits as defined by the Technical Specifications which have been exceeded (violated)

NOTE

If it is discovered that a condition existed which met the Emergency Plan criteria but no emergency was declared and the basis for the emergency class no longer exists at the time of discovery, an ENS notification (and notification of the Operations Duty Specialist), within one hour of discovery of the undeclared (or misclassified) event, will be made. However, actual declaration of the emergency class is not necessary in these circumstances.

2. §50.72(a)(1)(i) - The declaration of any of the Emergency classes specified in the licensee's approved Emergency Plan.

3. §50.72(b)(1) - Any deviation from the plant's Technical Specifications authorized pursuant to §50.54(x).

4. 10 CFR 73, Appendix G, paragraph I - Safeguards Events. The requirements of §73.71, Reporting of Safeguard Events, are also applicable. Refer to NSDP-1, "Safeguards Event Reporting Guidelines," for additional information.

   a. Any event in which there is reason to believe that a person has committed or caused, or attempted to commit or cause, or has made a credible threat to commit or cause:

      (1) A theft or unlawful diversion of special nuclear material; or

Attachment 1
(Page 5 of 18)

Reporting of Events or Conditions Affecting Licensed Nuclear Power Plants

3.1 Immediate Notification - NRC (continued)

(2) Significant physical damage to a power reactor or any facility possessing SSNM or its equipment or carrier equipment transporting nuclear fuel or spent nuclear fuel, or to the nuclear fuel or spent nuclear fuel a facility or carrier possesses; or

NOTE

A Confirmed Cyber Attack at any TVA Nuclear site is reported to the NRC in accordance with the requirements of 10 CFR 73.77 and NPG-SPP-12.8.8.

(3) Interruption of normal operation of a licensed nuclear power reactor through the unauthorized use of or tampering with its machinery, components, or controls including the security system.

b. An actual entry of an unauthorized person into a protected area, material access area, controlled access area, vital area, or transport.

c. Any failure, degradation, or the discovered vulnerability in a safeguard system that could allow unauthorized or undetected access to a protected area, material access area, controlled access area, vital area, or transport for which compensatory measures have not been employed.

d. The actual or attempted introduction of contraband into a protected area, material access area, vital area, or transport (refer to NSDP-1 Attachment 23).

C. The following criteria require 4-hour notification:

1. §50.72(b)(2)(i) - The initiation of any nuclear plant shutdown required by the plant's Technical Specifications.

2. §50.72(b)(2)(iv)(A) - Any event that results or should have resulted in Emergency Core Cooling System (ECCS) discharge into the reactor coolant system as a result of a valid signal except when the actuation results from and is part of a pre-planned sequence during testing or reactor operation.

3. §50.72(b)(2)(iv)(B) - Any event or condition that results in actuation of the reactor protection system (RPS) when the reactor is critical except when the actuation results from and is part of a pre-planned sequence during testing or reactor operation.

Attachment 1
(Page 6 of 18)

Reporting of Events or Conditions Affecting
Licensed Nuclear Power Plants

3.1 Immediate Notification - NRC (continued)

NOTES

1) NPG-SPP-05.14 provides additional instructions regarding addressing and informally communicating events to outside agencies involving radiological spills and leaks.

2) Routine or day-to-day communications between TVA organizations and state agencies typically do not constitute a formal notification to other government agencies that would require a report in accordance with §50.72(b)(2)(xi).

4. §50.72(b)(2)(xi) - Any event or situation, related to the health and safety of the public or onsite personnel, or protection of the environment, for which a news release is planned or notification to other government agencies has been or will be made. Such an event may include an onsite fatality or inadvertent release of radioactive contaminated materials.

D. The following criteria require 8-hour notification:

NOTE

With the exception of “Events or Conditions that Could Have Prevented Fulfillment of a Safety Function," ENS notifications are required for any event that occurred within three years of discovery, even if the event was not ongoing at the time of discovery.

1. §50.72(b)(3)(ii)(A) - Any event or condition that results in the condition of the nuclear power plant, including its principal safety barriers, being seriously degraded.

2. §50.72(b)(3)(ii)(B) - Any event or condition that results in the nuclear power plant being in an unanalyzed condition that significantly degrades plant safety.

3. §50.72(b)(3)(iv)(A) - Any event or condition that results in valid actuation of any of the systems listed in paragraph (b)(3)(iv)(B) [see list below], except when the actuation results from and is part of a pre-planned sequence during testing or reactor operation.

   a. The systems to which the requirements of paragraph §50.72(b)(3)(iv)(A) apply are:

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(Page 7 of 18)

Reporting of Events or Conditions Affecting Licensed Nuclear Power Plants

3.1 Immediate Notification - NRC (continued)

NOTE
Actuation of the RPS when the reactor is critical is also reportable under §50.72(b)(2)(iv)(B) above.

(1) Reactor protection system (RPS) including: reactor scram or reactor trip.

(2) General containment isolation signals affecting containment isolation valves in more than one system or multiple main steam isolation valves (MSIVs).

(3) Emergency core cooling systems (ECCS) for pressurized water reactors (PWRs) including: high-head, intermediate-head, and low-head injection systems and the low pressure injection function of residual (decay) heat removal systems.

(4) ECCS for boiling water reactors (BWRs) including: core spray systems; high-pressure coolant injection system; low pressure injection function of the residual heat removal system.

(5) BWR reactor core isolation cooling system; isolation condenser system; and feedwater coolant injection system.

(6) PWR auxiliary or emergency feedwater system.

(7) Containment heat removal and depressurization systems, including containment spray and fan cooler systems.

(8) Emergency ac electrical power systems, including: Emergency diesel generators (EDGs).

Attachment 1
(Page 8 of 18)

Reporting of Events or Conditions Affecting Licensed Nuclear Power Plants

3.1 Immediate Notification - NRC (continued)

NOTES

1) For systems within scope, the inadvertent TS inoperability of a system in a required mode of applicability constitutes an event or condition for which there is no longer reasonable expectation that equipment can fulfill its safety function. Therefore, such events or conditions are reportable as an "Event or Condition that Could Have Prevented Fulfillment of a Safety Function."

2) According to §50.72(b)(3)(vi) events covered by §50.72(b)(3)(v) “may include one or more procedural errors, equipment failures, and/or discovery of design, analysis, fabrication, construction, and/or procedural inadequacies. However, individual component failures need not be reported pursuant [this paragraph] if redundant equipment in the same system was operable and available to perform the required safety function.”

4. §50.72(b)(3)(v) - Any event or condition that at the time of discovery could have prevented the fulfillment of the safety function of structures or systems that are needed to:

   (A) Shut down the reactor and maintain it in a safe shutdown condition;

   (B) Remove residual heat;

   (C) Control the release of radioactive material; or

   (D) Mitigate the consequences of an accident.

5. §50.72(b)(3)(xii) - Any event requiring the transport of a radioactively contaminated person to an offsite medical facility for treatment.

NOTE

NPG-SPP-03.5.1, Reporting Requirements for Loss of Emergency Preparedness Capabilities, provides TVA site specific guidance for event notifications required by 10 CFR 50.72(b)(3)(xiii).

6. §50.72(b)(3)(xiii) - Any event that results in a major loss of emergency assessment capability, offsite response capability, or offsite communications capability (for example, significant portion of control room indication, emergency notification system, or offsite notification system).

E. Follow-up Notification (§50.72(c))

With respect to the telephone notifications made under paragraphs (a) and (b) [§50.72(a) and §50.72(b), respectively] of this section [§50.72], in addition to making the required initial notification, during the course of the event:

Attachment 1
(Page 9 of 18)

Reporting of Events or Conditions Affecting Licensed Nuclear Power Plants

3.1 Immediate Notification - NRC (continued)

1. Immediately report:

   (i) Any further degradation in the level of safety of the plant or other worsening plant conditions including those that require the declaration of the Emergency Classes, if such a declaration has not been previously made; or

   (ii) Any change from one Emergency Class to another, or

   (iii) A termination of the Emergency Class.

(1) Immediately report:

   (i) The results of ensuing evaluations or assessments of plant conditions,

   (ii) The effectiveness of response or protective measures taken, and

   (iii) Information related to plant behavior that is not understood.

(2) Maintain an open, continuous communication channel with the NRC Operations Center upon request by the NRC.

3.2 Twenty-Four Hour Notification - NRC

Any violation of the requirement contained in specific operating license conditions, shall be reported to NRC in accordance with the license condition.

3.3 Two-Day Notification - NRC

§50.9(b) - The NRC shall be notified of incomplete or inaccurate information which contains significant implications for the public health and safety or common defense and security. Notification shall be provided to the administrator of the appropriate regional office within two working days of identifying the information. Depending on where the information originates, either Corporate Nuclear Regulatory Affairs or Site Licensing is responsible for determining reportability (with input from affected organizations) and notifying NRC in accordance with §50.9.

Reporting of Events or Conditions Affecting Licensed Nuclear Power Plants

3.4 Sixty-Day Verbal Report

§50.73(a)(2)(iv)(A) requires that any event or condition that resulted in manual or automatic actuation of the specified systems be reported as a Licensee Event Report (LER [Refer to Attachment 1, Section 3.5]). This CFR section also allows that in the case of an invalid actuation, other than actuation of the reactor protection system when the reactor is critical, an optional telephone notification may be placed to the NRC Operations Center within 60 days after discovery of the event instead of submitting a written LER.

Reporting of Events or Conditions Affecting Licensed Nuclear Power Plants

3.4 Sixty-Day Verbal Report (continued)

A. Telephone Report Required Content:

If the telephone notification option is selected (NUREG 1022, Revision 3, Section 3.2.6, System Actuation), instead of an LER, the verbal report:

1. Is not considered an LER.

2. Should identify that the report is being made under §50.73(a)(2)(iv)(A).

3. Should provide the following information:

   a. The specific train(s) and system(s) that were actuated.

   b. Whether each train actuation was complete or partial.

   c. Whether or not the system started and functioned successfully.

NOTE
Licensing will ensure that the information that is provided to NRC during the 60-Day telephone report is verified in accordance with NPG-SPP-03.10.

B. Telephone Report Development and Review

Licensing will:

1. Develop (with input from responsible organization) the response (report summary) to address the required input.

2. Ensure that the reporting details are approved by site vice president or his designee prior to making the verbal report.

C. Telephone Report Timeliness

Operations will make the 60-day telephone report promptly after the response is approved by the site vice president or his designee.

3.5 Written Report - NRC

A. For events in which safety limits or limiting safety system settings are exceeded, reports are made as required by 10 CFR 50.72 and 50.73.

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Reporting of Events or Conditions Affecting Licensed Nuclear Power Plants

3.5 Written Report - NRC (continued)

B. Any violation of the requirements contained in the Operating license conditions in lieu of other reporting requirements requires a written follow-up report if specified in the license.

C. Reporting Radiation Injuries

1. §140.6(a) requires, as promptly as practicable, submittal of a written notice (report) in the event of:

    a. Bodily injury or property damage arising out of or in connection with the possession or use of the radioactive material at the licensee’s facility (location); or

    b. In the course of transportation; or

    c. In the event any radiation exposure claim is made. (Refer to RCDP-9, Radiological and Chemistry Control Radiological Exposure Inquiries)

2. The written notice shall contain particulars sufficient to identify the licensee and reasonably obtainable information with respect to time, place, and circumstances thereof, or the nature of the claim.

D. Licensee Event Reports

A written report shall be prepared in accordance with §50.73(a)(1) for items in the 60-day report criteria or Technical Specifications. The report shall be complete and accurate in accordance with the methods outlined in this procedure. The completed forms shall be submitted to the USNRC, Document Control Desk, Washington, DC 20555. NUREG 1022, Revision 3, contains the instructions for completion of the LER form. Licensing is responsible for developing (with input from affected organizations) and submitting the written reports or optional telephone reports (refer to Attachment 1, Section 3.4) required by §50.73.

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Reporting of Events or Conditions Affecting Licensed Nuclear Power Plants

3.5 Written Report - NRC (continued)

NOTE

Unless otherwise specified in the reporting criteria below, an event will be reported if it occurred within three years of the date of discovery regardless of the plant mode or power level, and regardless of the significance of the structure, system, or component that initiated the event.

E. Report Criteria

1. §50.73(a)(2)(i)(A) - The completion of any nuclear plant shutdown required by the plant's Technical Specifications.

2. §50.73(a)(2)(i)(B) - Any operation or condition which was prohibited by the plant's Technical Specifications, except when:

   a. The Technical Specification is administrative in nature;

   b. The event consisted solely of a case of a late surveillance test where the oversight was corrected, the test was performed, and the equipment was found to be capable of performing its specified safety functions; or

   c. The Technical Specification was revised prior to discovery of the event such that the operation or condition was no longer prohibited at the time of discovery of the event.

3. §50.73(a)(2)(i)(C) - Any deviation from the plant's Technical Specifications authorized pursuant to §50.54(x).

4. §50.73(a)(2)(ii)(A) - Any event or condition that resulted in the condition of the nuclear power plant, including its principal safety barriers, being seriously degraded.

5. §50.73(a)(2)(ii)(B) - Any event or condition that resulted in the nuclear power plant being in an unanalyzed condition that significantly degraded plant safety.

6. §50.73(a)(2)(iii) - Any natural phenomenon or other external condition that posed an actual threat to the safety of the nuclear power plant or significantly hampered site personnel in the performance of duties necessary for the safe operation of the nuclear power plant.

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Reporting of Events or Conditions Affecting Licensed Nuclear Power Plants

3.5 Written Report - NRC (continued)

NOTE

In the case of an invalid actuation, other than actuation of the reactor protection system (RPS) when the reactor is critical, a telephone notification to the NRC Operations Center within 60 days after discovery of the event may be provided instead of submitting a written LER (§50.73(a)). Refer to Attachment 1, Section 3.4.

7. §50.73(a)(2)(iv)(A) - Any event or condition that resulted in manual or automatic actuation of any of the systems listed in paragraph (a)(2)(iv)(B) [see list under Item 8 below], except when

   a. The actuation resulted from and was part of a pre-planned sequence during testing or reactor operation; or

   b. The actuation was invalid and

      (i) Occurred while the system was properly removed from service or

      (ii) Occurred after the safety function had been already completed.

8. §50.73(a)(2)(iv)(B) - The systems to which the requirements to paragraph (a)(2)(iv)(A) of this section apply are:

   a. Reactor protection system (RPS) including: reactor scram or reactor trip.

   b. General containment isolation signals affecting containment isolation valves in more than one system or multiple main steam isolation valves (MSIVs).

   c. Emergency core cooling systems (ECCS) for pressurized water reactors (PWRs) including: high-head, intermediate-head, and low-head injection systems and the low pressure injection function of residual (decay) heat removal systems.

   d. ECCS for boiling water reactors (BWRs) including: core spray systems; high-pressure coolant injection system; low pressure injection function of the residual heat removal system.

   e. BWR reactor core isolation cooling system; isolation condenser system; and feedwater coolant injection system.

   f. PWR auxiliary or emergency feedwater system.

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Reporting of Events or Conditions Affecting Licensed Nuclear Power Plants

3.5 Written Report - NRC (continued)

g. Containment heat removal and depressurization systems, including containment spray and fan cooler systems.

h. Emergency ac electrical power systems, including: emergency diesel generators (EDGs).

i. Emergency service water systems that do not normally run and that serve as ultimate heat sinks.

NOTE
Events reported below "may include one or more procedural errors, equipment failures, and/or discovery of design, analysis, fabrication, construction, and/or procedural inadequacies. However, individual component failures need not be reported pursuant to [this criterion] if redundant equipment in the same system was operable and available to perform the required safety function." [§50.73(a)(2)(vi)]

9. §50.73(a)(2)(v) - Any event or condition that could have prevented the fulfillment of the safety function of structures or systems that are needed to:

   (A) Shut down the reactor and maintain it in a safe shutdown condition;

   (B) Remove residual heat;

   (C) Control the release of radioactive material; or

   (D) Mitigate the consequences of an accident.

10. §50.73(a)(2)(vii) - Any event where a single cause or condition caused at least one independent train or channel to become inoperable in multiple systems or two independent trains or channels to become inoperable in a single system designed to:

   (A) Shut down the reactor and maintain it in a safe shutdown condition;

   (B) Remove residual heat;

   (C) Control the release of radioactive material; or

   (D) Mitigate the consequences of an accident.

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Reporting of Events or Conditions Affecting Licensed Nuclear Power Plants

3.5 Written Report - NRC (continued)

11. §50.73(a)(2)(viii)(A) - Any airborne radioactivity release that, when averaged over a time period of 1 hour, resulted in airborne radionuclide concentrations in an unrestricted area that exceeded 20 times the applicable concentration limits specified in Appendix B to Part 20, table 2, column 1.

12. §50.73(a)(2)(viii)(B) - "Any liquid effluent release that, when averaged over a time period of 1 hour, exceeds 20 times the applicable concentrations specified in Appendix B to Part 20, table 2, column 2, at the point of entry into the receiving waters (this is, unrestricted area) for all radionuclides except tritium and dissolved noble gases."

NOTE
Events covered below "may include cases of procedural error, equipment failure, and/or discovery of a design, analysis, fabrication, construction, and/or procedural inadequacy. However, licensees are not required to report an event pursuant to [this criterion] if the event results from a shared dependency among trains or channels that is a natural or expected consequence of the approved plant design or normal and expected wear or degradation." [§50.73(a)(2)(ix)(B)]

13. §50.73(a)(2)(ix)(A) - Any event or condition that as a result of a single cause could have prevented the fulfillment of a safety function for two or more trains or channels in different systems that are needed to:
    a. Shut down the reactor and maintain it in a safe shutdown condition;
    b. Remove residual heat;
    c. Control the release of radioactive material; or
    d. Mitigate the consequences of an accident.

14. §50.73(a)(2)(x) - Any event that posed an actual threat to the safety of the nuclear power plant or significantly hampered site personnel in the performance of duties necessary for the safe operation of the nuclear power plant including fires, toxic gas releases, or radioactive releases.

15. §73.1 and §73.77 security reporting requirements:
    a. 10 CFR 73, Appendix G, paragraph I (physical security events) - If a one hour notification is made in Attachment 1, section 3.1.B.4 of this procedure, then a written notification to the NRC is required within 60 days. Also refer to NSDP-1.

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Reporting of Events or Conditions Affecting Licensed Nuclear Power Plants

3.5 Written Report - NRC (continued)

b. 10 CFR 73.77 (cyber security events) - If a one hour or (specific) four hour notification is made in accordance with NPG-SPP-12.8.8, then a written security followup report (LER) is required to be submitted to the NRC within 60 days.

16. For reporting a defect found installed in the Plant's Safety Related Equipment, Radioactive Wastes System, and Special Nuclear Material within an LER, §Part 21, NRC Reporting of Defects and Noncompliance, see Attachment 5 in this procedure.

SQN and WBN Only

17. Non-radiological environmental reporting requirements to the NRC, as required from SQN and WBN Operating License (OL), Appendix B.

a. WBN or SQN shall record any occurrence of unusual or important environmental events. Unusual or important events are those that potentially could cause or indicate environmental impact causally related with station operation. The following are examples:

(1) Excessive bird impaction events;

(2) Onsite plant or animal disease outbreaks;

(3) Unusual mortality of any species protected by the Endangered Species Act of 1973;

(4) Fish kills near the plant site;

(5) Unanticipated or emergency discharges of waste water or chemical substances that exceeds the limits of, or is not authorized by, the NPDES permit and requires 24-hour notification to the County or State of Tennessee;

WBN Only - Examples 6 and 7

(6) Identification of any threatened or endangered species for which the NRC has not initiated consultation with the Federal Wildlife Service (FWS).

(7) Increase in nuisance organisms or conditions in excess of levels anticipated in station environmental impact appraisals.

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Reporting of Events or Conditions Affecting Licensed Nuclear Power Plants

3.5 Written Report - NRC (continued)

b. Once an unusual or important event has occurred, the required actions are:

(1) If required, SQN or WBN Site Licensing shall make a written report to the NRC in accordance with the NRC Non-routine Report, OL Appendix B, Subsections 5.4.2, within 30 days, in the event of a reportable occurrence in which a limit specified in a relevant permit or certificate issued by another Federal, State, or local agency is exceeded.

(2) For changes and renewals to permits and certifications, SQN and WBN Site Licensing shall make a written report to the NRC in accordance with the OL Appendix B, Subsections 5.5.2 (SQN and WBN Unit 2) and 3.2 (WBN Unit 1) within 30 days of approval.

3.6 Retraction or Cancellation of Event Reports

An ENS notification may be retracted via a follow-up telephone call. If an ENS notification is make and its later determined that the event or condition was not reportable, Plant Operations should call the NRC Operations Center on the ENS telephone to retract the notification and explain the rational for that decision. There is no set time limit for ENS telephone retractions. However, because most retractions occur following completion of engineering or management review, it is expected that retractions would occur shortly after such review.

Cancellation of LERs submitted should be made by letter. The letter should state that the LER is being cancelled (formally withdrawn). The bases for the cancellation should be explained so that the staff can review and understand the reasons supporting the determination.

BFN ILT 1407

NRC WRITTEN EXAM

KEY

LXR•TEST™ Response Form LXR-20020 Side 1

Name:

Signature:

Date:

READ CAREFULLY!

NOT OK

Use black ink only.

Mark responses darkly and fill completely.

Erase unwanted marks clearly.

Do NOT make any stray marks on the page

No credit will be given for improper marks

If Side 2 is used, fill in ID on both sides

DOCKET NUMBER

version

C1 C2 C3 C4 V

SECTION 1

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

SECTION 2