U. S. Nuclear Regulatory Commission **Site-Specific SRO Written Examination**	
Applicant Information	
Name:	
Date: 12/13/2016	Facility / Unit: Brunswick Unit 1/2
Region: I ☐ II ☒ III ☐ IV ☐	Reactor Type: W ☐ CE ☐ BW ☐ GE ☒
Start Time:	Finish Time:

Instructions

Use the answer sheets provided to document your answers. Staple this cover sheet on top of the answer sheets. To pass the examination you must achieve a final grade of at least 80 percent overall, with 70 percent or better on the SRO-only items if given in conjunction with the RO exam; SRO-only exams given alone require a final grade of 80 percent to pass. You have 8 hours to complete the combined examination, and 3 hours if you are only taking the SRO portion.

Applicant Certification

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

____
Applicant's Signature

Results	
RO/SRO-Only/Total Examination Values Points	____ / ____ / ____
Applicant's Score Points	____ / ____ / ____
Applicant's Grade Percent	____ / ____ / ____

1. Unit Two is operating at rated power when a control rod begins to drift out from position 24.

   Which one of the following identifies the **first** action to be taken by the operator at the controls (OATC)?

   A. Initiate a single rod scram.

   B. Initiate a manual reactor scram.

   C. Select and attempt to arrest the control rod.

   D. Select and fully insert the control rod to position 00.

2. Unit One is in an outage with the condensate system under clearance. An earthquake results in damage to the CST causing level to slowly lower.

Which one of the following completes the statement below with regards to the effect on the CRD system?

The CRD system will __(1)__ when the CST level reaches approximately __(2)__.

A. (1) trip
(2) 3 feet

B. (1) trip
(2) 11 feet

C. (1) transfer to the backup supply
(2) 3 feet

D. (1) transfer to the backup supply
(2) 11 feet

3. Unit One is at rated power.

   Which one of the following identifies the impact of inadvertently closing the 1A Reactor Recirculation Pump 1-B32-F031A, Pump A Disch Vlv?

   The 1A Reactor Recirculation pump speed will lower to approximately:

   A. 20%

   B. 34%

   C. 45.4%

   D. 48%

4. A line break has occurred in the Unit Two drywell with the following sequence of events:

1155	Drywell pressure rises above 1.7 psig
1202	RPV pressure drops below 410 psig
1203	RPV level drops to LL3

Which one of the following completes the statement below?

The **earliest** time that the operator can throttle the 2-E11-F048A, Loop 2A RHR Heat Exchanger Bypass Valve is at:

A. 1205.

B. 1206.

C. 1207.

D. 1208.

5. RHR Loop 2A is operating in the Shutdown Cooling mode of operation with the following parameters:

RHRSW Pump 2A	Operating
RHRSW Flow	4000 gpm
RHR Pump 2A	Operating
RHR Loop A Flow	6000 gpm

Which one of the following completes the statement below?

The required operator action to **lower** the cooldown rate IAW 2OP-17, *Residual Heat Removal System Operating Procedure*, is to throttle **closed**:

A. 2-E11-F003A, HX 2A Outlet Vlv.

B. 2-E11-F017A, Outboard Injection Vlv.

C. 2-E11-F048A, HX 2A Bypass Vlv.

D. 2-E11-PDV-F068A, HX 2A SW Disch Vlv.

6. A Group 1 isolation has occurred on Unit One.
HPCI has been placed in the pressure control mode of operation IAW 1OP-19, *High Pressure Coolant Injection System Operating Procedure.*
HPCI flow controller, E41-FIC-R600, is in manual with the output at midscale.

Which one of the following completes the statement below?

If the 1-E41-F008, Bypass To CST Valve, is throttled __(1)__ too far, this may result in HPCI __(2)__.

A. (1) open
(2) tripping on overspeed

B. (1) open
(2) operation below 2100 rpm

C. (1) closed
(2) tripping on overspeed

D. (1) closed
(2) operation below 2100 rpm

7. Unit Two is operating at rated power.
Due to a circuit malfunction an inadvertent LOCA initiation occurs in the Div II Core Spray logic causing A-03 (2-6), *CORE SPRAY SYSTEM II ACTUATED*, to alarm.

Which one of the following completes both statements below?

Core Spray Pump(s) __(1)__ will start.

__(2)__ will start.

A. (1) 2B ONLY
(2) All DGs

B. (1) 2B ONLY
(2) DG2 and DG4 ONLY

C. (1) 2A and 2B
(2) All DGs

D. (1) 2A and 2B
(2) DG2 and DG4 ONLY

8. Which one of the following completes the statement below concerning Core Spray Line Break Detection differential pressure instrument?

The __(1)__ leg of this DP instrument senses __(2)__ core plate pressure via the SLC/Core Differential Pressure penetration.

A. (1) variable
(2) below

B. (1) variable
(2) above

C. (1) reference
(2) below

D. (1) reference
(2) above

9. Which one of the following completes both statements below?

The normal power supply to RPS MG Set 2B is from 480V MCC __(1)__.

The **normal** alternate power supply to RPS B is from 480V Bus __(2)__.

A. (1) 2CA
   (2) E7

B. (1) 2CA
   (2) E8

C. (1) 2CB
   (2) E7

D. (1) 2CB
   (2) E8

10. Which one of the following identifies the LPRM detector level that provides input to the Rod Block Monitor system for indication ONLY, and is NOT used for the purpose of generating rod blocks?

    A. Level A

    B. Level B

    C. Level C

    D. Level D

11. Unit One is performing a startup with the reactor just declared critical. While ranging IRM G from range 1, the IRM will not change ranges and remains on Range 1.

Which one of the following completes both statements below?

When IRM G indication **first** exceeds __(1)__ on the 125 scale, annunciator A-05, 2-4, *IRM UPSCALE*, will alarm.

The action required IAW A-05, 2-4, *IRM UPSCALE*, is to __(2)__.

A. (1) 70
   (2) place the joystick on P603 for the IRM G to Bypass

B. (1) 70
   (2) withdraw the IRM G detector to maintain reading on scale

C. (1) 117
   (2) place the joystick on P603 for the IRM G to Bypass

D. (1) 117
   (2) withdraw the IRM G detector to maintain reading on scale

12. Which one of the following identifies the criteria for when SRM detectors can **first** begin to be withdrawn from the core IAW 0GP-02, *Approach To Criticality And Pressurization Of The Reactor*?

A. When all IRMs are above range 3.

B. When SRM counts reach 2 x 10^5 counts.

C. When RTRCT PERMIT light is illuminated.

D. When SRM/IRM overlap has been established.

13. Which one of the following identifies the power supply to the APRM channel NUMACs?

A. All APRM channels receive 120 VAC power from UPS

B. All APRM channels receive 120 VAC power from both RPS Bus A and RPS Bus B

C. APRM Channels 1 & 3 receive power from ONLY120 VAC RPS Bus A
APRM Channels 2 & 4 receive power from ONLY120 VAC RPS Bus B

D. APRM Channels 1 & 3 receive power from Division I 24/48 VDC
APRM Channels 2 & 4 receive power from Division II 24/48 VDC

14. Which one of the following completes the statement below?

An APRM must have at least __(1)__ of the assigned LPRMs operable with at least __(2)__ LPRM inputs per axial level operable.

A. (1) 18
   (2) 2

B. (1) 18
   (2) 3

C. (1) 17
   (2) 2

D. (1) 17
   (2) 3

15. Following a loss of feedwater, RCIC automatically initiated and subsequently tripped on low suction pressure.

Current plant status is:
Reactor water level is 150 inches
RCIC flow controller in Manual set at 200 gpm

Subsequently, the following actions are taken:
RCIC suction transferred to Torus
E51-V8, Turbine Trip and Throttle Valve is closed
E51-V8 is re-opened
PF push button on the RCIC flow controller is depressed

Which one of the following identifies the indicated flow on the RCIC flow controller that would be observed for these conditions?

A. 0 gpm

B. 200 gpm

C. 400 gpm

D. 500 gpm

16. Which one of the following completes both statements below concerning the Automatic Depressurization System (ADS) reactor water level inputs from the Nuclear Boiler System?

The __(1)__ instruments provide LL3 inputs to ADS initiation logic.

The __(2)__ range instruments provide LL1 inputs to ADS logic.

A. (1) Fuel Zone
   (2) Narrow

B. (1) Fuel Zone
   (2) Shutdown

C. (1) Wide range
   (2) Narrow

D. (1) Wide range
   (2) Shutdown

17. Unit One is operating at power with Core Spray Pump 1B under clearance.
A small break LOCA occurs simultaneously with a Loss of Off-site Power to both units.

DG1 and DG4 fail to start and tie onto their respective E bus.

The following plant conditions exist on Unit One:

A-03 (5-1) *Auto Depress Timers Initiated*	In alarm
A-03 (6-9) *Reactor Low Wtr Level Initiation*	In alarm
RPV pressure	600 psig
Drywell pressure	13 psig

Which one of the following completes both statements below?

ADS __(1)__ auto initiate.

After ADS is initiated (either automatically or manually), RPV water level __(2)__ be restored with **BOTH** RHR Loops.

A. (1) will
(2) will

B. (1) will
(2) will NOT

C. (1) will NOT
(2) will

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

18. Which one of the following completes the statement below concerning the Fuel Zone instruments, N036 and N037, during a loss of drywell cooling?

The reference leg density will __(1)__ causing the indicated level to read __(2)__ than actual level.

A. (1) rise
(2) higher

B. (1) rise
(2) lower

C. (1) lower
(2) higher

D. (1) lower
(2) lower

19. Unit One is at 75% power.
The 1A RPS MG set trips.
No operator actions have been taken.

Which one of the following identifies the Main Steam Line Isolation Valve (MSIV) logic lamp status on P601 panel?

	Inboard MSIV Logic		Outboard MSIV Logic	
	DC	AC	DC	AC
A.	LIT	OFF	LIT	OFF
B.	OFF	LIT	OFF	LIT
C.	LIT	OFF	OFF	LIT
D.	OFF	LIT	LIT	OFF

20. Which one of the following identifies the effect if both Refuel Bridge hoist grapple hooks are not open five seconds **after** placing the Engage/Release switch to Release?

A. Fuel Hoist Interlock is generated.

B. Engage amber light extinguishes.

C. Fault lockout is generated.

D. Grapple hooks will reclose.

21. Which one of the following identifies the SRV component that will prevent siphoning of water into the SRV discharge piping?

A. Vacuum breaker

B. Check Valve

C. T-Quencher

D. Sparger

22. Which one of the following identifies the criteria for tripping the main turbine IAW the Unit Two Scram Immediate Actions of 0EOP-01-UG, *Users Guide*?

A. When APRM's indicate downscale trip.

B. When steam flow is less than 3 Mlbs/hr.

C. When reactor water level is 160 inches and rising.

D. When reactor mode switch is placed in SHUTDOWN.

23. Which one of the following completes both statements below concerning the Main Generator Voltage Regulator?

The automatic voltage regulator maintains a constant generator __(1)__ voltage.

While in the automatic voltage regulation mode, the manual voltage regulator setting __(2)__ automatically follow the automatic setpoint.

A. (1) field
(2) does

B. (1) field
(2) does NOT

C. (1) terminal
(2) does

D. (1) terminal
(2) does NOT

24. Unit One Reactor Feed Pump 1B is operating in automatic DFCS control at 4500 RPM. The DFCS control signal to Reactor Feed Pump 1B woodward governor **immediately** fails downscale.

Which one of the following completes the statement below?

Reactor Feed Pump 1B speed will:

A. lower to 0 rpm.

B. lower to 1000 rpm.

C. lower to 2450 rpm.

D. remain at 4500 rpm.

25. Which one of the following completes both statements below concerning the reactor feed pump turbine (RFPT) DFCS controls?

During a RFPT startup, transfer to DFCS control is performed when RFPT speed is approximately __(1)__.

DFCS will automatically control the speed of the RFPT up to __(2)__.

A. (1) 1000 rpm
   (2) 5450 rpm

B. (1) 1000 rpm
   (2) 6150 rpm

C. (1) 2550 rpm
   (2) 5450 rpm

D. (1) 2550 rpm
   (2) 6150 rpm

26. Unit One primary containment venting is being performed IAW 1OP-10, *Standby Gas Treatment System Operating System,* with the following plant status:

1-VA-1F-BFV-RB, SBGT DW Suct Damper	Open
1-VA-1D-BFV-RB, Reactor Building SBGT Train 1A Inlet Valve	Closed
1-VA-1H-BFV-RB, Reactor Building SBGT Train 1B Inlet Valve	Closed

Which one of the following completes both statements below concerning the predicted SBGT response if drywell pressure rises to 1.9 psig?

1-VA-1F-BFV-RB __(1)__.

Both 1-VA-1D-BFV-RB and 1-VA-1H-BFV-RB __(2)__.

A. (1) auto closes
   (2) auto open

B. (1) auto closes
   (2) remain closed

C. (1) remains open
   (2) auto open

D. (1) remains open
   (2) remain closed

27. Unit One is operating at rated power.
Unit Two is in MODE 5 performing fuel movements.

Which one of the following completes both statements below IAW **Unit One** Tech Spec 3.8.1, AC Sources - Operating, LCO statement?

The Unit Two SAT __(1)__ required to be OPERABLE.

__(2)__ Diesel Generators are required to be OPERABLE.

A. (1) is
(2) Two

B. (1) is
(2) Four

C. (1) is NOT
(2) Two

D. (1) is NOT
(2) Four

28. Unit One is operating at rated power.

Subsequently, E1 breaker AU9, Feed to 480V Substation E5, trips.

Which one of the following completes the statement below?

120V UPS Distribution Panel 1A is:

A. de-energized.

B. energized from MCC 1CB.

C. energized from the Standby UPS.

D. energized from 250V DC SWBD A.

29. A reactor shutdown is in progress.
All IRMs on **range 1** reading between 15 and 20.
IRM B detector is failing downscale.

Which one of the following completes both statements below?

IAW A-05 (1-4) *IRM Downscale,* the alarm setpoint is __(1)__ on the 125 scale.

When the IRM downscale alarm is received, a rod block __(2)__ be generated.

A. (1) 3
(2) will

B. (1) 3
(2) will NOT

C. (1) 6.5
(2) will

D. (1) 6.5
(2) will NOT

30. Unit Two is operating at full power when a loss of DC Distribution Panel 4A occurs.

Which one of the following completes both statements below?

RCIC is __(1)__ for injection from the RTGB.

RCIC __(2)__ isolation logic has lost power.

A. (1) available
   (2) inboard

B. (1) available
   (2) outboard

C. (1) unavailable
   (2) inboard

D. (1) unavailable
   (2) outboard

31. Unit Two has lost off-site power.
DG3 started and tied to its respective E Bus.
Sequence of events:

1200 DG3 ties to E3
1205 DG3 lube oil temperature rises above 190°F
1206 DG3 lube oil pressure drops below 27 psig

Which one of the following identifies when DG3 will trip?

A. Immediately at 1205.

B. Immediately at 1206.

C. 45 seconds after 1205.

D. 45 seconds after 1206.

32. A Unit Two plant cooldown is being performed with the following plant conditions:

Reactor water level	175 inches, steady
Reactor pressure band	500 - 700 psig
Drywell ref leg temp	175°F

(REFERENCE PROVIDED)

Which one of the following completes both statements below?

The lowering of reactor pressure causes the N004A/B/C (Narrow Range) reactor water level instruments indicated level error to __(1)__.

The reactor water level that would correspond to Low level 4 (LL4) is __(2)__.

A. (1) increase
   (2) -60 inches

B. (1) increase
   (2) -65 inches

C. (1) decrease
   (2) -60 inches

D. (1) decrease
   (2) -65 inches

33. Unit Two is performing a startup IAW 0GP-02, *Approach to Criticality and Pressurization of the Reactor*.

IAW 0GP-02, which one of the following identifies the radiation monitor(s) that will require the alarm setpoints raised when HWC is placed in service?

A. D12-RM-K603A,B,C,D, Main Steam Line Rad Monitors

B. ARM Channel 2-9, U-2 Turbine Bldg Breezeway

C. D12-RR-4599-1,2,3, Main Stack Rad Monitors

D. ARM Channel 2-4, Cond Filter-Demin Aisle

34. Which one of the following identifies the power supply to 2D RHR Pump?

A. E1

B. E2

C. E3

D. E4

35. Unit One is operating at 70% power when the OATC observes indications for a failed jet pump. Subsequently, Recirc Pump 1A trips.

Which one of the following completes both statements below IAW 1AOP-04.0, *Low Core Flow*?

Performance of the jet pump operability surveillance for __(1)__ Loop Operation is required.

If it is determined that a jet pump has failed, the required action is to __(2)__.

A. (1) Single
(2) reduce reactor power below 25% rated thermal power

B. (1) Single
(2) commence unit shutdown IAW 0GP-05, Unit Shutdown

C. (1) Two
(2) reduce reactor power below 25% rated thermal power

D. (1) Two
(2) commence unit shutdown IAW 0GP-05, Unit Shutdown

36. Unit One is operating at rated power.
The load dispatcher reports degraded grid conditions with the following indications:

Generator frequency	59.7 hertz
230 KV Bus 1A voltage	205 KV
230 KV Bus 1B voltage	205 KV
E1 voltage	3690 volts
E2 voltage	3685 volts

Which one of the following completes both statements below?

The __(1)__ may be damaged with continued operation under these conditions.

IAW 0AOP-22.0, *Grid Instability,* the E-Bus master/slave breakers __(2)__ open.

A. (1) main turbine blades
(2) will

B. (1) main turbine blades
(2) will NOT

C. (1) emergency bus loads
(2) will

D. (1) emergency bus loads
(2) will NOT

37. Which one of the following completes both statements below?

IAW 0AOP-39.0, *Loss of DC Power*, before 125 VDC battery voltage reaches __(1)__, remove loads as directed by the Unit CRS.

IAW 1EOP-01-SBO, *Station Blackout*, if either division battery chargers can NOT be restored within __(2)__ then load strip the affected battery.

A. (1) 105 volts
   (2) 1 hour

B. (1) 105 volts
   (2) 2 hours

C. (1) 129 volts
   (2) 1 hour

D. (1) 129 volts
   (2) 2 hours

38. Which one of the following identifies the reason an operator is directed to trip the main turbine as an immediate action IAW 0AOP-32.0, Plant Shutdown From Outside Control Room?

A. To initiate a scram on TSV/TCV closure.

B. To prevent reverse power starts of the Diesel Generators.

C. The turbine cannot be tripped once the Control Room is evacuated.

D. To bring bypass valves into operation until Remote Shutdown Panel control is established.

39. Unit One has entered RSP with the following conditions:

Six control rods are at position 02, all others are fully inserted
B Recirc Pump has tripped

Which one of the following completes both statements below?

The control rods will be inserted by __(1)__ IAW 0EOP-01-LEP-02, *Alternate Control Rod Insertion*.

After the control rods are inserted, a CRD flow rate of approximately __(2)__ will be established.

A. (1) placing the individual scram test switches to the Scram position
   (2) 30 gpm

B. (1) placing the individual scram test switches to the Scram position
   (2) 45 gpm

C. (1) driving rods using RMCS
   (2) 30 gpm

D. (1) driving rods using RMCS
   (2) 45 gpm

40. A total loss of Unit One feedwater results in reactor water level lowering to 87 inches.
Drywell pressure is 2.1 psig.
Reactor water level is being restored with RCIC and CRD.

Which one of the following completes both statements below?

RVCP __(1)__ required to be entered.

The expected response of the G31-F001, Inboard RWCU Isolation Valve, and the G31-F004, Outboard RWCU Isolation Valve, is that __(2)__ should be closed.

A. (1) is
   (2) ONLY the G31-F004

B. (1) is
   (2) BOTH

C. (1) is NOT
   (2) ONLY the G31-F004

D. (1) is NOT
   (2) BOTH

41.

> **CAUTION**
>
> There are seven keylock *NORMAL/LOCAL* switches located on Diesel Generator 2 control panel. Six of these are located in a row. The seventh switch is located in the row above the six switches.

Which one of the following completes both statements below concerning the caution above from 0ASSD-02, *Control Building*?

The six switches in a row must be placed in *LOCAL* __(1)__ placing the seventh switch in *LOCAL*.

The purpose of this sequence is to prevent a loss of DG2 due to a loss of the redundant power supply fuses for the __(2)__ circuitry.

A. (1) before
   (2) output breaker

B. (1) before
   (2) engine run control

C. (1) after
   (2) output breaker

D. (1) after
   (2) engine run control

42. During accident conditions, the source term from the Unit One Reactor Building must be estimated. Three RB HVAC supply fans and three RB HVAC exhaust fans are running.

IAW 0PEP-03.6.1, *Release Estimates Based on Stack/Vent Readings*, which one of the following is the calculated release rate?

ATTACHMENT 2
Page 1 of 1
Source Term Calculation From #1 RX Gas (1-CAC-AQH-1264-3)

TIME	METER READING (cpm)	FLOW[1] (cfm)	EFFICIENCY[(2)] FACTOR	RELEASE[(3)] RATE (µCi/sec)
1 minute ago	4.0 E+3	43,200 CFM per exhaust fan	1.275 E-5	

(1) If not available use 43,200 cfm per exhaust fan times the number of fans operating.

(2) The efficiency factors can be obtained from 0E&RC-2020 (contact E&RC counting room).

(3) Release Rate = (cpm) x (cfm) x (Efficiency Factor)

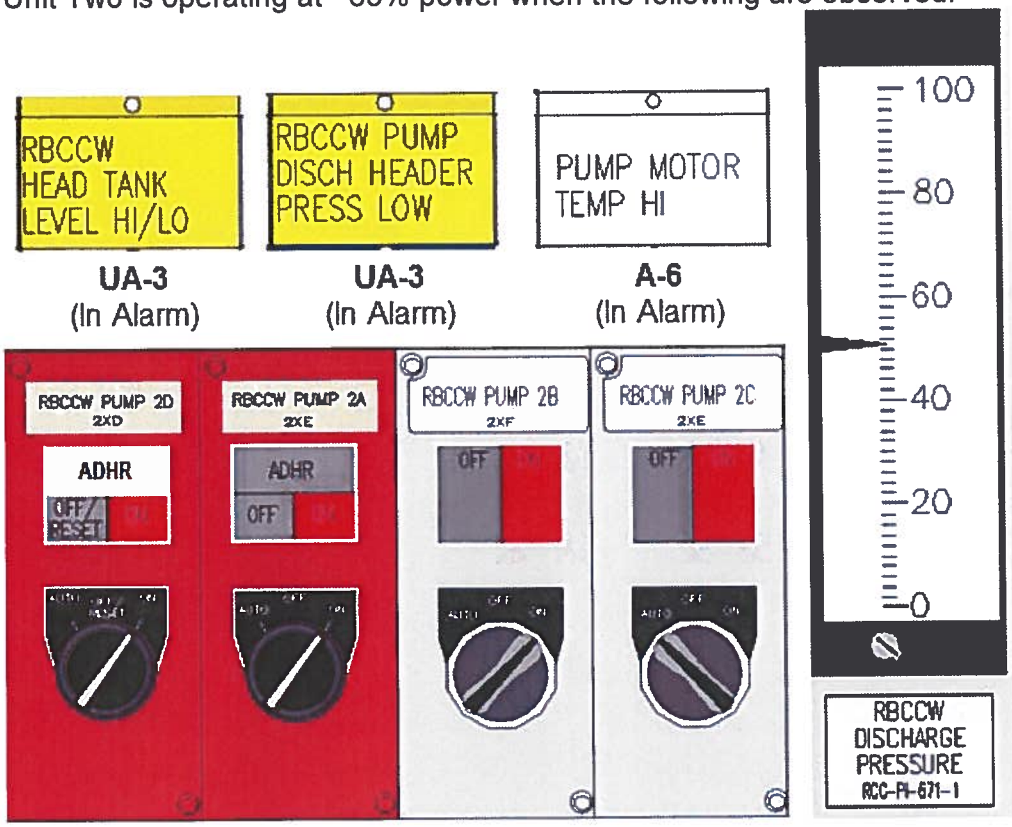
A. 2.2 E+3 µCi/sec.

B. 6.6 E+3 µCi/sec.

C. 1.3 E+4 µCi/sec.

D. 6.6 E+4 µCi/sec.

43. Unit Two is operating at ~65% power when the following are observed:



Which one of the following completes both statements below IAW 0AOP-16.0, *RBCCW System Failure*?

A complete loss of RBCCW __(1)__ occurred.

A reactor scram __(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

44. Unit Two has entered 0AOP-20.0, *Pneumatic (Air/Nitrogen) System Failures*, due to a loss of instrument air pressure with the following annunciator status:

UA-01 (1-1) *RB Instr Air Receiver 2A Press Low*	Alarm sealed in
UA-01 (1-2) *RB Instr Air Receiver 2B Press Low*	NOT in Alarm
UA-01 (3-2) *Air Compr D Trip*	Alarm sealed in
UA-01 (4-4) *Inst Air Press Low*	Alarm sealed in
UA-01 (5-4) *Service Air Press-Low*	Alarm sealed in

Which one of the following completes both statements below?

On a loss of instrument air, the RB HVAC Butterfly Isolation Valves will fail __(1)__.

IAW 0AOP-20.0, the reactor __(2)__ required to be scrammed.

A. (1) as-is
   (2) is

B. (1) as-is
   (2) is NOT

C. (1) open
   (2) is

D. (1) open
   (2) is NOT

45. I&C Techs inadvertently cause a low level 3 (LL3) signal.
Unit Two plant conditions are:

Reactor pressure	930 psig
Drywell pressure	1.7 psig, steady
Drywell temp (average)	140°F, slow rise
Drywell leak calculation	Normal

Which one of the following completes the statement below?

All Drywell Cooler Fans are:

A. tripped, but can be overridden on.

B. tripped, and cannot be overridden on.

C. running, but can be tripped at the RTGB.

D. running, and cannot be tripped at the RTGB.

46. Unit One in MODE 5.
The fuel pool gates are removed.
SDC Loop B is in service.
Fuel pool cooling assist is in operation.

The RHR Loop B pumps tripped and can NOT be restarted.

Which one of the following completes both statements below?
(consider each statement separately)

Fuel pool cooling assist __(1)__.

Fuel pool cooling assist __(2)__ capable of being aligned to the SDC Loop A IAW 1OP-17, *Residual Heat Removal System Operating Procedure*.

A. (1) remains in service
   (2) is

B. (1) remains in service
   (2) is NOT

C. (1) is lost
   (2) is

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

47. Unit Two is performing refueling operations when the refueling SRO reports that a spent fuel bundle has been dropped.
The following radiation monitoring alarms are received:

UA-03 (3-7) *Area Rad Refuel Floor High*
UA-03 (4-5) *Process Rx Bldg Vent Rad Hi*

Which one of the following identifies the "Immediate Action" that is required IAW 0AOP-05.0, *Radioactive Spills, High Radiation, and Airborne Activity*?

A. Verify Group 6 isolation.

B. Evacuate all personnel from the refuel floor.

C. Place Control Room Emergency Ventilation System in operation.

D. Isolate Reactor Building Ventilation and place Standby Gas Treatment trains in operation.

48. Unit Two is operating at rated power when high drywell pressure switch C72-PTM-N002A-1 fails high resulting in the annunciation of A-05-(5-6) *Pri Ctmt Press Hi Trip*.

Which one of the following completes the statement below?

RPS high drywell pressure relay C72-K4A will __(1)__.

The RSP __(2)__ be required to be entered.

A. (1) energize
   (2) will

B. (1) energize
   (2) will NOT

C. (1) de-energize
   (2) will

D. (1) de-energize
   (2) will NOT

49. Unit One was operating at power when a turbine trip occurred.
85 control rods fail to insert.
Reactor pressure peaks at 1145 psig.

Which one of the following completes both statements below?

The reactor recirc pumps __(1)__ tripped.

Tripping of the reactor recirc pumps results in a rapid decrease in reactor power due to __(2)__.

A. (1) must be manually
(2) voiding of the moderator

B. (1) must be manually
(2) a reduction in reactor water level

C. (1) have automatically
(2) voiding of the moderator

D. (1) have automatically
(2) a reduction in reactor water level

50. Unit One failed to scram following a loss of off-site power with the following plant conditions:

Reactor Power	5%
RPV Water Level	-55 inches (N036)
RPV Pressure	850 psig

Which one of the following completes both statements below?

This UA-12 (5-4) alarm is expected to be received when suppression pool water temperature **first** reaches __(1)__.

IAW 1OP-17, *Residual Heat Removal System Operating Procedure*, the RHR logic requirements to place torus cooling in service under the current plant conditions will require __(2)__.

A. (1) 95°F
(2) placing the CS-S17B "Think Switch" to Manual first and then bypassing the 2/3rd core height interlock

B. (1) 95°F
(2) bypassing the 2/3rd core height interlock first and then placing the CS-S17B "Think Switch" to Manual

C. (1) 105°F
(2) placing the CS-S17B "Think Switch" to Manual first and then bypassing the 2/3rd core height interlock

D. (1) 105°F
(2) bypassing the 2/3rd core height interlock first and then placing the CS-S17B "Think Switch" to Manual

51. Unit Two is in MODE 3 following a Station Blackout.
IAW 0EOP-01-SBO-01, *Plant Monitoring*, the AO has reported the following temperatures from the RSDP temperature recorder 2CAC-TR-778:

Point 1	290°F
Point 2	118°F
Point 3	255°F
Point 4	230°F
Point 5	191°F
Point 6	117°F

(REFERENCE PROVIDED)

Which one of the following represents the correct calculated Drywell temperature?

A. ~205°F

B. ~249°F

C. ~258°F

D. ~267°F

52. Unit Two is performing RVCP with HPCI in pressure control.

Subsequently, A-01 (1-5) *Suppression Chamber Level Hi Hi* is received.

Which one of the following completes both statements below?

The E41-F004, CST Suction Vlv, will __(1)__.

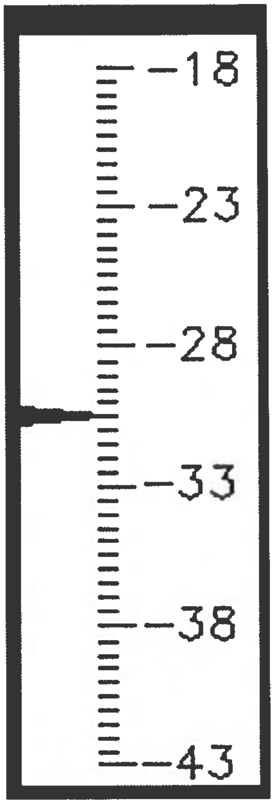
The E41-F008, Bypass to CST Vlv, will __(2)__.

A. (1) close
   (2) close

B. (1) close
   (2) remain open

C. (1) remain open
   (2) close

D. (1) remain open
   (2) remain open

53. 

Unit One is operating at rated power when A-01 (3-7) *Suppression Chamber Lvl Hi/Lo,* is received.

The BOP Operator verifies the alarm using CAC-LI-4177, Supp Pool Level, indicator on Panel XU-51. (indication provided to the left)

Which one of the following identifies the action that is required IAW A-01 (3-7) *Suppression Chamber Lvl Hi/Lo*?

The water level in the Unit One torus must be:

A. lowered by using Core Spray and routed to Radwaste.

B. lowered using RHR and routed to Radwaste.

C. raised by opening the HPCI suction from the CST.

D. raised by opening the Core Spray suction from the CST.

54. Unit One is executing the ATWS procedure with the following plant conditions:

Reactor power	12%
Reactor pressure	940 psig, controlled by EHC
Reactor water level	170 inches, controlled by feedwater

Which one of the following identifies the reason the ATWS procedure directs deliberately lowering RPV water level to 90 inches?

A. Reduces reactor power so that it will remain below the APRM downscale setpoint.

B. Provides heating of the feedwater to reduce potential for high core inlet subcooling.

C. Reduces challenges to primary containment if MSIVs close.

D. Promotes more efficient boron mixing in the core region.

55. Which one of the following identifies the reason for performing Emergency Depressurization due to exceeding Maximum Safe Operating Temperatures IAW 0OI-37.9, *Secondary Containment Control Procedure Basis Document*?

A. Prevent an unmonitored release.

B. Preserve personnel access into the reactor building.

C. Provide continued operability of equipment required for safe shutdown.

D. Ensure ODCM site boundary dose limits are not exceeded.

56. Which one of the following completes both statements below?

IAW 0AOP-5.4, *Radiological Releases*, RRCP is entered when the Turbine Building Vent Rad Monitor indication exceeds an __(1)__ EAL.

IAW RRCP, before the radioactivity release rate reaches a __(2)__ Emergency EAL, Emergency Depressurization is required.

A. (1) Unusual Event
   (2) Site Area

B. (1) Unusual Event
   (2) General

C. (1) Alert
   (2) Site Area

D. (1) Alert
   (2) General

57. Following an unisolable RWCU line break in the reactor building the following conditions exist:

South Core Spray Room temperature 155°F
South RHR Room temperature 300°F
UA-12 (2-3) *South Core Spray Room Flood Level Hi,* in alarm
UA-12 (2-4) *South RHR Room Flood Level Hi,* in alarm
UA-12 (1-4) *South RHR Room Flood Level Hi-Hi,* in alarm

(REFERENCE PROVIDED)

Which one of the following completes both statements below?

IAW 0EOP-01-UG, *User's Guide*, __(1)___equipment required for safe shutdown will fail.

IAW SCCP, Emergency Depressurization __(1)__ required.

A. (1) ONLY the South RHR room
(2) is

B. (1) ONLY the South RHR room
(2) is NOT

C. (1) the South RHR room AND Core Spray room
(2) is

D. (1) the South RHR room AND Core Spray room
(2) is NOT

58. The RO has attempted to manually scram Unit One with the following actions taken:

All rods are noted to be greater than position 02
Reactor mode switch is placed in shutdown
ARI was initiated.
Both recirculation pumps were tripped.
Reactor power reported at 12%
SLC is injecting
RPV level is 80 inches and stable
Rod insertion attempts are unsuccessful

Which one of the following completes both statements below?

Reactor power __(1)__ expected to be lowering.

Assuming no rod insertion, SLC injection __(2)__.

A. (1) is
(2) can be secured when all APRMs are downscale

B. (1) is
(2) must be continued until the reactor is shutdown under all conditions

C. (1) is NOT
(2) can be secured when all APRMs are downscale

D. (1) is NOT
(2) must be continued until the reactor is shutdown under all conditions

59. A radioactive release has occurred in the Turbine Building.

Which one of the following completes both statements below?

IAW 0AOP-05.4, *Radiological Releases*, the Unit Two turbine building ventilation must be in the __(1)__ operating mode.

This discharge will be monitored by the __(2)__.

A. (1) recirc
   (2) Main Stack Radiation Monitor

B. (1) recirc
   (2) Wide Range Gaseous Monitor (WRGM)

C. (1) once through
   (2) Main Stack Radiation Monitor

D. (1) once through
   (2) Wide Range Gaseous Monitor (WRGM)

60. Unit One is operating at rated power when the following alarms are received:

UA-01 (4-4) *Instr Air Press-Low*
UA-01 (5-1) *Air Dryer 1A Trouble*

The AO reports that the cause of the alarms is due to filter blockage.

Which one of the following completes both statements below?

The Service Air Dryer malfunction will cause SA-PV-5067, Service Air Dryer Bypass Valve, to open when pressure **first** lowers to __(1)__.

IAW 0AOP-20.0, Pneumatic (Air/Nitrogen) System Failures, the required action is to __(2)__.

A. (1) 105 psig
(2) place the 1B Service Air Dryer in service

B. (1) 105 psig
(2) set the service air dryer maximum sweep value to zero

C. (1) 98 psig
(2) place the 1B Service Air Dryer in service

D. (1) 98 psig
(2) set the service air dryer maximum sweep value to zero

61. Unit One is in MODE 3 following a seismic event and reactor scram with the following plant conditions:

Reactor level	55 inches
Reactor pressure	500 psig
Drywell pressure	9 psig
Division I PNS header pressure	93 psig
Division II PNS header pressure	98 psig

Which one of the following completes both statements below?

Div I Backup N2 Rack Isol Vlv, RNA-SV-5482 is ___(1)___.

Div II Backup N2 Rack Isol Vlv, RNA-SV-5481 is ___(2)___.

A. (1) open
(2) open

B. (1) open
(2) closed

C. (1) closed
(2) open

D. (1) closed
(2) closed

62. Unit One is operating at rated power with the following conditions:

CSW Pump 1A trips
Conventional header pressure lowers to 35 psig

Which one of the following completes both statements below?

If CSW header pressure remains at this pressure for __(1)__ seconds, the SW-V3, SW To TBCCW HXs Otbd Isol Vlv, and SW-V4, SW To TBCCW HXs Inbd Isol Vlv, will close to a throttled position.

IAW 0AOP-19, *Conventional Service Water System Failure*, the SW-V3 and SW-V4 are reopened __(2)__.

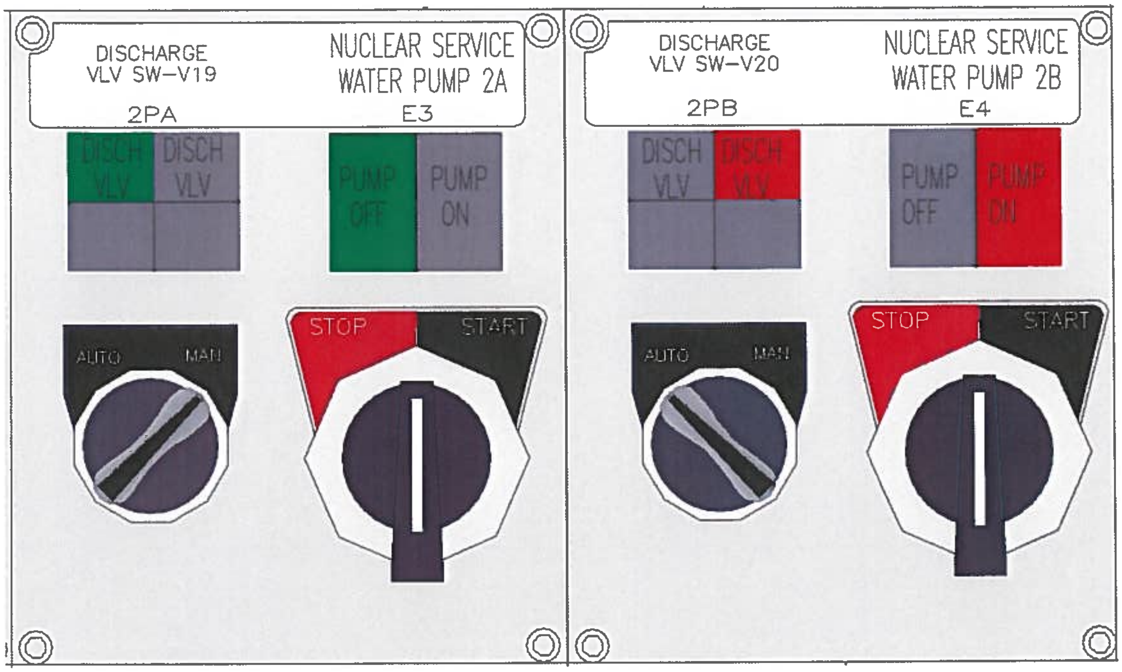
A. (1) 30
(2) ONLY after a reactor Scram is inserted

B. (1) 30
(2) if system pressure is restored by starting the standby CSW pump

C. (1) 70
(2) ONLY after a reactor Scram is inserted

D. (1) 70
(2) if system pressure is restored by starting the standby CSW pump

63. Unit Two Nuclear Service Water (NSW) pumps are aligned as follows in preparation for equipment realignment:



Subsequently, Off-site power is lost.

Which one of the following completes the statement below?

__(1)__ NSW pump(s) will auto start __(2)__ associated E Bus is re-energized.

A. (1) 2A and 2B
   (2) immediately when their

B. (1) 2A and 2B
   (2) five seconds after their

C. (1) 2B ONLY
   (2) immediately when its

D. (1) 2B ONLY
   (2) five seconds after its

64. Which one of the following identifies the potential consequence of failing to place backup nitrogen in service by placing RNA keylock switches in LOCAL IAW 0ASSD-02, *Control Building*?

<u>RNA keylock switch noun names:</u>
2-RNA-CS-001, Override Switch For Valve RNA-SV-5482
2-RNA-CS-002, Override Switch For Valve RNA-SV-5253

A. Misoperation of RCIC.

B. Loss of drywell cooling.

C. Inability to operate SRVs.

D. Spurious operation of MSIVs.

65. A grid disturbance occurs with the following Unit One plant parameters:

Generator Load	980 MWe
Generator Reactive Load	160 MVARs, out
Generator Gas Pressure	50 psig

(REFERENCE PROVIDED)

Which one of the following identifies both available options that will place the Unit within the Estimated Capability Curve?

A. Raise gas pressure to 58 psig or lower power to 940 MWe.

B. Raise gas pressure to 58 psig or raise reactive load to 240 MVARs.

C. Raise gas pressure to 58 psig or lower reactive load to 70 MVARs.

D. Lower power to 940 MWe or raise reactive load to 240 MVARs.

66. Which one of the following completes both statements below IAW AD-OP-ALL-1000, *Conduct of Operations*?

With the Unit operating at rated, steady state power, steam flow / feed flow __(1)__ a key parameter that the OATC must monitor to assure a constant awareness of its value and trend.

An end to end control panel walk down shall be performed every __(2)__ and documented in the Narrative Logbook.

A. (1) is NOT
   (2) one hour

B. (1) is NOT
   (2) two hours

C. (1) is
   (2) one hour

D. (1) is
   (2) two hours

67. Which one of the following completes the statement below?

1OP-10, *Standby Gas Treatment System Operating Procedure*, prohibits venting the drywell and the suppression pool chamber simultaneously with the reactor at power because this would cause the:

A. unnecessary cycling of reactor building to torus vacuum breakers.

B. unnecessary cycling of torus to drywell vacuum breaker.

C. SBGT Train water seal to blow out of the trough.

D. pressure suppression function to be bypassed.

68. A core reload is in progress during a refueling outage. The initial loading of fuel bundles around each SRM centered 4-bundle cell was completed with all four SRMs fully inserted and reading 50 cps.

It is now approximately half way through the core loading sequence and SRMs read 80 cps.

Which one of the following completes the statement below IAW 0FH-11, *Refueling*?

Fuel movement must be suspended when any SRM reading **first** rises to ______ upon insertion of the **next** fuel bundle.

A. 100 cps

B. 160 cps

C. 250 cps

D. 400 cps

69. Unit Two is conducting a routine power reduction for rod pattern improvement. The Reactivity Management Plan contains actions for the RO to insert a group of four rods from position 24 to position 12.

Which one of the following completes the statement below IAW AD-OP-ALL-0203, *Reactivity Management*?

The movement of these rods should be:

A. single notched for the entire movement.

B. continuously inserted to the final intended position.

C. continuously inserted to settle four notches prior to reaching the intended position and then single notched into the final intended position.

D. continuously inserted to settle one notch prior to reaching the intended position and then single notched into the final intended position.

70. Which one of the following identifies the Unit Two "Scram Immediate Operator Action" that utilizes a different criteria for performance than on Unit One?

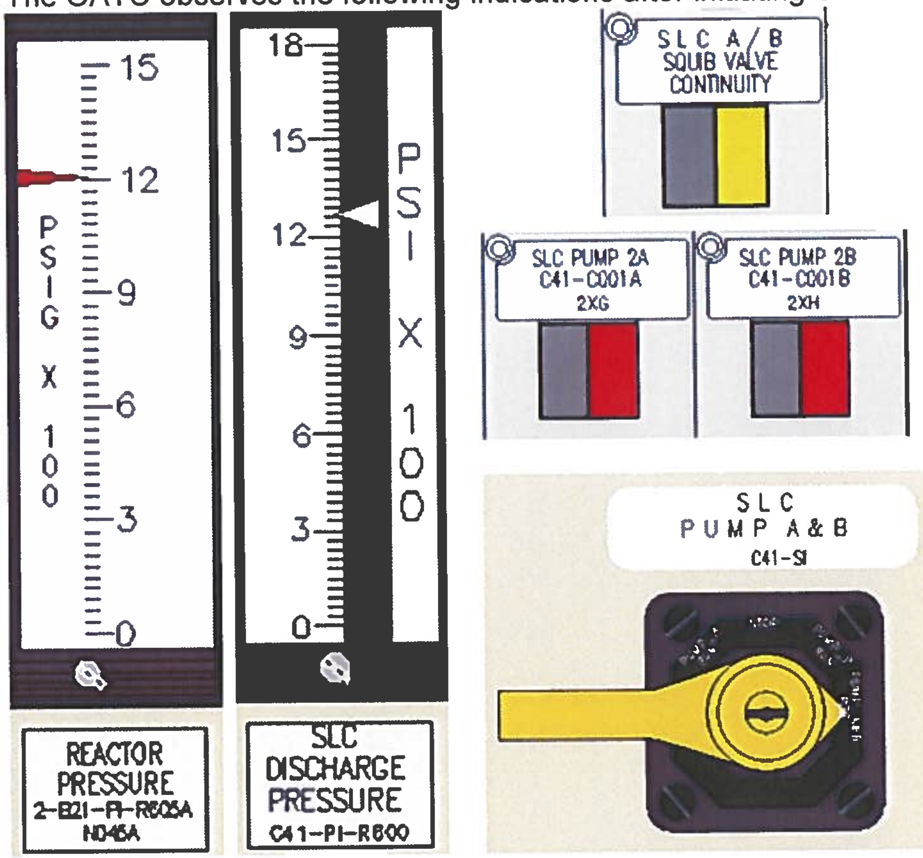
A. Tripping of the main turbine.

B. Tripping one of the running feed pumps.

C. Master level controller setpoint setdown.

D. Placing the reactor mode switch to Shutdown.

71. The OATC observes the following indications after initiating SLC during an ATWS.



Which one of the following completes both statements below?

Squib valve __(1)__ has failed to fire.

IAW 2OP-05, *Standby Liquid System Operating Procedure*, the OATC is required to __(2)__.

A. (1) A
   (2) place the CS-S1, SLC Pump A & B, in the PUMP B RUN position

B. (1) A
   (2) leave the CS-S1, SLC Pump A & B, in the PUMP A/B RUN position

C. (1) B
   (2) place the CS-S1, SLC Pump A & B, in the PUMP A RUN position

D. (1) B
   (2) leave the CS-S1, SLC Pump A & B, in the PUMP A/B RUN position

72. Two operators are required to enter a room that is posted as a Locked High Radiation Area (LHRA) to hang a clearance for **scheduled** work.

Which one of the following completes both statements below?

The radiation level at which a LHRA posting is required is __(1)__ in one hour at 30 centimeters from the radiation source.

The LHRA key is obtained from __(2)__.

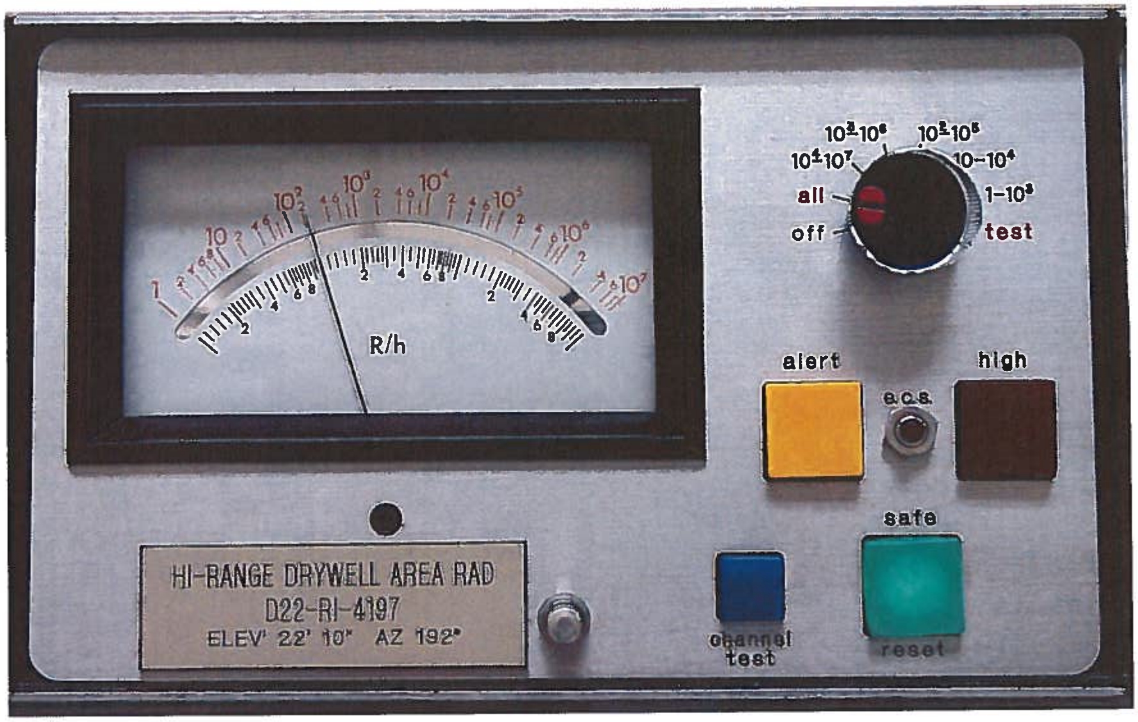
A. (1) >100 mrem
   (2) the Shift Manager

B. (1) >100 mrem
   (2) a RP Technician

C. (1) >1000 mrem
   (2) the Shift Manager

D. (1) >1000 mrem
   (2) a RP Technician

73.



Which one of the following identifies the DW radiation value indicated above?

A. ~ 10 R/hr

B. ~ 20 R/hr

C. ~ 100 R/hr

D. ~ 200 R/hr

74. A transient has occurred on Unit Two with the following plant conditions:

RPV pressure	1000 psig
Drywell ref leg area temp	197°F
Rx Bldg 50' temp	135°F
Wide Range Level	170 inches (N026A/B)
Shutdown Range Level	160 inches (N027A/B)

(REFERENCE PROVIDED)

Which one of the following completes both statements below concerning the level instruments that can be used to determine reactor water level IAW EOP Caution 1?

Wide Range Level instruments N026A/B __(1)__ be used.

Shutdown Range Level instruments N027A/B __(2)__ be used.

A. (1) can
(2) can

B. (1) can
(2) can NOT

C. (1) can NOT
(2) can

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

75. A fire has been reported and confirmed in the turbine building breezeway.
A fire hose is being used to control/suppress the fire.

Which one of the following completes both statements below IAW 0PFP-013, *General Fire Plan*?

The RO is required to sound the fire alarm and announce the location of the fire __(1)__.

A call for offsite assistance to the Brunswick County 911 Center __(2)__ required.

A. (1) ONLY once
(2) is

B. (1) ONLY once
(2) is NOT

C. (1) three times
(2) is

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

76. During a LOCA and LOOP on Unit One, the following plant conditions exist:

An Emergency Depressurization has been performed due to RPV water level

The Reactor Building -17 foot and 20 foot elevations are NOT accessible due to radiation levels.

ALL ECCS pumps are unavailable.

Which one of the following completes the statement below?

The CRS will direct demin water injection to the RPV, IAW 0EOP-01-LEP-01, *Alternate Coolant Injection,* Section:

A. 2.4.3.3a, *Demineralized Water Actions*, Inject demineralized water through Core Spray Loop A

B. 2.4.3.3c, *Demineralized Water Actions*, Inject demineralized water through RHR Loop A

C. 2.4.3.3d, *Demineralized Water Actions*, Inject demineralized water through HPCI

D. 2.4.3.3e, *Demineralized Water Actions,* Inject demineralized water through RCIC

77. Unit One is at rated power performing 0PT-01.1.6, *Reactor Protection System Manual Scram Test.*

The Reactor Scram System A pushbutton has been depressed.

RPS Trip System A Scram Groups light for groups one, two, three, and four are illuminated

(REFERENCE PROVIDED)

Which one of the following completes both statements below?

The scram pilot valve solenoids associated with these lights are __(1)__.

Tech Spec 3.3.1.1, *Reactor Protection System Instrumentation*, Condition B __(2)__ required to be entered.

A. (1) energized
(2) is

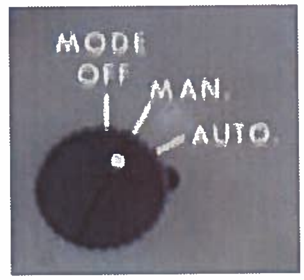
B. (1) energized
(2) is NOT

C. (1) de-energized
(2) is

D. (1) de-energized
(2) is NOT

78. Unit Two is at rated power. A TIP trace is in progress.

TIP D Valve Control Unit and Monitor indications are as follows:



(REFERENCE PROVIDED)

Which one of the following completes both statements below?

Tip Valves are Group __(1)__ PCIVs.

Tech Spec 3.6.1.3, *Primary Containment Isolation Valves (PCIVs)*, Condition(s) __(2)__ is/are required to be entered.

A. (1) 2
(2) A ONLY

B. (1) 2
(2) A and B

C. (1) 6
(2) A ONLY

D. (1) 6
(2) A and B

79. Unit One is operating at rated power.
Unit Two is in MODE 5 with UAT backfeed established.
A main generator backup lockout occurs on Unit One.

(REFERENCE PROVIDED)

Which one of the following completes both statements below?

All four diesels __(1)__ automatically start.

IAW Unit One Tech Spec 3.8.1, *AC sources Operating*, Condition E __(2)__ required to be entered.

A. (1) will
(2) is

B. (1) will
(2) is NOT

C. (1) will NOT
(2) is

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

80. Unit One was operating at power when a Group 1 isolation and reactor scram occurred.

Reactor pressure is 950 psig and being manually controlled by SRVs.

An SRV is stuck open with a stuck open **SRV tailpipe** vacuum breaker.

Torus and Drywell sprays have been initiated IAW PCCP

(REFERENCE PROVIDED)

Which one of the following completes both statements below?

The SRV is discharging through the open vacuum breaker directly into the __(1)__.

The **highest** EAL classification for this event is a(n) __(2)__.

A. (1) drywell
   (2) Alert

B. (1) drywell
   (2) Site Area Emergency

C. (1) suppression chamber air space
   (2) Alert

D. (1) suppression chamber air space
   (2) Site Area Emergency

81. Unit Two is operating at rated power with RHR Loop A operating in suppression pool cooling mode.

A-01 (2-8) *RHR Relay Logic Pwr Failure,* is in alarm due to a blown fuse affecting RHR Logic A ONLY.

(REFERENCE PROVIDED)

Which one of the following completes both statements below?

IAW Tech Spec 3.3.5.1, *ECCS Instrumentation*, ~~the~~ any required channels __(1)__ required to be placed in trip within 24 hours.

If a LOCA signal were to occur, 2-E11-F015A, Inboard Injection Vlv, __(2)__ open automatically on low reactor pressure.

A. (1) are
   (2) will

B. (1) are NOT
   (2) will

C. (1) are
   (2) will NOT

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

82. Unit Two is operating at rated power.

Subsequently, a Div I pneumatic leak occurs causing drywell pressure to rise to 1.9 psig.

Which one of the following completes both statements below?

The SBGT trains __(1)__ running.

The Div I pneumatics are required be isolated IAW __(2)__.

A. (1) are NOT
(2) 0EOP-01-SEP-16, *Drywell Systems Isolation*

B. (1) are NOT
(2) 0AOP-20.0, *Pneumatic (Air/Nitrogen) System Failures*

C. (1) are
(2) 0EOP-01-SEP-16, *Drywell Systems Isolation*

D. (1) are
(2) 0AOP-20.0, *Pneumatic (Air/Nitrogen) System Failures*

83. Unit Two is operating at rated power.

UA-48 (5-4) *AOG System Bypass*, has been alarming for 1 minute due to High-High off gas flow

(REFERENCE PROVIDED)

Which one of the following completes both statements below?

AOG-XCV-142, Guard Bed Isolation Valve, __(1)__ automatically close.

ODCM 7.3.10, *Gaseous Radwaste Treatment System*, Condition A entry __(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

84. Unit One is operating at 72% power with the following conditions:

Jet Pump Flow Loop A (B21-R611A)	25 Mlbs/hr
Jet Pump Flow Loop B (B21-R611B)	29 Mlbs/hr
Total Core Flow (U1CPWTCF)	54 Mlbs/hr

Which one of the following completes both statements below IAW Tech Spec 3.4.1, *Recirculation Loops Operating,* and Bases? (consider each statement separately)

The current Jet Pump Flow mismatch __(1)__.

If Jet Pump Flows are **not** matched within limits, then the loop with the __(2)__ must be considered not in operation.

A. (1) is within limits
(2) lower flow

B. (1) is within limits
(2) higher flow

C. (1) is not within limits
(2) lower flow

D. (1) is not within limits
(2) higher flow

85. Unit One was at full power when all offsite power was lost.
The following is the Emergency Diesel Generator status:

DG1	Locked out on fault
DG2	Running and loaded
DG3	Running and loaded
DG4	Locked out on fault

Which one of the following completes the statements below?

The __(1)__ CRD pump must be started to re-establish the CRD system.

0AOP-36.1, *Loss Of Any 4160V Buses or 480V E-Buses*, __(2)__ contain the step for placing the CRD Flow Control, C11-FC-R600, in manual with manual potentiometer at minimum setting following the loss of the CRD pump?

A. (1) 1A
(2) does

B. (1) 1A
(2) does NOT

C. (1) 1B
(2) does

D. (1) 1B
(2) does NOT

86. Unit One is performing the ATWS Procedure with the following conditions:

A-05 (2-6) *Reactor Vess Lo Level Trip,* is illuminated
A-06 (1-6) *Reactor Vess Lo Lo Water Level Sys A,* is NOT illuminated
A-06 (2-6) *Reactor Vess Lo Lo Water Level Sys B,* is NOT illuminated
MSIVs are closed
Reactor pressure peaked at 1141 psig and is now being controlled 800-1000 psig.
Torus water temperature is 105°F and rising
Reactor power is 25%

IAW 0OI-37.5, *ATWS Procedure Basis Document*, which one of the following identifies the action that will have the **highest** priority?

A. SLC initiation.

B. Inhibiting ADS.

C. Trip both Reactor Recirc Pumps.

D. Termination and prevention of RPV injection.

87. Which one of the following completes both statements below?

IAW Tech Spec 3.9.6, *Reactor Pressure Vessel (RPV) Water Level*, the minimum water level over the top of irradiated fuel assemblies seated within the RPV during movement of irradiated fuel assemblies in the RPV is __(1)__.

The Tech Spec bases for the minimum water level is to provide for __(2)__ during a fuel handling accident.

A. (1) 19 feet 11 inches
   (2) iodine retention

B. (1) 19 feet 11 inches
   (2) shielding of radioactive decay particles

C. (1) 23 feet
   (2) iodine retention

D. (1) 23 feet
   (2) shielding of radioactive decay particles

88. Unit Two is in an ATWS executing RXFP, with the following plant conditions:

Injection to the RPV has been terminated and prevented
The **Minimum Number of SRVs Required for Emergency Depressurization** are open.

Table P-3
Minimum Steam Cooling Pressure

Open SRVs	Pressure (psig)
7 or more	120
6	145
5	175
4	220
3	300
2	455
1	915

IAW RXFP, which one of the following completes the statement below?

The CRS should direct injection to the RPV when EITHER:

__(1)__ SRV remains open
OR
when reactor pressure lowers below the Minimum Steam Cooling Pressure of __(2)__.

A. (1) NO
(2) 175 psig

B. (1) NO
(2) 455 psig

C. (1) ONLY one
(2) 175 psig

D. (1) ONLY one
(2) 455 psig

89. An event on Unit One has resulted in the following plant conditions:

Reactor pressure:	1000 psig
Reactor Water Level	120 inches
Control Rod position	Unknown
APRMs	Downscale
Drywell pressure:	3 psig
Torus pressure:	2 psig
Torus water temp:	152°F
Torus water level:	-36 inches

(REFERENCE PROVIDED)

Which one of the following identifies the required actions for reactor pressure control?

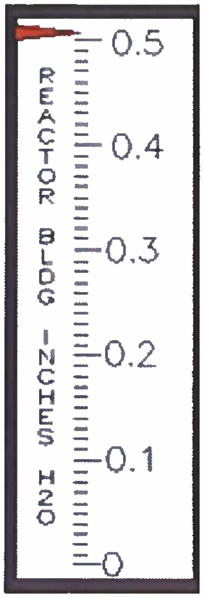
A. Exit the RC/P flowpath of ATWS, and go to 0EOP-01-EDP, *Emergency Depressurization.*

B. Exit the RC/P flowpath of RVCP, and go to 0EOP-01-EDP, *Emergency Depressurization.*

C. Remain in the RC/P flowpath of ATWS, and exceed 100°F/hr cooldown rate if necessary.

D. Remain in the RC/P flowpath of RVCP, and exceed 100°F/hr cooldown rate if necessary.

90.



Unit Two is operating at rated power.
PCCP has been entered due to high torus water temperature with the following plant conditions:

UA-12 (3-3) *Rx Bldg Diff Press High/Low,* is in alarm.

UA-05 (6-10) *Rx Bldg Isolated,* is in alarm.

Reactor Building Pressure (indication on the left)

Which one of the following completes both statements below?

Reactor Building pressure is __(1)__.

The CRS will direct Reactor Building HVAC restarted IAW __(2)__.

A. (1) positive
(2) 2OP-37.1, *Reactor Building Heating and Ventilation System Operating Procedure*

B. (1) positive
(2) 0EOP-01-SEP-04, *Reactor Building HVAC Restart Procedure*

C. (1) negative
(2) 2OP-37.1, *Reactor Building Heating and Ventilation System Operating Procedure*

D. (1) negative
(2) 0EOP-01-SEP-04, *Reactor Building HVAC Restart Procedure*

91. A release on Unit Two is occurring with the following plant conditions:

Main Stack Rad Monitor, D12-RM-23S, is reading 2.3E+08 µCI/sec

Turbine Building Vent Rad Monitor, D12-RM-23, is reading 2.5E+07 µCI/sec

Real-time dose assessment using actual meteorology indicates 0.92 Rem TEDE and 5.1 Rem thyroid CDE at the site boundary

(REFERENCE PROVIDED)

Which one of the following completes both statements below?

IAW RRCP, **Unit One** __(1)__ override and reset the main stack hi-hi isolation signal.

The **highest** EAL classification for this event is __(2)__.

A. (1) can
(2) Site Area Emergency

B. (1) can
(2) General Emergency

C. (1) can NOT
(2) Site Area Emergency

D. (1) can NOT
(2) General Emergency

92. Unit One and Unit Two are executing 0ASSD-01, *Alternative Safe Shutdown Procedure Index*, due to a fire in Main Control Room back panels requiring Main Control Room evacuation. Current plant conditions are:

Unit One and Two have scrammed
All MSIVs are shut

Which one of the following completes both statements below?

The CRS will enter 0ASSD-02, *Control Building,* and ___(1)___ 0ASSD-01.

The CRS will direct actions to achieve a safe shutdown using ___(2)___.

A. (1) exit
(2) HPCI

B. (1) exit
(2) RCIC

C. (1) concurrently perform
(2) HPCI

D. (1) concurrently perform
(2) RCIC

93. Unit One is in MODE 4, when a loss of SDC occurs due to RCS leakage.

CU1	**UNPLANNED loss of RPV inventory for 15 minutes or longer**				
			4	**5**	

CU1.1

UNPLANNED loss of reactor coolant results in RPV water level less than a required lower limit for ≥ 15 min. (Note 1)

Which one of the following completes both of the statements below?

The **minimum** required RPV water level to support natural circulation is __(1)__.

IAW 0PEP-02.2.1, *Emergency Action Level Technical Bases*, the Unusual Event required lower limit is defined as RPV water level less than__(2)__.

A. (1) 200 inches
(2) 105 inches

B. (1) 200 inches
(2) 166 inches

C. (1) 254 inches
(2) 105 inches

D. (1) 254 inches
(2) 166 inches

94. Which one of the following completes both statements below?
(Consider each statement separately.)

IAW Tech Spec 5.2.2, *Facility Staff*, the shift crew composition may be less than the minimum requirement for a period of time not to exceed __(1)__ for an unexpected absence of on-duty shift crew members.

IAW 0OI-01.01, *BNP Conduct of Operations Supplement*, the minimum required number of Auxiliary Operators for manning a shift at BNP is __(2)__.

A. (1) one hour
(2) three

B. (1) one hour
(2) nine

C. (1) two hours
(2) three

D. (1) two hours
(2) nine

95. Following the bypass of Unit Two feedwater heaters 4A and 5A, the following plant conditions exist:

Reactor Power is 60%
Feedwater Temperature is 330°F

Final Feedwater Temperature vs Power

RX PWR	Nominal FW Temp	Nominal FW Temp Reduced 10°F	110.3°F Reduced FFWT
65%	394.4	384.4	296.4
64%	393.1	383.1	295.5
63%	391.7	381.7	294.6
62%	390.4	380.4	293.7
61%	389.0	379.0	292.8
60%	387.6	377.6	291.9
[illegible]	[illegible]	[illegible]	[illegible]

(REFERENCE PROVIDED)

IAW 0OI-01.01, *BNP Conduct of Operations Supplement,* which one of the following completes both statements below? (consider each statement separately)

The CRS __(1)__ required to implement the thermal limit penalties for FHOOS (feedwater heater out of service).

Entry into **Tech Spec 3.0.3** __(2)__ required if final feedwater temperature is less than the 110.3°F reduced final feedwater temperature value.

A. (1) is
(2) is

B. (1) is
(2) is NOT

C. (1) is NOT
(2) is

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

96. Unit One is operating at rated power.

A-03 (2-2) *Auto Depress Control Pwr Failure*, is in alarm due to Fuse F5 being blown.

(REFERENCE PROVIDED)

Which one of the following completes both statements below?

Fuse F5 (D1 on 1-FP-05887) is located on ADS Logic __(1)__.

ADS __(2)__ operable.

A. (1) A
   (2) is

B. (1) A
   (2) is NOT

C. (1) B
   (2) is

D. (1) B
   (2) is NOT

97. Unit Two is operating at rated power.

While performing 0PT-07.2.4A, *Core Spray Loop A Operability*, Core Spray Room Cooler A fails to start when Core Spray Pump A is started.

The reactor building AO reports that the room cooler tripped on thermal overload.

IAW AD-OP-ALL-1000, *Conduct of Operations*, which one of the following completes both statements below? (consider each statement separately)

Core Spray Loop A is __(1)___.

A one time reset of the thermal overload __(2)___ allowed before a Maintenance and Engineering evaluation.

A. (1) OPERABLE
(2) is

B. (1) OPERABLE
(2) is NOT

C. (1) INOPERABLE
(2) is

D. (1) INOPERABLE
(2) is NOT

98. Following a small steam line break in the drywell plant conditions are as follows:

Drywell pressure:	25 psig and rising
Drywell hydrogen:	1.3%
Suppression Chamber hydrogen:	1.2%
Torus level:	42 inches

Which one of the following completes both statements below?

The CRS is required to direct venting containment IAW 0EOP-01-SEP-01, *Primary Containment Venting*, using ___(1)___.

Venting of the ___(2)___ will be directed **first**.

A. (1) Section 2.1, *Containment Pressure Control*
   (2) drywell

B. (1) Section 2.1, *Containment Pressure Control*
   (2) torus

C. (1) Section 2.2, *Containment Hydrogen Control*
   (2) drywell

D. (1) Section 2.2, *Containment Hydrogen Control*
   (2) torus

99. Unit Two is operating at rated power with LPCI A inoperable and the following sequence of events occurs:

0000	7 day completion time for LCO 3.5.1, *ECCS Operating*, Condition A expires and Condition C is entered requiring that the Unit be placed in MODE 3 in 12 hours.
0030	Plant shutdown is commenced per LCO 3.5.1, Condition C.
0050	LPCI A is repaired and declared operable; LCO 3.5.1 Conditions A and C are exited.
0100	Management decides to continue the plant shutdown as planned to complete other maintenance items.
0230	Unit Two in MODE 3

(REFERENCE PROVIDED)

Which one of the following completes both statements below?

IAW OI-01.07, *Notifications*, an Emergency Notification System (ENS) report to the NRC must be submitted no later than __(1)__.

IAW 10 CFR 50.73, *Licensee Event Reporting System*, an LER __(2)__ required.

A. (1) 0400
(2) is

B. (1) 0400
(2) is NOT

C. (1) 0430
(2) is

D. (1) 0430
(2) is NOT

100. Unit One and Unit Two have entered SBO procedures at time 1300 due to a loss of all onsite and offsite power.

Which one of the following completes both statements below?

IAW 1EOP-01-SBO, *Station Blackout*, opening the reactor building roof hatch is required to be performed **no later than** __(1)__.

IAW 0OI-37.14, *Station Blackout Procedure Basis Document*, the reactor building doors and roof hatch are opened to ensure __(2)__.

A. (1) 1330
(2) equipment availability

B. (1) 1330
(2) habitability

C. (1) 1500
(2) equipment availability

D. (1) 1500
(2) habitability

SRO Written Exam Reference Index

1. 0EOP-01-NL, EOP/SAMG Numerical Limits and Values, Attachment 3, Containment Parameters, Secondary Containment Area Temperature Limits, Table 3-B
2. 0EOP-01-SBO-01, Attachment 4, Drywell Temperature Calculation Using RSDP Recorder Inputs
3. 0EOP-01-UG, User’s Guide, Attachment 7, Heat Capacity Temperature Limit
4. 0EOP-01-UG, User’s Guide, Attachments 19 (RPV Saturation Limit), 22 Shutdown Range Level Instrument (N027A, B) Caution), and 31 (RPV Level Caution, pages 1 & 2)
5. 0EOP-01-UG, User’s Guide, Attachment 26, Unit 2 RPV Level at LL4
6. 0OI-01-07, Notifications, Attachment 1, Reportability Evaluation Checklist
7. NUREG 1022, Event Report Guidelines, Table 1, Reportable Events
8. 1OP-27, Attachment 2, Estimated Capability Curves
9. ODCM 7.3.10, Gaseous Radwaste Treatment System
10. Tech Spec 3.2.1, Average Planar Linear Heat Generation Rate
11. Tech Spec 3.2.2, Minimum Critical Power Ratio
12. Tech Spec 3.2.3, Linear Heat Generation Rate
13. Tech Spec 3.3.1.1, Reactor Protection System (RPS) Instrumentation
14. Tech Spec 3.3.5.1, Emergency Core Cooling System (ECCS) Instrumentation
15. Tech Spec 3.6.1.3, Primary Containment Isolation Valves (PCIVs)
16. Tech Spec 3.8.1, AC Sources - Operating
17. 1-FP-05887, Auto Depressurization System Elementary Diagram Unit
18. 10PEP-02.1, Brunswick Nuclear Plant Initial Emergency Actions

Containment Parameters

Secondary Containment Area Temperature Limits

Table 3-B

PLANT AREA	PLANT LOCATION DESCRIPTION	MAX NORM OPERATING VALUE (°F)	MAX SAFE OPERATING VALUE (°F)	AUTO GROUP ISOLATION
N CORE SPRAY	N CORE SPRAY ROOM	120	175	N/A
S CORE SPRAY	S CORE SPRAY ROOM	120	175	N/A
RWCU	PMP ROOM A PMP ROOM B HX ROOM	140	225	3
N RHR	N RHR EQUIP ROOM	175	295	N/A
S RHR	S RHR EQUIP ROOM RCIC EQUIP ROOM	175 165	295 295	N/A 5
HPCI	HPCI EQUIP ROOM	165	165	4
STEAM TUNNEL	RCIC STM TUNNEL HPCI STM TUNNEL	190 190	295 295	5 4
20 FT	2O FT NORTH 20 FT SOUTH	140 140	200 200	N/A N/A
50 FT	50 FT NW 50 FT SE	140 140	200 200	N/A N/A
REACTOR BLDG	MULTIPLE AREAS ANNUN. A-02 5-7	ALARM SETPOINT	N/A	3, 4, AND/OR 5
REACTOR BLDG	MSIV PIT ANNUN. A-06 6-7	ALARM SETPOINT	N/A	1

Drywell Temperature Calculation Using RSDP Recorder Inputs

Values obtained from Recorder CAC-TR-778

Above 70' Elevation

PT 1 ___________ x 0.141 = _________ °F

Between 28' and 45' Elevation

PT 3 ___________ x 0.404 = _________ °F

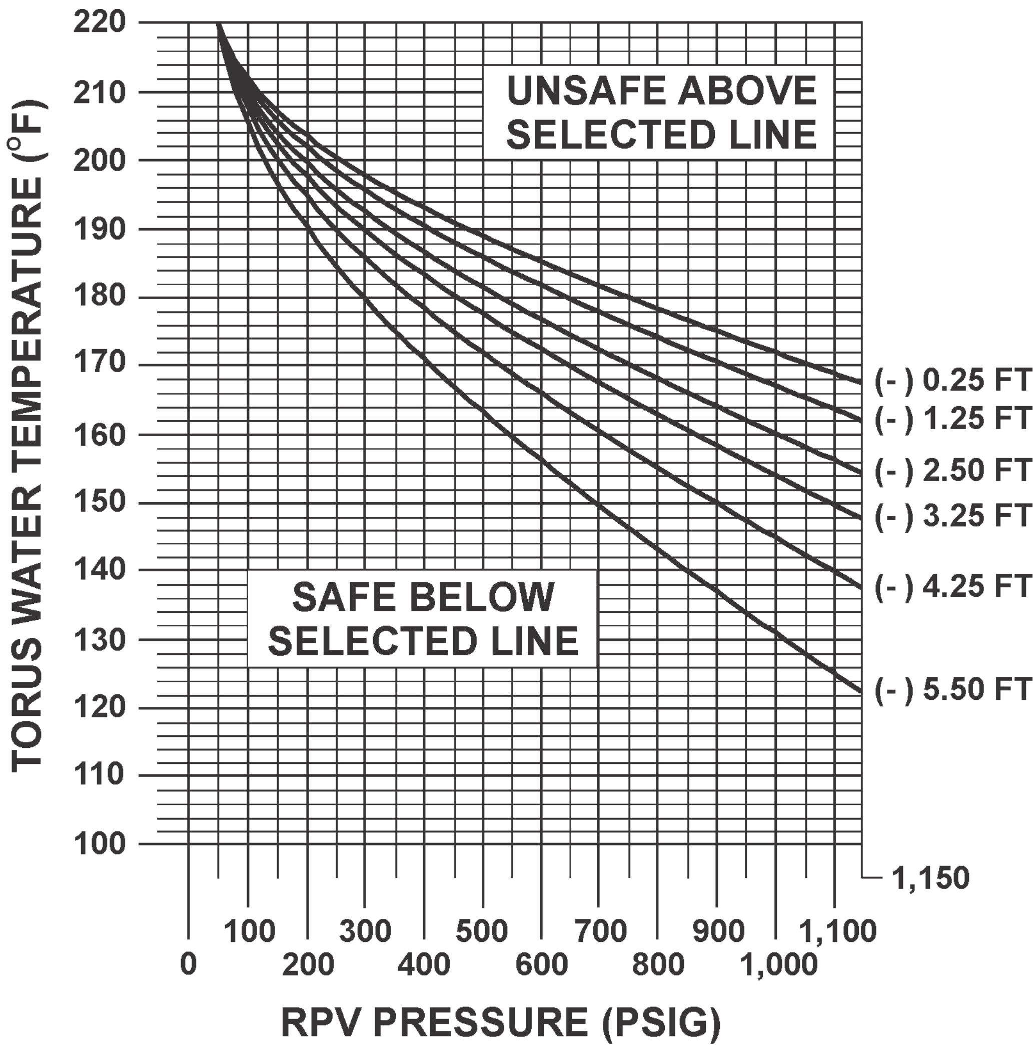
Between 10' and 23' Elevation

PT 4 ___________ x 0.455 = _________ °F

Average Drywell Temperature ______________ °F
(Sum of 3 Regional Weighted Areas)

ATTACHMENT 7
Page 1 of 1

<< Heat Capacity Temperature Limit >>



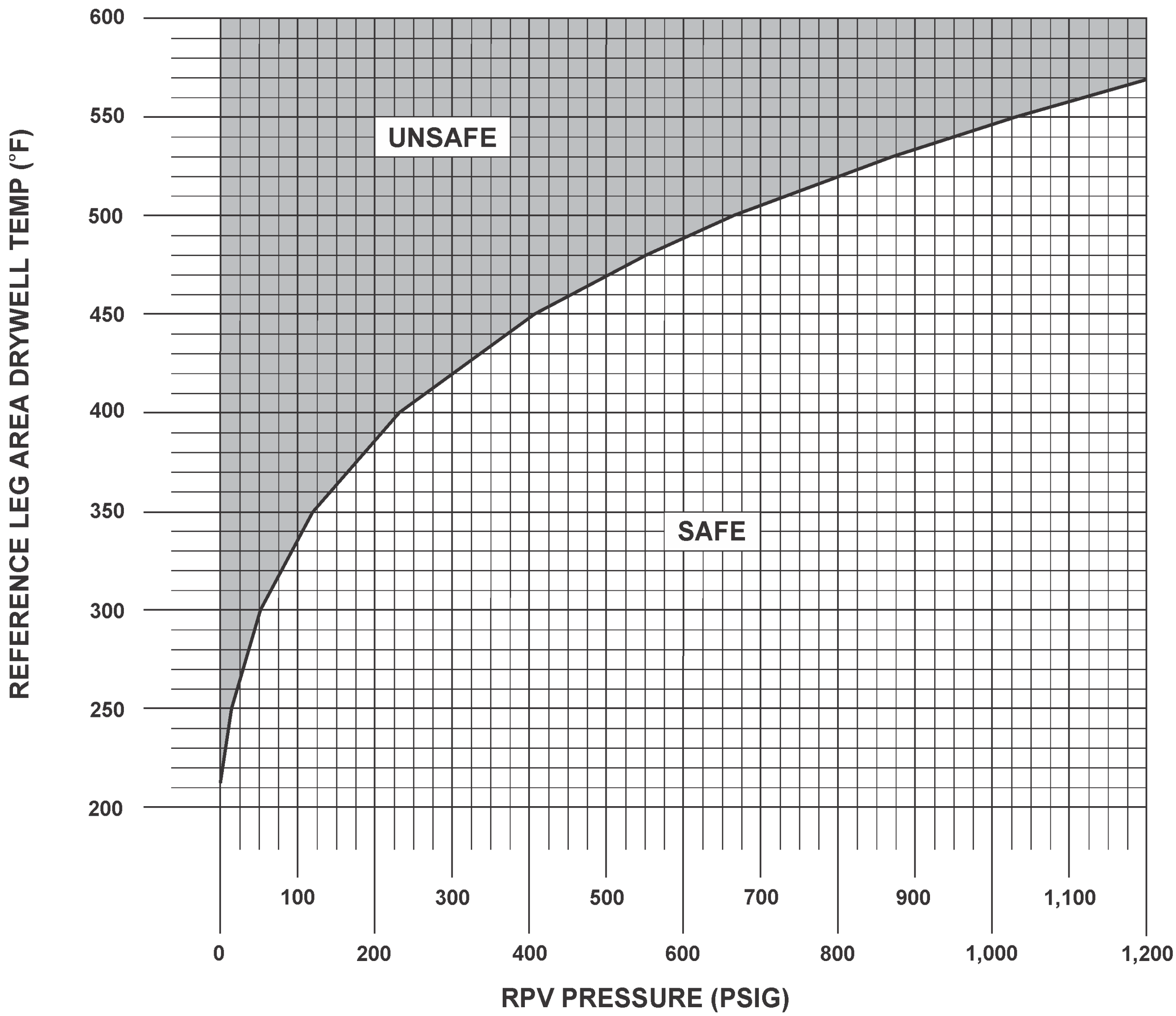
Torus water temperature is determined by:

- CAC-TR-4426-1A, Point Wtr Avg **OR**
- CAC-TR-4426-2A, Point Wtr Avg **OR**
- Computer point G050 **OR**
- Computer point G051 **OR**
- CAC-TY-4426-1 **OR**
- CAC-TY-4426-2

Select graph line immediately below torus water level as the limit.

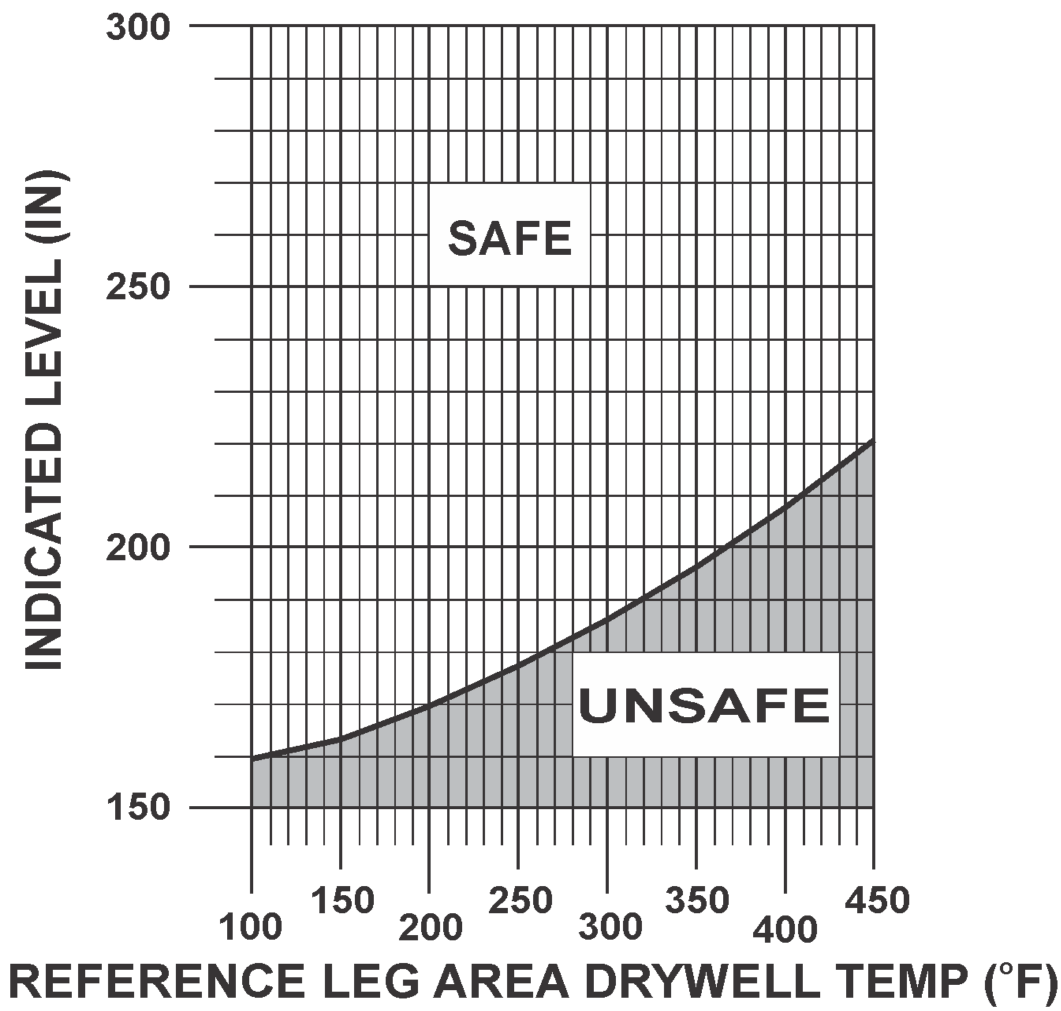
ATTACHMENT 19
Page 1 of 1

<< RPV Saturation Limit >>



ATTACHMENT 22
Page 1 of 1

<< Shutdown Range Level
Instrument (N027A, B) Caution >>



<< RPV Level Caution >>

Caution 1

A RPV level instrument may be used to determine RPV level only when the conditions for use specified below are satisfied for that instrument.

NOTE

- Reference leg area drywell temperature is determined using Attachment 18, Level Instrument Reference Leg Area Drywell Temperature Calculations, ERFIS or Instructional Aid based on Attachment 18. .... ☐
- If the temperature near any instrument run is in the UNSAFE region of the Attachment 19, RPV Saturation Limit, the instrument may be unreliable due to boiling in the run. .... ☐
- Immediate reference leg boiling is not expected to occur for short duration excursions into the unsafe region due to heating of the drywell. The thermal time constant associated with the mass of metal and water in the reference leg will prohibit immediate boiling of the reference leg. Reference leg boiling is an obvious phenomenon. Large scale oscillations of all water level instruments associated with the reference leg that is boiling will occur. This occurrence will be obvious and readily observable by the operator. Additionally, if the operator is not certain whether boiling has occurred, he can refer to plant history as provided on water level recorders or ERFIS. Reference leg boiling is indicated by level oscillations without corresponding pressure oscillations. .... ☐

Instrument	Conditions for Use
Narrow Range Level Instruments C32-LI-R606A, B, C (N004A, B, C) C32-LPR-R608 (N004A, B) Indicating Range 150-210 Inches Cold Reference Leg	**<u>Unit 1 Only</u>:** The indicated level is in the SAFE region of Attachment 20. **<u>Unit 2 Only</u>:** The indicated level is in the SAFE region of Attachment 21.
Shutdown Range Level Instruments B21-LI-R605A, B (N027A, B) Indicating Range 150-550 Inches Cold Reference Leg	The indicated level is in the SAFE region of Attachment 22. To determine RPV level at the Main Steam Line Flood Level (MSL), see Attachment 30. Attachment 30 has two curves: The upper curve is for reference leg area drywell temperature equal to or greater than 200°F. The lower curve is for reference leg area drywell temperature less than 200°F.

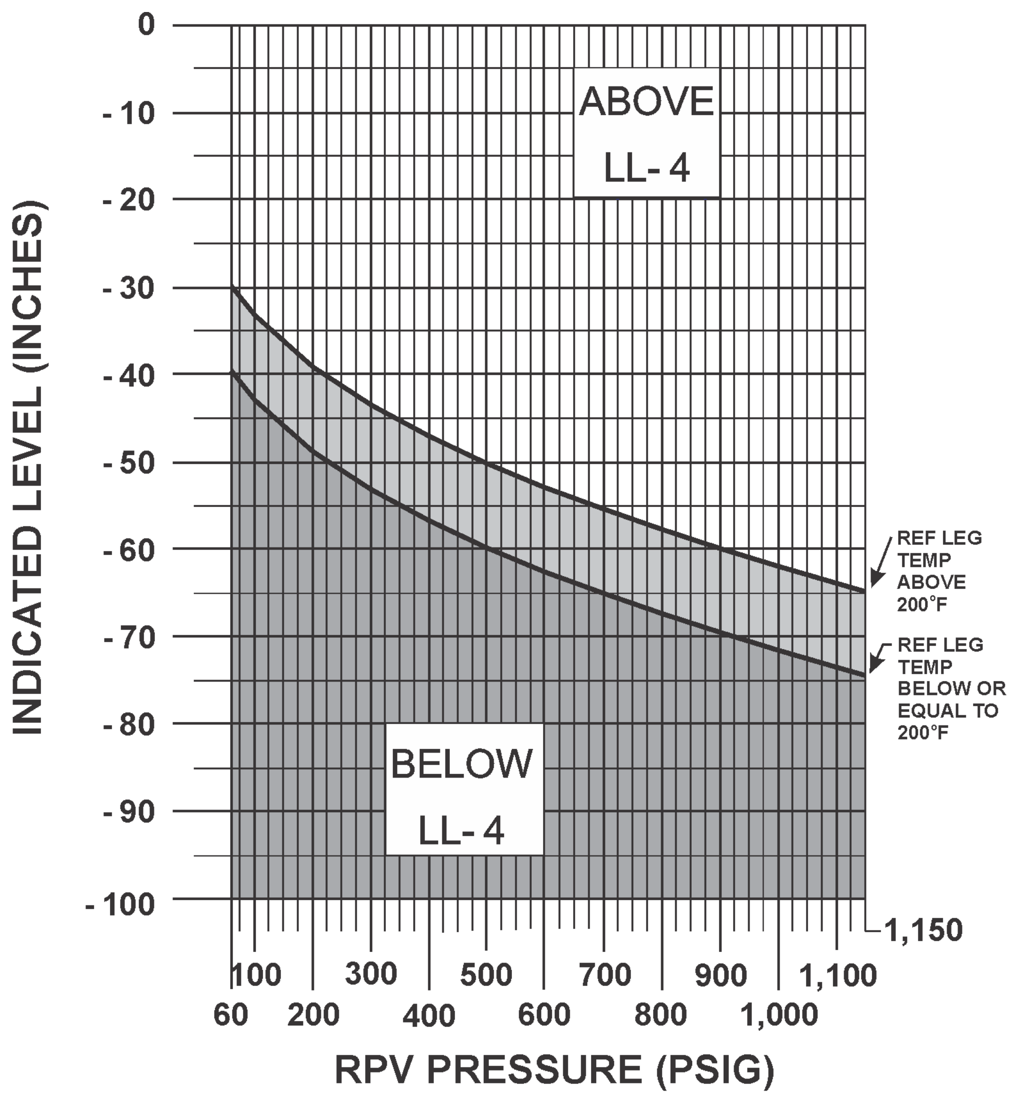
<< RPV Level Caution >>

Caution 1 (Continued)

Instrument	Conditions for Use
Wide Range Level Instruments B21-LI-R604A, B (N026A, B) C32-PR-R609 (N026B) Indicating Range 0-210 Inches Cold Reference Leg	• Temperature on the Reactor Building 50' below 140°F (B21-XY-5948A A2-4, B21-XY-5948B A2-4, ERFIS Computer Point B21TA102, **OR** B21TA103) **AND** • **IF** the reference leg area drywell temperature is in the UNSAFE region of Attachment 19, RPV Saturation Limit, **THEN** the indicated level is greater than 20 inches **OR** **IF** the reference leg area drywell temperature is in the SAFE region of Attachment 19, RPV Saturation Limit, **THEN** the indicated level is greater than 10 inches.

ATTACHMENT 26
Page 1 of 1

<< Unit 2 RPV Level at LL 4
(Minimum Steam Cooling RPV Level) >>



When RPV pressure is less than 60 psig, use indicated level. LL-4 is -27.5 inches.

<< Reportability Evaluation Checklist >>

NOTE

If the answer to the following question is YES, then Accelerated Verbal Notification to the NRC is required within 15 minutes. Reference 0AOP-40.0, Security Events, for notification content.

15 MINUTE REPORTABILITY			
ITEM #	**YES**	**NO**	**DESCRIPTIVE QUESTION**
NA			Has a Hostile Action occurred? [NRC Bulletin 2005-02]

NOTE

- NUREG-1022, Rev. 3 is a reference to provide additional guidance on reportability.
- If the answer to any of the following questions is YES, the event is reportable within 1 hour.
- If all answers to the following questions are NO, the event is not reportable within 1 hour.

1 HOUR REPORTABILITY			
ITEM #	**YES**	**NO**	**DESCRIPTIVE QUESTION**
1.1			Is the event a deviation from technical specifications as per 10 CFR 50.54(X)? [10 CFR 50.72(b)(1)]
1.2 1.2.1			Does the event involve by-product, source or special nuclear material possessed by the licensee that might have or threatens to cause: Any individual's exposure to reach or exceed 25 Rems total effective dose equivalent (TEDE); 75 Rems eye dose equivalent; or 250 Rads shallow-dose equivalent to the skin or extremities? [10 CFR 20.2202(a)(1)]
1.2.2			The release of radioactive material inside or outside of a restricted area, such that, had an individual been present for 24 hours, the individual could have received an intake 5 times the occupational annual limit on intake? [10 CFR 20.2202(a)(2)]
1.3			ISFSI - Does the event involve accidental criticality or loss of any special nuclear material? [10 CFR 72.74(a)]
1.4			Does the event involve the discovery of a cyber attack that adversely impacted safety related or important-to-safety functions, security functions, or emergency preparedness functions (including offsite communications); or that compromised support systems and equipment resulting in adverse impacts to safety, security, or emergency preparedness functions within the scope of 10 CFR 73.54? (Note 1) [10 CFR 73.77(a)(1)]

Notes:

1. Assistance with 10 CFR 73.77 reporting can be provided by the CSIRT.

<< Reportability Evaluation Checklist >>

NOTE

- If the answer to any of the following questions is YES, the event is reportable within 4 hours.
- If all answers to the following questions are NO, the event is not reportable within 4 hours.

4 HOUR REPORTABILITY

ITEM #	YES	NO	DESCRIPTIVE QUESTION
2.1			Is plant shutdown required by technical specifications being initiated? (Note 1) [10 CFR 50.72(b)(2)(i)]
2.2			Has the event resulted in or should have resulted in an Emergency Core Cooling System (ECCS) discharge into the Reactor Coolant System as a result of a valid signal, except when the actuation resulted from and was part of a pre-planned sequence during testing or reactor operation? [10 CFR 50.72(b)(2)(iv)(A)]
2.3			Did the event or condition result in actuation of the reactor protection system (RPS) when the reactor was critical, except when the actuation resulted from and was part of a pre-planned sequence during testing or reactor operation?[10 CFR 50.72(b)(2)(iv)(B)]
2.4			Is the event a situation, as related to the health and safety of the public or on-site 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? (Note 2) (Note 3) [10 CFR 50.72(b)(2)(xi)] [10 CFR 72.75(b)(2)]

Notes:

1. Includes any Safety Limit violation (Tech Spec 2.2)
2. Such an event may include an on-site fatality or an inadvertent release of radioactively contaminated materials.
3. The North Carolina Wildlife Commission's Sea Turtle Coordinator (NCSTC) is notified of each sea turtle recovery. A report per 10 CFR 50.72(b)(2)(xi) is required 1) when a dead turtle is recovered **OR** 2) when, after consultation with the NCSTC, it is determined that an injured turtle requires rehabilitation versus release. The NRC notification is required no later than 4 hrs after consultation with the NCSTC when either of these conditions is met.

<< Reportability Evaluation Checklist >>

4 HOUR REPORTABILITY			
ITEM #	**YES**	**NO**	**DESCRIPTIVE QUESTION**
2.5			Has any licensed material been lost, stolen, or missing in an aggregate quantity equal to or greater than 1,000 times the quantity specified in 10 CFR 20 Appendix C under such circumstances that it appears that an exposure could result to persons in unrestricted areas? (Note 1) [10 CFR 20.2201(a)(i)]
2.6			ISFSI – Departure from License Condition. Has an action been taken in an emergency that departs from a condition or a technical specification contained in a license or certificate of compliance issued under 10 CFR 72 when the action was immediately needed to protect the public health and safety and no action consistent with license conditions or technical specifications that could provide adequate or equivalent protection was immediately apparent as per 72.32(d)? [10 CFR 72.75(b)(1)]
2.7			Does the event involve discovery of a cyber attack that could have caused an adverse impact to safety related or important-to-safety functions, security functions, or emergency preparedness functions (including offsite communications); or that could have compromised support systems and equipment, which if compromised could have adversely impacted safety, security, or emergency preparedness functions within the scope of 10 CFR-73.54? (Note 2) [10 CFR 73.77(a)(2)(i)]
2.8			Does the event involve discovery of a suspected or actual cyber attack initiated by personnel with physical or electronic access to digital computer and communication systems and networks within the scope of 10 CFR 73.54? (Note 2) [10 CFR 73.77(a)(2)(ii)]
2.9			Does the event involve notification of a local, State, or other Federal agency (e.g., law enforcement, FBI, etc.) of an event related to implementation of the cyber security program for digital computer and communication systems and networks within the scope of 10 CFR 73.54 that does not otherwise require a notification under paragraph (a) of this section? (Note 2) [10 CFR 73.77(a)(2)(iii)]

Notes:

1. Further information is located in AD-SY-ALL-0150, Reporting Safeguards, Security, and Fitness for Duty Events.
2. Assistance with 10 CFR 73.77 reporting can be provided by the CSIRT.

<< Reportability Evaluation Checklist >>

NOTE

- If the answer to any of the following questions is YES, the event is reportable within 8 hours.
- If all the answers to the following questions are NO, the event is not reportable within 8 hours.

8 HOUR REPORTABILITY			
ITEM #	**YES**	**NO**	**DESCRIPTIVE QUESTION**
3.1			Has the event or condition resulted in the condition of the nuclear power plant, including its principal safety barriers, being seriously degraded? [10 CFR 50.72(b)(3)(ii)(A)]
3.2			Has the event or condition resulted in the nuclear power plant being in an unanalyzed condition that significantly degrades plant safety? [10 CFR 50.72(b)(3)(ii)(B)]
3.3			Did the event or condition result in valid actuation of any of the systems listed below except when the actuation resulted from and is part of a pre-planned sequence during testing or reactor operation? (Note 1) [10 CFR 50.72(b)(3)(iv)(A)]
3.3.1			These systems are: Reactor protection system (RPS) including: reactor scram and reactor trip. [10 CFR 50.72(b)(3)(iv)(B)(1)]

Notes:
1. Automatic **OR** Manual initiation of the system listed is reportable. NUREG-1022, Section 3.2.6 discussion, should be referenced for additional information.

<< Reportability Evaluation Checklist >>

8 HOUR REPORTABILITY			
ITEM #	**YES**	**NO**	**DESCRIPTIVE QUESTION**
3.3.2			General containment isolation signals affecting containment isolation valves in more than one system or multiple main steam isolation valves (MSIVs). • Main Steam Isolation. • Main Steam Line Drain Isolation. • HPCI Steam Line Isolation. • RCIC Steam Line Isolation. • RWCU Suction Isolation. • Primary Containment Isolation. • Secondary Containment Isolation. • SGTS Actuation. • Combustible Gas Control (CAD). [10 CFR 50.72(b)(3)(iv)(B)(2)]
3.3.3			Emergency core cooling systems (ECCS), including: • Core Spray (CS) • High Pressure Coolant Injection (HPCI) • Low Pressure Coolant Injection (LPCI) function of the • Residual Heat Removal (RHR) • Automatic Depressurization (ADS) System [10 CFR 50.72(b)(3)(iv)(B)(4)]
3.3.4			Reactor Core Isolation Cooling (RCIC) [10 CFR 50.72(b)(3)(iv)(B)(5)]
3.3.5			Containment heat removal and depressurization systems including containment spray and fan cooler systems. • RHR Suppression Pool Cooling. • Drywell Spray System Actuation. [10 CFR 50.72(b)(3)(iv)(B)(7)]
3.3.6			Emergency Diesel Generators (DGs) [10 CFR 50.72(b)(3)(iv)(B)(8)]

<< Reportability Evaluation Checklist >>

8 HOUR REPORTABILITY			
ITEM #	**YES**	**NO**	**DESCRIPTIVE QUESTION**
3.4			Could the event or condition at the time of discovery have prevented the fulfillment of the safety function of structures or systems that are needed to: [10 CFR 50.72(b)(3)(v)] These criteria cover an event or condition in which scoped in SSCs could have failed to perform their intended function because of one or more personnel errors, including procedure violations; equipment failures; inadequate maintenance; or design, analysis, fabrication, equipment qualification, construction, or procedural deficiencies and no redundant equipment in the same system was OPERABLE. However, individual component failures need not be reported if redundant equipment in the same system was OPERABLE and available to perform the required safety function. (Note 1) [10 CFR 50.72(b)(3)(vi)]
3.4.1			Shut down the reactor and maintain it in a safe shutdown condition? (Note 1) [10 CFR 50.72(b)(3)(v)(A)]
3.4.2			Remove residual heat? [10 CFR 50.72(b)(3)(v)(B)]
3.4.3			Control the release of radioactive material? [10 CFR 50.72(b)(3)(v)(C)]
3.4.4			Mitigate the consequences of an accident? (Note 2) [10 CFR 50.72(b)(3)(v)(D)]
3.5			Does the event require the transport of a radioactively contaminated person to an off-site medical facility for treatment? [10 CFR 50.72(b)(3)(xii)] [10 CFR 72.75(c)(3)]

Notes:

1. No Event Notification [i.e., per 10 CFR 50.72(b)(3)(v)] is required for conditions which could have prevented fulfillment of the safety function that are discovered when the affected system is INOPERABLE or when the affected system is INOPERABLE but considered available. If the condition is discovered when the system is OPERABLE, an EN will be made per 10 CFR 50.72(b)(3)(v).
2. RCIC INOPERABILITY is not reportable as a single train system per 10 CFR 50.72(b)(3)(v)(d). TS Basis 3.5.3 states that the RCIC System is not an ESF system and no credit is taken in the safety analysis for RCIC System operation. As such, consistent with Example 2 on NUREG 1022, Revision 3, RCIC Failure is not reportable under 10 CFR 50.72(b)(3)(v)(d).

<< Reportability Evaluation Checklist >>

NOTE

- Additional reportability guidance concerning loss of emergency preparedness capabilities is contained in Section 3.2.13 of NUREG-1022, Rev. 3. Consultation with an Emergency Preparedness representative is advised when assessing the significance of the loss of capability.
- 0PLP-37, Equipment Important to Emergency Preparedness and ERO Response, is a reference for assistance in determining equipment important to Emergency Preparedness and whether planned or unplanned OPERABILITY of the equipment may be reportable.

8 HOUR REPORTABILITY			
ITEM #	**YES**	**NO**	**DESCRIPTIVE QUESTION**
3.6			Has the event resulted in a major loss of emergency assessment capability, off-site response capability, or communications capability (i.e., significant portion of the Main Control Room indication, emergency notification system, or off-site notification system)? [10 CFR 50.72(b)(3)(xiii)] Major loss of emergency or off-site notification system is considered to be/but not limited to: a. Loss of: 1) DEMNET; OR NRC Emergency Notification System (ENS); AND 2) Commercial telephone network. b. INOPERABILITY for greater than or equal to 1 hour of: 1) Seven or more off-site sirens; OR 2) All off-site sirens in one county.
3.7			ISFSI –Important to Safety Defect Has a defect been discovered in any Independent Spent Fuel Storage structure, system, or component that is important to safety? [10 CFR 72.75(c)(1)]
3.8			ISFSI – Reduction in Effectiveness Has a condition been discovered which results in a significant reduction in the effectiveness of any Independent Spent Fuel Storage cask confinement system during use? [10 CFR 72.75(c)(2)]

<< Reportability Evaluation Checklist >>

NOTE

Additional cyber security event reportability guidance is contained in NEI 15-09, Revision 0. Consultation with Nuclear Information Technology and Licensing is advised when assessing the significance of cyber security events.

8 HOUR REPORTABILITY			
ITEM #	**YES**	**NO**	**DESCRIPTIVE QUESTION**
3.9			Does the event involve receipt or collection of information regarding observed behavior, activities, or statements that may indicate intelligence gathering or pre-operational planning related to a cyber attack against digital computer and communication systems and networks within the scope of 10 CFR 73.54? (Note 1) [10 CFR 73.77(a)(3)]

NOTE

If the answer to any of the following questions is YES, the event is reportable within 24 hours.

24 HOUR REPORTABILITY			
ITEM #	**YES**	**NO**	**DESCRIPTIVE QUESTION**
4.1 4.1.1			Does the incident involve the loss of control of licensed material possessed by BNP which might have caused or threatens to cause: Any individual's exposure in a period of 24 hours to exceed: 5 Rems total effective dose equivalent (TEDE); or 15 Rems eye dose equivalent; or 50 Rems shallow-dose equivalent to the skin or extremities? [10 CFR 20.2202(b)(1)]
4.1.2			The release of radioactive material inside or outside of a restricted area, so that, had an individual been present for 24 hours, the individual could have received an intake in excess of one occupational annual limit on intake? [10 CFR 20.2202(b)(2)]
4.2			ISFSI – Equipment Important to Safety Disabled or Failed to Function Does the event involve equipment important to safety which is disabled or fails to function as designed when: The equipment is required by certificate of compliance to be available and OPERABLE to prevent releases that could exceed regulatory limits, to prevent exposures to radiation or radioactive materials that could exceed regulatory limits, or to mitigate the consequences of an accident; and, No redundant equipment was available and OPERABLE to perform the required safety function. [10 CFR 72.75(d)(1)]

Notes:

1. Assistance with 10 CFR 73.77 reporting can be provided by the CSIRT.

NUREG-1022, Rev. 3

Event Report Guidelines 10 CFR 50.72 and 50.73

Final Report

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Prepared by:
Aron Lewin

Office of Nuclear Reactor Regulation

EXECUTIVE SUMMARY

Two of the many elements contributing to the safety of nuclear power are emergency response and the feedback from operating experience into plant operations. These are achieved partly by the licensee event reporting requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) 50.72, “Immediate Notification Requirements for Operating Nuclear Power Reactors,” and 10 CFR 50.73, “Licensee Event Report System.” In 10 CFR 50.72, the U.S. Nuclear Regulatory Commission (NRC) provides for immediate notification requirements through the emergency notification system, and in 10 CFR 50.73 provides for 60-day written licensee event reports.

The NRC staff uses the information reported under 10 CFR 50.72 and 10 CFR 50.73 in responding to emergencies, monitoring ongoing events, confirming licensing bases, studying potentially generic safety problems, assessing trends and patterns of operational experience, monitoring performance, identifying precursors of more significant events, and providing operational experience to the industry.

NUREG-1022 contains guidelines that the staff of the NRC considers acceptable for use in meeting the requirements of 10 CFR 50.72 and 10 CFR 50.73. Several identified reporting issues could not be quickly resolved given certain ambiguities in the guidance in Revision 2 of NUREG-1022. In developing Revision 3 to NUREG-1022, the NRC held numerous public and internal meetings to solicit stakeholder input and feedback. In resolving the ambiguities, the NRC considered the provisions of the rule itself, the associated statements of consideration, and other available guidance in that hierarchal order. Revision 3 to NUREG-1022 revises the event reporting guidelines to provide clearer guidance.

Table 1 Reportable Events

Declaration of an Emergency Class **(See Section 3.1.1 of this report)**	
§ 50.72(a)(1)(i) "The declaration of any of the Emergency Classes specified in the licensee's approved Emergency Plan."	
Plant Shutdown Required by Technical Specifications **(See Section 3.2.1 of this report)**	
§ 50.72(b)(2)(i) "The initiation of any nuclear plant shutdown required by the plant's Technical Specifications."	**§ 50.73(a)(2)(i)(A)** "The completion of any nuclear plant shutdown required by the plant's Technical Specifications."
Operation or Condition Prohibited by Technical Specifications **(See Section 3.2.2 of this report)**	
	§ 50.73(a)(2)(i)(B) "Any operation or condition which was prohibited by the plant's Technical Specifications except when: (1) The Technical Specification is administrative in nature; (2) 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 (3) 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."
Deviation from Technical Specifications Authorized under § 50.54(x) **(See Section 3.2.3 of this report)**	
§ 50.72(b)(1) "... any deviation from the plant's Technical Specifications authorized pursuant to § 50.54(x) of this part."	**§ 50.73(a)(2)(i)(C)** "Any deviation from the plant's Technical Specifications authorized pursuant to § 50.54(x) of this part."
Degraded or Unanalyzed Condition **(See Section 3.2.4 of this report)**	
§ 50.72(b)(3)(ii) "Any event or condition that	**50.73(a)(2)(ii)** "Any event or condition that

Table 1 Reportable Events (continued)

results in: (A) The condition of the nuclear power plant, including its principal safety barriers, being seriously degraded; or (B) The nuclear power plant being in an unanalyzed condition that significantly degrades plant safety."	resulted in: (A) The condition of the nuclear power plant, including its principal safety barriers, being seriously degraded; or (B) The nuclear power plant being in an unanalyzed condition that significantly degraded plant safety."
External Threat or Hampering (See Section 3.2.5 of this report)	
	§ 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."
System Actuation (See Section 3.2.6 of this report)	
§ 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." **§ 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." **§ 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) of this section, except when the actuation results from and is part of a pre-planned sequence during testing or reactor operation."	**§ 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) of this section, except when: (1) The actuation resulted from and was part of a pre-planned sequence during testing or reactor operation; or

Table 1 Reportable Events (continued)

	(2) 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.”
§ 50.72(b)(3)(iv)(B) “The systems to which the requirements of paragraph (b)(3)(iv)(A) of this section apply are: (1) Reactor protection system (RPS) including: reactor scram and reactor trip.[5] (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: high-pressure and low-pressure 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. [5] Actuation of the RPS when the reactor is critical is reportable under § 50.72(b)(2)(iv)(B).	**§ 50.73(a)(2)(iv)(B)** “The systems to which the requirements of paragraph (a)(2)(iv)(A) of this section apply are: (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: high-pressure and low-pressure 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.

Table 1 Reportable Events (continued)

<table>
<tr><td>(8) Emergency ac electrical power systems, including: emergency diesel generators (EDGs); hydroelectric facilities used in lieu of EDGs at the Oconee Station; and BWR dedicated Division 3 EDGs.”</td><td>(8) Emergency ac electrical power systems, including: emergency diesel generators (EDGs); hydroelectric facilities used in lieu of EDGs at the Oconee Station; and BWR dedicated Division 3 EDGs.

(9) Emergency service water systems that do not normally run and that serve as ultimate heat sinks.”</td></tr>
<tr><td colspan="2">Event or Condition that Could Have Prevented Fulfillment of a Safety Function
(See Section 3.2.7 of this report)</td></tr>
<tr><td>§ 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.”

§ 50.72(b)(3)(vi) “Events covered in paragraph (b)(3)(v) of this section 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 paragraph (b)(3)(v) of this section if redundant equipment in the same system was operable and available to perform the required safety function.”</td><td>§ 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.”

§ 50.73(a)(2)(vi) “Events covered in paragraph (a)(2)(v) of this section 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 paragraph (a)(2)(v) of this section if redundant equipment in the same system was operable and available to perform the required safety function.”</td></tr>
<tr><td colspan="2">Common Cause Inoperability of Independent Trains or Channels
(See Section 3.2.8 of this report)</td></tr>
<tr><td></td><td>§ 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;</td></tr>
</table>

Table 1 Reportable Events (continued)

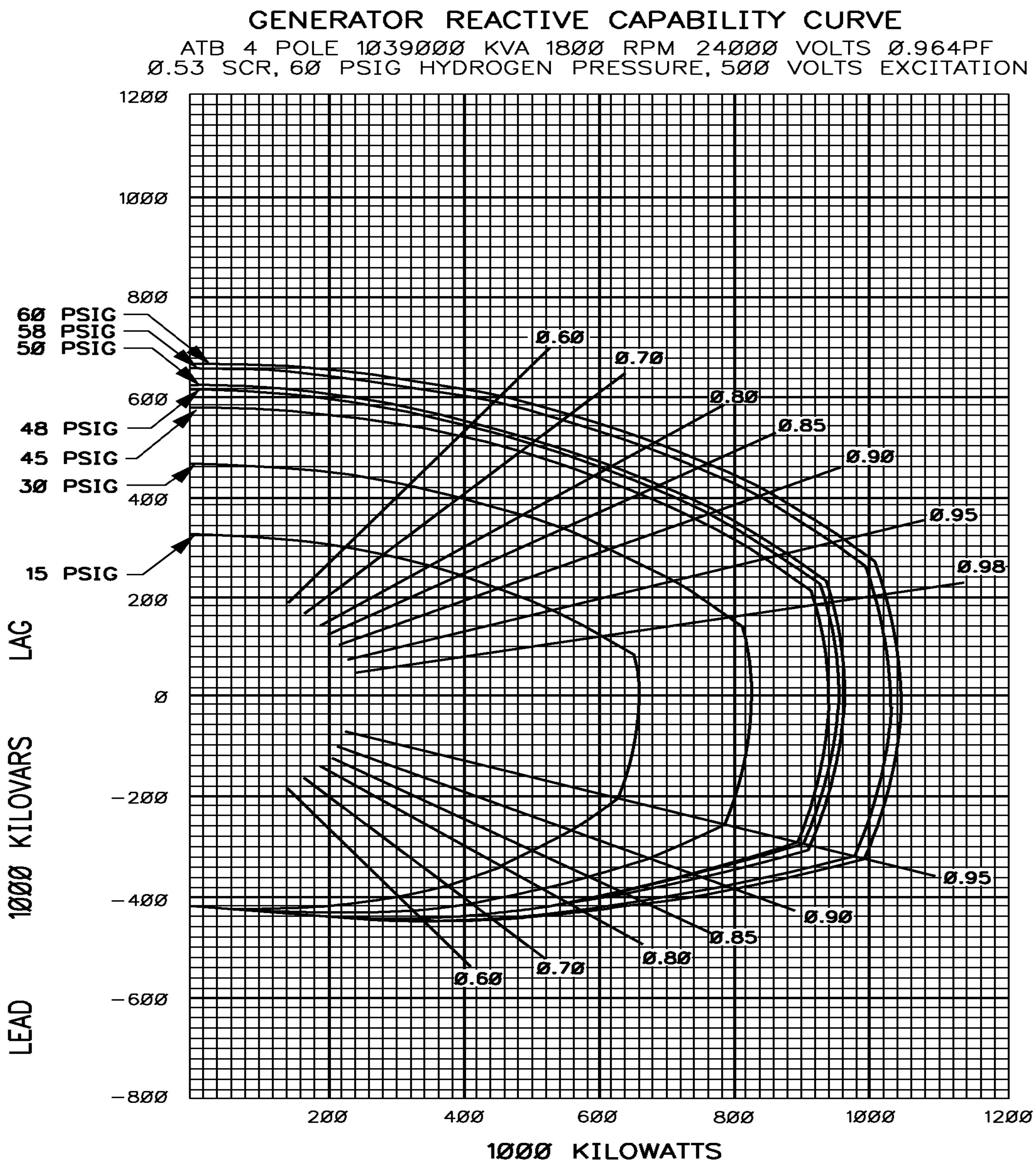
	(B) Remove residual heat; (C) Control the release of radioactive material; or (D) Mitigate the consequences of an accident."
Radioactive Release **(See Section 3.2.9 of this report)**	
	§ 50.73(a)(2)(viii)(A) "Any airborne radioactive release that, when averaged over a time period of 1 hour, resulted 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." **§ 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 (i.e., unrestricted area) for all radionuclides except tritium and dissolved noble gases."
Internal Threat or Hampering **(See Section 3.2.10 of this report)**	
	§ 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."
Transport of a Contaminated Person Offsite **(See Section 3.2.11 of this report)**	
§ 50.72(b)(3)(xii) "Any event requiring the transport of a radioactively contaminated person to an offsite medical facility for treatment."	
News Release or Notification of Other Government Agency **(See Section 3.2.12 of this report)**	
§ 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	

Table 1 Reportable Events (continued)

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 radioactively contaminated materials.”	
Loss of Emergency Preparedness Capabilities **(See Section 3.2.13 of this report)**	
§ 50.72(b)(3)(xiii) “Any event that results in a major loss of emergency assessment capability, offsite response capability, or offsite communications capability (e.g., significant portion of control room indication, emergency notification system, or offsite notification system).”	
Single Cause that Could Have Prevented Fulfillment of the Safety Functions of Trains or Channels in Different Systems (See Section 3.2.14 of this report)	
	§ 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: (1) Shut down the reactor and maintain it in a safe shutdown condition; (2) Remove residual heat; (3) Control the release of radioactive material; or (4) Mitigate the consequences of an accident.” **§ 50.73(a)(2)(ix)(B)** “Events covered in paragraph (ix)(A) of this section 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 paragraph (ix)(A) of this section if the event results from: (1) A shared dependency among trains or channels that is a natural or expected consequence of the approved plant design; or (2) Normal and expected wear or degradation.”

ATTACHMENT 2
Page 1 of 1

<< Estimated Capability Curves >>



7.3.10 GASEOUS RADWASTE TREATMENT SYSTEM

ODCMS 7.3.10 The GASEOUS RADWASTE TREATMENT SYSTEM shall be in operation.

APPLICABILITY: Whenever the Main Condenser Air Ejector (evacuation) System is in operation.

COMPENSATORY MEASURES

CONDITION	REQUIRED COMPENSATORY MEASURE	COMPLETION TIME
A. GASEOUS RADWASTE TREATMENT SYSTEM not in operation.	A.1 Place GASEOUS RADWASTE TREATMENT SYSTEM in operation.	7 days
B. NOTE: Required Compensatory Measure B.1 shall be completed if this Condition is entered. Required Compensatory measure and associated Completion Time not met.	B.1 Submit a Special Report to the NRC that identifies the required inoperable equipment and the reasons for the inoperability, corrective actions taken to restore the required inoperable equipment to OPERABLE status, and a summary description of the corrective actions taken to prevent recurrence.	30 days

TEST REQUIREMENTS

TEST		FREQUENCY
TR 7.3.10.1	Verify GASEOUS RADWASTE TREATMENT SYSTEM in operation by checking the readings of the relevant instruments.	12 hours

3.2 POWER DISTRIBUTION LIMITS

3.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)

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

APPLICABILITY: THERMAL POWER ≥ 23% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Any APLHGR not within limits.	A.1 Restore APLHGR(s) to within limits.	4 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to < 23% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.2.1.1 Verify all APLHGRs are less than or equal to the limits specified in the COLR.	Once within 12 hours after ≥ 23% RTP <u>AND</u> 24 hours thereafter

3.2 POWER DISTRIBUTION LIMITS

3.2.2 MINIMUM CRITICAL POWER RATIO (MCPR)

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

APPLICABILITY: THERMAL POWER $\geq$ 23% RTP.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Any MCPR not within limits.	A.1	Restore MCPR(s) to within limits.	4 hours
B. Required Action and associated Completion Time not met.	B.1	Reduce THERMAL POWER to < 23% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.2.1	Verify all MCPRs are greater than or equal to the limits specified in the COLR.	Once within 12 hours after $\geq$ 23% RTP AND 24 hours thereafter

(continued)

3.2 POWER DISTRIBUTION LIMITS

3.2.3 LINEAR HEAT GENERATION RATE (LHGR)

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

APPLICABILITY: THERMAL POWER ≥ 23% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Any LHGR not within limits.	A.1 Restore LHGR(s) to within limits.	4 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to < 23% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.3.1	Verify all LHGRs are less than or equal to the limits specified in the COLR.	Once within 12 hours after ≥ 23% RTP <u>AND</u> 24 hours thereafter

3.3 INSTRUMENTATION

3.3.1.1 Reactor Protection System (RPS) Instrumentation

LCO 3.3.1.1 The RPS instrumentation for each Function in Table 3.3.1.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.1.1-1.

ACTIONS

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

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more required channels inoperable.	A.1	Place channel in trip.	12 hours
	<u>OR</u>		
	A.2	---------------NOTE------------- Not applicable for Functions 2.a, 2.b, 2.c, 2.d, or 2.f. ------------------------------------ Place associated trip system in trip.	12 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. ------------NOTE---------------- Not applicable for Functions 2.a, 2.b, 2.c, 2.d, or 2.f. ------------------------------------ One or more Functions with one or more required channels inoperable in both trip systems.	B.1	Place channel in one trip system in trip.	6 hours
	OR		
	B.2	Place one trip system in trip.	6 hours
C. One or more Functions with RPS trip capability not maintained.	C.1	Restore RPS trip capability.	1 hour
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1	Enter the Condition referenced in Table 3.3.1.1-1 for the channel.	Immediately
E. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	E.1	Reduce THERMAL POWER to < 26% RTP.	4 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
F. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	F.1	Be in MODE 2.	6 hours
G. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	G.1	Be in MODE 3.	12 hours
H. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	H.1	Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately
I. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	I.1	Initiate alternate method to detect and suppress thermal hydraulic instability oscillations.	12 hours
J. Required Action and associated Completion Time of Condition I not met.	J.1	Reduce THERMAL POWER to < 20% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

--NOTES--

1. Refer to Table 3.3.1.1-1 to determine which SRs apply for each RPS Function.
2. When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains RPS trip capability.

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.1	(Not used.)	12 hours
SR 3.3.1.1.2	Perform CHANNEL CHECK.	24 hours
SR 3.3.1.1.3	------------------------------NOTE------------------------------ Not required to be performed until 12 hours after THERMAL POWER ≥ 23% RTP. --- Adjust the average power range monitor (APRM) channels to conform to the calculated power while operating at ≥ 23% RTP.	7 days
SR 3.3.1.1.4	------------------------------NOTE------------------------------ Not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. --- Perform CHANNEL FUNCTIONAL TEST.	7 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.3.1.1.5	Perform a functional test of each automatic scram contactor.	7 days
SR 3.3.1.1.6	Verify the source range monitor (SRM) and intermediate range monitor (IRM) channels overlap.	Prior to withdrawing SRMs from the fully inserted position
SR 3.3.1.1.7	-------------------------------NOTE------------------------------- Only required to be met during entry into MODE 2 from MODE 1. -- Verify the IRM and APRM channels overlap.	7 days
SR 3.3.1.1.8	Calibrate the local power range monitors.	2000 effective full power hours
SR 3.3.1.1.9	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.1.1.10	Calibrate the trip units.	92 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.11	-------------------------------NOTES------------------------------ 1. For Function 2.a, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. 2. For Functions 2.b and 2.f, the CHANNEL FUNCTIONAL TEST includes the recirculation flow input processing, excluding the flow transmitters. --- Perform CHANNEL FUNCTIONAL TEST.	184 days
SR 3.3.1.1.12	Perform CHANNEL FUNCTIONAL TEST.	24 months
SR 3.3.1.1.13	-------------------------------NOTES------------------------------ 1. Neutron detectors are excluded. 2. For Function 1, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. 3. For Functions 2.b and 2.f, the recirculation flow transmitters that feed the APRMs are included. --- Perform CHANNEL CALIBRATION.	24 months
SR 3.3.1.1.14	(Not used.)	
SR 3.3.1.1.15	Perform LOGIC SYSTEM FUNCTIONAL TEST.	24 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.16	Verify Turbine Stop Valve—Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure—Low Functions are not bypassed when THERMAL POWER is ≥ 26% RTP.	24 months
SR 3.3.1.1.17	-------------------------------NOTES--------------------------- 1. Neutron detectors are excluded. 2. For Functions 3 and 4, the sensor response time may be assumed to be the design sensor response time. 3. For Function 5, "n" equals 4 channels for the purpose of determining the STAGGERED TEST BASIS Frequency. 4. For Function 2.e, "n" equals 8 channels for the purpose of determining the STAGGERED TEST BASIS Frequency. Testing of APRM and Oscillation Power Range Monitor (OPRM) outputs shall alternate. -- Verify the RPS RESPONSE TIME is within limits.	24 months on a STAGGERED TEST BASIS
SR 3.3.1.1.18	Adjust the flow control trip reference card to conform to reactor flow.	Once within 7 days after reaching equilibrium conditions following refueling outage

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.19	Verify OPRM is not bypassed when APRM Simulated Thermal Power is ≥ 25% and recirculation drive flow is ≤ 60%.	24 months

Table 3.3.1.1-1 (page 1 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Intermediate Range Monitors					
a. Neutron Flux—High	2	3	G	SR 3.3.1.1.2 SR 3.3.1.1.4 SR 3.3.1.1.5 SR 3.3.1.1.6 SR 3.3.1.1.7 SR 3.3.1.1.13 SR 3.3.1.1.15	
	5[(a)]	3	H	SR 3.3.1.1.2 SR 3.3.1.1.4 SR 3.3.1.1.5 SR 3.3.1.1.13 SR 3.3.1.1.15	
b. Inop	2	3	G	SR 3.3.1.1.4 SR 3.3.1.1.5 SR 3.3.1.1.15	
	5[(a)]	3	H	SR 3.3.1.1.4 SR 3.3.1.1.5 SR 3.3.1.1.15	
2. Average Power Range Monitors					
a. Neutron Flux—High (Setdown)	2	3[(c)]	G	SR 3.3.1.1.2 SR 3.3.1.1.5 SR 3.3.1.1.7 SR 3.3.1.1.8 SR 3.3.1.1.11 SR 3.3.1.1.13	
b. Simulated Thermal Power—High	1	3[(c)]	F	SR 3.3.1.1.2 SR 3.3.1.1.3 SR 3.3.1.1.5 SR 3.3.1.1.8 SR 3.3.1.1.11 SR 3.3.1.1.13 SR 3.3.1.1.18	[(b)]

(continued)

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.
(b) ≤ [0.55 (W - ΔW) + 62.6% RTP] when reset for single loop operation per LCO 3.4.1, "Recirculation Loops Operating." The value of ΔW is defined in plant procedures.
(c) Each APRM channel provides inputs to both trip systems.

Table 3.3.1.1-1 (page 2 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. Average Power Range Monitors (continued)					
c. Neutron Flux—High	1	3[(c)]	F	SR 3.3.1.1.2 SR 3.3.1.1.3 SR 3.3.1.1.5 SR 3.3.1.1.8 SR 3.3.1.1.11 SR 3.3.1.1.13	
d. Inop	1,2	3[(c)]	G	SR 3.3.1.1.5 SR 3.3.1.1.11	
e. 2-Out-Of-4 Voter	1,2	2	G	SR 3.3.1.1.2 SR 3.3.1.1.5 SR 3.3.1.1.11 SR 3.3.1.1.15 SR 3.3.1.1.17	
f. OPRM Upscale	≥ 20% RTP	3[(c)]	I	SR 3.3.1.1.2 SR 3.3.1.1.5 SR 3.3.1.1.8 SR 3.3.1.1.11 SR 3.3.1.1.13 SR 3.3.1.1.18 SR 3.3.1.1.19	
3. Reactor Vessel Steam Dome Pressure—High	1,2	2	G	SR 3.3.1.1.2 SR 3.3.1.1.5 SR 3.3.1.1.9 SR 3.3.1.1.10 SR 3.3.1.1.13 SR 3.3.1.1.15 SR 3.3.1.1.17	
4. Reactor Vessel Water Level—Low Level 1	1,2	2	G	SR 3.3.1.1.2 SR 3.3.1.1.5 SR 3.3.1.1.9 SR 3.3.1.1.10 SR 3.3.1.1.13 SR 3.3.1.1.15 SR 3.3.1.1.17	
5. Main Steam Isolation Valve—Closure	1	8	F	SR 3.3.1.1.5 SR 3.3.1.1.9 SR 3.3.1.1.13 SR 3.3.1.1.15 SR 3.3.1.1.17	
6. Drywell Pressure—High	1,2	2	G	SR 3.3.1.1.2 SR 3.3.1.1.5 SR 3.3.1.1.9 SR 3.3.1.1.10 SR 3.3.1.1.13 SR 3.3.1.1.15	

(continued)

(c) Each APRM channel provides inputs to both trip systems.

(d) See COLR for OPRM period based detection algorithm (PBDA) setpoint limits.

Table 3.3.1.1-1 (page 3 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
7. Scram Discharge Volume Water Level—High	1,2	2	G	SR 3.3.1.1.5 SR 3.3.1.1.9 SR 3.3.1.1.13 SR 3.3.1.1.15	
	5[(a)]	2	H	SR 3.3.1.1.5 SR 3.3.1.1.9 SR 3.3.1.1.13 SR 3.3.1.1.15	
8. Turbine Stop Valve—Closure	≥ 26% RTP	4	E	SR 3.3.1.1.5 SR 3.3.1.1.9 SR 3.3.1.1.13 SR 3.3.1.1.15 SR 3.3.1.1.16 SR 3.3.1.1.17	
9. Turbine Control Valve Fast Closure, Control Oil Pressure—Low	≥ 26% RTP	2	E	SR 3.3.1.1.5 SR 3.3.1.1.9 SR 3.3.1.1.13 SR 3.3.1.1.15 SR 3.3.1.1.16 SR 3.3.1.1.17	
10. Reactor Mode Switch— Shutdown Position	1,2	1	G	SR 3.3.1.1.12 SR 3.3.1.1.15	
	5[(a)]	1	H	SR 3.3.1.1.12 SR 3.3.1.1.15	
11. Manual Scram	1,2	1	G	SR 3.3.1.1.9 SR 3.3.1.1.15	
	5[(a)]	1	H	SR 3.3.1.1.9 SR 3.3.1.1.15	

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

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.	Immediately
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 feature(s) inoperable when its redundant feature ECCS initiation capability is inoperable. <u>AND</u>	1 hour from discovery of loss of initiation capability for feature(s) in both divisions (continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. (continued)	B.2	--------------NOTE-------------- Only applicable for Functions 3.a and 3.b. ------------------------------------	
		Declare High Pressure Coolant Injection (HPCI) System inoperable.	1 hour from discovery of loss of HPCI initiation capability
	<u>AND</u>		
	B.3	Place channel in trip.	24 hours
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.d, 2.c, 2.d, and 2.f. -------------------------------------	
		Declare supported feature(s) inoperable when its redundant feature ECCS initiation capability is inoperable.	1 hour from discovery of loss of initiation capability for feature(s) in both divisions
	<u>AND</u>		
	C.2	Restore channel to OPERABLE status.	24 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
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.	1 hour from discovery of loss of HPCI initiation capability
	AND		
	D.2.1	Place channel in trip.	24 hours
	OR		
	D.2.2	Align the HPCI pump suction to the suppression pool.	24 hours

(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	Declare Automatic Depressurization System (ADS) valves inoperable.	1 hour from discovery of loss of ADS initiation capability in both trip systems
	AND		
	E.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
F. As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	F.1	Declare ADS valves inoperable.	1 hour from discovery of loss of ADS initiation capability in both trip systems
	<u>AND</u>		
	F.2	Restore channel to OPERABLE status.	96 hours from discovery of inoperable channel concurrent with HPCI or RCIC inoperable <u>AND</u> 8 days
G. Required Action and associated Completion Time of Condition B, C, D, E, or F not met.	G.1	Declare associated supported feature(s) inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

--NOTES--

1. Refer to Table 3.3.5.1-1 to determine which SRs apply for each ECCS Function.
2. When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed as follows: (a) for up to 6 hours for Function 3.c; and (b) for up to 6 hours for Functions other than 3.c provided the associated Function or the redundant Function maintains ECCS initiation capability.

SURVEILLANCE		FREQUENCY
SR 3.3.5.1.1	Perform CHANNEL CHECK.	24 hours
SR 3.3.5.1.2	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.5.1.3	Calibrate the trip unit.	92 days
SR 3.3.5.1.4	Perform CHANNEL CALIBRATION.	24 months
SR 3.3.5.1.5	Perform LOGIC SYSTEM FUNCTIONAL TEST.	24 months
SR 3.3.5.1.6	Perform CHANNEL FUNCTIONAL TEST.	24 months

Table 3.3.5.1-1 (page 1 of 4)
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 Level 3	1,2,3, 4[(a)], 5[(a)]	4	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.4 SR 3.3.5.1.5	≥ 13 inches
b. Drywell Pressure—High	1,2,3	4	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.4 SR 3.3.5.1.5	≤ 1.8 psig
c. Reactor Steam Dome Pressure—Low	1,2,3	4	C	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.4 SR 3.3.5.1.5	≥ 402 psig and ≤ 425 psig
	4[(a)], 5[(a)]	4	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.4 SR 3.3.5.1.5	≥ 402 psig and ≤ 425 psig
d. Core Spray Pump Start—Time Delay Relay	1,2,3, 4[(a)], 5[(a)]	2 1 per pump	C	SR 3.3.5.1.4 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 14 seconds and ≤ 16 seconds
2. Low Pressure Coolant Injection (LPCI) System					
a. Reactor Vessel Water Level—Low Level 3	1,2,3, 4[(a)], 5[(a)]	4	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.4 SR 3.3.5.1.5	≥ 13 inches
b. Drywell Pressure—High	1,2,3	4	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.4 SR 3.3.5.1.5	≤ 1.8 psig

(continued)

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

Table 3.3.5.1-1 (page 2 of 4)
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)					
c. Reactor Steam Dome Pressure—Low	1,2,3	4	C	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.4 SR 3.3.5.1.5	≥ 402 psig and ≤ 425 psig
	$4^{(a)}$, $5^{(a)}$	4	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.4 SR 3.3.5.1.5	≥ 402 psig and ≤ 425 psig
d. Reactor Steam Dome Pressure—Low (Recirculation Pump Discharge Valve Permissive)	$1^{(b)}$,$2^{(b)}$, $3^{(b)}$	4	C	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.4 SR 3.3.5.1.5	≥ 302 psig
e. Reactor Vessel Shroud Level	1,2,3	2	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.4 SR 3.3.5.1.5	≥ -50 inches
f. RHR Pump Start—Time Delay Relay	1,2,3, $4^{(a)}$, $5^{(a)}$	4 1 per pump	C	SR 3.3.5.1.4 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 9 seconds and ≤ 11 seconds
3. High Pressure Coolant Injection (HPCI) System					
a. Reactor Vessel Water Level—Low Level 2	1, $2^{(c)}$, $3^{(c)}$	4	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.4 SR 3.3.5.1.5	≥ 101 inches
b. Drywell Pressure—High	1, $2^{(c)}$,$3^{(c)}$	4	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.4 SR 3.3.5.1.5	≤ 1.8 psig

(continued)

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

(b) With associated recirculation pump discharge valve or recirculation pump discharge bypass valve open.

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

Table 3.3.5.1-1 (page 3 of 4)
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. HPCI System (continued)					
c. Reactor Vessel Water Level—High	1, $2^{(c)}$, $3^{(c)}$	2	C	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.4 SR 3.3.5.1.5	≤ 207 inches
d. Condensate Storage Tank Level—Low	1, $2^{(c)}$, $3^{(c)}$	2	D	SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.5	≥ 23 feet 4 inches
e. Suppression Chamber Water Level—High	1, $2^{(c)}$, $3^{(c)}$	2	D	SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.5	≤-2 feet
4. Automatic Depressurization System (ADS) Trip System A					
a. Reactor Vessel Water Level—Low Level 3	1, $2^{(c)}$, $3^{(c)}$	2	E	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.4 SR 3.3.5.1.5	≥ 13 inches
b. ADS Timer	1, $2^{(c)}$, $3^{(c)}$	1	F	SR 3.3.5.1.4 SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 108 seconds
c. Reactor Vessel Water Level—Low Level 1	1, $2^{(c)}$, $3^{(c)}$	1	E	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.4 SR 3.3.5.1.5	≥ 153 inches
d. Core Spray Pump Discharge Pressure—High	1, $2^{(c)}$, $3^{(c)}$	2	F	SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.5	≥ 102 psig and ≤ 130 psig
e. RHR (LPCI Mode) Pump Discharge Pressure—High	1, $2^{(c)}$, $3^{(c)}$	4 2 per pump	F	SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.5	≥ 102 psig and ≤ 130 psig

(continued)

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

Table 3.3.5.1-1 (page 4 of 4)
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					
a. Reactor Vessel Water Level—Low Level 3	1, 2(c), 3(c)	2	E	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.4 SR 3.3.5.1.5	≥ 13 inches
b. ADS Timer	1, 2(c), 3(c)	1	F	SR 3.3.5.1.4 SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 108 seconds
c. Reactor Vessel Water Level—Low Level 1	1, 2(c), 3(c)	1	E	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.4 SR 3.3.5.1.5	≥ 153 inches
d. Core Spray Pump Discharge Pressure—High	1, 2(c), 3(c)	2	F	SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.5	≥ 102 psig and ≤ 130 psig
e. RHR (LPCI Mode) Pump Discharge Pressure—High	1, 2(c), 3(c)	4 2 per pump	F	SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.5	≥ 102 psig and ≤ 130 psig

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

3.6 CONTAINMENT SYSTEMS

3.6.1.3 Primary Containment Isolation Valves (PCIVs)

LCO 3.6.1.3 Each PCIV, except reactor building-to-suppression chamber vacuum breakers, shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,

When associated instrumentation is required to be OPERABLE per LCO 3.3.6.1, "Primary Containment Isolation Instrumentation."

ACTIONS

--NOTES--

1. Penetration flow paths may be unisolated intermittently under administrative controls.
2. Separate Condition entry is allowed for each penetration flow path.
3. Enter applicable Conditions and Required Actions for systems made inoperable by PCIVs.
4. Enter applicable Conditions and Required Actions of LCO 3.6.1.1, "Primary Containment," when PCIV leakage results in exceeding overall containment leakage rate acceptance criteria.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. --------------NOTE------------- Only applicable to penetration flow paths with two PCIVs. ------------------------------------ One or more penetration flow paths with one PCIV inoperable except for MSIV leakage not within limit.	A.1 AND	Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured.	8 hours (continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2 --------------NOTE-------------- Isolation devices in high radiation areas may be verified by use of administrative means. ----------------------------------- Verify the affected penetration flow path is isolated.	Once per 31 days for isolation devices outside primary containment <u>AND</u> Prior to entering MODE 2 or 3 from MODE 4, if primary containment was de-inerted while in MODE 4, if not performed within the previous 92 days, for isolation devices inside primary containment

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. --------------NOTE-------------- Only applicable to penetration flow paths with two PCIVs. ----------------------------------- One or more penetration flow paths with two PCIVs inoperable except for MSIV leakage not within limit.	B.1	Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.	2 hours
C. --------------NOTE-------------- Only applicable to penetration flow paths with only one PCIV. ----------------------------------- One or more penetration flow paths with one PCIV inoperable.	C.1	Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.	8 hours except for excess flow check valves (EFCVs) AND 12 hours for EFCVs
	AND		
	C.2	--------------NOTE-------------- Isolation devices in high radiation areas may be verified by use of administrative means. ----------------------------------- Verify the affected penetration flow path is isolated.	Once per 31 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. One or more penetration flow paths with one or more MSIVs not within MSIV leakage rate limits.	D.1 Restore leakage rate to within limit.	8 hours
E. Required Action and associated Completion Time of Condition A, B, C, or D not met in MODE 1, 2, or 3.	E.1 Be in MODE 3. <u>AND</u> E.2 Be in MODE 4.	12 hours 36 hours
F. Required Action and associated Completion Time of Condition A, B, C, or D not met for PCIV(s) required to be OPERABLE during MODE 4 or 5.	F.1 Initiate action to suspend operations with a potential for draining the reactor vessel (OPDRVs). <u>OR</u> F.2 Initiate action to restore valve(s) to OPERABLE status.	Immediately Immediately

3.8 ELECTRICAL POWER SYSTEMS

3.8.1 AC Sources—Operating

LCO 3.8.1

APPLICABILITY:

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ----------------NOTE------------- Only applicable when Unit 2 is in MODE 4 or 5. ------------------------------------- One Unit 2 offsite circuit inoperable.	A.1 Restore Unit 2 offsite circuit to OPERABLE status.	45 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. --------------NOTES-------------- 1. Only applicable when Unit 2 is in MODE 4 or 5. 2. Condition B shall not be entered in conjunction with Condition A. -------------------------------------- Two Unit 2 offsite circuits inoperable due to one Unit 2 balance of plant circuit path to the downstream 4.16 kV emergency bus inoperable for planned maintenance. <u>AND</u> DG associated with the affected downstream 4.16 kV emergency bus inoperable for planned maintenance.	B.1	Declare required feature(s) with no power available inoperable when the redundant required feature(s) are inoperable.	Immediately from discovery of Condition B concurrent with inoperability of redundant required feature(s)
	<u>AND</u>		
	B.2	Perform SR 3.8.1.1 for OPERABLE offsite circuit(s).	2 hours <u>AND</u> Once per 12 hours thereafter
	<u>AND</u>		
	B.3	Restore both Unit 2 offsite circuits and DG to OPERABLE status.	7 days <u>AND</u> 10 days from discovery of failure to meet LCO 3.8.1.a or b

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. One offsite circuit inoperable for reasons other than Condition A or B.	C.1	Perform SR 3.8.1.1 for OPERABLE offsite circuit(s).	2 hours AND Once per 12 hours thereafter
	AND		
	C.2	Declare required feature(s) with no offsite power available inoperable when the redundant required feature(s) are inoperable.	24 hours from discovery of no offsite power to one 4.16 kV emergency bus concurrent with inoperability of redundant required feature(s)
	AND		
	C.3	Restore offsite circuit to OPERABLE status.	72 hours AND 17 days from discovery of failure to meet LCO 3.8.1.a or b

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. One DG inoperable for reasons other than Condition B.	D.1	Perform SR 3.8.1.1 for OPERABLE offsite circuit(s).	2 hours AND Once per 12 hours thereafter
	AND		
	D.2	Evaluate availability of supplemental diesel generator (SUPP-DG)	2 hours AND Once per 12 hours thereafter
	AND		
	D.3	Declare required feature (s), supported by the inoperable DG, inoperable when the redundant required feature (s) are inoperable.	4 hours from discovery of Condition D concurrent with inoperability of redundant required feature (s)
	AND		
	D.4.1	Determine OPERABLE DG(s) are not inoperable due to common cause failure.	24 hours
	OR		
	D.4.2	Perform SR 3.8.1.2 for OPERABLE DG(s).	24 hours
	AND		

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. (continued)	D.5	Restore DG to OPERABLE status.	7 days from discovery of unavailability of SUPP-DG <u>AND</u> 24 hours from discovery of Condition D entry ≥ 6 days concurrent with unavailability of SUPP-DG <u>AND</u> 14 days <u>AND</u> 17 days from discovery of failure to meet LCO 3.8.1.a or b
E. Two or more offsite circuits inoperable for reasons other than Condition B.	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 all but one offsite circuit to OPERABLE status.	24 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. One offsite circuit inoperable for reasons other than Condition B. <u>AND</u> One DG inoperable for reasons other than Condition B.	---------------------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 emergency bus. --- F.1 Restore offsite circuit to OPERABLE status. <u>OR</u> F.2 Restore DG to OPERABLE status.	 12 hours 12 hours
G. Two or more DGs inoperable.	G.1 Restore all but one DG to OPERABLE status.	2 hours
H. Required Action and associated Completion Time of Condition A, B, C, D, E, F or G not met.	H.1 Be in MODE 3. <u>AND</u> H.2 Be in MODE 4.	12 hours 36 hours
I. One or more offsite circuits and two or more DGs inoperable. <u>OR</u> Two or more offsite circuits and one DG inoperable for reasons other than Condition B.	I.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.1.1	Verify correct breaker alignment and indicated power availability for each offsite circuit.	7 days
SR 3.8.1.2	-------------------------------NOTES------------------------------- 1. All DG starts may be preceded by an engine prelube period. 2. A modified DG start involving idling and gradual acceleration to synchronous speed may be used for this SR. When modified start procedures are not used, the time, voltage, and frequency tolerances of SR 3.8.1.7 must be met. 3. A single test at the specified Frequency will satisfy this Surveillance for both units. --- Verify each DG starts from standby conditions and achieves steady state voltage ≥ 3750 V and ≤ 4300 V and frequency ≥ 58.8 Hz and ≤ 61.2 Hz.	 31 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.3	-------------------------------NOTES------------------------------ 1. DG loadings may include gradual loading. 2. Momentary transients outside the load range do not invalidate this test. 3. This Surveillance shall be conducted on only one DG at a time. 4. This SR shall be preceded by and immediately follow, without shutdown, a successful performance of SR 3.8.1.2 or SR 3.8.1.7. 5. A single test at the specified Frequency will satisfy this Surveillance for both units. --- Verify each DG is synchronized and loaded and operates for ≥ 60 minutes at a load ≥ 2800 kW and ≤ 3500 kW.	31 days
SR 3.8.1.4	Verify each engine mounted tank contains ≥ 150 gal of fuel oil.	31 days
SR 3.8.1.5	Check for and remove accumulated water from each engine mounted tank.	31 days
SR 3.8.1.6	Verify the fuel oil transfer system operates to transfer fuel oil from the day fuel oil storage tank to the engine mounted tank.	31 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.7	-------------------------------NOTES------------------------------- 1. All DG starts may be preceded by an engine prelube period. 2. A single test at the specified Frequency will satisfy this Surveillance for both units. -- Verify each DG starts from standby condition and achieves, in ≤ 10 seconds, voltage ≥ 3750 V and frequency ≥ 58.8 Hz, and after steady state conditions are reached, maintains voltage ≥ 3750 V and ≤ 4300 V and frequency ≥ 58.8 Hz and ≤ 61.2 Hz.	184 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.8	------------------------------NOTES------------------------------ 1. SR 3.8.1.8.a shall not be performed in MODE 1 or 2 for the Unit 1 offsite circuits. However, credit may be taken for unplanned events that satisfy this SR. 2. SR 3.8.1.8.a is not required to be met if the unit power supply is from the preferred offsite circuit. 3. A single test at the specified Frequency will satisfy this Surveillance for both units. ---	
	Verify: a. Automatic transfer capability of the unit power supply from the normal circuit to the preferred offsite circuit; and b. Manual transfer of the unit power supply from the preferred offsite circuit to the alternate offsite circuit.	24 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.8.1.9	------------------------------NOTES------------------------------ 1. This Surveillance shall not be performed in MODE 1, 2, or 3 for DG 1 and DG 2. However, credit may be taken for unplanned events that satisfy this SR. 2. If performed with the DG synchronized with offsite power, it shall be performed at a power factor ≤ 0.9. 3. A single test at the specified Frequency will satisfy this Surveillance for both units. --- Verify each DG rejects a load greater than or equal to its associated core spray pump without tripping.	24 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.10	-------------------------------NOTE-------------------------------- A single test at the specified Frequency will satisfy this Surveillance for both units. -- Verify each DG's automatic trips are bypassed on an actual or simulated ECCS initiation signal except: a. Engine overspeed; b. Generator differential overcurrent; c. Low lube oil pressure; d. Reverse power; e. Loss of field; and f. Phase overcurrent (voltage restrained).	24 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.11	------------------------------NOTES------------------------------ 1. Momentary transients outside the load and power factor ranges do not invalidate this test. 2. A single test at the specified Frequency will satisfy this Surveillance for both units. --- Verify each DG operating at a power factor ≤ 0.9 operates for ≥ 60 minutes loaded to ≥ 3500 kW and ≤ 3850 kW.	24 months
SR 3.8.1.12	------------------------------NOTE------------------------------- A single test at the specified Frequency will satisfy this Surveillance for both units. --- Verify an actual or simulated ECCS initiation signal is capable of overriding the test mode feature to return each DG to ready-to-load operation.	24 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.13	-------------------------------NOTE-------------------------------- This Surveillance shall not be performed in MODE 1, 2, or 3 for the load sequence relays associated with DG 1 and DG 2. However, credit may be taken for unplanned events that satisfy this SR. --- Verify interval between each sequenced load block is within ± 10% of design interval for each load sequence relay.	24 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.14	-------------------------------NOTES------------------------------- 1. All DG starts may be preceded by an engine prelube period. 2. This Surveillance shall not be performed in MODE 1, 2, or 3 for DG 1 and DG 2. However, credit may be taken for unplanned events that satisfy this SR. --	
	Verify, on actual or simulated loss of offsite power signal in conjunction with an actual or simulated ECCS initiation signal: a. De-energization of emergency buses; b. Load shedding from emergency buses; and c. DG auto-starts from standby condition and: 1. energizes permanently connected loads in $\leq$ 10.5 seconds, 2. energizes auto-connected emergency loads through load sequence relays, 3. maintains steady state voltage $\geq$ 3750 V and $\leq$ 4300 V, 4. maintains steady state frequency $\geq$ 58.8 Hz and $\leq$ 61.2 Hz, and 5. supplies permanently connected and auto-connected emergency loads for $\geq$ 5 minutes.	24 months

U. S. Nuclear Regulatory Commission Site-Specific SRO Written Examination	
Applicant Information	
Name: Answer Key	
Date: 12/13/2016	Facility / Unit: Brunswick Unit 1/2
Region: I ☐ II ☒ III ☐ IV ☐	Reactor Type: W ☐ CE ☐ BW ☐ GE ☒
Start Time:	Finish Time:

Instructions

Use the answer sheets provided to document your answers. Staple this cover sheet on top of the answer sheets. To pass the examination you must achieve a final grade of at least 80 percent overall, with 70 percent or better on the SRO-only items if given in conjunction with the RO exam; SRO-only exams given alone require a final grade of 80 percent to pass. You have 8 hours to complete the combined examination, and 3 hours if you are only taking the SRO portion.

Applicant Certification

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

____
Applicant's Signature

Results	
RO/SRO-Only/Total Examination Values Points	____ / ____ / ____
Applicant's Score Points	____ / ____ / ____
Applicant's Grade Percent	____ / ____ / ____

1. 201003 1
Unit Two is operating at rated power when a control rod begins to drift out from position 24.

Which one of the following identifies the **first** action to be taken by the operator at the controls (OATC)?

A. Initiate a single rod scram.

B. Initiate a manual reactor scram.

C. Select and attempt to arrest the control rod.

D. Select and fully insert the control rod to position 00.

Answer: C
K/A:
201003 Control Rod and Drive Mechanism
G2.4.49Ability to perform without reference to procedures those actions that require immediate operation of system components and controls. (CFR: 41.10 / 43.2 / 45.6)

RO/SRO Rating: 4.6/4.4

Tier 2 / Group 2

K/A Match: This meets the K/A because the question is testing the operator action required to control a drifting control rod
(Chief Examiner agreed that operation of the RMCS for a rod drift would meet this K/A)

Pedigree: New

Objective: LOI-CLS-LP-07. Obj. 11b
Describe the possible cause(s) and required operator actions for the following alarms:
A-5 3-2. Control Rod Drift

Reference: None

Cog Level: Fundamental

Explanation: This abnormal positive reactivity addition requires response from the APP before entering the AOP. The APP requires that the operator attempt to arrest the drift at the intended position first, if it cannot be arrested but responds to RMCS to insert to 00, if it does not respond to RMCS to perform a single rod scram. If more than 1 rod drifts then a manual scram is required.

Distractor Analysis:

Choice A: Plausible because if the rod does not move then this is the appropriate action.

Choice B: Plausible because if more than 1 rod is drifting then this would be correct

Choice C: Correct Answer, see explanation.

Choice D: Plausible because this is the correct action if the rod is drifting in or the rod continues to drift after attempting to arrest.

SRO Basis: N/A

ROD DRIFT

AUTO ACTIONS

1. RWM withdraw or insert errors possibly causing rod block if reactor power is below the low power setpoint.
2. The reactor power will respond to the drifting rod depending upon the direction of the rod drift and rod worth, and could result in a reactor Scram if the plant is at low power operation.

CAUSE

1. Rod in uneven position due to:
   a. Leaking Scram valve.
   b. High cooling water pressure.
   c. Failure of directional control valves.
   d. Slow to settle due to fuel bundle channel bow.
2. Malfunction in alarm circuit.

OBSERVATIONS

1. "Rod Drift" indication on the full core display.
2. RWM error indications and Rod Block if reactor power is below the low power setpoint.
3. A change in neutron monitoring system meter readings as a result of the drifting rod with possible high flux alarms.
4. If drifting rod is selected, the four-rod group display will indicate an odd control rod position, a blank window, or a changing control rod position in the direction of the drift.
5. High control rod cooling water pressure and/or flow.
6. If control rod drifts to the full in position, a green backlight on the full core display will illuminate with no position readout on RTGB.
7. ROD OUT BLOCK alarm A-05 (2-2) and no withdraw permissive light.
8. Greater than normal settle times causing an odd or no-position to be present when the RMCS timer times out.

ACTIONS

1. Determine if the affected control rod(s) is drifting or if the rod(s) has scrammed using full core display, RPIS, and RWM.
2. Select the drifting rod and determine direction of drift.
   a. Attempt to arrest the drift and latch rod by performing the following:
      1) Apply appropriate insert or withdrawal signals to the rod using RMCS.
      2) If RWM is causing rod blocks, then bypass RWM if directed by Unit CRS.

ACTIONS (Continued)

3. If the rod continues to DRIFT OUT, then perform the following:

CAUTION

A control rod collett piston stuck in the withdraw (unlatched) position will allow the rod to drift full out due to its own weight when insert pressure is removed either by the RMCS or by closing Valve C12-101.

a. Notify Reactor Engineer.
b. Monitor core parameters, main steam line radiation monitors, and off-gas activity.
c. If rod responds to an RMCS insert signal, then fully insert the rod to position 00.
d. If rod fails to latch at position 00, then reapply insert signal to drive the rod full in.
e. If rod fails to respond to RMCS, then initiate a single control rod scram.
f. Refer to 0AOP-03.0.
g. Refer to Technical Specifications 3.1.3.

4. If rod continues to DRIFT IN, then perform the following:

a. Apply an RMCS insert signal and fully insert rod to position 00.

3.0 AUTOMATIC ACTIONS

1. Possible rod block or select block from a failed reed switch or a loss of power

2. CRD pumps trip after a 3 second delay on low suction pressure

4.0 OPERATOR ACTIONS

NOTE

The following should be considered for establishment as critical parameters during performance of this procedure: .... ☐

- Reactor power
- Control rod position
- Thermal limits

4.1 Immediate Actions

1. **Stop** any power changes in progress. .... ☐

NOTE

Detected control rod motion without a withdraw or insert command will cause annunciator A-05 3-2, Rod Drift, to alarm. **IF** the annunciator alarms **AND NO** blue scram light(s) are lit on the full core display, the conservative assumption is that rod(s) are drifting........ ☐

2. **IF** more than one control rod is drifting,
   THEN insert a manual scram **AND enter** 1EOP-01-RSP(2EOP-01-RSP), Reactor Scram Procedure........ ☐

2. 201001 1

Unit One is in an outage with the condensate system under clearance.
An earthquake results in damage to the CST causing level to slowly lower.

Which one of the following completes the statement below with regards to the effect on the CRD system?

The CRD system will ___(1)___ when the CST level reaches approximately ___(2)___ .

A. (1) trip
   (2) 3 feet

B. (1) trip
   (2) 11 feet

C. (1) transfer to the backup supply
   (2) 3 feet

D. (1) transfer to the backup supply
   (2) 11 feet

Answer: B

K/A:

201001 Control Rod Drive Hydraulic System

K6 Knowledge of the effect that a loss or malfunction of the following will have on the CONTROL ROD DRIVE HYDRAULIC System: (CFR: 41.7 / 45.7)

02 Condensate storage tanks

RO/SRO Rating: 3.0/3.1

Tier 2 / Group 2

K/A Match: This meets the K/A because the student has determine the effect of the loss of the CST on the CRD system.

Pedigree: New

Objective: LOI-CLS-LP-008, Obj. 8
Given plant conditions, predict the effect that a loss or malfunction of the following will have on the CRDH System: b. Condensate Storage Tank

Reference: None

Cog Level: High

Explanation: Under normal system operations the CRD system suction is from the condensate system. The alternate supply is from the CST, which will transfer automatically. With the condensate system under clearance these valves would be isolated. The standpipe for the CRD suction is at ~11 feet. The auto transfer for the suction for ECCS is at ~3 feet.

Distractor Analysis:

Choice A: Plausible because the pumps will lose NPSH and trip but the suction is at 11 feet not 3 feet.

Choice B: Correct Answer, see explanation.

Choice C: Plausible because an auto transfer to the CST would occur but in this case an auto transfer to the condensate system is not possible. 3 feet is the suction height for the ECCS system.

Choice D: Plausible because an auto transfer to the CST would occur but in this case an auto transfer to the condensate system is not possible. The second part is correct.

SRO Basis: N/A

CONTROL ROD DRIVE HYDRAULIC SYSTEM OPERATING PROCEDURE	1OP-08
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3.0 PRECAUTIONS AND LIMITATIONS

1. This procedure is Reactivity Management related per AD-OP-ALL-0203, Reactivity Management. Those portions of this procedure that move control rods in MODES 1 or 2 are considered a Direct Reactivity manipulation and Reactivity Evolution Category R2 (Reactivity Manipulation, R2). .... ☐

2. CST level is maintained greater than 11 feet to prevent CRD pumps from losing suction. .... ☐

3. 202002 1
Unit One is at rated power.

Which one of the following identifies the impact of inadvertently closing the 1A Reactor Recirculation Pump 1-B32-F031A, Pump A Disch Vlv?

The 1A Reactor Recirculation pump speed will lower to approximately:

A. 20%

B. 34%

C. 45.4%

D. 48%

Answer: B
K/A:
202002 Recirculation Flow Control System
K6 Knowledge of the effect that a loss or malfunction of the following will have on the RECIRCULATION FLOW CONTROL SYSTEM: (CFR: 41.7 / 45.7)
03 Recirculation system

RO/SRO Rating: 2.8/2.8

Tier 2 / Group 2

K/A Match: This meets the K/A because the student has to determine the effect of closing the discharge valve (which causes a loss of recirc) will have on the recirc flow control system.

Pedigree: new

Objective: LOI-CLS-LP-002.1, Obj. 17
Explain the operation of the following VFD limiters and controls: a. Limiter #1 b. Limiter #2

Reference: None

Cog Level: Fundamental

Explanation: Closing of the discharge valve will cause the pump to runback to limiter #1 (34%).

Distractor Analysis:

Choice A: Plausible because this is the minimum speed setting.

Choice B: Correct Answer, see explanation.

Choice C: Plausible because this is limiter #2 setting for Unit One

Choice D: Plausible because this is limiter #2 setting for Unit Two

SRO Basis: N/A

4. **RFCS VFD Runback #1 Logic (Figure 02-18D and 02-18E)**

The logic for VFD A Runback #1 is shown on Figure 02-18D; the logic for VFD B Runback #1 shown on Figure 02-18E is functionally identical to that for VFD A. The initiating conditions for Runback #1 are:

- Recirculation Pump A Discharge Valve B32-F031A Limit Switch LS-2 opens (equivalent to Discharge Valve Not Full Open);
- Total Feedwater Flow as sensed by DFCS is less than 16.4% for 15 seconds or more.

Unit 1 Specific VFD Parameters

Parameter	Value	Function
1170	98.9% (1661.5 rpm)	VFD Over Speed Trip (Over Speed Alarm at 93.95% or 1578.4 rpm)
2080	92.5% (1554.0 rpm)	Maximum running motor speed (based upon achieving 104.5% Core Flow)
2120	45.4% (762.7 rpm)	Runback #2 Active Maximum Motor Speed
4250	50.8% (853.0 rpm)	Manual Runback Motor Speed Low Limit

Unit 2 Specific VFD Parameters

Parameter	Value	Function
1170	103.7% (1742.2 rpm)	VFD Over Speed Trip (Over Speed Alarm at 98.5% or 1655.1 rpm)
2080	97.9% (1644.7 rpm)	Maximum running motor speed (based upon achieving 104.5% Core Flow)
2120	48% (806.4 rpm)	Runback #2 Active Maximum Motor Speed
4250	53.6% (900.5 rpm)	Manual Runback Motor Speed Low Limit

VFD Parameters Common to Both Units

Parameter	Value	Function
2090	20% (336 rpm)	Minimum Running Motor Speed
2100	34% (571.2 rpm)	Runback #1 Active Maximum Motor Speed

4. 203000 1

A line break has occurred in the Unit Two drywell with the following sequence of events:

1155	Drywell pressure rises above 1.7 psig
1202	RPV pressure drops below 410 psig
1203	RPV level drops to LL3

Which one of the following completes the statement below?

The **earliest** time that the operator can throttle the 2-E11-F048A, Loop 2A RHR Heat Exchanger Bypass Valve is at:

A. 1205.

B. 1206.

C. 1207.

D. 1208.

Answer: A

K/A:

203000 RHR / LPCI: Injection Mode

A4 Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8)

04 Heat exchanger cooling flow

RO/SRO Rating: 3.6/3.6

Tier 2 / Group 1

K/A Match: This meets the K/A because the student has to determine when the HX cooling flow can be operated.

Pedigree: Bank

Objective: LOI-CLS-LP-017, Obj. 09
Given an RHR pump or valve, list the interlocks, permissives and/or automatic actions associated with the RHR pump or valve, including setpoints.

Reference: None

Cog Level: High

Explanation: The heat exchanger bypass valve has a 3 minute timer that starts on a LOCA signal. Drywell pressure greater than 1.7# and reactor pressure is less than 410# is the first LOCA signal. The injection valve has a 5 minute interlock initiated by the same conditions. Another LOCA signal is introduced when reactor water level less than LL3 which provides the plausibility of the distractors.

Distractor Analysis:

Choice A: Correct Answer, see explanation.

Choice B: Plausible because this is 3 minutes from the LL3 LOCA signal.

Choice C: Plausible because this is 5 minutes from the low pressure setpoint which is time limit interlock for the injection valve.

Choice D: Plausible because this is 5 minutes from the LL3 LOCA signal.

SRO Basis: N/A

After an initiation signal is received the following actions will occur:

- all four RHR pumps will start 10 seconds after power is available to the E-buses.
- Recirculation pumps are tripped via LL#2
- All valves not needed for LPCI injection automatically isolate and are interlocked shut as previously described.
- Heat exchanger bypass valve F048A/B opens and cannot be throttled for 3 minutes after an initiation signal is received. This ensures a discharge path for the RHR pumps.
- Permissives sent to ADS as RHR pump pressure is sensed > 100 psig. Both pumps in either loop are required to satisfy the ADS permissive, or one core spray loop.
- Minimum flow valve opens if injection flow in loop is < 1000 gpm decreasing after a 10 sec time delay. It automatically shuts as injection valves open and injection flow raises to > 3000 gpm increasing.
- Reactor pressure decreases through the break and/or with actuation of ADS.
- As reactor pressure decreases to 410 psig, the LPCI injection valves F015A(B) auto open. The outboard injection valve F017A(B) can be throttled 5 minutes after the RPV pressure is below 410 psig.
- As pressure reaches 310 psig, recirculation pump discharge and discharge bypass valves shut and are interlocked shut in the attempt to re-flood the core.
- As pressure reaches 200 psig, the RHR system injects into both recirculation system loops by lifting the check valves and overcoming reactor pressure.

5. 205000 1

RHR Loop 2A is operating in the Shutdown Cooling mode of operation with the following parameters:

RHRSW Pump 2A	Operating
RHRSW Flow	4000 gpm
RHR Pump 2A	Operating
RHR Loop A Flow	6000 gpm

Which one of the following completes the statement below?

The required operator action to **lower** the cooldown rate IAW 2OP-17, *Residual Heat Removal System Operating Procedure*, is to throttle **closed**:

A. 2-E11-F003A, HX 2A Outlet Vlv.

B. 2-E11-F017A, Outboard Injection Vlv.

C. 2-E11-F048A, HX 2A Bypass Vlv.

D. 2-E11-PDV-F068A, HX 2A SW Disch Vlv.

Answer: D

K/A:

205000 Shutdown Cooling System (RHR Shutdown Cooling Mode)

K5 Knowledge of the operational implications of the following concepts as they apply to SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE): (CFR: 41.5 / 45.3)

03 Heat removal mechanisms

RO/SRO Rating: 2.8/3.1

Tier 2 / Group 1

K/A Match: This meets the K/A because the student has to know which valve would need to be operated to control the heat removal for SDC.

Pedigree: New

Objective: LOI-CLS-LP-017, Obj. 15
Describe how the reactor cool down rate is controlled when the RHR system is in the Shutdown Cooling mode

Reference: None

Cog Level: High

Explanation: The procedure allows throttling closed the F003, F017 or F068 or throttling open the F048. Throttling open the F048 will bypass some of the RHR flow around the heat exchanger thereby lowering cooldown rate. RHR flow is limited to greater than 6000 gpm, so closing the F017 or F003 is not an option.

Distractor Analysis:

Choice A: Plausible because if the valve was throttled closed this would lower cooldown, but flow must be greater than 6000 gpm.,

Choice B: Plausible because if the valve was throttled closed this would lower cooldown, but flow must be greater than 6000 gpm..

Choice C: Plausible because if the valve was throttled open this would be correct.

Choice D: Correct Answer, see explanation

SRO Basis: N/A

5. IF less cooling is desired,
THEN perform the following, in order of preference:............................ ________

CAUTION
When E11-F003A(B) [Hx 2A(2B) Outlet Vlv] is CLOSED, RHR Heat Exchanger 2A(2B) inlet temperature, located on E41-TR-R605 (HPCI Turb Brg Oil Temp Recorder), Point 1(2), is **NOT** a valid indication of reactor coolant temperature. .... ☐

a. **Throttle close** E11-F003A(B) [Hx 2A(2B) Outlet Vlv] not lower than 6000 gpm, as necessary.. ________

b. **Reduce** RHRSW loop flow by performing the following as necessary:

- **Throttle closed** E11-PDV-F068A(B) [Hx 2A(2B) SW Disch Vlv] to reduce RHRSW flow rate............................ ________
- IF both RHRSW loop pumps are running,
THEN secure one pump per 2OP-43, Service Water System Operating Procedure .... ________

c. **Bypass** a portion of RHR loop flow around the HX as follows:

(1) **Station** an operator at E11-F048A(B) [Hx 2A(2B) Bypass Vlv] to monitor for severe vibration/cavitation during throttling evolutions.. ________

(2) **Adjust** E11-F003A(B) [Hx 2A(2B) Outlet Vlv] as required to achieve 6500 gpm RHR loop flow .... ________

(3) **Throttle close** E11-F017A(B) (Outboard Injection Vlv) as required to achieve 6000 gpm RHR loop flow .... ________

(4) **Throttle open** E11-F048A(B) [Hx 2A(2B) Bypass Vlv] as necessary .... ________

6. 206000 1

A Group 1 isolation has occurred on Unit One.

HPCI has been placed in the pressure control mode of operation IAW 1OP-19, *High Pressure Coolant Injection System Operating Procedure.*

HPCI flow controller, E41-FIC-R600, is in manual with the output at midscale.

Which one of the following completes the statement below?

If the 1-E41-F008, Bypass To CST Valve, is throttled ____(1)____ too far, this may result in HPCI ____(2)____.

A. (1) open
   (2) tripping on overspeed

B. (1) open
   (2) operation below 2100 rpm

C. (1) closed
   (2) tripping on overspeed

D. (1) closed
   (2) operation below 2100 rpm

Answer: B

K/A:

206000 High Pressure Coolant Injection System

K1 Knowledge of the physical connections and/or cause-effect relationships between HIGH PRESSURE COOLANT INJECTION SYSTEM and the following:
(CFR: 41.2 to 41.9 / 45.7 to 45.8)

10 Condensate storage and transfer system

RO/SRO Rating: 2.8/3.1

Tier 2 / Group 1

K/A Match: This meets the K/A because this is testing the cause-effect relationship between HPCI and the flowpath to the CST.

Pedigree: Bank

Objective: LOI-CLS-LP-019. Obj. 8
Describe the methods available for controlling RPV pressure and/or RPV cooldown when operating the HPCI System in the Pressure Control mode. (LOCT)

Reference: None

Cog Level: high

Explanation: Opening F008 will increase flow, causing turbine speed control to lower turbine speed to maintain desired flow. Opening valve too far can result in RPM below 2100 (OP-19, Section 8.2). Closing F008 will cause turbine speed to increase, but the governor limits turbine speed to a maximum value (4100 RPM) below the overspeed trip.

Distractor Analysis:

Choice A: Plausible because opened is correct and an overspeed condition may be thought correct if the flowpath is considered incorrectly.

Choice B: Correct Answer, see explanation.

Choice C: Plausible because throttling closed will increase the speed of the turbine.

Choice D: Plausible because may be thought correct if the flowpath is considered incorrectly.

SRO Basis: N/A

From OP-19:

CAUTION

Throttling *E41-F008* open may cause turbine speed reduction to less than 2100 rpm, if opened too far.

From the SD:

Operation of the HPCI Turbine below the minimum rated speed of 2100 rpm may result in a failure of the auxiliary oil pump from repeated startup cycles. A loss of the auxiliary oil pump will prevent starting of the HPCI Turbine.

7. 209001 1

Unit Two is operating at rated power.

Due to a circuit malfunction an inadvertent LOCA initiation occurs in the Div II Core Spray logic causing A-03 (2-6), *CORE SPRAY SYSTEM II ACTUATED*, to alarm.

Which one of the following completes both statements below?

Core Spray Pump(s) ___(1)___ will start.

___(2)___ will start.

A. (1) 2B ONLY
   (2) All DGs

B. (1) 2B ONLY
   (2) DG2 and DG4 ONLY

C. (1) 2A and 2B
   (2) All DGs

D. (1) 2A and 2B
   (2) DG2 and DG4 ONLY

Answer: A

K/A:

209001 Low Pressure Core Spray System

K3 Knowledge of the effect that a loss or malfunction of the LOW PRESSURE CORE SPRAY SYSTEM will have on following: (CFR: 41.7 / 45.4)

03 Emergency generators

RO/SRO Rating: 2.9/3.0

Tier 2 / Group 1

K/A Match: This meets the K/A because it is testing the knowledge of a malfunction of the CS logic has on the EDG.

Pedigree: New

Objective: LOI-CLS-LP-018, Obj. 14
List three systems, other than the Core Spray System, which are initiated or isolated by the Core Spray System logic.

Reference: None

Cog Level: High

Explanation: For CS the logic will only start that divisions pump (RHR would start the other divisions pump) for the CS logic to the DGs either divisions signal will start all DGs.

Distractor Analysis:

Choice A: Correct Answer, see explanation.

Choice B: Plausible because the first part is correct and since it is divisional for the pump starts the student may think that it would start only the Div II DGs. There are signals that would start divisional DGs.

Choice C: Plausible because the student may think the CS logic is similar to the RHR logic for pump starts and the second part is correct.

Choice D: Plausible because the student may think the CS logic is similar to the RHR logic for pump starts and since it is a Div II logic the student may think that it would start only the Div II DGs. There are signals that would start divisional DGs.

SRO Basis: N/A

Unit 2
APP A-03 2-6
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CORE SPRAY SYS II ACTUATED

AUTO ACTIONS

1. If E bus was not deenergized, Core Spray Pump 2B starts 15 seconds after receipt of initiation signal
2. If E bus was deenergized, Core Spray pump 2B starts 15 seconds after diesel generator ties onto E bus
3. If open, Full Flow Test Byp Vlv, E21-F015B, closes
4. If closed, Outboard Injection Vlv, E21-F004B, opens
5. When reactor pressure drops to 410 psig, Inboard Injection Vlv, E21-F005B, opens
6. When loop flow is greater than 1500 gpm, Min Flow Bypass Vlv, E21-F031B, closes
7. Div II Non-Intrpt RNA, RNA-SV-5261, and Div I Non-Intrpt RNA, RNA-SV-5262, close
8. Div II Backup N2 Rack Isol Vlv, RNA-SV-5461, and Div I Backup N2 Rack Isol Vlv, RNA-SV-5462, open
9. Fans for Drywell Coolers B and C trip
10. All diesel generators start
11. Nuclear Service Water To Vital Header Valve, SW-V117, opens
12. RBCCW HX Service Water Inlet Valve, SW-V103, closes

CAUSE

1. Reactor low level three (45 inches)
2. High drywell pressure (1.7 psig) in conjunction with low reactor pressure (410 psig)
3. Circuit malfunction

8. 211000 1

Which one of the following completes the statement below concerning Core Spray Line Break Detection differential pressure instrument?

The ____(1)____ leg of this DP instrument senses ____(2)____ core plate pressure via the SLC/Core Differential Pressure penetration.

A. (1) variable
   (2) below

B. (1) variable
   (2) above

C. (1) reference
   (2) below

D. (1) reference
   (2) above

Answer: D

K/A:

211000 Standby Liquid Control System

K1 Knowledge of the physical connections and/or cause-effect relationships between STANDBY LIQUID CONTROL SYSTEM and the following: (CFR: 41.2 to 41.9 / 45.7 to 45.8)

01 Core spray line break detection

RO/SRO Rating: 3.0/3.3

Tier 2 / Group 1

K/A Match: This meets the K/A because it is testing the physical connection of SBLC and CS line break detection.

Pedigree: Last used on 10-1 NRC Exam

Objective: CLS-LP-18, Obj. 10
Explain the principle of operation of the CS Line Break Detection Instrumentation

Reference: None

Cog Level: fundamental

Explanation: This system is comprised of a differential pressure detector which provides Control Room annunciation on detected high DP. The high pressure reference leg of this instrument is exposed to above core plate pressure via the SLC/Core Differential Pressure penetration. The low pressure of this instrument is normally exposed to above core pressure via the Core Spray injection line. This results in the instrument normally measuring core DP (not including core plate DP).

Distractor Analysis:

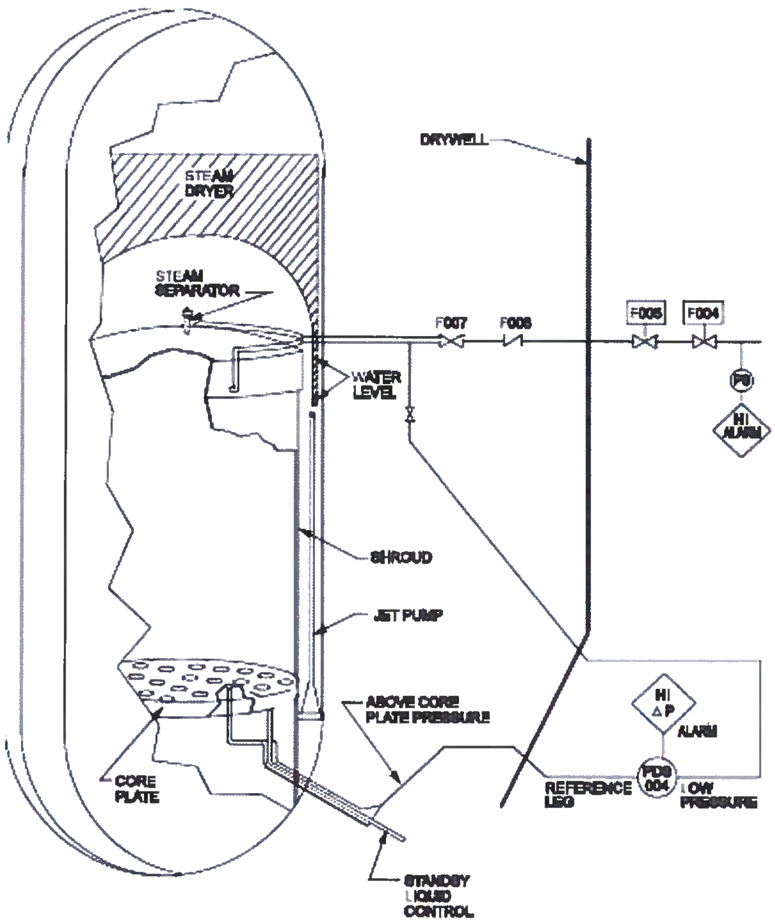
Choice A: Plausible because the examinee may confuse the reference and variable legs and SLC does discharge below the core plate

Choice B: Plausible because the examinee may confuse the reference and variable legs

Choice C: Plausible because it is the reference leg and SLC does discharge below the core plate.

Choice D: Correct Answer, see explanation

SRO Basis: N/A



This system is comprised of a differential pressure detector which provides Control Room annunciation on detected high ΔP. The high pressure reference leg of this instrument is exposed to above core plate pressure via the SLC/Core Differential Pressure penetration. The low pressure of this instrument is normally exposed to above core pressure via the Core Spray injection line. This results in the instrument normally measuring core ΔP (not including core plate ΔP).

A break in the Core Spray injection line between the reactor vessel penetration and the core shroud would expose the low pressure side of the instrument to the lower pressure of the region outside the shroud. This would be sensed as an <u>increased</u> differential pressure and Control Room annunciator would alert the Operator. Although other indications would be available, this alarm would also indicate a break in the line between the E21-F006B(A) check valve and the reactor vessel penetration.

The Core Spray pipe break detection instruments are located on the Reactor Building 20' elevation.

9. 212000 1
Which one of the following completes both statements below?

The normal power supply to RPS MG Set 2B is from 480V MCC ___(1)___.

The **normal** alternate power supply to RPS B is from 480V Bus ___(2)___.

A. (1) 2CA
(2) E7

B. (1) 2CA
(2) E8

C. (1) 2CB
(2) E7

D. (1) 2CB
(2) E8

Answer: C
K/A:
212000 Reactor Protection System
K2 Knowledge of electrical power supplies to the following: (CFR: 41.7)
01 RPS motor-generator sets

RO/SRO Rating: 3.2/3.3

Tier 2 / Group 1

K/A Match: This meets the K/A because it is testing the power supply to the RPS MG Set

Pedigree: New

Objective: CLS-LP-03, Obj 18b
State the power supplies for the following: RPS MG Set B

Reference: None

Cog Level: Fundamental

Explanation: Power for the Unit 2B Motor Generator Sets is tapped off two phases of the normal 480 VAC MCC 2CB power supply for the motor through a stepdown transformer (480V to 120V) from E8 (the 2A MG Set is powered from 2CA). Normal alternate power to the RPS Bus is provided from E7 with Alternate alternate power to the RPS Bus provided from E8. In the event that either RPS M-G Set fails to operate, the alternate power sources must be manually selected.

Distractor Analysis:

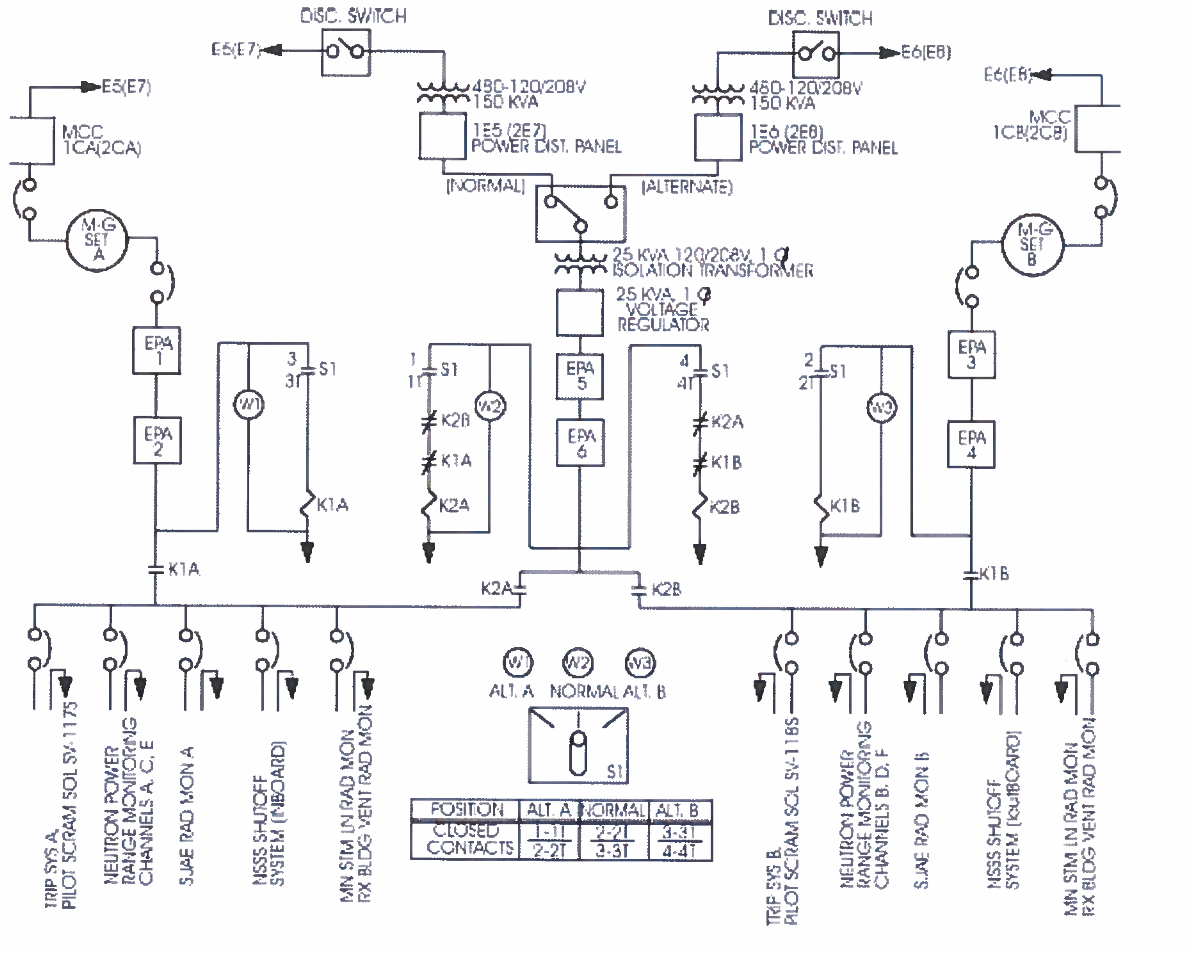
Choice A: Plausible because 2CA supplies RPS MG Set A and E7 is the normal alternate power supply.

Choice B: Plausible because 2CA supplies RPS MG Set A and E8 is the alternate alternate power supply.

Choice C: Correct Answer, see explanation.

Choice D: Plausible because 2CB is the RPS MG Set B power supply and E8 is the alternate alternate power supply.

SRO Basis: N/A



POSITION	ALT. A	NORMAL	ALT. B
CLOSED CONTACTS	1-1T / 2-2T	2-2T / 3-3T	3-3T / 4-4T

10. 215002 1

Which one of the following identifies the LPRM detector level that provides input to the Rod Block Monitor system for indication ONLY, and is NOT used for the purpose of generating rod blocks?

A. Level A

B. Level B

C. Level C

D. Level D

Answer: A

K/A:

215002 Rod Block Monitor System

K1 Knowledge of the physical connections and/or cause-effect relationships between ROD BLOCK MONITOR SYSTEM and the following: (CFR: 41.2 to 41.9 / 45.7 to 45.8)

02 LPRM

RO/SRO Rating: 3.2/3.1

Tier 2 / Group 2

K/A Match: This meets the K/A because it is testing the connection between RBM and LPRMs

Pedigree: Bank

Objective: LOI-CLS-LP-09.6, Obj 5a
List the PRNMS system signals/conditions that will cause the following actions: APRM / RBM Rod Blocks

Reference: None

Cog Level: Fundamental

Explanation: The level A inputs are sent to RBM-A for processing/output to the LPRM Display Meters on the 4-Rod Display. Level A is for indication only

RBM-A Receives
all four level C inputs
lower left and upper right level B inputs
upper left and lower right level D inputs

RBM-B Receives
all four level C inputs
upper left and lower right level B inputs
lower left and upper right level D inputs

Distractor Analysis:

Choice A: Correct Answer, see explanation.

Choice B: Plausible because LPRMs have a B level that input to the RBMs Rod Blocks.

Choice C: Plausible because LPRMs have a C level that input to the RBMs Rod Blocks.

Choice D: Plausible because LPRMs have a D level that input to the RBMs Rod Blocks.

SRO Basis: N/A

The "A" level LPRM detectors are not used for RBM input processing, while both RBM channels use all "C" level detectors. This gives an accurate representation of actual power around the control rod. The "B" and "D" detectors are distributed evenly between the two RBM channels. An example of LPRM input to a both RBM channels with a four-string rod selected is two "B" level LPRMs, four "C" level LPRMs, and two "D" level LPRMs for each channel.

The RBM circuitry undergoes a nulling and filtering sequence when a rod is selected and therefore a delay of at least 2.5 seconds must be allowed between selection and rod movement. A Rod Inhibit signal is

11. 215003 1

Unit One is performing a startup with the reactor just declared critical.
While ranging IRM G from range 1, the IRM will not change ranges and remains on Range 1.

Which one of the following completes both statements below?

When IRM G indication **first** exceeds ___(1)___ on the 125 scale, annunciator A-05, 2-4, *IRM UPSCALE*, will alarm.

The action required IAW A-05, 2-4, *IRM UPSCALE*, is to ___(2)___.

A. (1) 70
   (2) place the joystick on P603 for the IRM G to Bypass

B. (1) 70
   (2) withdraw the IRM G detector to maintain reading on scale

C. (1) 117
   (2) place the joystick on P603 for the IRM G to Bypass

D. (1) 117
   (2) withdraw the IRM G detector to maintain reading on scale

Answer: A

K/A:
215003 Intermediate Range Monitor System

A2 Ability to (a) predict the impacts of the following on the INTERMEDIATE RANGE MONITOR (IRM) SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5 / 45.6)

06 Faulty range switch

RO/SRO Rating: 3.0/3.2

Tier 2 / Group 1

K/A Match: THIS QUESTION WAS PRE-SUBMITTED FOR APPROVAL.
This meets the K/A because it is testing what will happen with a faulty range switch and the action required.

Pedigree: New

Objective: LOI-CLS-LP-009.1, Obj. 3a
List the SRM/IRM system signals/conditions that will cause the following actions and the conditions under which each is bypassed: Rod Blocks (LOCT)
LOI-CLS-LP-009.1, Obj. 14a
Given plant conditions and one of the following events, use plant procedures to determine the actions required to control and/or mitigate the consequences of the event: SRM/IRM Upscale alarm (LOCT)

Reference: None

Cog Level: High

Explanation: With the reactor critical the indication will continue to rise. The Upscale alarm will come in at 70 on the 0-125 scale. The Upscale Hi/Inop alarm comes in at 117 on the 0-125 scale. IAW with the APP the action to take is to bypass the IRM. In the case of the SRMs an action to take could be to withdraw the SRM to maintain on scale readings.

Distractor Analysis:

Choice A: Correct Answer, see explanation.

Choice B: Plausible because the first part is correct and the second part could be correct if this was an SRM.

Choice C: Plausible because 117 is an alarm setpoint for the IRMs and the second part is correct.

Choice D: Plausible because 117 is an alarm setpoint for the IRMs and the second part could be correct if this was an SRM.

SRO Basis: N/A

IRM UPSCALE

AUTO ACTIONS

1. Rod withdrawal block (bypassed when Reactor Mode Switch in RUN)

CAUSE

1. IRM channel indicates greater than or equal to 70 on 0-125 scale
2. Improper ranging of IRM channels during reactor startup or shutdown
3. IRM detector failure
4. During refuel outages, IRM spiking due to noise generation from work activities in drywell, such as welding
5. Circuit malfunction

OBSERVATIONS

1. IRM channel indicating greater than or equal to 70 on 0-125 scale
2. IRM channel upscale (UPSC ALARM) amber indicating light on
3. ROD OUT BLOCK (A-05 2-2) alarms
4. Rod withdrawal permissive indicating light off

ACTIONS

1. If in progress, stop withdrawal of control rods.
2. Monitor IRM indications to determine affected channel(s).

CAUTION

IRM range switches should be repositioned carefully in order to prevent a reactor scram.

3. Reposition affected IRM range switch to next higher range.
4. If a sudden rise in indicated reactor power occurred on more than one IRM channel, verify correct rod withdrawal sequence is being used and insert in-sequence control rods as necessary to turn power rise.
5. If IRM detector failure or circuit malfunction is suspected, perform the following:
   a. Refer to Technical Specification 3.3.1.1 and TRM 3.3 for IRM channel operability requirements.
   b. Notify Unit CRS.
   c. Bypass affected channel using IRM bypass switch.
   d. Ensure a WR is prepared.

12. 215004 1

Which one of the following identifies the criteria for when SRM detectors can **first** begin to be withdrawn from the core IAW 0GP-02, *Approach To Criticality And Pressurization Of The Reactor*?

A. When all IRMs are above range 3.

B. When SRM counts reach 2 x 10^5 counts.

C. When RTRCT PERMIT light is illuminated.

D. When SRM/IRM overlap has been established.

Answer: D

K/A:

215004 Source Range Monitor System

K5 Knowledge of the operational implications of the following concepts as they apply to SOURCE RANGE MONITOR (SRM) SYSTEM: (CFR: 41.5 / 45.3)

03 Changing detector position

RO/SRO Rating: 2.8/2.8

Tier 2 / Group 1

K/A Match: This meets the K/A because it is testing when SRMs can change detector positions.

Pedigree: New

Objective: LOI-CLS-LP-009-A, Obj 3b
List the SRM/IRM system signals/conditions that will cause the following actions and the conditions under which each is bypassed: Retract Permissive (SRM) only (LOCT)

Reference: None

Cog Level: Fundamental

Explanation: When SRM/IRM overlap has been established then SRM can be withdrawn to maintain an indicated SRM count rate between 100 cps and 200,000 cps.

Distractor Analysis:

Choice A: Plausible because this is the logic setpoint at which the SRM can be fully withdrawn

Choice B: Plausible because this is the point at which the SRM must be fully withdrawn.

Choice C: Plausible because this is an indication that is used during the withdrawal of the SRMs

Choice D: Correct Answer, see explanation.

SRO Basis: N/A

NOTE

- SRM/IRM overlap is required to be demonstrated for all operable IRM channels prior to withdrawing SRMs from the fully inserted position. SRM/IRM overlap exists when IRM channels show an increase to at least twice their pre-startup levels and indicate at least 10% of scale (i.e., 12.5 on the digital readout 0-125 scale) before the first SRM channel reaches 5×10^5 cps (Technical Specifications, SR 3.3.1.1.6).. ☐
- If desired, the level of the highest reading IRM (pre-startup) may be doubled and that value used as overlap criteria for all IRMs. This method will allow the operator to compare IRM channel response to a single value which is at least twice the pre-startup levels of the individual IRMs.. ☐

6.2 Pulling Rods To Achieve Criticality (continued)

NOTE

- With IRM channels below Range 3, the SRM channels will initiate a rod withdrawal block when either of the following conditions exists:
  - ◊ SRM channel indicates greater than 2×10^5 cps .... ☐
  - ◊ SRM channel indicates less than 10^2 cps with its detector **NOT** full in .... ☐
- SRM detectors are withdrawn two at a time so that the reactor flux level conditions are being monitored by channels that are **NOT** being affected by detector movement.. ☐

32. **WHEN** SRM/IRM overlap has been confirmed,
    THEN withdraw SRM detectors as required to maintain an indicated SRM count rate between 10^2 cps and 2×10^5 cps................. ________

CAUTION

Repositioning IRM range switches is performed by one operator, using one hand, on one trip system at a time {8.1.6}.. ☐

33. As reactor power rises, **reposition** the IRM range switches to maintain IRM indication on recorders between 15 and 50 on the 0-125 scale. .... ________

34. **WHEN** all OPERABLE IRM channels are above Range 3 **AND** prior to reaching Range 7,
    THEN fully **withdraw** all SRM detectors .... ________

13. 215005 1

Which one of the following identifies the power supply to the APRM channel NUMACs?

A. All APRM channels receive 120 VAC power from UPS

B. All APRM channels receive 120 VAC power from both RPS Bus A and RPS Bus B

C. APRM Channels 1 & 3 receive power from ONLY120 VAC RPS Bus A
APRM Channels 2 & 4 receive power from ONLY120 VAC RPS Bus B

D. APRM Channels 1 & 3 receive power from Division I 24/48 VDC
APRM Channels 2 & 4 receive power from Division II 24/48 VDC

Answer: B

K/A:
215005 Average Power Range Monitor/Local Power Range Monitor
K2 Knowledge of electrical power supplies to the following: (CFR: 41.7)
02 APRM channels

RO/SRO Rating: 2.6/2.8

Tier 2 / Group 1

K/A Match: This meets the K/A because it is testing the power supply to the NUMACs.

Pedigree: Modified from 2015 NRC Exam

Objective: LOI-CLS-LP-09.6, Objective 7a
Describe the operational relationships between the PRNMS and the following:
Reactor Protection System

Reference: None

Cog Level: Fundamental

Explanation: Each APRM channel NUMAC is equipped with a dual power supply arrangement with one supply from RPS Bus A and the other supply from RPS Bus B. All four APRM channels maintain power on loss of either supply as long as the other supply is available

Distractor Analysis:

Choice A: Plausible because UPS supplies power to the APRM ODA and recorder

Choice B: Correct Answer, see explanation.

Choice C: Plausible because this is the power supply arrangement for the voters.

Choice D: Plausible because other ranges of nuclear instrumentation (SRM/IRM) receive their power from here.

SRO Basis: N/A

2015 Exam Question:

Which one of the following is the power supply to APRM Channel 4 NUMAC on P608?

A✓ 120 VAC RPS

B. 120 VAC UPS

C. 24/48 VDC Div I

D. 24/48 VDC Div II

2.8.8 PRNMS Power Supplies

The Power Range Neutron monitoring System uses one Quadruple Voltage Power Supply (QLVPS) chassis and four Dual Low Voltage Power Supplies (DLVPS), one for each bay of the PRNMS panel, to provide redundant power to the NUMAC instruments. These LVPS convert 120 VAC to low voltage DC. See Figure 09.6-14.

Each APRM instrument receives power from two power supplies, LVPS 1 and LVPS 4. LVPS 1 is fed from RPS Bus A while LVPS 4 is fed from RPS Bus B. Therefore, a loss of an RPS Bus will not affect operation of the APRM NUMACS. Each RBM instrument also

4.3.1 Reactor Protection System

APRM channels provide signals to open contacts in the scram trip logic of the RPS System under various conditions discussed previously.

The RPS System provides power to each of the four APRM instruments, which in turn provide power to all subsystems driven from the APRM instruments or NUMAC. Both RPS busses, A and B, provide power to each APRM instrument, as well as, each RBM. Therefore, a loss of one RPS bus will not affect operation of the PRNMS.

The reactor mode switch provides input to each APRM instrument to determine when to enforce the fixed or flow biased scram trip and rod block settings. OPRM circuitry is enabled only when power/flow conditions are met and the mode switch in RUN.

14. 215005 2
Which one of the following completes the statement below?

An APRM must have at least ___(1)____ of the assigned LPRMs operable with at least ___(2)____ LPRM inputs per axial level operable.

A. (1) 18
(2) 2

B. (1) 18
(2) 3

C. (1) 17
(2) 2

D. (1) 17
(2) 3

Answer: D
K/A:
215005 Average Power Range Monitor/Local Power Range Monitor
K5 Knowledge of the operational implications of the following concepts as they apply to AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM: (CFR: 41.5 / 45.3)
04 LPRM detector location and core symmetry

RO/SRO Rating: 2.9/3.2

Tier 2 / Group 1

K/A Match: This meets the K/A because it is testing the knowledge of the LPRM inputs per axial level that are required and the minimum number of inputs for core symmetry that are required.

Pedigree: New

Objective: LOI-CLS-LP-09.6, Obj. 13b
Given plant conditions, predict the effect of a single or multiple LPRM failure on the following: APRM

Reference: None

Cog Level: Fundamental

Explanation: An APRM channel must have a minimum of 3 LPRM inputs per level and a total of 17 LPRM inputs to be operable

Distractor Analysis:

Choice A: Plausible because an OPRM requires 18 LPRMs with at least 2 LPRM inputs to each cell.

Choice B: Plausible because an OPRM requires 18 LPRMs and 3 per level is correct for APRMS.

Choice C: Plausible because 17 is correct for APRMs and OPRMs require at least 2 LPRM inputs to each cell.

Choice D: Correct Answer, see explanation.

SRO Basis: N/A

4.2.1 LPRM

LPRM System failure, depending on the extent or failure type, can cause the loss of LPRM functions including the loss of indication, incorrect operation of rod block or scram protection. Generally, the following symptoms are exhibited for LPRM failure for the affected LPRM:

- Indicates upscale, accompanied by an upscale alarm.
- Indicates downscale, accompanied by a downscale alarm.
- Indicator reads erratically.

The results of an LPRM failure may lead to an APRM or OPRM becoming inoperable. An APRM channel must have a minimum of 3 LPRM inputs per level and a total of 17 LPRM inputs to be operable.

An OPRM cell must have a minimum of 2 LPRM inputs to each cell and a total of 18 cells to be operable.

15. 217000 1

Following a loss of feedwater, RCIC automatically initiated and subsequently tripped on low suction pressure.

Current plant status is:
Reactor water level is 150 inches
RCIC flow controller in Manual set at 200 gpm

Subsequently, the following actions are taken:
RCIC suction transferred to Torus
E51-V8, Turbine Trip and Throttle Valve is closed
E51-V8 is re-opened
PF push button on the RCIC flow controller is depressed

Which one of the following identifies the indicated flow on the RCIC flow controller that would be observed for these conditions?

A. 0 gpm

B. 200 gpm

C. 400 gpm

D. 500 gpm

changed per exam Feedback

Answer: ~~C~~ A

K/A:

217000 Reactor Core Isolation Cooling System

A1 Ability to predict and/or monitor changes in parameters associated with operating the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) controls including: (CFR: 41.5 / 45.5)

01 RCIC flow

RO/SRO Rating: 3.7/3.7

Tier 2 / Group 1

K/A Match: This meets the K/A because it is testing the prediction of what RCIC flow will be when operating the RCIC system.

Pedigree: New

Objective: CLS-LP-016-A, Obj.16c
Describe how the following evolutions are performed during operation of the RCIC System: Adjusting RCIC flow in the Reactor Level Control mode.

Reference: None

Cog Level: high

Explanation: The RCIC Turbine is provided with a solenoid operated remote electrical tripping device, which when actuated (in this case by low suction pressure), will close the Turbine Trip and Throttle Valve, E51-V8. Resetting of the remote electrical tripping device may be accomplished from the RTGB. The RCIC system is restarted after auto initiation and turbine trip by fully closing the V-8, and re-opening the V-8. Located on the controller face is a PF (programmable function) pushbutton which when depressed an automatic transfer from manual to automatic at a predetermined setpoint of 400 GPM will result. This button (PF) has no function if the controller is already in automatic.

Distractor Analysis:

Choice A: This is plausible because this answer would be correct for these actions following a high RPV water level trip of RCIC

Choice B: Plausible because this would be correct if the operator did not depress the PF pushbutton.

Choice C: Correct Answer, see explanation.

Choice D: Plausible because the PF push button would raise RCIC flow to rated (400 gpm) and not maximum per procedure (500 gpm). Achieving 500 gpm would require the flow control setpoint to be manually raised.

SRO Basis: N/A

From SD-16:

Also located on the controller face is a PF (programmable function) pushbutton. When depressed an automatic transfer from MANUAL to AUTOMATIC at a predetermined setpoint of 400 GPM will result. NOTE: This button (PF) has no function if the controller is already in AUTOMATIC.

For various internal processing failures, the controller is designed to hold the last output and automatically switch to MANUAL giving the operator manual control capability. Barring operator intervention, this failure could result in rising or lowering RCIC flow and would be indicated by the red FAIL lamp on the controller face. Failure display code can then be checked using the side panel keypad. A down scale failure of the controller is possible and would result in turbine operation at well below the normal minimum speed of 2000 rpm. An upscale failure is highly unlikely but would result in turbine speed at or above the maximum running speed of 4600 rpm. Failures associated with the dynamic response are also highly unlikely but would produce either excessively sluggish responses or dynamic instability (full scale oscillations) when in the Automatic mode. Programmable settings internal to the controller are maintained during a loss of 24 Vdc power supply by a lithium battery. If this battery voltage drops to a pre-determined low value, the yellow ALARM light will flash. If the input signals are not within the limits of -6.3% to 106.3% or if the input or output signals are not intact, the Yellow ALARM light will come on solid.

<< RCIC Instructional Aid for EOPs >>

RESTARTING RCIC AFTER AUTO INITIATION AND TURBINE TRIP

(2OP-16 Section 8.7)

1. **ENSURE** THE E51-V8 (VALVE POSITION) **AND** E51-V8 (MOTOR OPERATOR) ARE CLOSED. .... ☐
2. **PLACE** RCIC FLOW CONTROL IN MANUAL (M) **AND ADJUST** OUTPUT TO 0%.. .... ☐
3. **JOG OPEN** E51-V8 UNTIL THE TURBINE SPEED IS CONTROLLED BY THE GOVERNOR............ ☐
4. **FULLY OPEN** E51-V8. .... ☐
5. **SLOWLY RAISE** TURBINE SPEED UNTIL FLOW RATE OF AT LEAST 120 GPM. .... ☐
6. **ENSURE** E51-F019 IS CLOSED WITH FLOW GREATER THAN 80 GPM............ ☐
7. **WHEN** SYSTEM CONDITIONS ARE STABLE, **THEN ADJUST** SETPOINT, **AND TRANSFER** RCIC FLOW CONTROL TO AUTO (A). .... ☐
8. **SLOWLY ADJUST** FLOW RATE USING RCIC FLOW CONTROL IN AUTO (A)............ ☐
9. **ENSURE** THE FOLLOWING:

   BAROMETRIC CNDSR VACUUM PUMP HAS STARTED .... ☐

   SBGT STARTED (2OP-10)............ ☐

   SGT-V8 AND SGT-V9 ARE OPEN............ ☐

16. 218000 1

Which one of the following completes both statements below concerning the Automatic Depressurization System (ADS) reactor water level inputs from the Nuclear Boiler System?

The ___(1)___ instruments provide LL3 inputs to ADS initiation logic.

The ___(2)___ range instruments provide LL1 inputs to ADS logic.

A. (1) Fuel Zone
   (2) Narrow

B. (1) Fuel Zone
   (2) Shutdown

C. (1) Wide range
   (2) Narrow

D. (1) Wide range
   (2) Shutdown

Answer: C

K/A:

217000 Automatic Depressurization System

K1 Knowledge of the physical connections and/or cause-effect relationships between AUTOMATIC DEPRESSURIZATION SYSTEM and the following: (CFR: 41.2 to 41.9 / 45.7 to 45.8)

03 Nuclear boiler instrument system

RO/SRO Rating: 3.7/3.8

Tier 2 / Group 1

K/A Match: This meets the K/A because it is testing the connection between ADS and level indicators.

Pedigree: New

Objective: LOI-CLS-LP-0001.2, Obj 4a
List the systems which receive input from the Vessel Instrumentation system for the following: Level signal

Reference: None

Cog Level: Fundamental

Explanation: B21-LT-N031(Wide Range) provide LL3 initiation from N031A and C for Logic B and from N031B and D for Logic A.
B21-LT-N042 (Narrow Range) provide LL1 confirmatory from N042A for Logic B and from N042B for Logic A.

Distractor Analysis:

Choice A: Plausible because the fuel zone instruments covers LL3 (45 inches) and the second part is correct.

Choice B: Plausible because fuel zone instruments covers LL3 (45 inches) and the shutdown range covers the LL1 setpoint (166 inches).

Choice C: Correct Answer, see explanation.

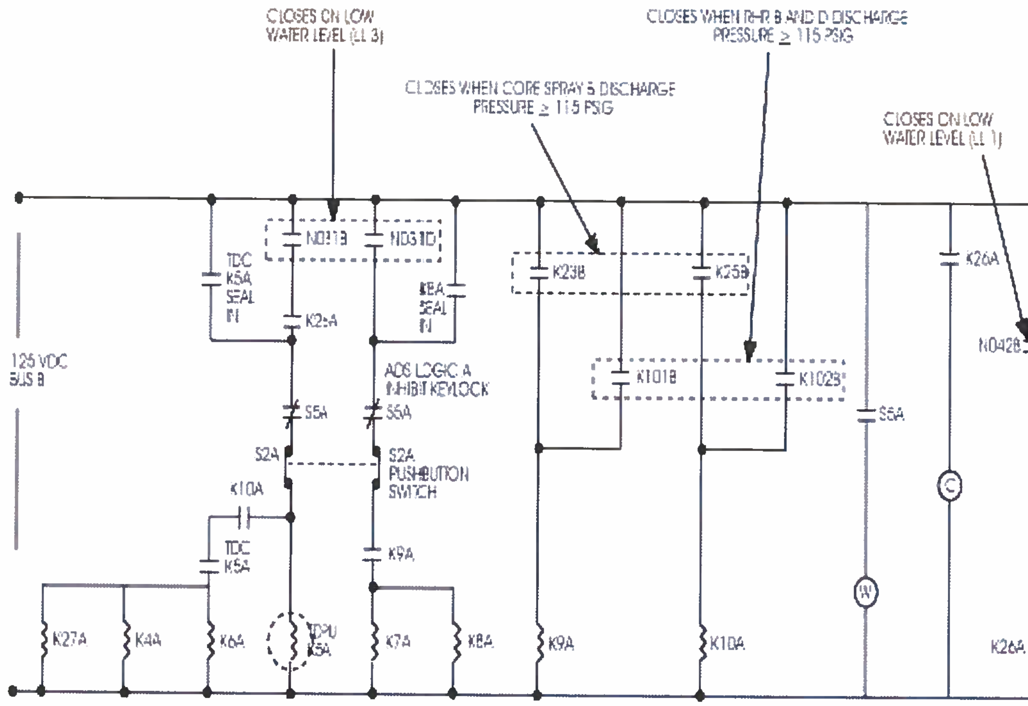
Choice D: Plausible because the wide range is correct and the shutdown range covers the LL1 setpoint (166 inches).

SRO Basis: N/A

4.1.2 Automatic Operation

The ADS logic automatically opens the ADS valves in the event the HPCI System fails to maintain reactor level during a LOCA. The seven ADS valves open automatically when all the following conditions are met on either of two logic channels (A or B) associated with ADS:

- Reactor low water level (LL3 from B21-LTS-N031A and C or B and D).
- Reactor confirmatory low water level (LL1 from B21-LTS-N042A or B).
- Operation of both pumps of an RHR loop or one Core Spray pump as indicated by a pump discharge pressure of 115 psig (either E11-PS-N016A AND C or B AND D or E11-PS-N020A AND C or B AND D for RHR or either E21-PS-N008A AND E11-PS-N009A or E21-PS-N008B AND E21-PS-N009B for CS).
- A time delay of 83 seconds has elapsed (timer B21-TDPU-K5A or B).
- AUTO/INHIBIT switches in AUTO for either or both logic channels A and B.



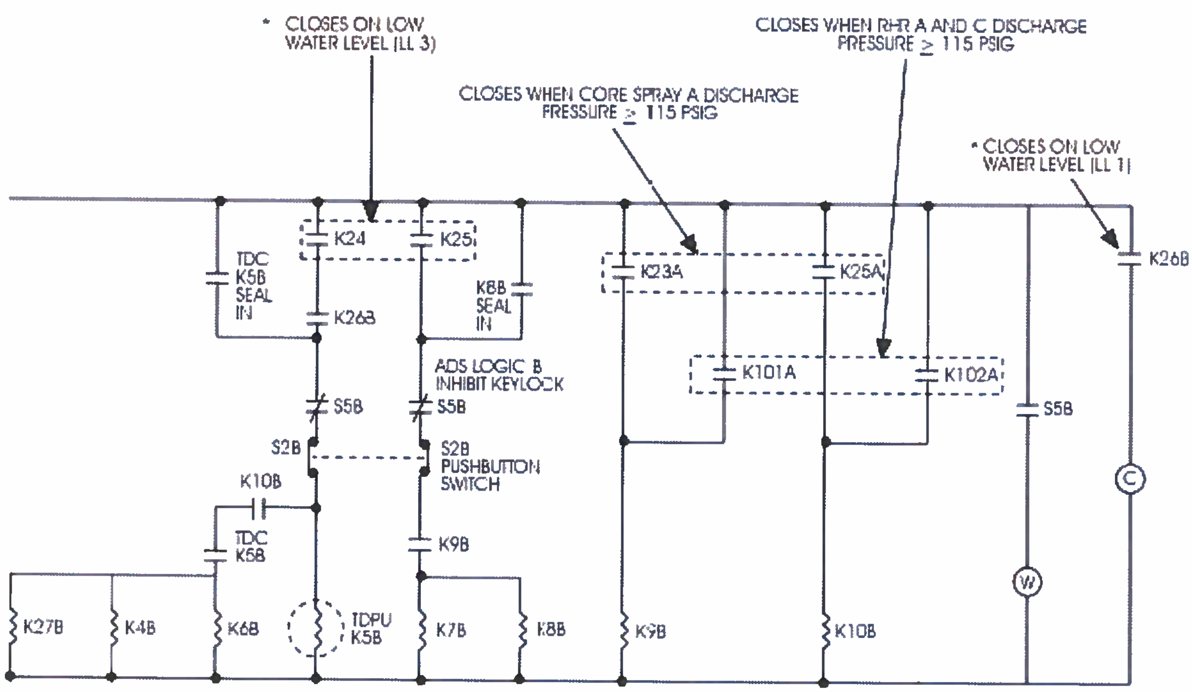


FIGURE 20-10
ADS Logic Channel "A"

17. 218000 2

Unit One is operating at power with Core Spray Pump 1B under clearance.
A small break LOCA occurs simultaneously with a Loss of Off-site Power to both units.

DG1 and DG4 fail to start and tie onto their respective E bus.

The following plant conditions exist on Unit One:

A-03 (5-1) *Auto Depress Timers Initiated*	In alarm
A-03 (6-9) *Reactor Low Wtr Level Initiation*	In alarm
RPV pressure	600 psig
Drywell pressure	13 psig

Which one of the following completes both statements below?

ADS ___(1)___ auto initiate.

After ADS is initiated (either automatically or manually), RPV water level ___(2)___ be restored with **BOTH** RHR Loops.

A. (1) will
(2) will

B. (1) will
(2) will NOT

C. (1) will NOT
(2) will

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

Answer: D

K/A:

218000 Automatic Depressurization System

K3 Knowledge of the effect that a loss or malfunction of the AUTOMATIC DEPRESSURIZATION SYSTEM will have on following: (CFR: 41.7 / 45.4)

01 Restoration of reactor water level after a break that does not depressurize the reactor when required

RO/SRO Rating: 4.4/4.4

Tier 2 / Group 1

K/A Match: This meets the K/A because it is testing the knowledge of the effect of the malfunction on auto initiation of ADS and how level will be restored.

Pedigree: Last used on 10-1 NRC exam

Objective: CLS-LP-20 Obj. 16b
Given plant conditions, predict how the following will be affected by a loss or malfunction of ADS/SRVs: Reactor water level

Reference: None

Cog Level: high

Explanation: With the loss of offsite power and 1B CS pump under clearance this would leave only one pump available in each RHR loop. Therefore ADS logic is lost. Level will continue to lower until the ADS valves are manually opened (emergency depressurization) at which time the running low pressure pumps will be able to add water. Injection would be from the A Loop of RHR as the B Loop injection valves do not have power.

Distractor Analysis:

Choice A: Plausible because ADS does have initiation conditions except that the logic will not have the appropriate pumps lined up for injection. B Loop of RHR does not have power to the injection valves

Choice B: Plausible because ADS does have initiation conditions except that the logic will not have the appropriate pumps lined up for injection. B Loop of RHR does not have power to the injection valves

Choice C: Plausible because ADS will not auto initiate but the B Loop of RHR does not have power to the injection valves. B Loop of RHR does not have power to the injection valves

Choice D: Correct Answer, see explanation.

SRO Basis: N/A

SD-20

4.1.2 Automatic Operation

The ADS logic automatically opens the ADS valves in the event the HPCI System fails to maintain reactor level during a LOCA. The seven ADS valves open automatically when all the following conditions are met on either of two logic channels (A or B) associated with ADS:

- Reactor confirmatory low water level (LL1 from B21-LTS-N042A or B).
- Operation of both pumps of an RHR loop or one Core Spray pump as indicated by a pump discharge pressure of 115 psig (either E11-PS-N016A AND C or B AND D or E11-PS-N020A AND C or B AND D for RHR or either E21-PS-N008A AND E11-PS-N009A or E21-PS-N008B AND E21-PS-N009B for CS).
- A time delay of 83 seconds has elapsed (timer B21-TDPU-K5A or B).
- AUTO/INHIBIT switches in AUTO for either or both logic channels A and B. Reactor low water level (LL3 from B21-LTS-N031A and C or B and D).

18. 223001 1
Which one of the following completes the statement below concerning the Fuel Zone instruments, N036 and N037, during a loss of drywell cooling?

The reference leg density will ___(1)___ causing the indicated level to read ___(2)___ than actual level.

A. (1) rise
(2) higher

B. (1) rise
(2) lower

C. (1) lower
(2) higher

D. (1) lower
(2) lower

Answer: C
K/A:
223001 Primary Containment System and Auxiliaries
K3 Knowledge of the effect that a loss or malfunction of the PRIMARY CONTAINMENT SYSTEM AND AUXILIARIES will have on following: (CFR: 41.7 / 45.4)
09 Nuclear boiler instrumentation
RO/SRO Rating: 2.8/3.1
Tier 2 / Group 2

K/A Match: This meets the K/A because it is testing the knowledge of a loss of DW cooling has on instrumentation.

Pedigree: Bank

Objective: LOI-CLS-LP-001.2, Obj. 05c
Explain the effect that the following will have on reactor vessel level and/or pressure indications: High containment (primary and secondary) temperatures.

Reference: None

Cog Level: High

Explanation: The reference leg length is longer than the variable leg length, therefore secondary temp increasing makes the instrument read higher than actual level.

Distractor Analysis:

Choice A: Plausible because density is a function of temperature and the temperature is rising. The second part is correct.

Choice B: Plausible because density is a function of temperature and the temperature is rising. The second part is plausible because if the first part is seen as right then this would be correct.

Choice C: Correct Answer, see explanation.

Choice D: Plausible because the first part is correct and the second part is the opposite of the answer.

SRO Basis: N/A

19. 223002 1
Unit One is at 75% power.
The 1A RPS MG set trips.
No operator actions have been taken.

Which one of the following identifies the Main Steam Line Isolation Valve (MSIV) logic lamp status on P601 panel?

	Inboard MSIV Logic DC	Inboard MSIV Logic AC	Outboard MSIV Logic DC	Outboard MSIV Logic AC
A.	LIT	OFF	LIT	OFF
B.	OFF	LIT	OFF	LIT
C.	LIT	OFF	OFF	LIT
D.	OFF	LIT	LIT	OFF

Answer: C
K/A:
223002 Primary Containment Isolation System/Nuclear Steam Supply Shut-Off
A1 Ability to predict and/or monitor changes in parameters associated with operating the PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF controls including: (CFR: 41.5 / 45.5)
01 System indicating lights and alarms

RO/SRO Rating: 3.5/3.5

Tier 2 / Group 1

K/A Match: This meets the K/A because it is testing the ability to predict the light status on a loss of a power supply

Pedigree: New

Objective: LOI-CLS-LP-012, Objective 12
Given plant conditions, determine how the following will affect PCIS:
c. Loss of RPS

Reference: None

Cog Level: High

Explanation: See Notes Section. RPS A provides power to PCIS Logic A. PCIS Logic A is Inboard AC and Outboard DC indicating lights on P601.

Distractor Analysis:

Choice A: Plausible because first part is correct. Outboard light is DC.

Choice B: Plausible because second part is correct. Inboard light is AC.

Choice C: Correct Answer, see explanation.

Choice D: Plausible because the lights are just reversed. This would be true for Loss of RPS B.

SRO Basis: N/A

4.3.10 AC Distribution

RPS MG sets supply power to the following PCIS related components:

RPS Bus A

PCIS Trip System A logic
PCIS Trip Channels A1 and A2 logic
Inboard isolation logic for valves:
- Inboard reactor water sample valve
- Main Steam Line drains
- Shutdown cooling suction
- RWCU
- Inboard RHR Sample valves
- Drywell floor and equipment drains
- CAC/CAMS/PASS for LL1 and High Drywell pressure

Valve operating power:
- Inboard reactor water sample valve
- Inboard RHR Sample valves
- Drywell floor and equipment drains
- Inboard "AC" MSIV solenoids
- Reactor Building Vent Exh Rad Monitor N010A
- Main Steam Line Rad Monitors A and C (alarm function only)

P601 panel. These lights are arranged above the MSIV control switches as follows:

TABLE 25-3, MSIV ISOLATION SIGNAL STATUS

Light	INBD DC	INBD AC	OUTBD DC	OUTBD AC
Solenoid Power	125 VDC "A"	RPS "A"	125 VDC "B"	RPS "B"
PCIS Logic	B	A	A	B

20. 234000 1

Which one of the following identifies the effect if both Refuel Bridge hoist grapple hooks are not open five seconds **after** placing the Engage/Release switch to Release?

A. Fuel Hoist Interlock is generated.

B. Engage amber light extinguishes.

C. Fault lockout is generated.

D. Grapple hooks will reclose.

Answer: D

K/A:

234000 Fuel Handling

A3 Ability to monitor automatic operations of the FUEL HANDLING EQUIPMENT including: (CFR: 41.7 / 45.7)

01 Crane/refuel bridge movement

RO/SRO Rating: 2.6/3.1

Tier 2 / Group 2

K/A Match: This meets the K/A because it is testing the ability to monitor the crane grapple hooks auto re-close feature.

Pedigree: New

Objective: LOI-CLS-LP-58.1, Obj 13
Describe the operation of the grapple if the ENGAGE/RELEASE Switch is positioned to RELEASE and both grapple hooks are not open within 5 seconds when the main hoist is loaded.

Reference: None

Cog Level: Fundamental

Explanation: If the grapple does not indicate released (open) within 5 seconds, the solenoid is de-energized and the grapple hooks re-close. The switch must then be taken to the ENGAGE position to reset the logic prior to making another attempt to release the grapple.

Distractor Analysis:

Choice A: Plausible because a Fuel Hoist Interlock is generated for a number of reasons.

Choice B: Plausible because this is an indication of operation of the grapple hooks.

Choice C: Plausible because a fault lockout is generated for a number of reasons.

Choice D: Correct Answer, see explanation.

SRO Basis: N/A

21. 239002 1

Which one of the following identifies the SRV component that will prevent siphoning of water into the SRV discharge piping?

A. Vacuum breaker

B. Check Valve

C. T-Quencher

D. Sparger

Answer: A

K/A:

239002 Safety Relief Valves

K4 Knowledge of RELIEF/SAFETY VALVES design feature(s) and/or interlocks which provide for the following: (CFR: 41.7)

03 Prevents siphoning of water into SRV discharge piping and limits loads on subsequent actuation of SRV's

RO/SRO Rating: 3.1/3.3

Tier 2 / Group 1

K/A Match: THIS QUESTION WAS PRE-SUBMITTED FOR APPROVAL.
This meets the K/A because it is testing the knowledge of the design feature that prevents siphoning of water.

Pedigree: New

Objective: LOI-CLS-LP-020, Obj. 7d
State the purpose of the following: SRV tailpipe vacuum breakers

Reference: None

Cog Level: Fundamental

Explanation: Following operation of the valve, a vacuum is created in the SRV tailpipe as the steam condenses. Water in the line above the suppression pool water level would cause excessive pressure at the SRVs discharge when and if the valve reopened. For this reason, a vacuum relief valve is provided on each SRV tailpipe to prevent drawing water up into the line due to this steam condensation following SRV operation.

Distractor Analysis:

Choice A: Correct Answer, see explanation.

Choice B: Plausible because this is a component that is typically used to provide an anti-siphon break.

Choice C: Plausible because this is a component on the SRV that the steam discharges through and has holes in the pipe which could be thought of an anti-siphon type break.

Choice D: Plausible because this is a component on the SRV that the steam discharges through and has holes throughout the pipe which could be thought of an anti-siphon type break. (The supplemental fuel pool cooling sparger has this design to prevent siphoning of water)

SRO Basis: N/A

22. 241000 1

Which one of the following identifies the criteria for tripping the main turbine IAW the Unit Two Scram Immediate Actions of 0EOP-01-UG, *Users Guide*?

A. When APRM's indicate downscale trip.

B. When steam flow is less than 3 Mlbs/hr.

C. When reactor water level is 160 inches and rising.

D. When reactor mode switch is placed in SHUTDOWN.

Answer: A

K/A:

241000 Reactor/Turbine Pressure Regulating System

A4 Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8)

14 Turbine trip

RO/SRO Rating: 3.8/3.7

Tier 2 / Group 2

K/A Match: This meets the K/A because it is testing the ability of tripping the turbine from the control room.

Pedigree: New

Objective: LOI-CLS-LP-300-C, Obj. 2
List the immediate operator actions for a reactor scram.

Reference: None

Cog Level: Fundamental

Explanation: The main turbine is tripped after reactor power is below 2% which is indicated by APRM downscale trip lights illuminated.

Distractor Analysis:

Choice A: Correct Answer, see explanation.

Choice B: Plausible because this is a criteria for placing the mode switch to shutdown which is an immediate operator action.

Choice C: Plausible because this is a criteria for a reactor feed pump which is an immediate operator action.

Choice D: Plausible because this is an immediate operator action that is performed on the scram.

SRO Basis: N/A

Unit 2 Scram Immediate Actions (0EOP-01-UG)

SCRAM IMMEDIATE ACTIONS

1. Ensure SCRAM valves OPEN by manual SCRAM or ARI initiation.

2. **WHEN** steam flow less than 3 x 10^6 lb/hr,
   THEN place reactor mode switch in SHUTDOWN.

3. **IF** reactor power below 2% (APRM downscale trip),
   THEN trip main turbine.

4. Ensure master RPV level controller setpoint at +170 inches.

5. **IF:**

   - Two reactor feed pumps running

     AND

   - RPV level above +160 inches

     AND

   - RPV level rising,

   THEN trip one.

23. 245000 1

Which one of the following completes both statements below concerning the Main Generator Voltage Regulator?

The automatic voltage regulator maintains a constant generator ___(1)___ voltage.

While in the automatic voltage regulation mode, the manual voltage regulator setting ___(2)___ automatically follow the automatic setpoint.

A. (1) field
(2) does

B. (1) field
(2) does NOT

C. (1) terminal
(2) does

D. (1) terminal
(2) does NOT

Answer: D

K/A:

245000 Main Turbine Generator and Auxiliary Systems

K4 Knowledge of MAIN TURBINE GENERATOR AND AUXILIARY SYSTEMS design feature(s) and/or interlocks which provide for the following: (CFR: 41.7)

07 Generator voltage regulation

RO/SRO Rating: 2.5/2.6

Tier 2 / Group 2

K/A Match: This meets the K/A because this is testing the design of the auto regulator as to what it controls and whether the manual regulator automatically follows the auto regulator.

Pedigree: Bank

Objective: LOI-CLS-LP-027.0, Obj 7c

Given a simplified diagram of the Main Generator Voltage Regulator, explain how:

a. the MANUAL regulator controls Generator output voltage

b. the AUTOMATIC regulator controls Generator output voltage

c. to transfer from one Voltage Regulator to the other

Reference: None

Cog Level: Fundamental

Explanation: The AVR controls terminal voltage while the manual regulator controls field voltage. The manual voltage regulator does not track the setpoint of the AVR, this must be manually adjusted in the control room.

Distractor Analysis:

Choice A: Plausible because the MVR controls field voltage and the DG manual voltage regulator does track the auto regulator setpoint.

Choice B: Plausible because the MVR controls field voltage and the second part is correct.

Choice C: Plausible because the first part is correct and the DG manual voltage regulator does track the auto regulator setpoint.

Choice D: Correct Answer, see explanation.

SRO Basis: N/A

2.15 Excitation Control (Refer to Figure 27-12)

The Silicon Controlled Rectifier (SCR) bridge circuit is used as a variable DC voltage source to control the exciter field current as required by the AC or DC regulator. The source of the control signal for the SCRs is determined by the Regulator Mode Selector Switch (43CS) located on Panel XU-1. When Manual is selected, the DC regulator maintains a constant generator field voltage that is determined by the Manual Volts Adjust Rheostat. When the Automatic regulator is selected, the AC regulator maintains a constant generator terminal voltage.

2.17.8 Generator Voltage Regulator Differential Voltmeter

This is a standard voltmeter that measures the magnitude and polarity of the difference between the DC regulator output signal and the AC regulator output signal. When shifting control from the DC voltage regulator to the AC regulator or back, it is important to ensure that the signals are the same. As an example, if the meter reads to the clockwise of zero, then the manual regulator output is less than the automatic regulator. If the meter reads counter clockwise of zero, then the manual signal is larger than the automatic signal. The meter indicates 0-10 volts in both directions.

Failure to have the regulator control signals matched when shifting regulator modes may result in transients on the generator output. The severity of the transient would be determined by the direction and magnitude of the mismatch.

24. 259001 1

Unit One Reactor Feed Pump 1B is operating in automatic DFCS control at 4500 RPM. The DFCS control signal to Reactor Feed Pump 1B woodward governor **immediately** fails downscale.

Which one of the following completes the statement below?

Reactor Feed Pump 1B speed will:

A. lower to 0 rpm.

B. lower to 1000 rpm.

C. lower to 2450 rpm.

D. remain at 4500 rpm.

Answer: D

K/A:

259001 Reactor Feedwater System

A1 Ability to predict and/or monitor changes in parameters associated with operating the REACTOR FEEDWATER SYSTEM controls including: (CFR: 41.5 / 45.5)

04 RFP turbine speed: Turbine-Driven-Only

RO/SRO Rating: 2.8/2.7

Tier 2 / Group 2

K/A Match: This meets the K/A because it is testing the ability to predict the response in parameters.

Pedigree: New

Objective: LOI-CLS-LP-032.2, Obj. 13d
Given plant conditions and one of the following events, use plant procedures to determine the actions required to control and/or mitigate the consequences of the event:
Loss of signal interface between controllers and processor.

Reference: None

Cog Level: High

Explanation: If RFPT A(B) *MAN/DFCS* selector switch is in *DFCS*, and DFCS control signal subsequently drops below 2450 rpm, or increases to greater than 5450 rpm, then Woodward 5009 digital controls will automatically assume RFPT speed control and maintain current pump speed.

Distractor Analysis:

Choice A: Plausible if the student believes that a loss of input signal will cause the controller to use 0 as the input for the speed of the pump. (i.e. HPCI/RCIC controllers will fail to zero)

Choice B: Plausible because an idled RFP is maintained at 1000 rpm.

Choice C: Plausible because 2450 is the low end of the controller function.

Choice D: Correct Answer, see explanation.

SRO Basis: N/A

NOTE: If RFPT A(B) *MAN/DFCS* selector switch is in *DFCS*, and DFCS control signal subsequently drops below 2450 rpm, or increases to greater than 5450 rpm, then Woodward 5009 digital controls will automatically assume RFPT speed control and maintain current pump speed. In this condition, the RFPT will only respond to *LOWER/RAISE* speed control switch commands until *MAN/DFCS* selector switch is placed in *MAN*, DFCS *CTRL RESET* pushbutton is depressed, and *MAN/DFCS* selector switch returned to *DFCS*.

3.13 Plant management has recommended one RFPT be idled at 1000 rpm with the discharge valve closed, during conditions with one RFPT in service.

25. 259002 1
Which one of the following completes both statements below concerning the reactor feed pump turbine (RFPT) DFCS controls?

During a RFPT startup, transfer to DFCS control is performed when RFPT speed is approximately ___(1)___.

DFCS will automatically control the speed of the RFPT up to ___(2)___.

A. (1) 1000 rpm
(2) 5450 rpm

B. (1) 1000 rpm
(2) 6150 rpm

C. (1) 2550 rpm
(2) 5450 rpm

D. (1) 2550 rpm
(2) 6150 rpm

Answer: C
K/A:
259002 Reactor Water Level Control System
A3 Ability to monitor automatic operations of the REACTOR WATER LEVEL CONTROL SYSTEM including: (CFR: 41.7 / 45.7)
01 Runout flow control

RO/SRO Rating: 3.0/3.0

Tier 2 / Group 1

K/A Match: This meets the K/A because this is testing the upper limit of the auto (DFCS) controls which in essence prevent pump runout of the reactor feed pumps.

Pedigree: new

Objective: LOI-CLS-LP-032.2, Obj. 5d
Describe the operation of the DFCS in the following operating modes:
Master Level Control Mode (auto and manual)

Reference: None

Cog Level: Fundamental

Explanation: DFCS will be placed into service with the manual output set at 2550 RPM. The DFCS system will control the RFPT speed from 2450 - 5450 RPMs

Distractor Analysis:

Choice A: Plausible because 1000 RPM is the idle speed of the RFPT and the second part is correct.

Choice B: Plausible because 1000 RPM is the idle speed of the RFPT and 6150 is the overspeed setpoint of the woodward controls.

Choice C: Correct Answer, see explanation.

Choice D: Plausible because the first part is correct and 6150 is the overspeed setpoint of the woodward controls.

SRO Basis: N/A

6.1.5 Reactor Feed Pump Startup from Idle Speed to Injection at Low Pressure Conditions (continued)

NOTE

When using RFPT A(B) Lower/Raise speed control switch, reactor feed pump turbine speed will change at a rate of 50 rpm per second. If switch is held in LOWER or RAISE for greater than 3 seconds, the rate of change will rise to 375 rpm per second. .... ☐

6. **Maintain** RFPT A(B) discharge pressure at least 100 psig greater than reactor pressure by adjusting RFPT A(B) Lower/Raise speed control switch <u>until</u> RFPT speed is approximately 2550 rpm. .... ______

END R.M. LEVEL R3 REACTIVITY EVOLUTION

7. **Direct** Radwaste Operator to monitor effluent conductivity for each in service CDD. .... ______

8. **<u>WHEN</u>** RFPT A(B) speed is approximately 2550 rpm,
   <u>THEN</u> raise C32-SIC-R601A(B) [RFPT A(B) Sp Ctl] output to match DFCS Stpt and Speed Stpt on Panel P603 to within 100 rpm. .... ______

NOTE

- When RFPT A(B) Man/DFCS control switch is placed in DFCS, C32-SIC-R601A(B) [RFPT A(B) Sp Ctl] will control RFPT speed. .... ☐
- When RFPT A(B) Man/DFCS control switch is placed in DFCS, and DFCS is in control, the RFPT A(B) DFCS Ctrl light will be ON. .... ☐
- If RFPT A(B) Man/DFCS selector switch is in DFCS and DFCS control signal subsequently drops to less than 2450 rpm or rises to greater than 5450 rpm, Woodward 5009 digital controls will automatically assume RFPT speed control and maintain current pump speed. In this condition, the RFPT will only respond to Lower/Raise speed control switch commands <u>until</u> the Man/DFCS selector switch is placed in MAN, DFCS Ctrl Reset pushbutton is depressed, and the Man/DFCS selector switch returned to DFCS. .... ☐

9. **Confirm** the following RFPT A(B) speed signals on Panel P603 agree within 100 rpm:
   - DFCS Stpt (speed demand from DFCS) .... ______
   - Speed Stpt (speed demand from 5009 control) .... ______
   - Act Spd (actual RFPT speed) .... ______

10. **Place** Man/DFCS control switch in DFCS. .... ______

3.0 PRECAUTIONS AND LIMITATIONS (continued)

5. Any of the following conditions will automatically trip a reactor feed pump turbine:

   - RFPT Woodward 5009 overspeed greater than or equal to 6150 rpm .... ☐

26. 261000 1

Unit One primary containment venting is being performed IAW 1OP-10, *Standby Gas Treatment System Operating System,* with the following plant status:

1-VA-1F-BFV-RB, SBGT DW Suct Damper	Open
1-VA-1D-BFV-RB, Reactor Building SBGT Train 1A Inlet Valve	Closed
1-VA-1H-BFV-RB, Reactor Building SBGT Train 1B Inlet Valve	Closed

Which one of the following completes both statements below concerning the predicted SBGT response if drywell pressure rises to 1.9 psig?

1-VA-1F-BFV-RB ____(1)____.

Both 1-VA-1D-BFV-RB and 1-VA-1H-BFV-RB ____(2)____.

A. (1) auto closes
   (2) auto open

B. (1) auto closes
   (2) remain closed

C. (1) remains open
   (2) auto open

D. (1) remains open
   (2) remain closed

Answer: A

K/A:
261000 Standby Gas Treatment System
A4 Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8)
02 Suction valves

RO/SRO Rating: 3.1/3.1

Tier 2 / Group 1

K/A Match: This meets the K/A because it is testing the ability to monitor SBGT suction valves.

Pedigree: Last used on 2014 NRC Exam

Objective: LOI-CLS-LP-004.1, Obj 5
List the signals and setpoints that will cause a Secondary Containment isolation

Reference: None

Cog Level: High

Explanation: The filter train fans will automatically start on High Drywell Pressure. The following actions occur: 1) SBGT Reactor Building suction dampers (1D-BFV-RB and 1H-BFV-RB) open, 2) SBGT DW Suct Damper (F-BFV-RB) closes. The SBGT Train A/B Suction & Discharge Valves on U1 do not auto open. These valves on U2 do auto open, so there could be a misconception on these valves (inlet vs. suction dampers).

Distractor Analysis:

Choice A: Correct Answer, see explanation

Choice B: Plausible because 1F does auto close and SBGT Train 1A/B Suction Valves (1C & 1E) on Unit One only do not auto open

Choice C: Incorrect since SBGT will auto realign from primary containment to the Reactor Building on system initiation

Choice D: Incorrect since SBGT will auto realign from primary containment to the Reactor Building on system initiation and SBGT Train 1A/B Suction Valves (1C & 1E) on Unit One only do not auto open

SRO Basis: N/A

2.1.6 Fan

A 100% capacity, heavy-duty, industrial type Fan and motor assembly is provided in each SBGT filter train. Each Fan will produce the required 2700 - 3300 scfm flow through its associated filter train.

Each Fan is driven by a direct-drive AC motor which is energized from a redundant and separate emergency power supply. The Unit 1 A and B Fans are powered from 480 VAC MCCs 1XE and 1XF respectively and Unit 2 A and B Fans from 2XE and 2XF.

The filter train fans may be operated manually from controls located at RTGB XU-51.

The filter train fans will automatically start if any of the following Secondary Containment isolation conditions exist: (Figure 10-2)

1. Low Reactor Water Level, LL #2
2. High Drywell Pressure
3. Reactor Building Ventilation Radiation (Figure 10-3)

3.2.6 Automatic

1. Upon receipt of an automatic initiation signal both trains of SBGT will start.

   Unit 1 ONLY

   The dampers associated with Unit 1 SBGT System will receive automatic open signals when an initiation signal is received EXCEPT for the train inlet and outlet dampers, (BFVs-1B,1C,1E,and 1G). Should these normally open dampers be manually closed locally via their CLOSE/OPEN pushbuttons, **they will NOT automatically reopen and the associated SBGT will not automatically start.**

27. 262001 1

Unit One is operating at rated power.
Unit Two is in MODE 5 performing fuel movements.

Which one of the following completes both statements below IAW **Unit One** Tech Spec 3.8.1, AC Sources - Operating, LCO statement?

The Unit Two SAT ___(1)___ required to be OPERABLE.

___(2)___ Diesel Generators are required to be OPERABLE.

A. (1) is
(2) Two

B. (1) is
(2) Four

C. (1) is NOT
(2) Two

D. (1) is NOT
(2) Four

Answer: B

K/A:
262001 A.C. Electrical Distribution
G2.2.40 Ability to apply Technical Specifications for a system. (CFR: 41.10 / 43.2 / 43.5 / 45.3)

RO/SRO Rating: 3.4/4.7

Tier 2 / Group 1

K/A Match: This meets the K/A because this is testing the items above the line for TS 3.8.1.

Pedigree: New

Objective: LOI-CLS-LP-050, Obj. 16
Given plant conditions, determine whether given plant conditions meet minimum Technical Specifications requirements associated with the 230 KV Electrical Distribution system.

Reference: None

Cog Level: Fundamental

Explanation: Unit One Tech Specs require with Unit One in Mode 1, both SAT's and both UAT's and all four DGs are required to be operable. This would change if Unit One was not in Mode 1, 2, or 3. Unit Two Tech Specs do not require the SAT, it only requires one offsite circuit.

Distractor Analysis:

Choice A: Plausible because the first part is correct and there are only two Unit One DGs but all four are required for the LCO to be met.

Choice B: Correct Answer, see explanation.

Choice C: Plausible because this is the Unit 2 SAT and asking if it is required for Unit 1 TS (it is not required for the Unit Two TS) and whether only the 2 Unit One DGs are required or all four of the DGs.

Choice D: Plausible because this is the Unit 2 SAT and asking if it is required for Unit 1 TS (it is not required for the Unit Two TS) and the second part is correct.

SRO Basis: N/A

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

b. Four diesel generators (DGs); and

c. Two Unit 2 qualified circuits between the offsite transmission network and the onsite Class 1E AC Electrical Power Distribution System.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

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

28. 262002 1
Unit One is operating at rated power.

Subsequently, E1 breaker AU9, Feed to 480V Substation E5, trips.

Which one of the following completes the statement below?

120V UPS Distribution Panel 1A is:

A. de-energized.

B. energized from MCC 1CB.

C. energized from the Standby UPS.

D. energized from 250V DC SWBD A.

Answer: D
K/A:
262002 Uninterruptable Power Supply (A.C. /D.C.)
A3 Ability to monitor automatic operations of the UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.) including: (CFR: 41.7 / 45.7)
01 Transfer from preferred to alternate source

RO/SRO Rating: 2.8/3.1
Tier 2 / Group 1

K/A Match: This meets the K/A because it is testing the ability to monitor the transfer to the alternate power source.

Pedigree: New

Objective: LOI-CLS-LP-052, Obj. 5
Given plant conditions, determine the lineup of the primary UPS, the Standby UPS, and their reserve sources.

Reference: None

Cog Level: High

Explanation: The UPS system is normally aligned such the primary inverter is powering UPS loads. The standby inverter is energized but bypassed with the Manual Bypass switch in Bypass Test. The static transfer switch of the Primary inverter (and also the Standby inverter) is receiving an input from the alternate (hard) source. If the primary power source is lost (in this case the loss of E5 which powers MCC CA) the alternate power source from the 250V batteries will keep the loads energized with no need for the inverter to swap to the hard source.

Distractor Analysis:

Choice A: Plausible because the normal power source is lost.

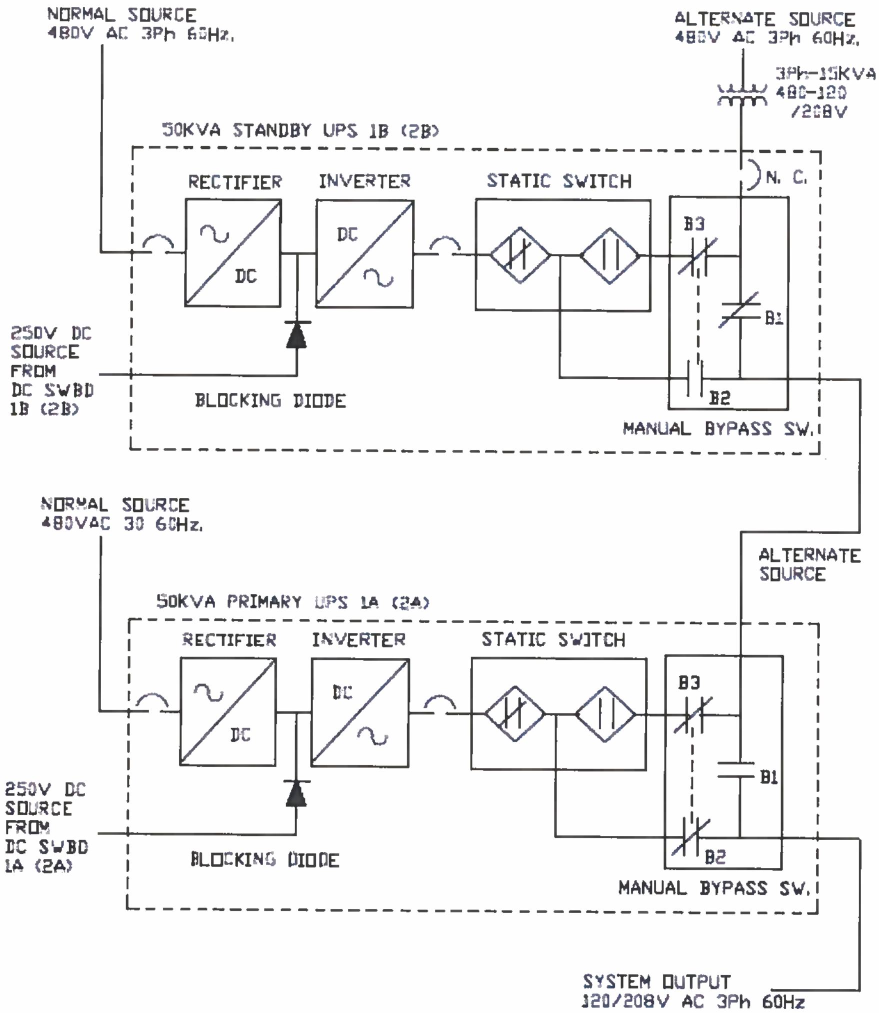
Choice B: Plausible because this is the hard source for the Distribution Panel.

Choice C: Plausible because this is an available power source for the Distribution Panel.

Choice D: Correct Answer, see explanation.

SRO Basis: N/A

FIGURE 52-7
Basic Vital UPS System



29. 215003 2

A reactor shutdown is in progress.
All IRMs on **range 1** reading between 15 and 20.
IRM B detector is failing downscale.

Which one of the following completes both statements below?

IAW A-05 (1-4) *IRM Downscale,* the alarm setpoint is ___(1)___ on the 125 scale.

When the IRM downscale alarm is received, a rod block ___(2)___ be generated.

A. (1) 3
   (2) will

B. (1) 3
   (2) will NOT

C. (1) 6.5
   (2) will

D. (1) 6.5
   (2) will NOT

Answer: D

K/A:
215003 Intermediate Range Monitor (IRM) System
K6 Knowledge of the effect that a loss or malfunction of the following will have on the INTERMEDIATE RANGE MONITOR (IRM) SYSTEM : (CFR: 41.7 / 45.7)
04 Detectors

RO/SRO Rating: 3.0/3.0

Tier 2 / Group 1

K/A Match: This meets the K/A because this is testing a failure/malfunction of a detector effect on the IRM system (whether it generates a rod block)

Pedigree: New

Objective: LOI-CLS-LP-009-A, Obj. 3a
List the SRM/IRM system signals/conditions that will cause the following actions and the conditions under which each is bypassed: Rod Blocks

Reference: None

Cog Level: High

Explanation: The downscale setpoint for the IRMs is 6.5 on the 125 scale. The rod block is bypassed under these conditions because the IRMs are all on Range 1.

Distractor Analysis:

Choice A: Plausible because 3 is the downscale tech spec setpoint for SRMs and if the IRMs were not all on range 1 a rod block would be generated.

Choice B: Plausible because 3 is the downscale tech spec setpoint for SRMs and the second part is correct.

Choice C: Plausible because the first part is correct and if the IRMs were not all on range 1 a rod block would be generated.

Choice D: Correct Answer, see explanation.

SRO Basis: N/A

IRM DOWNSCALE

AUTO ACTIONS

1. Rod withdrawal block (bypassed when IRM range switch for the affected channel is on Range 1 or when the reactor mode switch is in RUN).
2. Computer printout.

CAUSE

1. IRM channel(s) indicating less than or equal to 6.5 on the 0-125 scale when its range switch is not on Range 1.
2. Improper ranging of IRM channels during reactor startup or shutdown.
3. IRM detector not fully inserted.
4. IRM detector failure.
5. Circuit malfunction.

OBSERVATIONS

1. IRM channel indicating less than or equal to 6.5 on the 0-125 scale.
2. IRM downscale (DNSC) white indicating light is on.
3. ROD OUT BLOCK (A-05 2-2) alarm, if affected IRM channel is not on Range 1.
4. If the affected IRM channel(s) is not on Range 1, the rod withdrawal permissive indicating light will be off.

30. 263000 1

Unit Two is operating at full power when a loss of DC Distribution Panel 4A occurs.

Which one of the following completes both statements below?

RCIC is ___(1)___ for injection from the RTGB.

RCIC ___(2)___ isolation logic has lost power.

A. (1) available
   (2) inboard

B. (1) available
   (2) outboard

C. (1) unavailable
   (2) inboard

D. (1) unavailable
   (2) outboard

Answer: A

K/A:

263000 D.C. Electrical Distribution

G2.2.37 Ability to determine operability and/or availability of safety related equipment.
(CFR: 41.7 / 43.5 / 45.12)

RO/SRO Rating: 3.6/4.6

Tier 2 / Group 1

K/A Match: This meets the K/A because it is testing the operability of RCIC/ADS.

Pedigree: New

Objective: LOI-CLS-LP-051, Obj. 7
Given plant conditions, determine the effect that a loss of DC power will have on the following:
d. Reactor Core Isolation Cooling.
e. Automatic Depressurization System.

Reference: None

Cog Level: High

Explanation: RCIC will not shutdown on reactor water level and the inboard isolation logic is powered from Division II 125 VDC panels 4B for Unit 2.

Distractor Analysis:

Choice A: Correct Answer, see explanation

Choice B: Plausible because the first part is correct and a loss of 4B would cause the outboard logic to be lost.

Choice C: Plausible because a loss of 4B would cause RCIC to be unavailable for injection and the second part is correct.

Choice D: Plausible because this would be correct for a loss of 4B.

SRO Basis: N/A

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ATTACHMENT 6
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<< Plant Effects from Loss of DC Distribution Panel 3B(4B) >>

RCIC: Will **NOT** auto initiate, outboard isolation logic INOPERABLE (E51-F008, -F029, and -F066 will **NOT** auto close), RCIC turbine will **NOT** trip except on overspeed, RCIC flow controller and EGM INOPERABLE (no flow control or indication), E51-F045 will **NOT** auto close on high water level, E51-F004, -F054, and -F026 fail closed. RCIC isolation is required in accordance with APP 1(2)-A-03 1-4.

<< Plant Effects from Loss of DC Distribution Panel 3A(4A) >>

RCIC: Will **NOT** shutdown on reactor high water level, inboard isolation logic INOPERABLE (E51-F007, -F031, and -F062 will **NOT** auto close). Valves E51-F005 and -F025 fail closed.

31. 264000 1

Unit Two has lost off-site power.
DG3 started and tied to its respective E Bus.
Sequence of events:

1200 DG3 ties to E3
1205 DG3 lube oil temperature rises above 190°F
1206 DG3 lube oil pressure drops below 27 psig

Which one of the following identifies when DG3 will trip?

A. Immediately at 1205.

B. Immediately at 1206.

C. 45 seconds after 1205.

D. 45 seconds after 1206.

Answer: B

K/A:
264000 Emergency Generators (Diesel/Jet)
K6 Knowledge of the effect that a loss or malfunction of the following will have on the EMERGENCY GENERATORS (DIESEL/JET): (CFR: 41.7 / 45.7)
03 Lube oil pumps

RO/SRO Rating: 3.5/3.7

Tier 2 / Group 1

K/A Match: This meets the K/A because it is testing the effect of a loss of lube oil on the EDG.

Pedigree: Bank

Objective: LOI-CLS-LP-039, Obj. 4a
Given plant conditions, determine if EDGs will trip: After an auto start (LOCT)

Reference: None

Cog Level: High

Explanation: Hi lube oil temperature bypassed by auto start signal (LOOP and LOCA). Low lube oil pressure trip never bypassed. On a start of the DG the low lube oil trip is bypassed for 45 seconds.

Distractor Analysis:

Choice A: Plausible because hi lube oil temperature is a trip, but it is bypassed on the LOOP.

Choice B: Correct Answer, see explanation.

Choice C: Plausible because there is a 45 second time delay associated with the lube oil trip on an initial start of the EDG.

Choice D: Plausible because there is a 45 second time delay associated with the lube oil trip on an initial start of the EDG.

SRO Basis: N/A

3.3.3 Automatic Stop Control (Figure 39-14)

Under conditions where continued Diesel Generator operation may cause damage to the Diesel itself automatic shutdowns are provided. The shutdown signals will vary dependent upon whether the engine has been started manually or automatically.

When operating due to receipt of an automatic start signal the following trips and lockout are provided:

- Low lube oil pressure 27 psig
- Overspeed 575 (561 to 589) rpm

When operating as a result of an initiation from a normal non-emergency start the following trips and lockouts are enforced in addition to those listed above:

- High lube oil temperature 190°F
- High jacket water temperature 200°F
- Jacket Water Low pressure 12 psig

The low lube oil pressure, and low jacket water pressure shutdowns are blocked for the first forty-five second on initiation of an engine start sequence (auto or manual). This permits the conditions to be established which will prevent these shutdowns during engine operation

32. 216000 1

A Unit Two plant cooldown is being performed with the following plant conditions:

Reactor water level	175 inches, steady
Reactor pressure band	500 - 700 psig
Drywell ref leg temp	175°F

(REFERENCE PROVIDED)

Which one of the following completes both statements below?

The lowering of reactor pressure causes the N004A/B/C (Narrow Range) reactor water level instruments indicated level error to ___(1)___.

The reactor water level that would correspond to Low level 4 (LL4) is ___(2)___.

A. (1) increase
   (2) -60 inches

B. (1) increase
   (2) -65 inches

C. (1) decrease
   (2) -60 inches

D. (1) decrease
   (2) -65 inches

Answer: A

K/A:

216000 Nuclear Boiler Instrumentation

A2 Ability to (a) predict the impacts of the following on the NUCLEAR BOILER INSTRUMENTATION; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5 / 45.6)

11 Heatup or cooldown of the reactor vessel

RO/SRO Rating: 3.2/3.3

Tier 2 / Group 2

K/A Match: THIS QUESTION WAS PRE-SUBMITTED FOR APPROVAL.
The first part of the question deals with predicting the effect of a cooldown on indicated level error while the second part has the student has to determine based on the lowering pressure what LL4 value would be which is the value that emergency depressurization would be required. They must utilize the lower end of the pressure band to determine LL4. If LL4 cannot be maintained then ED is required.

Pedigree: New

Objective: LOI-CLS-LP-001.2, Objective 5a
Explain the effect that the following will have on reactor vessel level and/or pressure indications: Plant heatup/cooldown

Reference: 0EOP-01-UG, Attachment 26

Cog Level: High

Explanation: The indicated level error is sensitive to changes in the saturation density of the bulk water as a function of system pressure. The amount of the indicated level error is also a function of the difference in the actual water level and the variable leg instrument tap elevation. As the saturation density increases (pressure decreases) the indicated level error will increase for the narrow and wide range instruments and decrease for the fuel zone and shutdown range instruments due to calibration criteria.
From OI-37.11, TAF, LL4, and LL5 values should be determined based on the reference leg area temperature and RPV pressure compensation curves, using RPV pressure at the low end of the established RPV pressure control band. Based on the low end of the band of 500 psig and < 200°F in the drywell the LL4 value would be -60 inches.

Distractor Analysis:

Choice A: Correct Answer, see explanation

Choice B: Plausible because the first part is correct and the second part would be correct for 700 psig.

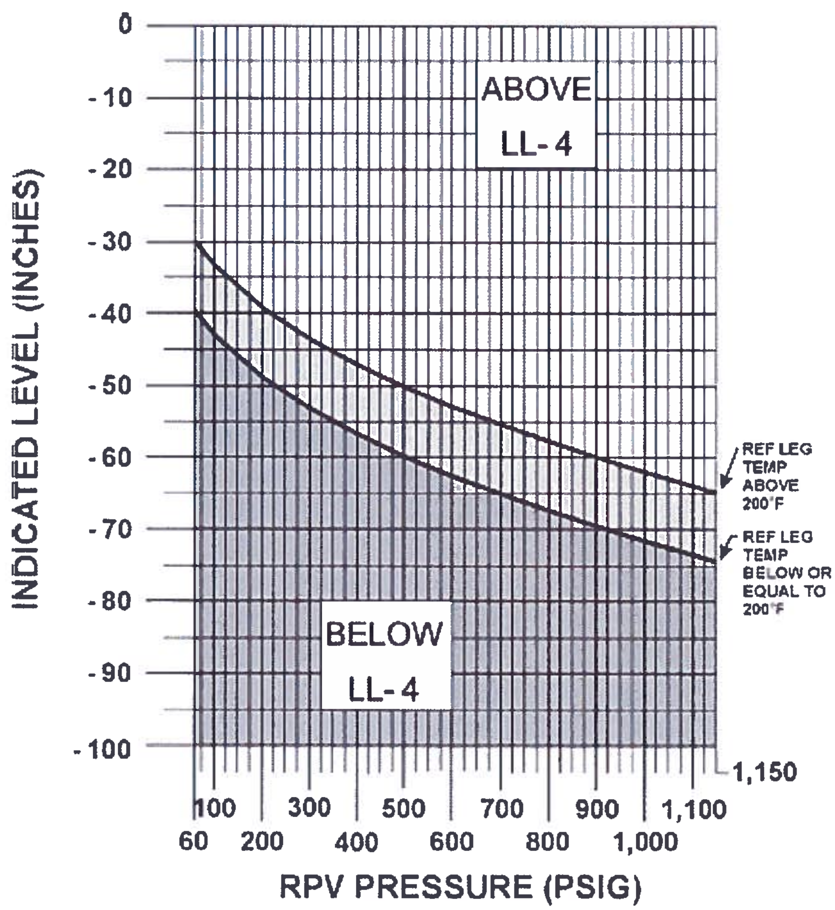
Choice C: Plausible because this would be correct for the fuel zone or shutdown range instruments and the second part is correct.

Choice D: Plausible because this would be correct for the fuel zone or shutdown range instruments and the second part would be correct for 700 psig.

SRO Basis: N/A

ATTACHMENT 26
Page 1 of 1

<< Unit 2 RPV Level at LL 4
(Minimum Steam Cooling RPV Level) >>



When RPV pressure is less than 60 psig,
use indicated level. LL-4 is -27.5 inches.

4.1.2 System Pressure (Heat-up and Cool-down)

The indicated level error is sensitive to changes in the saturation density of the bulk water as a function of system pressure. The amount of the indicated level error is also a function of the difference in the actual water level and the variable leg instrument tap elevation. As the saturation density increases (pressure decreases) the indicated level error will increase for the narrow and wide range instruments and decrease for the fuel zone and shutdown range instruments due to calibration criteria. As actual water level decreases, the amount of error will decrease because less vessel water level is acting on the instrument.

5.3.3 1(2)EOP-01-RVCP, Reactor Vessel Control Procedure

1. Level Leg

   a. The CRS directs an initial RPV level band of +166 to +206 inches. The reactor operator actually maintains a RPV level band of +170 to +200 inches to provide additional margin to the reactor scram and turbine trip set points. The CRS may direct a widened band based on plant conditions and other controlling procedures associated with the transient.

   b. If RPV level is above TAF, injection flow should be controlled so as to control the cooldown rate below 100°F/hr.

   c. If RPV level is below TAF, RPV level should be rapidly restored to above TAF, and then injection flow reduced so as to control the cooldown rate below 100°F/hr.

   d. TAF, LL4, and LL5 values should be determined based on the reference leg area temperature and RPV pressure compensation curves, using RPV pressure at the low end of the established RPV pressure control band.

33. 272000 1

Unit Two is performing a startup IAW 0GP-02, *Approach to Criticality and Pressurization of the Reactor*.

IAW 0GP-02, which one of the following identifies the radiation monitor(s) that will require the alarm setpoints raised when HWC is placed in service?

A. D12-RM-K603A,B,C,D, Main Steam Line Rad Monitors

B. ARM Channel 2-9, U-2 Turbine Bldg Breezeway

C. D12-RR-4599-1,2,3, Main Stack Rad Monitors

D. ARM Channel 2-4, Cond Filter-Demin Aisle

Answer: A

K/A:

272000 Radiation Monitoring System

K5 Knowledge of the operational implications of the following concepts as they apply to RADIATION MONITORING SYSTEM: (CFR: 41.7 / 45.4)

01 Hydrogen injection operation's effect on process radiation indications

RO/SRO Rating: 3.2/3.5

Tier 2 / Group 2

K/A Match: THIS QUESTION WAS PRE-SUBMITTED FOR APPROVAL.
This meets the K/A because this is testing the operational implication as to which rad monitor, if asked as the operational effect on the individual rad monitor this would provide no discrimatory value.

Pedigree: New

Objective: LOI-CLS-LP-059, Obj. 8
Explain why Chemistry must be notified when starting and securing the HWC System.

Reference: None

Cog Level: Fundamental

Explanation: The excess Hydrogen injected into the reactor coolant creates the driving force to shift the Nitrogen-16 distribution ratio, resulting in a larger fraction of the Nitrogen-16 forming volatile Ammonia and a smaller fraction forming Nitrites and Nitrates. This additional volatile Ammonia is then carried over in the reactor steam resulting in higher background radiation levels. Any increase in Hydrogen injection rates will result in a proportional increase in background radiation levels and vise-versa.
0GP-02 has a step for ensuring that the rad monitors are adjusted based on this background rad level increase.

Distractor Analysis:

Choice A: Correct Answer, see explanation.

Choice B: Plausible because when HWC is placed in service the rad levels will increase minimally and HWC H2 is injected in the reactor feed pumps.

Choice C: Plausible because sufficient decay time is available for N-16 such that radiation levels wouldn't raise that much in this area.

Choice D: Plausible because when HWC is placed in service the rad levels will increase minimally and this is downstream of the HWC O2 injection point.

SRO Basis: N/A

3.0 PRECAUTIONS AND LIMITATIONS (continued)

15. B21-F032A and B21-F032B (Feedwater Supply Line Isolation Valves), are stop-check valves. These valves are designed to prevent leakage from the reactor into the feedwater system. These valves are not designed to positively close against condensate system pressure. As such, with the reactor depressurized and the condensate system in service, these valves may leak by, causing reactor water level to rise. ☐

16. The Main Steam Line Radiation Monitor (MSLRM) High-High Radiation setpoint is adjusted assuming HWC is in service. If HWC is removed from service for an extended period of time (greater than one week), 1(2)OP-59, Hydrogen Water Chemistry System Operating Procedure requires BESS determine if a MSLRM High-High Radiation setpoint adjustment is required ☐

17. The HWC System will normally be placed in service immediately after establishing the following conditions:

    - At least one Condensate Booster Pump feeding the reactor with minimum flow valve closed ☐

    - At least one SJAE operating at greater than or equal to half-load ☐

34. 226001 1

Which one of the following identifies the power supply to 2D RHR Pump?

A. E1

B. E2

C. E3

D. E4

Answer: B

K/A:

226001 RHR/LPCI: Containment Spray System Mode

K2 Knowledge of electrical power supplies to the following: (CFR: 41.7)

03 Pumps

RO/SRO Rating: 2.9/2.9

Tier 2 / Group 2

K/A Match: This meets the K/A because this is testing the power supply to RHR pumps which are the pumps for the containment sprays.

Pedigree: Modified from the 2012 NRC Exam (changed to the D RHR Pump)

Objective: LOI-CLS-LP-017-A Obj. 17a
List the normal and emergency power sources for the following: RHR Pumps.

Reference: None

Cog Level: Fundamental

Explanation: 2D RHR pump is a Div II pump with a power supply from E2.

Distractor Analysis:

Choice A: Plausible because E1 is a Unit One bus that supplies power to Unit One and Unit Two loads. RHR Pumps 1C and 2C are supplied from this bus.

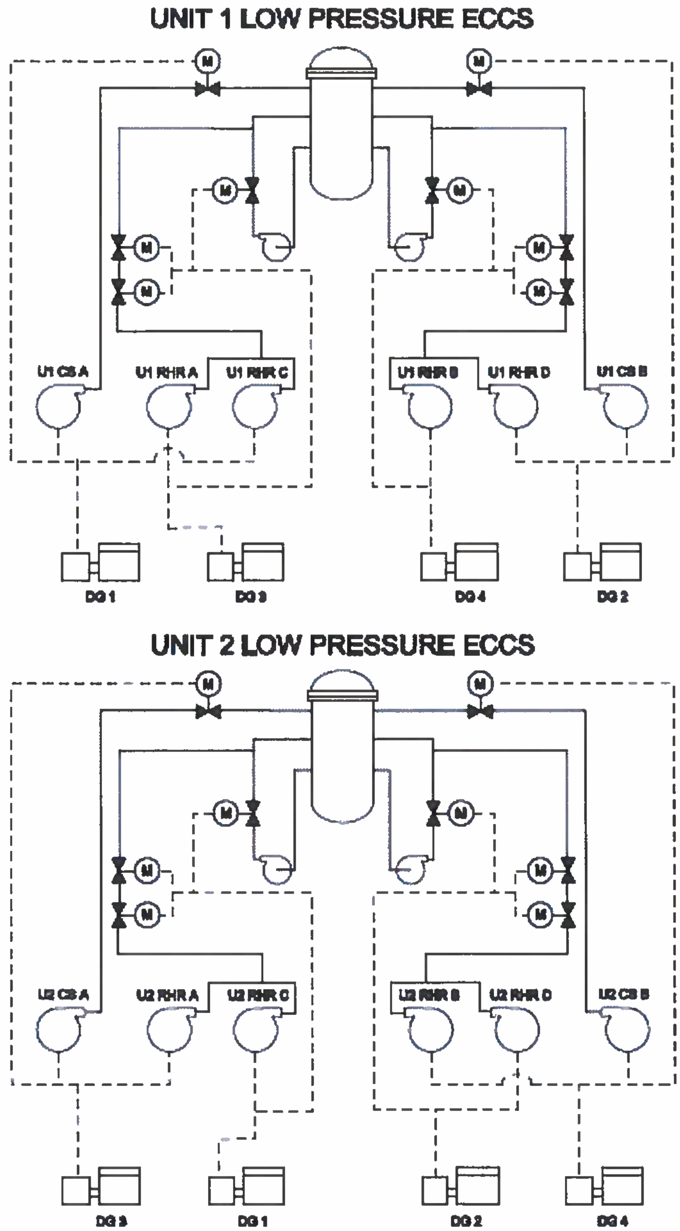
Choice B: Correct Answer, see explanation

Choice C: Plausible because E3 is a Unit Two bus that supplies power to Unit One and Unit Two loads. RHR Pumps 1A and 2A are supplied from this bus.

Choice D: Plausible because E4 is a Unit Two bus that supplies power to Unit One and Unit Two loads. RHR Pumps 1B and 2B are supplied from this bus

SRO Basis: N/A

FIGURE 17-2B
Low Pressure Injection Systems and Power



NOTE: INJECTION FLOW PATH AND POWER SUPPLIES SHOWN.
LOGIC & OTHER FLOW PATHS NOT SHOWN.

35. 295001 1

Unit One is operating at 70% power when the OATC observes indications for a failed jet pump. Subsequently, Recirc Pump 1A trips.

Which one of the following completes both statements below IAW 1AOP-04.0, *Low Core Flow*?

Performance of the jet pump operability surveillance for ___(1)___ Loop Operation is required.

If it is determined that a jet pump has failed, the required action is to ___(2)___.

A. (1) Single
   (2) reduce reactor power below 25% rated thermal power

B. (1) Single
   (2) commence unit shutdown IAW 0GP-05, Unit Shutdown

C. (1) Two
   (2) reduce reactor power below 25% rated thermal power

D. (1) Two
   (2) commence unit shutdown IAW 0GP-05, Unit Shutdown

Answer: B

K/A:
295001 Partial or Complete Loss of Forced Core Flow Circulation
G2.2.12Knowledge of surveillance procedures. (CFR: 41.10 / 45.13)

RO/SRO Rating: 3.7/4.1

Tier 1 / Group 1

K/A Match: This meets the K/A because this is testing knowledge of which surv. is required and the action if it has failed the surv.

Pedigree: New

Objective: LOI-CLS-LP-302-C, Obj 4
Given plant conditions and AOP-04.0, determine the required supplementary actions.

Reference: None

Cog Level: High

Explanation: The indications given are for a failed jet pump which IAW the AOP require the surveillance performed for determination of a failed jet pump. Unlike the selection of the power to flow map the PT only looks at the recirc pumps for determination of single loop or two loop operation. The power to flow maps for single loop are not used until the APRM setpoint adjustments are made.

Distractor Analysis:

Choice A: Plausible because single loop is correct and 25% is the requirement for when the PT is required to be performed.

Choice B: Correct Answer, see explanation.

Choice C: Plausible because APRM setpoint adjustments have not been made which is a determination of how to use the power to flow maps and 25% is the requirement for when the PT is required to be performed.

Choice D: Plausible because APRM setpoint adjustments have not been made which is a determination of how to use the power to flow maps and the second part is correct.

SRO Basis: N/A

4.2 Supplementary Actions (continued)

NOTE

Jet pump failure is indicated by the following: .... ☐

- Reduction in generator megawatt output on GEN-WMR-760 (Net Unit Megawatts)
- Reduction in core thermal power
- Rise in indicated total core flow on B21-R613 (Core Δ Pressure/Core Flow) recorder
- Reduction in core plate differential pressure on B21-R613 (Core Δ Pressure/Core Flow) recorder
- Rise in recirculation loop flow in the loop with a failed jet pump on B32-R614 (Recirculation Flow) recorder

CAUTION

Under conditions of jet pump failure, indicated core flow on Process Computer Point U2CPWTCF and B21-R613 (Core Δ Pressure/Core Flow) recorder, will **NOT** be accurate. Accurate core flow is available from Process Computer Point U2NSSWDP (Core Plate Differential Pressure) or Attachment 1, Estimated Total Core Flow vs. Core Support Plate Delta P for B2C22. Until Step 23.b(1), the operating point on the Power-to-Flow Map will **NOT** be accurate. Indicated total core flow on B21-R613 (Core Δ Pressure/Core Flow) recorder will continue to be inaccurate until the failed jet pump is repaired. .... ☐

23. **IF** jet pump failure is suspected, **THEN** perform the following:

    a. **IF** reactor power is greater than or equal to 25%, **THEN** ensure the following:

    - 0PT-13.1, Reactor Recirculation Jet Pump Operability, is performed for two loop operation .... ☐

      OR

    - 0PT-13.4, Reactor Recirculation Jet Pump Operability for Single Loop Operation, is performed for single loop operation .... ☐

b. <u>IF</u> any jet pump is determined to be INOPERABLE,
<u>**THEN**</u> **perform** the following:

(1) **Ensure** the input to the Power-to-Flow Map has been changed from WTCF to core plate differential pressure. .... ☐

(2) **Notify** the Duty Reactor Engineer the input to the Power-to-Flow Map has been changed from WTCF to core plate differential pressure. .... ☐

(3) **Commence** unit shutdown in accordance with <u>0GP-05</u>, Unit Shutdown, in compliance with Technical Specification 3.4.2. .... ☐

36. 295003 1

Unit One is operating at rated power.

The load dispatcher reports degraded grid conditions with the following indications:

Generator frequency	59.7 hertz
230 KV Bus 1A voltage	205 KV
230 KV Bus 1B voltage	205 KV
E1 voltage	3690 volts
E2 voltage	3685 volts

Which one of the following completes both statements below?

The ____(1)____ may be damaged with continued operation under these conditions.

IAW 0AOP-22.0, *Grid Instability,* the E-Bus master/slave breakers ____(2)____ open.

A. (1) main turbine blades
   (2) will

B. (1) main turbine blades
   (2) will NOT

C. (1) emergency bus loads
   (2) will

D. (1) emergency bus loads
   (2) will NOT

Answer: C

K/A:

295003 Partial or Complete Loss of A.C. Power

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

03 Under voltage/degraded voltage effects on electrical loads

RO/SRO Rating: 2.9/3.2

Tier 1 / Group 1

K/A Match: THIS QUESTION WAS PRE-SUBMITTED FOR APPROVAL.
This meets the K/A because this is testing the degraded voltage conditions.

Pedigree: Last used on the 10-2 NRC Exam

Objective: LOI-CLS-LP-302-G, Obj. 4b
Given plant conditions and any of the following AOPs, determine the required supplemental actions: AOP-22.0, Grid Instability

Reference: None

Cog Level: High

Explanation: There are frequency based criteria in AOP-22.0 (Caution directly preceding step 3.2.1) for tripping the turbine to prevent resonance vibration of low pressure blades due to off frequency operation. Time limits include, 5 minute ranges and 1 minute ranges. At this current frequency, the Main Turbine can be operated indefinitely, which will not cause turbine damage. Sustained low voltage provides for higher running currents which will damage running ESF motors. Per the automatic actions section of AOP-22.0, the degraded voltage relays will actuate when emergency bus voltage has dropped below 3700 VAC for 10 seconds. This trips the Master/Slave breakers (BOP bus supply to E Buses) and the DGs start and load.

Distractor Analysis:

Choice A: Plausible because turbine blade damage can occur due to off frequency operation and the second part is correct.

Choice B: Plausible because turbine blade damage can occur due to off frequency operation and the second part is plausible because the frequency is within range.

Choice C: Correct Answer, see explanation

Choice D: Plausible because damage to E Bus loads is correct and the second part is plausible because the frequency is within range.

SRO Basis: N/A

asymmetrical short circuit current of 200kA RMS.

2.4 Protective Relaying

Protective relaying is designed to isolate any faulted component or portion of the electrical system, while maintaining continuity of power to the unfaulted portion of the system. The most commonly used protective devices include:

1. Undervoltage (27 Device) Relays. Undervoltage relays actuate on a low voltage condition, and usually are time delayed to account for momentary transient conditions, such as fault clearing and bus transfers. The degraded grid voltage relays are provided with a substantially longer time delay to prevent actuation due to motor starting transients. Undervoltage relays provide a variety of protective functions including supply breaker trips and closure permissives, large motor breaker trips and closure permissives, and automatic starting of the Emergency Diesel Generators.

4.2 Abnormal Operation

4.2.1 Abnormal Frequency Conditions

When system frequency reaches 59.8 hertz, Annunciator, UA-06, window 1-2, "GEN BUS UNDER FREQ RELAY" is activated. Operators are directed to respond per AOP-22.0, Generator Abnormal Frequency Conditions. This is done to stabilize loads on the system. One of the most probable causes of an under frequency condition would be the loss of another large generating unit, or units, when the on-line reserve capacity is inadequate for current system loads. Rapid response and close coordination with the load dispatcher are required to ensure system stability.

Abnormal frequency operation can develop resonant frequencies that may induce vibrations in the low pressure turbine blades. The vibration can cause turbine blades to fatigue and possibly fail during operation. The effect increases proportionally in relation to the magnitude of the frequency difference, and the length of time at the abnormal frequency.

3.0 AUTOMATIC ACTIONS

1. **IF** emergency bus voltage has lowered to less than 3700 volts (approximately equal to BOP bus voltage) for greater than 10 seconds,
   THEN the master/slave breakers to the E bus open and associated diesel generator starts and loads. .... ☐

4.2 Supplementary Actions

NOTE

A sudden rise in system frequency may be observed due to additional generation or load shedding. Automatic load shedding (10% of system load) occurs at each of the following frequencies: 59.3, 59.0, and 58.5 Hz. ☐

CAUTION

The maximum allowable time at a given frequency is as follows: ☐

- Below 58.1 Hz, operation is prohibited
- Between 58.1-58.5 Hz, operation for 1 minute is allowed
- Between 58.6-59.3 Hz, operation for 5 minutes is allowed
- Between 59.4-60.6 Hz, operation is allowed indefinitely
- Between 60.7-61.4 Hz, operation for 5 minutes is allowed
- Between 61.5-61.9 Hz, operation for 1 minute is allowed
- Above 61.9 Hz, operation is prohibited

CAUTION

- Off-frequency operation can stimulate resonance vibration in low pressure blades. ☐
- A total loss of off-site power (LOOP) should be anticipated if the turbine is tripped. ☐
- With grid voltage or frequency unstable or grid vulnerability identified, diesel generators should **NOT** be paralleled with any E bus connected to the grid since severe load swings may occur and possibly overload the diesel generators. ☐

1. **IF** the maximum allowable time at a given frequency is exceeded, **THEN** perform the following:

   a. **IF** reactor power is greater than or equal to 26%, **THEN insert a manual scram**. ☐

   b. **Trip the main turbine**. ☐

   c. **IF** the unit was scrammed, **THEN enter** 1EOP-01-RSP(2EOP-01-RSP), Reactor Scram Procedure. ☐

4.2 Supplementary Actions (continued)

10. IF system frequency is high,
    THEN:

    a. **Establish** communication with the Load Dispatcher. ☐

    b. **Continue** unit generation as directed by the Unit CRS AND **coordinate** with the Load Dispatcher. ☐

    c. IF tripping the turbine becomes imminent,
       THEN **rapidly reduce** power in an attempt to lower frequency to less than 60.7 Hz prior to tripping the main turbine. ☐

11. IF notified by the Load Dispatcher system voltage is unable
    OR will be unable to support a LOCA,
    OR abnormal frequency conditions persist,
    THEN **follow** the guidelines in 0OI-01.01, BNP Conduct of Operations Supplement. ☐

12. IF any diesel generator is loaded to an E bus connected to the grid,
    THEN **restore** the diesel generator to standby in accordance with applicable procedures. ☐

13. IF system voltage is less than 3700 volts for greater than 10 seconds,
    THEN **ensure**:

    - The affected E bus master/slave breakers OPEN ☐
    - The affected diesel generator starts and loads ☐

37. 295004 1

Which one of the following completes both statements below?

IAW 0AOP-39.0, *Loss of DC Power*, before 125 VDC battery voltage reaches ___(1)___, remove loads as directed by the Unit CRS.

IAW 1EOP-01-SBO, *Station Blackout*, if either division battery chargers can NOT be restored within ___(2)___ then load strip the affected battery.

A. (1) 105 volts
   (2) 1 hour

B. (1) 105 volts
   (2) 2 hours

C. (1) 129 volts
   (2) 1 hour

D. (1) 129 volts
   (2) 2 hours

Answer: A

K/A:

295004 Partial or Complete Loss of D.C. Power

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

01 Battery charger

RO/SRO Rating: 3.1/3.1

Tier 1 / Group 1

K/A Match: This meets the K/A because this is testing knowledge of the relationship between the loss of DC power and time requirement to re-energize the battery charger.

Pedigree: New

Objective: LOI-CLS-LP-051, Obj. 14
Describe the consequences/problems associated with the following: a. Battery chargers remaining out of service during a loss of off-site power / station blackout.

Reference: None

Cog Level: Fundamental

Explanation: AOP-39.0 directs to load strip before reaching 105 VDC to prevent cell reversal. The alarm for undervoltage comes in at 129 VDC. The station Blackout procedure states that if the battery charger is not energized in 1 hour to load strip the batteries. There is a time critical 2 hour action in the SBO procedure for opening the Reactor Building doors.

Distractor Analysis:

Choice A: Correct Answer, see explanation.

Choice B: Plausible because the first part is correct and the second part is a time critical action time limit in the SBO procedure.

Choice C: Plausible because 129 volts is the annunciator setpoint for the batteries and the second part is correct.

Choice D: Plausible because 129 volts is the annunciator setpoint for the batteries and the second part is a time critical action time limit in the SBO procedure.

SRO Basis: N/A

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4.2 Supplementary Actions

1. Loss of Battery Chargers:

   a. **Monitor** 125V and 24V DC battery voltages. ☐

   b. **IF** power has been removed from the battery chargers for greater than 1 hour, **THEN remove** selected loads from the battery based on 0OI-50, 125/250 and 24/48 VDC Electrical Load List and Unit CRS direction. ☐

   c. Before 125V DC battery voltage reaches the low voltage limit of 105 volts, **remove** loads as directed by the Unit CRS as necessary to maintain battery voltage greater than 105 volts. ☐

   d. Before 24V battery voltage reaches the low voltage limit of 21 volts, **remove** loads as directed by the Unit CRS as necessary to maintain battery voltage greater than 21 volts. ☐

   e. **IF** battery charger AC power has been lost due to Station Blackout, **THEN enter** 1EOP-01-SBO(2EOP-01-SBO), Station Blackout. ☐

Time Critical

Battery load stripping required within 60 minutes.

Time ____________________

IF either division battery chargers **CANNOT** be restored,

THEN load strip affected battery per EOP-01-SBO-10.

SBO-21

ELAP

WHEN

Determined E1 **OR** E2 will **NOT** be energized within 60 minutes

THEN continue

SBO-22

Defeat RCIC automatic logic per EOP-01-SEP-10.

SBO-23

IF RCIC is **NOT** available

THEN **defeat** HPCI automatic logic per EOP-01-SEP-10.

SBO-24

Open reactor building doors per EOP-01-SBO-04.

SBO-25

Time Sensitive

RB roof hatch required open within 2 hours.

Time ____________________

Stage alternate fuel pool makeup/spray equipment per EOP-01-SEP-12.

SBO-26

IF directed by ERO,

THEN **isolate** containment per EOP-01-SBO-15.

SBO-27

38. 295005 1

Which one of the following identifies the reason an operator is directed to trip the main turbine as an immediate action IAW 0AOP-32.0, Plant Shutdown From Outside Control Room?

A. To initiate a scram on TSV/TCV closure.

B. To prevent reverse power starts of the Diesel Generators.

C. The turbine cannot be tripped once the Control Room is evacuated.

D. To bring bypass valves into operation until Remote Shutdown Panel control is established.

Answer: B

K/A:

295005 Main Turbine Generator Trip

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

04 Main generator trip

RO/SRO Rating: 3.2/3.2

Tier 1 / Group 1

K/A Match: This question requires the operator to have knowledge of the reason for turbine/generator trip. AOP-32 was used to include plausibility of distractors.

Pedigree: Bank

Objective: LOI-CLS-LP-302E, Obj. 6
Given plant conditions and entry into 0AOP-32.0, Plant Shutdown From Outside Control Room, explain the basis for a specific caution, note, or series of procedure steps.

Reference: None

Cog Level: Fundamental

Explanation: Following a reactor scram, the turbine control valves throttle shut in an effort to control RPV pressure at the setpoint of 928 psig. Without operator action, the turbine control valves will fully close, causing the generator to motor. Reverse power on the generator will cause a generator primary lockout and auto start of the diesel generators. The main turbine is therefore manually tripped to prevent it from automatically tripping on generator reverse power. This also reduces the number of cold start demands on the diesel generators.

Distractor Analysis:

Choice A: Plausible because a reactor scram is inserted as a step in the AOP, but it is performed earlier.

Choice B: Correct Answer, see explanation

Choice C: Plausible because the procedure states to perform the step prior to exiting the control room but it could still be done at the turbine front standard.

Choice D: Plausible because this would allow use of the bypass valves, but MSIVs are manually closed prior to leaving the control room. This brings SRVs into operation. If MSIVs are not closed prior to leaving the control room, RPS EPA breakers are opened prior to establishing control at Remote Shutdown panel, which would close MSIVs.

SRO Basis: N/A

5.2 Step RSP-2

Perform scram immediate actions.

RSP-2

Step RSP-2 includes the potential for multiple sensor and sensor relay failures in the automatic RPS logic where an automatic reactor scram should have initiated but did not. If needed a manual scram is inserted to accomplish an automatic action which should have taken place. A manual reactor scram is also required when directed from other EOPs and no condition exists which would have automatically initiated a reactor scram (e.g., entry from PCCP because of high torus temperature).

Step RSP-2 also addresses other Reactor Operator scram immediate actions and includes:

- ARI initiation is an additional means of inserting control rods if needed.

- Placing the reactor mode switch to shutdown. When the reactor mode switch is placed in SHUTDOWN position, a diverse and redundant reactor scram signal is generated by the RPS logic. If the mode switch is taken out of RUN prior to RPV pressure decreasing to 835 psig, the MSIV closure due to low main steam line pressure is prevented.

  For Unit 2 only, if the mode switch is taken out of RUN when steam flow is above 33%, the MSIVs will close. Therefore, for Unit 2 the mode switch is placed in SHUTDOWN after steam flow is below 3x10^6 lb/hr.

- Following a reactor scram, the turbine control valves throttle shut in an effort to control RPV pressure at the setpoint of 928 psig. Without operator action, the turbine control valves will fully close, causing the generator to motor. Reverse power on the generator will cause a generator primary lockout and auto start of the diesel generators. The main turbine is therefore manually tripped to prevent it from automatically tripping on generator reverse power. This also reduces the number of cold start demands on the diesel generators.

39. 295006 1

Unit One has entered RSP with the following conditions:

Six control rods are at position 02, all others are fully inserted
B Recirc Pump has tripped

Which one of the following completes both statements below?

The control rods will be inserted by ___(1)___ IAW 0EOP-01-LEP-02, *Alternate Control Rod Insertion*.

After the control rods are inserted, a CRD flow rate of approximately ___(2)___ will be established.

A. (1) placing the individual scram test switches to the Scram position
(2) 30 gpm

B. (1) placing the individual scram test switches to the Scram position
(2) 45 gpm

C. (1) driving rods using RMCS
(2) 30 gpm

D. (1) driving rods using RMCS
(2) 45 gpm

Answer: C

K/A:
295006 Scram
AA1 Ability to operate and/or monitor the following as they apply to SCRAM: (CFR: 41.7 / 45.6)
06 CRD hydraulic system

RO/SRO Rating: 3.5/3.6

Tier 1 / Group 1

K/A Match: This meets the K/A because this is testing operation of CRD controls after a scram.

Pedigree: new

Objective: LOI-CLS-LP-300-C, Obj. 10
Given plant conditions and the Reactor Scram Procedure, determine the required operator actions

Reference: None

Cog Level: High

Explanation: Even if the reactor will remain shutdown under all conditions without boron the LEP is used to insert the control rods using RMCS. If more control rods were out then the scram test switches would be an option. If a recirc pump is tripped then CRD flow is set to 30 gpm to minimize the stratification in the bottom head region.

Distractor Analysis:

Choice A: Plausible because this is an option used to insert the control rods in the LEP. The second part is correct.

Choice B: Plausible because this is an option used to insert the control rods in the LEP. The second part is the nominal setting for the CRD flowrate.

Choice C: Correct Answer, see explanation.

Choice D: Plausible because the first part is correct and the second part is the nominal setting for the CRD flowrate.

SRO Basis: N/A

7. **WHEN** either:

- All control rods in .... ☐ RO
- Only one control rod **NOT** fully inserted .... ☐ RO
- **NO** more than 10 control rods withdrawn to position 02 **AND** **NO** control rod withdrawn beyond position 02 .... ☐ RO
- Reactor engineering has determined the reactor will remain shutdown under all conditions without boron .... ☐ RO

THEN perform Section 2.2, Control Rod Insertion Verification on Page 7. .... ☐ RO

10. **IF** any control rod **NOT** fully inserted, **THEN insert** control rods:

a. **Record** in Control Room log the control rod number and position of any rods **NOT** fully inserted. .... ________ RO

2.2.3 Control Rod Verification Actions (continued)

b. **Bypass** RWM. .... ________ RO

c. **Insert** control rods with Emergency Rod In Notch Override switch. .... ________ RO

2.6.3 Scram Individual Control Rods Actions (continued)

10. **Unit 1 Only: Insert** control rods with individual scram test switches:

a. **Identify** any control rod **NOT** inserted to or beyond Position 00. .... ☐ RO

NOTE
• A sound powered phone jack is located on the column beside Panel XU-76 and in Panels XU-12, 58, 49 and 61 .... ☐
• The preferred sound-powered phone switchboard bus for use is Bus 1 .... ☐

b. **Establish** communication between Panel P610 and Control Room. .... ☐ RO

NOTE
The individual scram test switch SCRAM position is down. .... ☐

c. **Place** individual scram test switch to SCRAM position for any control rod **NOT** inserted to or beyond Position 00. .... ☐ RO

2.2.3 Control Rod Verification Actions (continued)

(3) **IF** both CRD pumps running,
THEN stop one CRD pump. .... ________ RO

(4) **Set** the setpoint tape on C11(C12)-FC-R600 (CRD Flow Control) to 30 gpm. .... ________ RO

> **NOTE**
>
> The actions in Section 2.2.3 Step 9.h(5) may be repeated as necessary. .... ☐

(5) **Adjust** cooling water differential pressure, CRD flow rate and drive pressure:

- C11(C12)-FC-R600 (CRD Flow Control) to maintain cooling water differential pressure between 10 and 26 psid. .... ________ RO
- **IF** a reactor recirculation pump is tripped, **THEN establish** a CRD flow rate of approximately 30 gpm. .... ________ RO
- **IF** both reactor recirculation pumps running, **THEN establish** a CRD flow rate between 30 and 60 gpm. .... ________ RO

40. 295009 1

A total loss of Unit One feedwater results in reactor water level lowering to 87 inches.
Drywell pressure is 2.1 psig.
Reactor water level is being restored with RCIC and CRD.

Which one of the following completes both statements below?

RVCP ___(1)___ required to be entered.

The expected response of the G31-F001, Inboard RWCU Isolation Valve, and the G31-F004, Outboard RWCU Isolation Valve, is that ___(2)___ should be closed.

A. (1) is
   (2) ONLY the G31-F004

B. (1) is
   (2) BOTH

C. (1) is NOT
   (2) ONLY the G31-F004

D. (1) is NOT
   (2) BOTH

Answer: B

K/A:

295009 Low Reactor Water Level

AK2 Knowledge of the interrelations between LOW REACTOR WATER LEVEL and the following: (CFR: 41.7 / 45.8)

04 Reactor water cleanup

RO/SRO Rating: 2.6/2.6

Tier 1 / Group 2

K/A Match: This meets the K/A because this is testing the LL2 relationship to Group 3 (RWCU) isolation.

Pedigree: New

Objective: LOI-CLS-LP-014, Obj 8
Given plant conditions, determine if the RWCU system should have isolated, including expected changes in RWCU System components

Reference: None

Cog Level: High

Explanation: Based on conditions RVCP should be entered. By knowing the entry conditions for RVCP (2# DW pressure) this eliminates the RSP. The low level condition will isolate the F001 and F004. There are some signals that will isolate only the F004 only.

Distractor Analysis:

Choice A: Plausible because the first part is correct and some of the Group 3 signals do only close the F004.

Choice B: Correct Answer, see explanation.

Choice C: Plausible because the RSP would be entered, but there is an entry condition for RVCP (2# in the DW). Some of the Group 3 signals do only close the F004.

Choice D: Plausible because the RSP would be entered, but there is an entry condition for RVCP (2# in the DW). The second part is correct.

SRO Basis: N/A

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ATTACHMENT 1
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<< Group Isolation Checklist >>

Group 3 Isolation Signals

Signal	**Tech Spec Value**	**Setpoint Value**
Low Level 2	+101 inches	+105 inches
High Differential Flow	73 gpm	43 gpm (after 28.5 minute time delay)
Area High Temperature	150°F	140°F
Area Ventilation ΔT High	50°F	47°F
Non-Regen Hx Outlet Temp Hi	N/A	135°F
SLC Initiation	N/A	N/A
RWCU Outside Pump/Hx Rms	120°F	115°F
RWCU Differential Flow High Time Delay	30 minutes	28.5 minutes

Group 3 Isolation Valves
Control Room - RTGB - Panel H12-P601

Valve Number	**Power Supply Unit 1(Unit 2)**	**Normal Position**	**Fail Position**	**Checked**
[Note 1] G31-F001	1XC(2XC)/E1(E3)	NO	[Note 2] FAI	
G31-F004	1XDB(2XDB) [DC]	NO	FAI	

Note 1: SLC Initiation and RWCU Non-Regen Hx Outlet Temperature Hi signals do **NOT** isolate the RWCU Inlet Inboard Isolation Valve, G31-F001.

41. 295016 1

CAUTION
There are seven keylock *NORMAL/LOCAL* switches located on Diesel Generator 2 control panel. Six of these are located in a row. The seventh switch is located in the row above the six switches.

Which one of the following completes both statements below concerning the caution above from 0ASSD-02, *Control Building*?

The six switches in a row must be placed in *LOCAL* ___(1)___ placing the seventh switch in *LOCAL*.

The purpose of this sequence is to prevent a loss of DG2 due to a loss of the redundant power supply fuses for the ___(2)___ circuitry.

A. (1) before
(2) output breaker

B. (1) before
(2) engine run control

C. (1) after
(2) output breaker

D. (1) after
(2) engine run control

Answer: B

K/A:
295016 Control Room Abandonment
AK3 Knowledge of the reasons for the following responses as they apply to CONTROL ROOM ABANDONMENT: (CFR: 41.5 / 45.6)
03 Disabling control room controls

RO/SRO Rating: 3.5/3.7

Tier 1 / Group 1

K/A Match: This meets the K/A because the six local switches remove control room controls and the seventh switch supplies an alternate power supply to the equipment.

Pedigree: Bank

Objective: LOI-CLS-LP-304, Obj. 21
Explain why the Diesel Generator NORMAL/LOCAL switches must be placed in LOCAL in a particular sequence.

Reference: None

Cog Level: Fundamental

Explanation: The six switches in a row isolate DG2 engine and generator control circuitry from the control room (the fire area) since a fire induced fault in wiring in the fire area may result in loss of the DG. The seventh switch inserts redundant control power fuses to the circuitry that has been isolated in the event a fault has already resulted in blowing the normal fuses. This seventh switch must be turned last with the potentially faulted circuitry already isolated or the alternate fuses may also blow making the DG unavailable. The DG engine lockout is already tripped if the DG had been running since the operator is directed to trip the DG using emergency stop. Of the first six switches, they include: - Diesel START/STOP (2 switches) - Diesel Governor (2 switches) - Generator Voltage Regulation (2 switches)

Distractor Analysis:

Choice A: Plausible because the six switches are placed in local first and the output breaker does have redundant control power fuses.

Choice B: Correct Answer, see explanation.

Choice C: Plausible because this is the opposite of the correct sequence and the output breaker does have redundant control power fuses.

Choice D: Plausible because this is the opposite of the correct sequence and the second part is correct.

SRO Basis: N/A

The Governor Control At Setpoint indicator light provides a status of the DRU speed reference for the 2301A governor. The light is an indicator that the Governor Control System is ready to operate at the Setpoint "speed". During actual operation of the DG, the Governor Control At Setpoint indicator light may or may not be illuminated depending on the speed of the DG.

Voltage Adjust Switches

Two three position (RAISE-NEUT-LOWER) spring return to NEUT switches are provided per engine to permit the adjustment of voltage regulators from the local panel regardless of EDG mode of operation. The auto adjust switch is normally used.

ASSD Keylock Switches

Brass handled two-position NORM - LOCAL ASSD keylock switches on the local engine panels permit the operator to transfer control of the engine and generator to the local control panel. ASSD operations are performed when a fire exists in the plant and components required to be operated may be damaged by the fire.

These switches isolate control room controls and indications to isolate the EDG control circuitry from potential fire induced faults. There are six ASSD switches (2 for EDG run/stop controls, 2 for governor controls, and 2 for voltage regulation controls) located on each local EDG panel. When in the "ASSD" mode, operation of the Diesel engine can only be accomplished by the LOCAL EMERGENCY STOP and LOCAL EMERGENCY START pushbuttons.

In addition to the six ASSD switches, for EDG 2 and 4 only, there is a seventh ASSD switch located above the other six switches. This switch provides an alternate set of control power fuses for EDG control circuitry. This may be necessary since fire induced faults may have blown normal control fuses. When operating the ASSD switches for EDGs 2 or 4, the seventh switch **must be turned last** after the potentially faulted circuitry has been isolated to prevent blowing the alternate fuses, making the EDG unavailable to provide power to Safe Shutdown loads.

42. 295017 1

During accident conditions, the source term from the Unit One Reactor Building must be estimated. Three RB HVAC supply fans and three RB HVAC exhaust fans are running.

IAW 0PEP-03.6.1, *Release Estimates Based on Stack/Vent Readings*, which one of the following is the calculated release rate?

ATTACHMENT 2
Page 1 of 1
Source Term Calculation From #1 RX Gas (1-CAC-AQH-1264-3)

TIME	METER READING (cpm)	FLOW[1] (cfm)	EFFICIENCY[(2)] FACTOR	RELEASE[(3)] RATE (μCi/sec)
1 minute ago	4.0 E+3	43,200 CFM per exhaust fan	1.275 E-5	

(1) If not available use 43,200 cfm per exhaust fan times the number of fans operating.

(2) The efficiency factors can be obtained from 0E&RC-2020 (contact E&RC counting room).

(3) Release Rate = (cpm) x (cfm) x (Efficiency Factor)

A. 2.2 E+3 μCi/sec.

B. 6.6 E+3 μCi/sec.

C. 1.3 E+4 μCi/sec.

D. 6.6 E+4 μCi/sec.

Answer: B

K/A:
295017 High Off-Site Release Rate
AA2 Ability to determine and/or interpret the following as they apply to HIGH OFF-SITE RELEASE RATE: (CFR: 41.10 / 43.5 / 45.13)
03 Radiation levels

RO/SRO Rating: 3.1/3.9

Tier 1 / Group 2

K/A Match: This meets the K/A because it is testing the source term for a release off-site.

Pedigree: Bank

Objective: LOI-CLS-LP-301A, Obj. 6
Determine data required for offsite dose projection in accordance with AD-EP-ALL-0202, Emergency Response Offsite Dose Assessment, and PEP-03.6.1, Release Estimates Based Upon Stack/Vent Readings.

Reference: None

Cog Level: High

Explanation: Per Attachment 2 the calculated release rate is:
Meter reading (CPM) X Flow (43,200 per fan X no of discharge fans) X efficiency factor
or
(4 E+3) (43,200 X 3) (1.275 E-5) = 6.6 E+3 mCi/sec

Distractor Analysis:

Choice A: Plausible because it is the calculation without multiplying times the number of running exhaust fans.

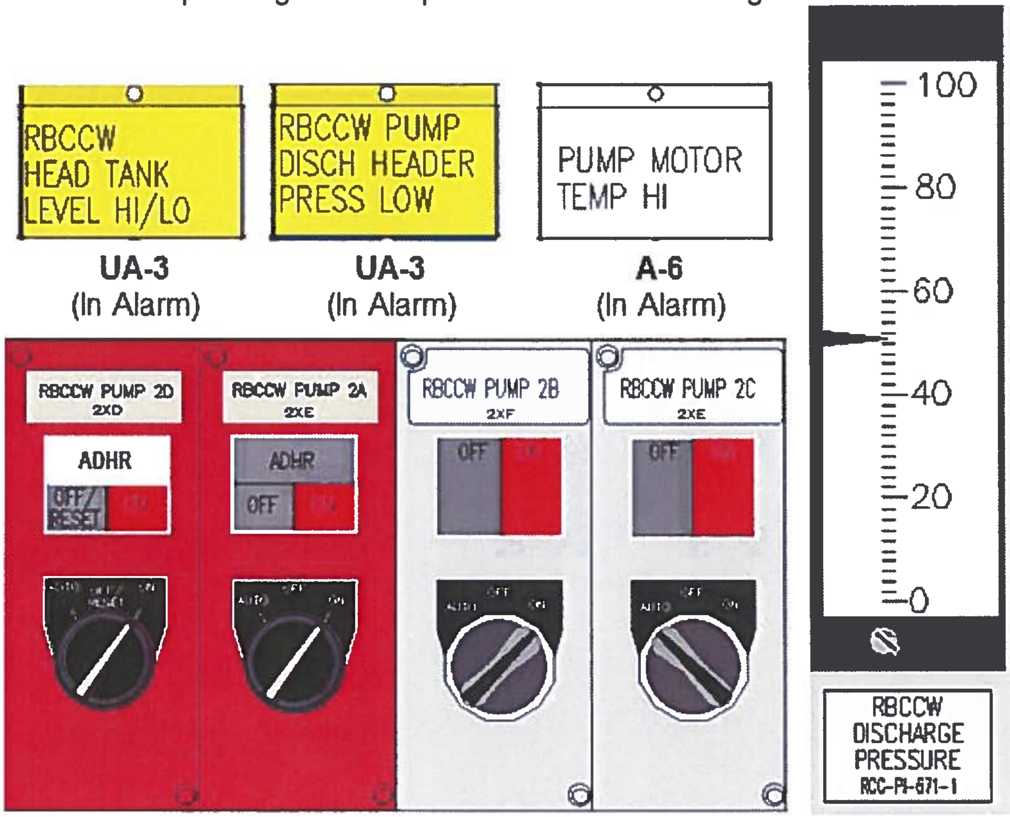
Choice B: Correct Answer, see explanation.

Choice C: Plausible because it uses the total number of fans running vs. the number of exhaust fans.

Choice D: Plausible because it is the correct numerical value but is off by a factor of 10.

SRO Basis: N/A

43. 295018 1
Unit Two is operating at ~65% power when the following are observed:



Which one of the following completes both statements below IAW 0AOP-16.0, *RBCCW System Failure*?

A complete loss of RBCCW ____(1)____ occurred.

A reactor scram ____(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

Answer: A
K/A:
295018 Partial or Complete Loss of Component Cooling Water
AK2 Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER and the following: (CFR: 41.7 / 45.8)
02 Plant operations

RO/SRO Rating: 3.4/3.6

Tier 1 / Group 1

K/A Match: This meets the K/A because it is testing the relationship of the loss of RBCCW and the actions required for plant operations

Pedigree: Last used on the 04 NRC Exam

Objective: LOI-CLS-LP-302-H, Obj. 4a
Given plant conditions, determine the required supplementary actions in accordance with the following AOPs: 0AOP-16.0, RBCCW System Failure

Reference: None

Cog Level: High

Explanation: A complete loss of RBCCW is defined as discharge header pressure below 60 psig and all available RBCCW pumps running (AOP-16.0). A complete loss requires a manual scram.

Distractor Analysis:

Choice A: Correct Answer, see explanation.

Choice B: Plausible because the first part is correct and for a loss of TBCCW only a performing power reduction is allowed.

Choice C: Plausible because the pumps are running and system pressure is available and a reactor scram could be thought of to be needed due to alarms.

Choice D: Plausible because the pumps are running and system pressure is available and for a loss of TBCCW only a performing power reduction is allowed.

SRO Basis: N/A

4.2 Supplementary Actions (continued)

b. IF ADHR Mode piping is **NOT** the source of the leakage, **THEN re-align** RBCCW Pumps A and D from ADHR Mode to RBCCW Mode, as necessary ☐

NOTE

A complete loss of RBCCW is defined as discharge header pressure less than 60 psig, high temperature alarms on components supplied by RBCCW, and all available (no more than three) RBCCW pumps operating on the RBCCW header ☐

4. **IF** there is a complete loss of RBCCW, **THEN**: ☐

a. **Trip** all RBCCW pumps (including RBCCW Drywell HVAC Cooling Pump if operating on the affected unit and pumps operating in ADHR Mode). ☐

b. **Close** the following valves:

- RCC-V28 (RBCCW To DW Isol Vlvs) ☐
- RCC-V52 (RBCCW TO DW Isol Vlvs) ☐

c. **Trip** RWCU pump(s). ☐

d. **Isolate** RWCU System by closing the following valves:

- G31-F001 (RWCU Inboard Isol Vlv) ☐
- G31-F004 (RWCU Outboard isol Vlv) ☐

e. **Reduce** reactor power with recirc flow in accordance with 0ENP-24.5, Form 2, Immediate Reactor Power Reduction Instructions ☐

f. **Insert** a manual scram ☐

g. **Enter** 1EOP-01-RSP(2EOP-01-RSP), Reactor Scram Procedure, **AND perform** concurrently with this procedure. ☐

h. **Trip** both reactor recirculation pumps by performing the following:

(1) **Depress** VFD A Emerg Stop ☐

(2) **Depress** VFD B Emerg Stop ☐

44. 295019 1

Unit Two has entered 0AOP-20.0, *Pneumatic (Air/Nitrogen) System Failures*, due to a loss of instrument air pressure with the following annunciator status:

UA-01 (1-1) *RB Instr Air Receiver 2A Press Low*	Alarm sealed in
UA-01 (1-2) *RB Instr Air Receiver 2B Press Low*	NOT in Alarm
UA-01 (3-2) *Air Compr D Trip*	Alarm sealed in
UA-01 (4-4) *Inst Air Press Low*	Alarm sealed in
UA-01 (5-4) *Service Air Press-Low*	Alarm sealed in

Which one of the following completes both statements below?

On a loss of instrument air, the RB HVAC Butterfly Isolation Valves will fail ____(1)____.

IAW 0AOP-20.0, the reactor ____(2)____ required to be scrammed.

A. (1) as-is
   (2) is

B. (1) as-is
   (2) is NOT

C. (1) open
   (2) is

D. (1) open
   (2) is NOT

Answer: B

K/A:

295019 Partial or Complete Loss of Instrument Air

AA2 Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR: (CFR: 41.10 / 43.5 / 45.13)

02 Status of safety-related instrument air system loads

RO/SRO Rating: 3.6/3.7

Tier 1 / Group 1

K/A Match: This meets the K/A because it is testing status of equipment on a loss of air and the action that is required from the AOP

Pedigree: New

Objective: LOI-CLS-LP-302K, Objective 6
Summarize the consequences associated with improper equipment operation specified in 0AOP-20.0, Pneumatic (Air/Nitrogen) System Failures

Reference: None

Cog Level: Fundamental

Explanation: A loss of instrument air the BFIVs fail as-is. Other equipment will fail open or closed. A reactor scram is required if unable to maintain at least one division non-interruptible instrument air pressure greater than 95 psig.

Distractor Analysis:

Choice A: Plausible because the first part is correct and a scram is required if both divisions are pressure cannot be maintained. The BFIV's are closed if either divisions air pressure cannot be maintained.

Choice B: Correct Answer, see explanation.

Choice C: Plausible because air operated valves can be designed to fail open, closed or as-is. Part 2 because a reactor scram is required if unable to maintain at least one division non-interruptible instrument air pressure greater than 95 psig.

Choice D: Plausible because air operated valves can be designed to fail open, closed or as-is. Part 2 is correct.

SRO Basis: N/A

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4.0 OPERATOR ACTIONS

NOTE

The following should be considered for establishment as critical parameters during performance of this procedure: ☐

- Instrument air pressure
- PNS pressure
- MSIV position
- Condensate and Feedwater System minimum flow valve status
- Control rod positions

4.1 Immediate Actions

1. IF any of the following conditions exist: ☐

   - Unable to maintain at least one division non-interruptible instrument air pressure greater than 95 psig ☐
   - Unable to maintain at least one division drywell pneumatic pressure greater than 95 psig ☐
   - Instrumentation indicates unsafe reactor operation ☐

   THEN:

   a. **Insert** a manual scram. ☐

   b. **Enter** 1EOP-01-RSP(2EOP-01-RSP), Reactor Scram Procedure and **perform** concurrently with this procedure. ☐

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e. **<u>IF</u>** UA-01 1-1, RB Instr Air Receiver 1A(2A) Press Low
<u>OR</u> UA-01 1-2, RB Instr Air Receiver 1B(2B) Press Low,
alarm is received,
<u>THEN</u> perform the following:

NOTE
Isolation of the reactor building supply and exhaust valves renders the building ventilation system INOPERABLE. Standby Gas Treatment System operation may be required to maintain reactor building negative pressure. ☐

(1) **<u>IF</u>** required to maintain reactor building negative pressure,
<u>THEN</u> start the Standby Gas Treatment System in accordance with 1OP-10(2OP-10), Standby Gas Treatment System. ☐

(2) **Close** the following valves:

<u>Unit 1 Only</u>:

- 1A-BFIV-RB and 1C-BFIV-RB (RB Vent Inbd Valves) ☐
- 1B-BFIV-RB and 1D-BFIV-RB (RB Vent Outbd Valves) ☐

<u>Unit 2 Only</u>:

- 2A-BFIV-RB and 2C-BFIV-RB (RB Vent Inbd Valves) ☐
- 2B-BFIV-RB and 2D-BFIV-RB (RB Vent Outbd Valves) ☐

45. 295020 1

I&C Techs inadvertently cause a low level 3 (LL3) signal.
Unit Two plant conditions are:

Reactor pressure	930 psig
Drywell pressure	1.7 psig, steady
Drywell temp (average)	140°F, slow rise
Drywell leak calculation	Normal

Which one of the following completes the statement below?

All Drywell Cooler Fans are:

A. tripped, but can be overridden on.

B. tripped, and cannot be overridden on.

C. running, but can be tripped at the RTGB.

D. running, and cannot be tripped at the RTGB.

Answer: A

K/A:
295020 Inadvertent Containment Isolation
AA1 Ability to operate and/or monitor the following as they apply to INADVERTENT CONTAINMENT ISOLATION: (CFR: 41.7 / 45.6)
02 Drywell ventilation/cooling system
RO/SRO Rating: 3.2/3.2
Tier 1 / Group 2

K/A Match: This meets the K/A because it is testing what the DW coolers do on an isolation signal.

Pedigree: Bank

Objective: LOI-CLS-LP-04, Obj. 20
Given plant conditions determine if the drywell coolers should auto start or trip

Reference: None

Cog Level: High

Explanation: LOCA signal on LL3 closes Group 10 which fails dampers open, but also trips fan motors. Override for LOCA trip can be performed as long as a LOCA does not really exist which is overridden in back panels (XU-27/XU-28). The low level condition also is a scram signal which provides an auto start signal for the DW Coolers which is prioritized by the trip signal.

Distractor Analysis:
Choice A: Correct Answer, see explanation.

Choice B: Plausible because the fans do trip and if the conditions were different they would not be able to be overridden.

Choice C: Plausible because the fans do auto start on a scram signal or usually when the dampers are opened and under different conditions they would be able to be tripped from the RTGB.

Choice D: Plausible because the fans do auto start on a scram signal or usually when the dampers are opened and under different conditions they would not be able to be tripped from the RTGB.

SRO Basis: N/A

Placing a Unit 2 Drywell Cooling Fan control switch in *START* causes the fan's discharge damper to open. WHEN the discharge damper is full open, the fan will start. The control switch should be held in the *START* position until the discharge damper is full open. The RBCCW cooling water valve to the coils will open concurrently with a fan start.

Placing the Drywell Cooling 1B Fan control switch in *START* causes the fan's discharge damper to open. WHEN the discharge damper is full open, the fan will start. The control switch should be held in the *START* position until the fan starts. The common air inlet damper and the RBCCW cooling water valve to the coils will open concurrently with a fan start.

WHEN the control switch for Drywell Cooling Fan 1A, 1C, or 1D is placed in *START*, the associated fan starts and the discharge backdraft damper opens from the fan air flow. The discharge damper position indication does not input to the start logic for Drywell Cooling Fans 1A, 1C, and 1D. The RBCCW cooling water valve to the coils will open concurrently with a fan start.

The Drywell Lower Vent dampers can be positioned to either MIN or MAX position by a two-position control switch on Panel XU-3. Normal plant operating position for these dampers is the MIN position. Placing these dampers to MAX position during plant operation may produce extreme temperature excursions in the upper drywell regions. Low scram air header pressure will reposition these dampers to the MAX position and automatically start any idle drywell cooling fan selected for AUTO.

Drywell Cooler Override Switches, VA-CS-5993/5994, are provided in Panels XU-27/28 to facilitate various modes of Drywell cooler operation as required by the EOPs.

The Pneumatic Nitrogen System or Reactor Building Non-Interruptible Instrument Air pneumatically operates the drywell cooling fans discharge dampers. These dampers will fail open on loss of pneumatics. Unit 2 and 1B drywell cooling fans discharge dampers fail closed on loss of the associated 120 VAC distribution panel.

A contactor in the associated fan's 480 VAC breaker provides drywell cooler FAN ON indication on RTGB Panel XU-3.

The drywell coolers receive a LOCA trip signal from the Core Spray initiation relays.

46. 295021 1

Unit One in MODE 5.
The fuel pool gates are removed.
SDC Loop B is in service.
Fuel pool cooling assist is in operation.

The RHR Loop B pumps tripped and can NOT be restarted.

Which one of the following completes both statements below?
(consider each statement separately)

Fuel pool cooling assist ___(1)___.

Fuel pool cooling assist ___(2)___ capable of being aligned to the SDC Loop A IAW 1OP-17, *Residual Heat Removal System Operating Procedure*.

A. (1) remains in service
   (2) is

B. (1) remains in service
   (2) is NOT

C. (1) is lost
   (2) is

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

Answer: D

K/A:
295021 Loss of Shutdown Cooling
AK2 Knowledge of the interrelations between LOSS OF SHUTDOWN COOLING and the following: (CFR: 41.7 / 45.8)
05 Fuel pool cooling and cleanup system

RO/SRO Rating: 2.7/2.8

Tier 1 / Group 1

K/A Match: This meets the K/A because it is testing the relationship of using SDC and the Fuel Pool.

Pedigree: New

Objective: LOI-CLS-LP-017, Obj 5
Given a drawing of the RHR system, trace the flow path for all of the six (6) modes of operation.

Reference: None

Cog Level: High

Explanation: Fuel pool cooling assist mode utilizes the B Loop of RHR so that when it is lost so too will the fuel pool cooling assist operations. If the gates were installed then the A Loop of SDC could be used with the B loop discharge flowpath, but with the gates removed this is NOT an option.

Distractor Analysis:

Choice A: Plausible because the students may think that the FPC pumps provide the motive force for this mode of operation and if the gates were installed then this would be correct.

Choice B: Plausible because the students may think that the FPC pumps provide the motive force for this mode of operation and the second part is correct.

Choice C: Plausible because the first part is correct and if the gates were installed then this would be correct.

Choice D: Correct Answer, see explanation

SRO Basis: N/A

8.11 Fuel Pool Cooling Assist Mode With Fuel Pool Gates Removed U

CAUTION

The following section has the potential to significantly raise area dose rates.

8.11.1 Initial Conditions Date/Time Started ______________

Initials

1. Reactor in Mode 5 with fuel pool gates removed. _____

2. Fuel pool temperature can **NOT** be maintained less than 125°F. _____

3. 0PT-08.0C has been completed satisfactorily within previous 92 days. _____

4. Fuel Pool Cooling system in operation in accordance with 1OP-13 with available fuel pool cooling heat exchangers in operation. _____

5. RHR Loop B is operating in shutdown cooling in accordance with Section 5.7 or 5.8. _____

47. 295023 1

Unit Two is performing refueling operations when the refueling SRO reports that a spent fuel bundle has been dropped.
The following radiation monitoring alarms are received:

UA-03 (3-7) *Area Rad Refuel Floor High*
UA-03 (4-5) *Process Rx Bldg Vent Rad Hi*

Which one of the following identifies the "Immediate Action" that is required IAW 0AOP-05.0, *Radioactive Spills, High Radiation, and Airborne Activity*?

A. Verify Group 6 isolation.

B. Evacuate all personnel from the refuel floor.

C. Place Control Room Emergency Ventilation System in operation.

D. Isolate Reactor Building Ventilation and place Standby Gas Treatment trains in operation.

Answer: C
K/A:
295023 Refueling Accidents
AA1 Ability to operate and/or monitor the following as they apply to REFUELING ACCIDENTS: (CFR: 41.7 / 45.6)
04 Radiation monitoring equipment

RO/SRO Rating: 3.4/3.7

Tier 1 / Group 1

K/A Match: This meets the K/A because it is testing the immediate operator actions for a radiation event.

Pedigree: Bank

Objective: LOI-CLS-LP-302J, Obj. 5
List the immediate operator actions required to be performed in accordance with 0AOP-05, Radioactive Spills, High Radiation, and Airborne Activity

Reference: None

Cog Level: Fundamental

Explanation: This is an Immediate Action identified in AOP-05.0.

Distractor Analysis:

Choice A: Plausible because this is an auto action not an immediate operator action of the AOP

Choice B: Plausible because IAW 0AOP-05.0 this is the first supplemental action.

Choice C: Correct Answer, see explanation.

Choice D: Plausible because RBHVAC isolation and SBGT start requires *PROCESS RX BLDG VENT RAD HI-HI* (UA-03 3-5) in alarm and these are supplementary actions in the procedure.

SRO Basis: N/A

4.0 OPERATOR ACTIONS

NOTE

The following should be considered for establishment as critical parameters during performance of this procedure: ☐

- Area radiation levels
- Personnel habitability in the affected area

4.1 Immediate Actions

1. **IF** a fuel assembly was dropped or damaged, **THEN ensure** the Control Room Emergency Ventilation System (CREVS) is in operation. {7.1.1} ☐

48. 295024 1

Unit Two is operating at rated power when high drywell pressure switch C72-PTM-N002A-1 fails high resulting in the annunciation of A-05-(5-6) *Pri Ctmt Press Hi Trip*.

Which one of the following completes the statement below?

RPS high drywell pressure relay C72-K4A will ___(1)___.

The RSP ___(2)___ be required to be entered.

A. (1) energize
   (2) will

B. (1) energize
   (2) will NOT

C. (1) de-energize
   (2) will

D. (1) de-energize
   (2) will NOT

Answer: D

K/A:

295024 High Drywell Pressure

EA1 Ability to operate and/or monitor the following as they apply to HIGH DRYWELL PRESSURE: (CFR: 41.7 / 45.6)

05 RPS

RO/SRO Rating: 3.9/4.0

Tier 1 / Group 1

K/A Match: This meets the K/A because it is testing the ability to monitor RPS (half scram condition) for a high DW pressure condition

Pedigree: New

Objective: LOI-CLS-LP-003, Objectives:
7.g Given plant conditions state the Normal, Initiation, and Fail position/condition of the following components: (Open/Closed Energized/De Energized) RPS Logic
9. Given any scram signal, describe the logic arrangement for the signal including what combination of signals will cause a Full Scram.

Reference: None

Cog Level: High

Explanation: The RPS relays are de-energize to actuate and a single relay actuates the alarm and will cause a half scram.

Distractor Analysis:

Choice A: Plausible because there are logics that are energize to actuate and there are also logics that only require one instrument to actuate (Nuclear instrumentation).

Choice B: Plausible because there are logics that are energize to actuate and the half scram is the result.

Choice C: Plausible because the first part is correct and some logics do cause a full scram (Nuclear instrumentation).

Choice D: Correct Answer, see explanation.

SRO Basis: N/A

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PRI CTMT PRESS HI TRIP

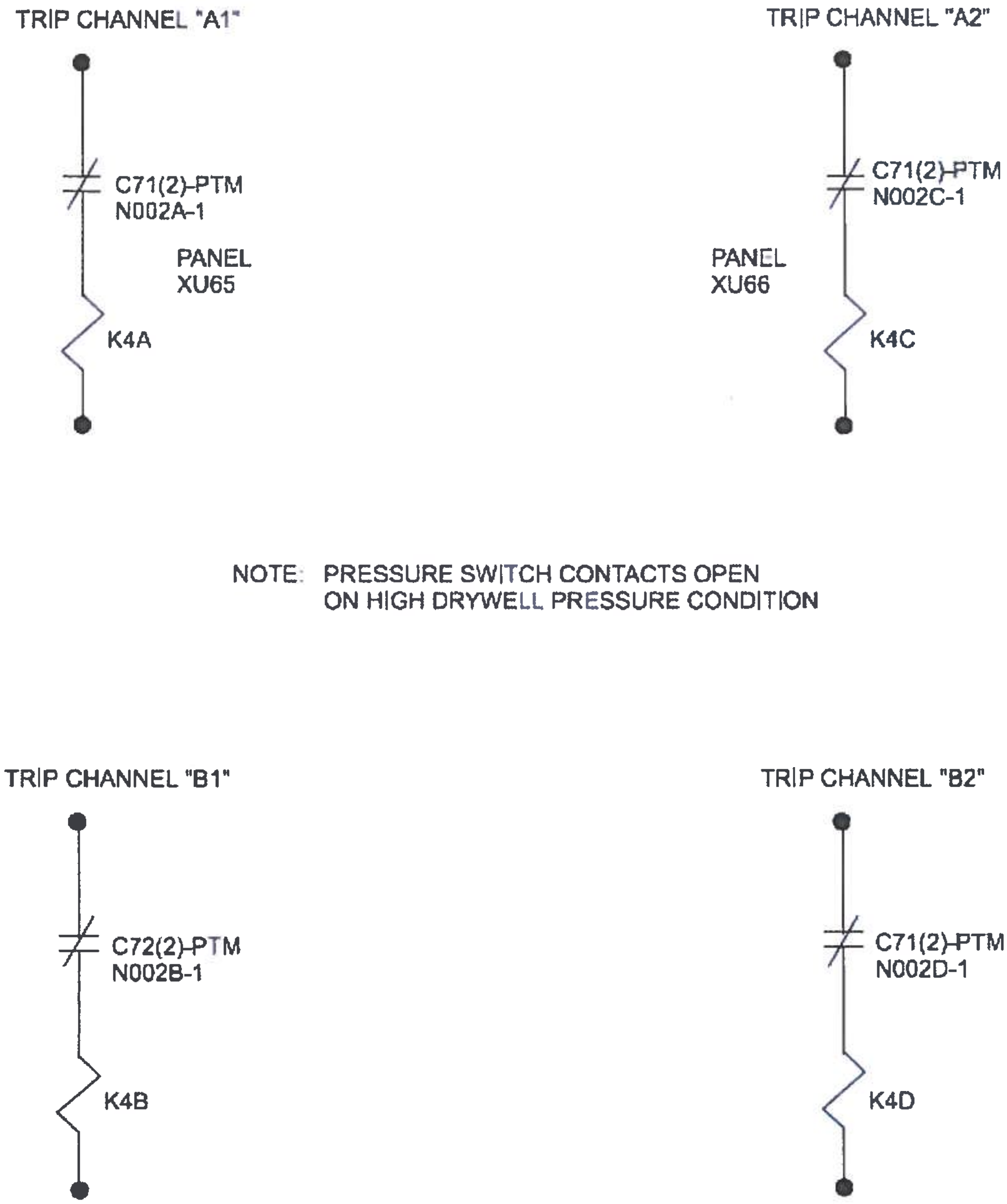
AUTOMATIC ACTIONS

1. If the primary containment pressure high trip signal is received in only one RPS Trip System, a half Scram will occur.
2. If the primary containment pressure high trip signal is received in both RPS Trip Systems, a reactor Scram will occur.

DEVICE/SETPOINTS

Relay C72-K4A-D	Deenergized
Pressure Switch C72-PTM-N002A-1, B-1, C-1, or D-1	1.7 psig

FIGURE 03-15
High Drywell Pressure Trip



49. 295025 1

Unit One was operating at power when a turbine trip occurred.
85 control rods fail to insert.
Reactor pressure peaks at 1145 psig.

Which one of the following completes both statements below?

The reactor recirc pumps ___(1)___ tripped.

Tripping of the reactor recirc pumps results in a rapid decrease in reactor power due to ___(2)___.

A. (1) must be manually
   (2) voiding of the moderator

B. (1) must be manually
   (2) a reduction in reactor water level

C. (1) have automatically
   (2) voiding of the moderator

D. (1) have automatically
   (2) a reduction in reactor water level

Answer: C

K/A:
295025 High Reactor Pressure
EK3 Knowledge of the reasons for the following responses as they apply to HIGH REACTOR PRESSURE: (CFR: 41.5 / 45.6)
02 Recirculation pump trip

RO/SRO Rating: 3.9/4.1

Tier 1 / Group 1

K/A Match: This meets the K/A because it is testing the reason the recirc pump is tripped.

Pedigree: Bank

Objective: LOI-CLS-LP-002, Obj. 30
Given Plant conditions determine if the ATWS-RPT protection logic should have actuated

Reference: None

Cog Level: Fundamental

Explanation: The Anticipated Transient Without Scram circuit provides an alternate means of reducing reactor power in the unlikely event that the control rods fail to insert into the core following a Reactor Protection System actuation signal. Tripping of the VFD Input Circuit Breakers (ICB) will rapidly reduce recirculation flow. This results in a rapid decrease in reactor power because of the voiding of the moderator. Setpoints for ATWS trip are high reactor pressure 1137.8 psig and low reactor level LL2 105"

Distractor Analysis:

Choice A: Plausible because the ATWS procedure directs the pumps to be tripped and the second part is correct.

Choice B: Plausible because the ATWS procedure directs the pumps to be tripped and level is reduced in the ATWS procedure which does lower power.

Choice C: Correct Answer, see explanation.

Choice D: Plausible because the first part is correct and level is reduced in the ATWS procedure which does lower power.

SRO Basis: N/A

3.2.6 Anticipated Transient Without Scram Recirculation Pump Trip (ATWS-RPT)

The Anticipated Transient Without Scram circuit provides an alternate means of reducing reactor power in the unlikely event that the control rods fail to insert into the core following a Reactor Protection System actuation signal. Tripping of the VFD Input Circuit Breakers (ICB) will rapidly reduce recirculation flow. This results in a rapid decrease in reactor power because of the voiding of the moderator.

Two signals are used for the initiation of ATWS-RPT. These signals are LL2 reactor vessel water level and high reactor vessel pressure. Each of these parameters is monitored by four sensors. Two level or pressure instruments in one of two logic trains are required to energize relays which trip both Recirculation Pumps.

50. 295026 1

Unit One failed to scram following a loss of off-site power with the following plant conditions:

Reactor Power	5%
RPV Water Level	-55 inches (N036)
RPV Pressure	850 psig

Which one of the following completes both statements below?

This UA-12 (5-4) alarm is expected to be received when suppression pool water temperature **first** reaches ___(1)___.

IAW 1OP-17, *Residual Heat Removal System Operating Procedure*, the RHR logic requirements to place torus cooling in service under the current plant conditions will require ___(2)___.

A. (1) 95°F
   (2) placing the CS-S17B "Think Switch" to Manual first and then bypassing the 2/3rd core height interlock

B. (1) 95°F
   (2) bypassing the 2/3rd core height interlock first and then placing the CS-S17B "Think Switch" to Manual

C. (1) 105°F
   (2) placing the CS-S17B "Think Switch" to Manual first and then bypassing the 2/3rd core height interlock

D. (1) 105°F
   (2) bypassing the 2/3rd core height interlock first and then placing the CS-S17B "Think Switch" to Manual

Answer: B

K/A:

295026 Suppression Pool High Water Temperature

G2.4.50Ability to verify system alarm setpoints and operate controls identified in the alarm response manual. (CFR: 41.10 / 43.5 / 45.3)

RO/SRO Rating: 4.2/4.0

Tier 1 / Group 1

K/A Match: This meets the K/A because it is testing when the torus temperature alarm setpoint and what controls need to be operated to establish cooling.

Pedigree: New

Objective: LOI-CLS-LP-017, Obj 09
Given an RHR pump or valve, list the interlocks, permissives and/or automatic actions associated with the RHR pump or valve, including setpoints.

Reference: None

Cog Level: High

Explanation: LOCA signal is sealed in due to being less than LL3 (45 inches) RPV water level is less than 2/3rd core height (-47 inches) therefore the keylock switch and then the Think switch is required (sequencing is essential). When the torus reaches 95°F this alarm will come in, 105°F is the TMax alarm.

Distractor Analysis:

Choice A: Plausible because the first part is correct and the second part is opposite of the required actions.

Choice B: Correct Answer, see explanation.

Choice C: Plausible because this is the alarm setpoint for the *SPTMS DIV I BULK WTR TEMP SETPT TMAX,* and the second part is opposite of the required actions.

Choice D: Plausible because this is the alarm setpoint for the *SPTMS DIV I BULK WTR TEMP SETPT TMAX,* and the second part is correct

SRO Basis: N/A

SPTMS DIV I BULK WTR TEMP SETPOINT TS1

NOTE:	Inoperability of this annunciator may result in a TRM Required Compensatory Measure.

AUTO ACTIONS

NONE

CAUSE

1. High suppression pool bulk average water temperature.

OBSERVATIONS

1. Recorder Channel 1 on CAC-TR-4426-1A indicates increasing suppression pool temperature.
2. TS1 indicator illuminated (CAC-TY-4426-1).

ACTIONS

1. If suppression pool temperature is approaching 95°F and no testing is in progress that could add heat to the suppression pool, then refer to AOP-14.0, Abnormal Primary Containment Conditions and AOP-30.0, Safety/Relief Valve Failures.
2. If suppression pool temperature is greater than 95°F due to adding heat to the suppression pool from approved testing procedures, then refer to the appropriate test procedure to maintain suppression pool temperature below 105°F.
3. If suppression pool temperature is greater than 95°F and no testing is in progress that could add heat to the suppression pool, then enter EOP-02-PCCP, Primary Containment Control, and AOP-14.0, Abnormal Primary Containment Conditions.
4. If a circuit or equipment malfunction is suspected, ensure that a WR/WO is prepared.

DEVICE/SETPOINTS

SPTMS Microprocessor CAC-TY-4426-1	95°F

SPTMS DIV I BULK WTR TEMP SETPT TMAX

AUTO ACTIONS

NONE

CAUSE

1. High suppression pool bulk average water temperature.

OBSERVATIONS

1. Recorder Channel 1 on CAC-TR-4426-1A indicates increasing suppression pool temperature.
2. TMAX indicator illuminated (CAC-TY-4426-1).

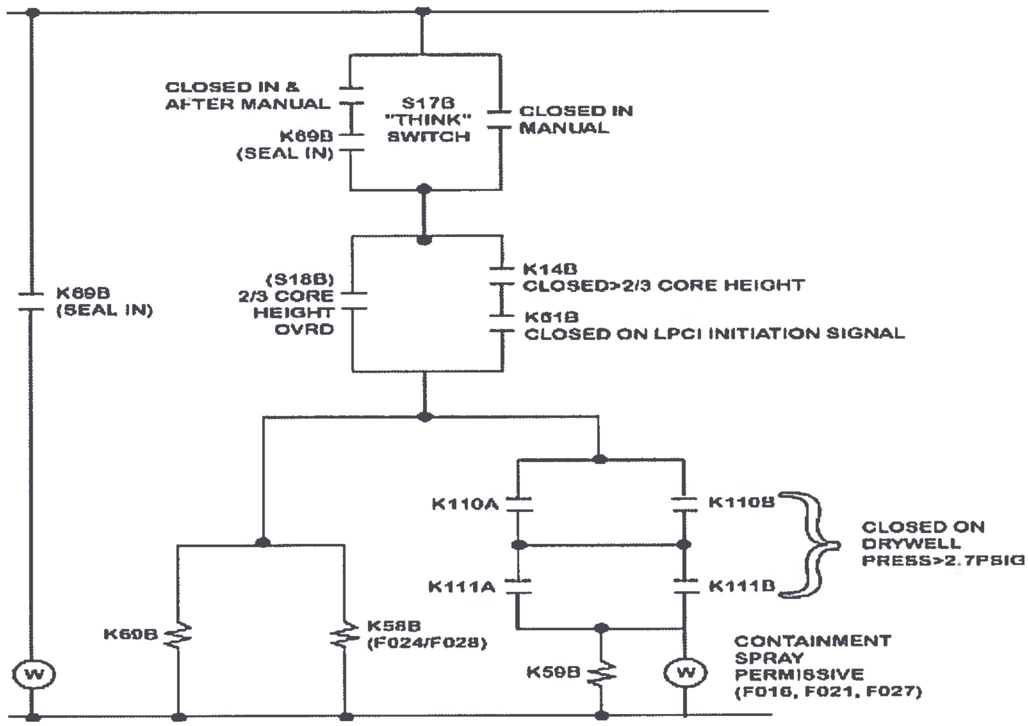
ACTIONS

1. If suppression pool temperature is greater than 95°F and no testing is in progress that could add heat to the suppression pool, then enter EOP-02-PCCP, Primary Containment Control, and AOP-14.0, Abnormal Primary Containment Conditions, if not already entered.
2. If suppression pool temperature is approaching 105°F due to adding heat to the suppression pool from approved testing procedures, then refer to the appropriate test procedure to maintain suppression pool temperature below 105°F.
3. If suppression pool temperature is greater than 105°F, then stop all testing and enter EOP-02-PCCP, Primary Containment Control, and AOP-14.0, Abnormal Primary Containment Conditions.
4. If a circuit or equipment malfunction is suspected, then ensure that a WR/WO is prepared.

DEVICE/SETPOINTS

SPTMS Microprocessor CAC-TY-4426-1	105°F

FIGURE 17-12
Cooling/Spray Permissive Logic



51. 295028 1

Unit Two is in MODE 3 following a Station Blackout.

IAW 0EOP-01-SBO-01, *Plant Monitoring*, the AO has reported the following temperatures from the RSDP temperature recorder 2CAC-TR-778:

Point 1	290°F
Point 2	118°F
Point 3	255°F
Point 4	230°F
Point 5	191°F
Point 6	117°F

(REFERENCE PROVIDED)

Which one of the following represents the correct calculated Drywell temperature?

A. ~205°F

B. ~249°F

C. ~258°F

D. ~267°F

Answer: B

K/A:

295028 High Drywell Temperature

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

01 Drywell temperature

RO/SRO Rating: 4.0/4.1

Tier 1 / Group 1

K/A Match: This meets the K/A because it is testing the students ability to determine drywell temperature.

Pedigree: Bank

Objective: LOI-CLS-LP-303-B, Obj. 3
Given plant conditions, control room or remote shutdown panel indications, and SBO-04, calculate the following parameters: a. Drywell Temperature

Reference: Attachment 4 of 0EOP-01-SBO-01, Plant Monitoring

Cog Level: Fundamental

Explanation: Attachment 4 of 0EOP-01-SBO-01, Plant Monitoring, has a calculation worksheet for figuring Drywell temperature from RSDP temperature recorder readings.

$$\begin{aligned} 290 * 0.141 &= 40.89 \\ 255 * 0.404 &= 103.02 \\ 230 * 0.455 &= \underline{104.65} \\ & \quad 248.56 \end{aligned}$$

Distractor Analysis:

Choice A: Plausible because this is the average of points 1 - 3 used in calculation.

Choice B: Correct Answer, see explanation.

Choice C: Plausible because this is the average of points 1, 3, & 4.

Choice D: Plausible because this is performing the calculation backwards (points 4, 3, 1)

SRO Basis: N/A

Drywell Temperature Calculation Using RSDP Recorder Inputs

Values obtained from Recorder CAC-TR-778

Above 70' Elevation

PT 1 _290______ x 0.141 = _40.89__ °F

Between 28' and 45' Elevation

PT 3 _255______ x 0.404 = _103.02__ °F

Between 10' and 23' Elevation

PT 4 _230______ x 0.455 = _104.65__ °F

Average Drywell Temperature _248.56______ °F
(Sum of 3 Regional Weighted Areas)

52. 295029 1

Unit Two is performing RVCP with HPCI in pressure control.

Subsequently, A-01 (1-5) *Suppression Chamber Level Hi Hi* is received.

Which one of the following completes both statements below?

The E41-F004, CST Suction Vlv, will ___(1)___.

The E41-F008, Bypass to CST Vlv, will ___(2)___.

A. (1) close
   (2) close

B. (1) close
   (2) remain open

C. (1) remain open
   (2) close

D. (1) remain open
   (2) remain open

Answer: A

K/A:

295029 High Suppression Pool Water Level

EA1 Ability to operate and/or monitor the following as they apply to HIGH SUPPRESSION POOL WATER LEVEL: (CFR: 41.7 / 45.6)

01 HPCI

RO/SRO Rating: 3.4/3.5

Tier 1 / Group 2

K/A Match: This meets the K/A because it is testing operation of HPCI on high torus level

Pedigree: New

Objective: LOI-CLS-LP-019. Obj. 3p
Given plant conditions, predict how the HPCI System will respond to the following events: High/low Suppression Pool water level

Reference: None

Cog Level: High

Explanation: The torus water high level condition (> -25 inches) will cause the torus suction valves to open. When either valve is full open the F008 and F004 will close.

Distractor Analysis:

Choice A: Correct Answer, see explanation.

Choice B: Plausible because the first part is correct. The F008 will get a close signal when the F041 or F042 is full open.

Choice C: Plausible because the high/low level alarm does not affect the valves (annunciation only). The second part is correct.

Choice D: Plausible because the high/low level alarm does not affect the valves (annunciation only).

SRO Basis: N/A

FIGURE 19-7
CST Suction Valve, E41-F004, Control Logic

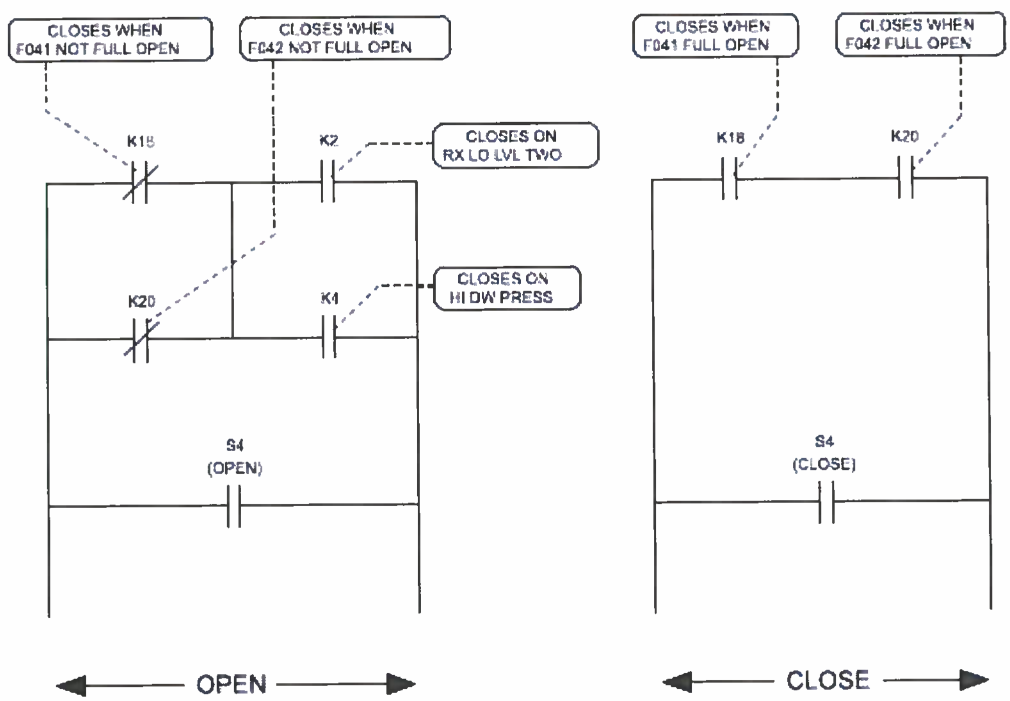
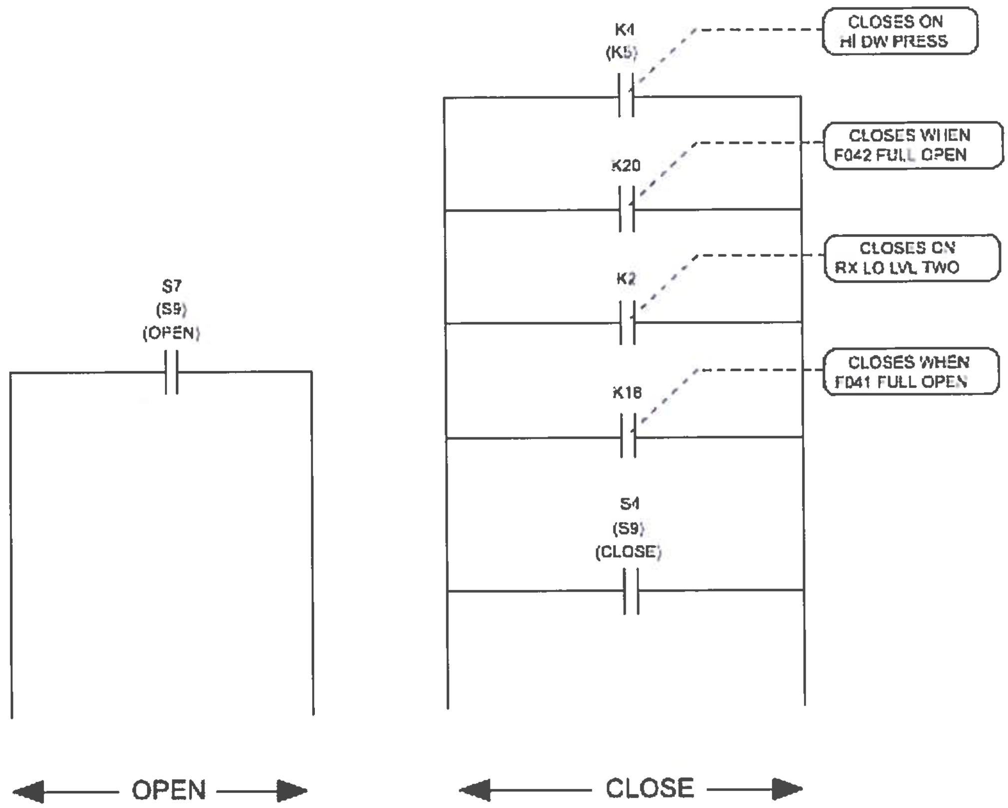
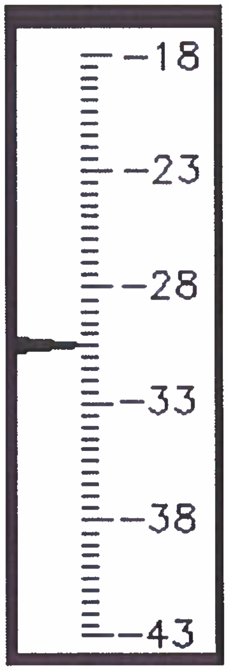


FIGURE 19-15
Test Return Isolation Valve, E41-F008 (E41-F011) Control Logic



53. 295030 1



Unit One is operating at rated power when A-01 (3-7) *Suppression Chamber Lvl Hi/Lo,* is received.

The BOP Operator verifies the alarm using CAC-LI-4177, Supp Pool Level, indicator on Panel XU-51. (indication provided to the left)

Which one of the following identifies the action that is required IAW A-01 (3-7) *Suppression Chamber Lvl Hi/Lo*?

The water level in the Unit One torus must be:

A. lowered by using Core Spray and routed to Radwaste.

B. lowered using RHR and routed to Radwaste.

C. raised by opening the HPCI suction from the CST.

D. raised by opening the Core Spray suction from the CST.

Answer: D

K/A:
295030 Low Suppression Pool Water Level
G2.4.50Ability to verify system alarm setpoints and operate controls identified in the alarm response manual. (CFR: 41.10 / 43.5 / 45.3)

RO/SRO Rating: 4.2/4.0

Tier 1 / Group 1

K/A Match: This meets the K/A because it is testing ability to know whether the alarm is due to high or low level and knowledge of how to correct.

Pedigree: New

Objective: LOI-CLS-LP-302-D, Obj 2
Given plant conditions and AOP-14.0, determine the required supplementary actions.

Reference: None

Cog Level: High

Explanation: The student will verify that level is low using the provided indication and then will determine that the level must be raised IAW the APP. The low level alarm comes in at -30.5 inches and the high level alarm comes in at -27.5 inches. Level can be raised using RHR or the Core Spray systems.

Distractor Analysis:

Choice A: Plausible because it is a combined alarm and if it is assumed that a high water level condition exists the CS system can take a suction from the torus to correct the level condition, but is not allowed in the procedure.

Choice B: Plausible because it is a combined alarm and if it is assumed that a high water level condition exists the RHR system is utilized in the procedure to lower level.

Choice C: Plausible because level is low requiring it to be raised and the HPCI system could gravity drain to the torus, but is not allowed by the procedure.

Choice D: Correct Answer, see explanation.

SRO Basis: N/A

SUPPRESSION CHAMBER LVL HI/LO

AUTO ACTIONS

NONE

CAUSE

1. Suppression pool water level high (-27½ inches)
2. Suppression pool water level low (-30½ inches)
3. Circuit malfunction

OBSERVATIONS

1. Suppression pool water level (CAC-LI-2601-1, CAC-LI-4177, CAC-LR-2602)

NOTE:	Rapid changes in suppression pool pressure due to conditions such as inerting or air in-leakage can cause level fluctuations in suppression pool up to 1 inch or more.

ACTIONS

NOTE:	ECCS keepfill stations makeup flow to the suppression pool is approximately 27 gpm.

1. If the cause of the annunciator is a planned evolution, then refer to the appropriate operating procedure to maintain suppression pool water level.
2. If the cause of the annunciator is not a planned evolution, then determine the cause of addition or loss of water to suppression pool and minimize evolutions which add or remove water to or from the suppression pool.
3. If suppression pool water level is high or low, then enter 0AOP-14.0 to drain or fill the suppression pool as necessary.
4. If suppression pool water level is greater than -27 inches or less than -31 inches, then enter 0EOP-02-PCCP.
5. If a circuit malfunction is suspected, ensure a WO is prepared.

4.2.4 Suppression Pool Level High/Low

1. IF suppression pool level is approaching -27 inches, THEN **lower** suppression pool level to Radwaste in accordance with 1OP-17(2OP-17), Residual Heat Removal System Operating Procedure. ☐

2. IF suppression pool level is approaching -31 inches, THEN **raise** suppression pool level in accordance with the following applicable procedure: ☐

   Unit 1 Only:

   - 1OP-17, Residual Heat Removal System Operating Procedure ☐
   - 1OP-18, Core Spray System Operating Procedure ☐

   Unit 2 Only:

   - 2OP-17, Residual Heat Removal System Operating Procedure ☐
   - 2OP 18, Core Spray System Operating Procedure ☐

54. 295031 1

Unit One is executing the ATWS procedure with the following plant conditions:

Reactor power	12%
Reactor pressure	940 psig, controlled by EHC
Reactor water level	170 inches, controlled by feedwater

Which one of the following identifies the reason the ATWS procedure directs deliberately lowering RPV water level to 90 inches?

A. Reduces reactor power so that it will remain below the APRM downscale setpoint.

B. Provides heating of the feedwater to reduce potential for high core inlet subcooling.

C. Reduces challenges to primary containment if MSIVs close.

D. Promotes more efficient boron mixing in the core region.

Answer: B

K/A:

295031 Reactor Low Water Level

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

03 Water level effects on reactor power

RO/SRO Rating: 3.7/4.1

Tier 1 / Group 1

K/A Match: This meets the K/A because it is testing the knowledge of why level is lowered in a ATWS

Pedigree: Bank

Objective: LOI-CLS-LP-300-E, Obj 7
Explain the reason for lowering reactor water level while performing the Anticipated Transient Without Scram Procedure.

Reference: None

Cog Level: fundamental

Explanation: To prevent or mitigate the consequences of any large irregular neutron flux oscillations induced by neutronic/thermal-hydraulic instabilities, reactor water level is lowered sufficiently below the elevation of the feedwater sparger nozzles. This places the feedwater spargers in the steam space providing effective heating of the relatively cold feedwater and eliminating the potential for high core inlet subcooling. For conditions that are susceptible to oscillations, the initiation and growth of oscillations is principally dependent upon the subcooling at the core inlet; the greater the subcooling, the more likely oscillations will commence and increase in magnitude. Twenty-four inches below the lowest nozzle in the feedwater sparger (i.e. 90 inches) has been selected as the upper bound of the reactor water level control band. This water level is sufficiently low that steam heating of the injected water will be at least 65% to 75% effective (i.e., the temperature of the injected water will be increased to 65% to 75% of its equilibrium value in the steam environment). This water level is sufficiently high that even without bypassing the low reactor water level MSIV isolation, reactor water level can be controlled with the feedwater pumps to preclude the isolation.

Distractor Analysis:

Choice A: Plausible because since the operator can re-establish injection at 90 inches irrespective of power level. Power will lower as level is lowered but 90 inches will not guarantee APRMs are downscale

Choice B: Correct Answer, see explanation.

Choice C: Plausible because since there is no current challenge to containment from heat input. If level is lowered due to containment heat input, 90 inches is not specified as the top of the level band. This would be either TAF or the level at which downscales are received

Choice D: Plausible because since lowering level will reduce natural circulation and reduce boron mixing. ATWS procedure directs raising level back to the normal band (170-200 inches) once hot shutdown boron weight is injected

SRO Basis: N/A

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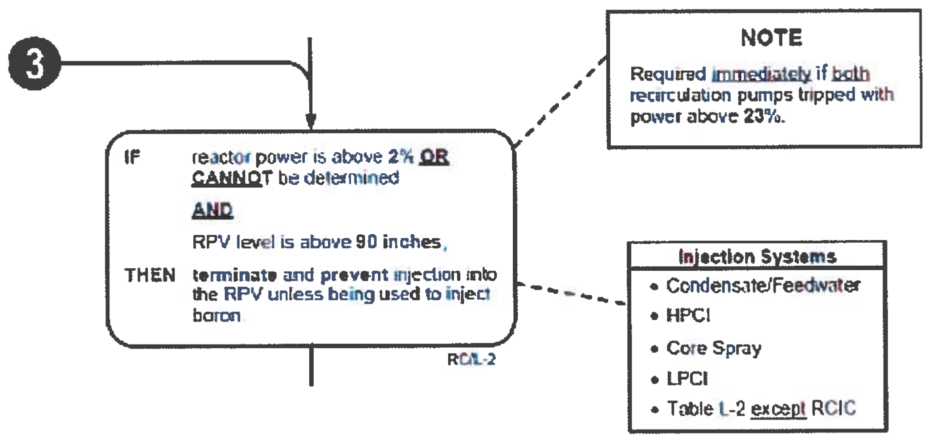
5.4 **Step RC/L-2**



If reactor power is greater than 23% with both reactor recirculation pumps tripped and RPV level above 90 inches, RPV level needs to be promptly reduced below the feedwater nozzles, to avoid thermal hydraulic instabilities. This is accomplished by termination and prevention of injection systems, from identified systems, particularly feedwater, within 120 seconds.

To prevent or mitigate the consequences of any large irregular neutron flux oscillations induced by neutronic/thermal-hydraulic instabilities, RPV level is initially lowered sufficiently below the elevation of the feedwater sparger nozzles. This places the feedwater spargers in the steam space providing effective heating of the relatively cold feedwater and eliminating the potential for high core inlet subcooling. For conditions that are susceptible to oscillations, initiation and growth of oscillations is principally dependent upon subcooling at the core inlet; the greater the subcooling, the more likely oscillations will commence and increase in magnitude.

55. 295032 1

Which one of the following identifies the reason for performing Emergency Depressurization due to exceeding Maximum Safe Operating Temperatures IAW 0OI-37.9, *Secondary Containment Control Procedure Basis Document*?

A. Prevent an unmonitored release.

B. Preserve personnel access into the reactor building.

C. Provide continued operability of equipment required for safe shutdown.

D. Ensure ODCM site boundary dose limits are not exceeded.

Answer: C

K/A:

295032 High Secondary Containment Area Temperature

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

01 Emergency/normal depressurization

RO/SRO Rating: 3.5/3.8

Tier 1 / Group 2

K/A Match: This meets the K/A because it is testing the reason ED is performed for high secondary containment temperatures.

Pedigree: Bank

Objective: LOI-CLS-LP-300-M, Obj 13a
Given plant conditions and the SCCP, determine the required actions if the following limits are exceeded: Maximum Safe operating values with a primary system discharging into secondary containment.

Reference: None

Cog Level: Fundamental

Explanation: The MSOT values are the area temperatures above which equipment necessary for the safe shutdown of the plant will fail. These area temperatures are utilized in establishing the conditions which reactor depressurization is required. The criteria of more than one area specified in this step identifies the rise in reactor building parameters as a wide spread problem which may pose a direct and immediate threat to secondary containment integrity, equipment located in the RB, and continued safe operation of the plant.

Distractor Analysis:

Choice A: Plausible because this is a purpose of SCCP not the reason for ED on Temperature.

Choice B: Plausible because this is the reason for max safe operating rad levels.

Choice C: Correct Answer, see explanation.

Choice D: Plausible because this is a purpose of SCCP not the reason for ED on Temperature.

SRO Basis: N/A

5.1 Step SCCP-1

The conditions which require entry to SCCP are symptomatic of conditions which, if not corrected, could degrade into an emergency. Adverse effects on the operability of equipment located in the reactor building and conditions directly challenging secondary containment integrity or spent fuel pool cooling were specifically considered in the selection of these entry conditions. In addition, personnel accessibility to some of the areas may be required to perform certain actions specified in the procedure. This was also considered in making these determinations.

An area temperature or area differential temperature above its maximum normal operating level is an indication that steam from a primary system may be discharging into the reactor building. As temperatures continue to increase, the continued operability of equipment needed to carry out EOP actions may be compromised.

56. 295034 1

Which one of the following completes both statements below?

IAW 0AOP-5.4, *Radiological Releases*, RRCP is entered when the Turbine Building Vent Rad Monitor indication exceeds an ___(1)___ EAL.

IAW RRCP, before the radioactivity release rate reaches a ___(2)___ Emergency EAL, Emergency Depressurization is required.

A. (1) Unusual Event
   (2) Site Area

B. (1) Unusual Event
   (2) General

C. (1) Alert
   (2) Site Area

D. (1) Alert
   (2) General

Answer: D

K/A:

295034 Secondary Containment Ventilation High Radiation

G2.4.08 Knowledge of how abnormal operating procedures are used in conjunction with EOPs. (CFR: 41.10 / 43.5 / 45.13)

RO/SRO Rating: 3.8/4.5

Tier 1 / Group 2

K/A Match: This question matches the KA because it tests the knowledge if the AOP and EOP are performed in conjunction with each other.

Pedigree: new

Objective: LOI-CLS-LP-302-J, Obj. 3c
Given plant conditions, determine the required Supplementary Actions in accordance with: 0AOP-05.4, Radiological Release

Reference: None

Cog Level: Fundamental

Explanation: The AOP states that when an Alert EAL is entered then ENTER RRCP. Before a GE is declared ED is required to be performed. (A scram is required before a SAE is declared)

Distractor Analysis:

Choice A: Plausible because an Unusual Event is the first declaration in the EAL network and a SAE is the criteria for a scram in RRCP.

Choice B: Plausible because an Unusual Event is the first declaration in the EAL network and the second part is correct.

Choice C: Plausible because the first part is correct and the SAE is the criteria for a scram in RRCP.

Choice D: Correct Answer, see explanation

SRO Basis: N/A

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3.0 AUTOMATIC ACTIONS (continued)

- SBGT starts ☐
- Group 6 isolation valves close ☐

4. IF UA-03 2-8, Radwaste Effluent Rad Hi Hi, in ALARM, THEN D12-V27A(B) (RW Liq Effluent Disch Vlvs) close ☐

5. IF UA-23 3-6, Main Steam Line Rad Hi-Hi/Inop, in ALARM, THEN:
   - Mechanical vacuum pumps trip ☐
   - OG-V7 (Cndsr Hogging Valve) closes ☐

4.0 OPERATOR ACTIONS

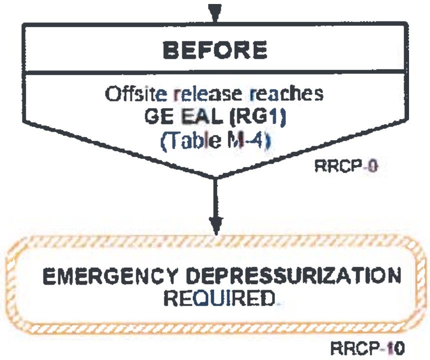
4.1 Immediate Actions

None

4.2 Supplementary Actions

1. IF AT ANY TIME elevated radiation levels are determined to be from resin injection only, THEN go to 0AOP-26.0, High Reactor Coolant or Condensate Conductivity. ☐

2. IF AT ANY TIME gaseous release rate exceeds an Alert level, THEN enter 0EOP-04-RRCP, Radioactivity Release Control Procedure. ☐


BEFORE
Offsite release reaches
GE EAL (RG1)
(Table M-4)
RRCP-0
EMERGENCY DEPRESSURIZATION
REQUIRED
RRCP-10

57. 295036 1

Following an unisolable RWCU line break in the reactor building the following conditions exist:

South Core Spray Room temperature 155°F
South RHR Room temperature 300°F
UA-12 (2-3) *South Core Spray Room Flood Level Hi,* in alarm
UA-12 (2-4) *South RHR Room Flood Level Hi,* in alarm
UA-12 (1-4) *South RHR Room Flood Level Hi-Hi,* in alarm

(REFERENCE PROVIDED)

Which one of the following completes both statements below?

IAW 0EOP-01-UG, *User's Guide,* ___(1)___ equipment required for safe shutdown will fail.

IAW SCCP, Emergency Depressurization ___(1)___ required.

A. (1) ONLY the South RHR room
   (2) is

B. (1) ONLY the South RHR room
   (2) is NOT

C. (1) the South RHR room AND Core Spray room
   (2) is

D. (1) the South RHR room AND Core Spray room
   (2) is NOT

Answer: B

K/A:

295036 Secondary Containment High Sump / Area Water Level

EK1 Knowledge of the operational implications of the following concepts as they apply to SECONDARY CONTAINMENT HIGH SUMP/AREA WATER LEVEL: (CFR: 41.8 to 41.10)

02 Electrical ground/ circuit malfunction

RO/SRO Rating: 2.6/2.8

Tier 1 / Group 2

K/A Match: This meets the K/A because this is testing the implication of high water level on equipment and whether ED is required.

Pedigree: New

Objective: LOI-CLS-LP-300-M, Obj, 13a
Given plant conditions and the Secondary Containment Control Procedure, determine the required action if the following limits are exceeded: Maximum Safe operating values WITH a primary system discharging into Secondary Containment

Reference: 0EOP-01-NL, *EOP/SAMG Numerical Limits And Values,* Attachment 3, *Containment Parameters*, Table 3-B, *Secondary Containment Area Temperature Limits*

Cog Level: High

Explanation:

Distractor Analysis:

Choice A: Plausible because the first part is correct and for ED two areas in the same parameter must be at max safe conditions, while this question has two parameters in the same area.

Choice B: Correct Answer, see explanation.

Choice C: Plausible because both areas have a max normal condition and for ED two areas in the same parameter must be at max safe conditions, while this question has two parameters in the same area.

Choice D: Plausible because both areas have a max normal condition and the second part is correct.

SRO Basis: N/A

3.0 DEFINITIONS (continued)

- Core Spray Loop A
- Core Spray Loop B
- RHR Loop A (one or two pumps running)
- RHR Loop B (one or two pumps running)

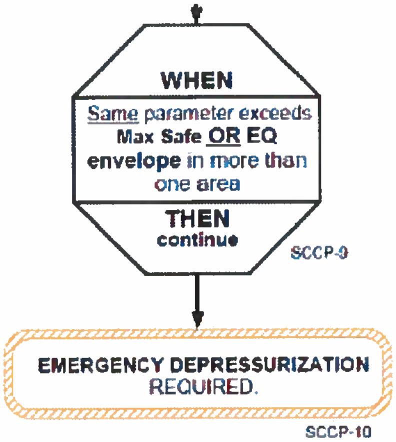
32. **Maximum Normal Operating (Parameter):** The highest value of the identified parameter expected to occur during normal plant operating conditions with all directly associated support and control systems functioning properly.

33. **Maximum Pressure Suppression Primary Containment Water Level:** The highest primary containment water level at which the pressure suppression capability of the containment can be maintained. This corresponds to the bottom of the ring header.

34. **Maximum Safe Operating Radiation Level:** The radiation level above which personnel access necessary for the safe shutdown of the plant will be precluded. If the maximum safe operating radiation level is exceeded in an area (but is within the EQ envelope as contained in DR-227, Document Reference for Environmental Qualification Service Conditions) and then later clears and is subsequently followed by another area exceeding maximum safe operating radiation level, action for one area exceeding maximum safe operating radiation level should be taken.

35. **Maximum Safe Operating Temperature:** The temperature above which equipment necessary for the safe shutdown of the plant may fail. This temperature is utilized in establishing the conditions under which RPV depressurization is required. Separate temperatures are provided for each Secondary Containment area. If the maximum safe operating temperature is exceeded in an area and then later clears and is subsequently followed by another area exceeding maximum safe operating temperature, action for two areas exceeding maximum safe operating temperature should be taken.

36. **Maximum Safe Operating Water Level:** The water level above which equipment necessary for the safe shutdown of the plant may fail. This water level is utilized in establishing the conditions under which RPV depressurization is required. Separate water levels are provided for each Secondary Containment area. If the maximum safe operating water level is exceeded in an area and then later clears and is subsequently followed by another area exceeding maximum safe operating water level, action for two areas exceeding maximum safe operating water level should be taken.



Containment Parameters

Secondary Containment Area Temperature Limits

Table 3-B

PLANT AREA	PLANT LOCATION DESCRIPTION	MAX NORM OPERATING VALUE (°F)	MAX SAFE OPERATING VALUE (°F)	AUTO GROUP ISOLATION
N CORE SPRAY	N CORE SPRAY ROOM	120	175	N/A
S CORE SPRAY	S CORE SPRAY ROOM	120	175	N/A
RWCU	PMP ROOM A PMP ROOM B HX ROOM	140	225	3
N RHR	N RHR EQUIP ROOM	175	295	N/A
S RHR	S RHR EQUIP ROOM RCIC EQUIP ROOM	175 165	295 295	N/A 5
HPCI	HPCI EQUIP ROOM	165	165	4
STEAM TUNNEL	RCIC STM TUNNEL HPCI STM TUNNEL	190 190	295 295	5 4
20 FT	20 FT NORTH 20 FT SOUTH	140 140	200 200	N/A N/A
50 FT	50 FT NW 50 FT SE	140 140	200 200	N/A N/A
REACTOR BLDG	MULTIPLE AREAS ANNUN. A-02 5-7	ALARM SETPOINT	N/A	3, 4, AND/OR 5
REACTOR BLDG	MSIV PIT ANNUN. A-06 6-7	ALARM SETPOINT	N/A	1

58. 295037 1

The RO has attempted to manually scram Unit One with the following actions taken:

- All rods are noted to be greater than position 02
- Reactor mode switch is placed in shutdown
- ARI was initiated.
- Both recirculation pumps were tripped.
- Reactor power reported at 12%
- SLC is injecting
- RPV level is 80 inches and stable
- Rod insertion attempts are unsuccessful

Which one of the following completes both statements below?

Reactor power ___(1)___ expected to be lowering.

Assuming no rod insertion, SLC injection ___(2)___.

A. (1) is
(2) can be secured when all APRMs are downscale

B. (1) is
(2) must be continued until the reactor is shutdown under all conditions

C. (1) is NOT
(2) can be secured when all APRMs are downscale

D. (1) is NOT
(2) must be continued until the reactor is shutdown under all conditions

Answer: B

K/A:

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

EK1 Knowledge of the operational implications of the following concepts as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN : (CFR: 41.8 to 41.10)

03 Boron effects on reactor power (SBLC)

RO/SRO Rating: 4.2/4.4

Tier 1 / Group 1

K/A Match: This meets the K/A because the student will have to know the effects that boron has shutting down the reactor during an ATWS.

Pedigree: Bank

Objective: LOI-CLS-LP-005, Obj 3
List the positive reactivity effects that must be overcome by SLC injection

Reference: None

Cog Level: Fundamental

Explanation: Injection of the CSBW into the RPV will provide adequate assurance that the reactor is and will remain shutdown. It is the least weight of soluble boron which, if injected into the RPV and mixed uniformly, will maintain the reactor shutdown under all conditions. This weight is utilized to assure the reactor will remain shutdown irrespective of control rod position or RPV water temperature. Boron injection is continued until the entire tank is injected or all rods are inserted.

Distractor Analysis:

Choice A: Plausible because the first part is correct and the second part is plausible because in some procedures the reactor is called shutdown if power is downscale on the APRM's.

Choice B: Correct Answer, see explanation

Choice C: Plausible because rods are not being inserted and the second part is plausible because in some procedures the reactor is called shutdown if power is downscale on the APRM's.

Choice D: Plausible because rods are not being inserted and the second part is correct.

SRO Basis: N/A

1.2 System Design Basis

The design basis for the SLC System is as follows:

1.2.1 Backup capability for reactivity control is provided, independent of the normal reactivity control provisions in the nuclear reactor, to permit shutdown of the reactor if the normal control ever becomes inoperative.

1.2.2 To assure complete shutdown from the most reactive condition at any time in core life, this backup system has the capacity to control the reactivity difference between the steady state rated operating condition of the reactor with voids and the cold shutdown condition, including shutdown margin.

1.2.3 The time required to actuate and effect the backup control is consistent with the nuclear reactivity rate of change predicted between rated operating and cold shutdown conditions. A fast scram of the reactor or operational control of fast reactivity transients is not specified for this system.

1.2.4 Means are provided by which the functional performance capability of the backup control system components can be verified under conditions approaching actual use requirements.

1.2.5 The neutron absorber is dispersed within the reactor core in sufficient quantity to provide a reasonable margin for leakage, dilution, or

59. 295038 1

A radioactive release has occurred in the Turbine Building.

Which one of the following completes both statements below?

IAW 0AOP-05.4, *Radiological Releases*, the Unit Two turbine building ventilation must be in the ___(1)____ operating mode.

This discharge will be monitored by the __(2)__.

A. (1) recirc
   (2) Main Stack Radiation Monitor

B. (1) recirc
   (2) Wide Range Gaseous Monitor (WRGM)

C. (1) once through
   (2) Main Stack Radiation Monitor

D. (1) once through
   (2) Wide Range Gaseous Monitor (WRGM)

Answer: B

K/A:

295038 High Off-Site Release Rate

EA1 Ability to operate and/or monitor the following as they apply to HIGH OFF-SITE RELEASE RATE: (CFR: 41.7 / 45.6)

01 Stack-gas monitoring system

RO/SRO Rating: 3.9/4.2

Tier 1 / Group 1

K/A Match: This meets the K/A because the student will have to determine the procedural requirement for the turbine building ventilation operational mode and the rad monitor that monitors it. (ability to monitor)

Pedigree: New

Objective: LOI-CLS-LP-302-J, Obj 3c
Given plant conditions, determine the required Supplementary Actions in accordance with:: c. 0AOP-05.4, Radiological Release

Reference: None

Cog Level: Fundamental

Explanation: The turbine building ventilation can be lined up for once through or recirc mode, the AOP has the operator ensure that it is lined up in the recirc mode. The discharge is monitored by the turbine building WRGM.

Distractor Analysis:

Choice A: Plausible because the first part is correct and the second part is a common radiation monitor for other ventilation systms.

Choice B: Correct Answer, see explanation.

Choice C: Plausible because once through is a mode of operation for the TB Ventilation system and the second part is a common radiation monitor for other ventilation systms.

Choice D: Plausible because once through is a mode of operation for the TB Ventilation system and the second part is correct.

SRO Basis: N/A

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4.2 Supplementary Actions (continued)

NOTE

- Turbine Building habitability should be considered for establishment as critical parameters during performance of this procedure ☐
- Emergency Plan requirements mandate securing once through ventilation for any site radiological release. ☐

10. IF any site radiological release occurring, **THEN ensure** Unit 2 turbine building ventilation in recirculation mode per 2OP-37.3, Turbine Building Ventilation System Operating Procedure. ☐

60. 300000 1
Unit One is operating at rated power when the following alarms are received:

UA-01 (4-4) *Instr Air Press-Low*
UA-01 (5-1) *Air Dryer 1A Trouble*

The AO reports that the cause of the alarms is due to filter blockage.

Which one of the following completes both statements below?

The Service Air Dryer malfunction will cause SA-PV-5067, Service Air Dryer Bypass Valve, to open when pressure **first** lowers to ___(1)___.

IAW 0AOP-20.0, Pneumatic (Air/Nitrogen) System Failures, the required action is to ___(2)___.

A. (1) 105 psig
(2) place the 1B Service Air Dryer in service

B. (1) 105 psig
(2) set the service air dryer maximum sweep value to zero

C. (1) 98 psig
(2) place the 1B Service Air Dryer in service

D. (1) 98 psig
(2) set the service air dryer maximum sweep value to zero

Answer: C
K/A:
300000 Instrument Air System
A2 Ability to (a) predict the impacts of the following on the INSTRUMENT AIR SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation: (CFR: 41.5 / 45.6)
01 Air dryer and filter malfunctions

RO/SRO Rating: 2.9/2.8

Tier 2 / Group 1

K/A Match: This meets the KA because it is predicting the response on the system and then using procedure (AOP-20) determine the action required.

Pedigree: New

Objective: LOI-CLS-LP-046, Obj. 6
Given plant conditions, determine if the following automatic actions should occur:
a. Service Air Isolation g. Air Dryer bypass.

Reference: None

Cog Level: High

Explanation: 98 psig is when the bypass valve auto opens, the 105 psig is the isolation setpoint for Service Air. The AOP will direct placing the standby Air Dryer in service.

Distractor Analysis:

Choice A: Plausible because 105 is the isolation setpoint for the service air system and the second part is correct.

Choice B: Plausible because 105 is the isolation setpoint for the service air system and this is an action in the AOP but would not be performed for this failure. If there is a high demand then this is performed to limit the amount of air that is used for the blowdown of the air dryer filter when cycling filters.

Choice C: Correct Answer, see explanation.

Choice D: Plausible because the first part is correct and this is an action in the AOP but would not be performed for this failure. If there is a high demand then this is performed to limit the amount of air that is used for the blowdown of the air dryer filter when cycling filters.

SRO Basis: N/A

4.2 Supplementary Actions (continued)

NOTE

- Service Air System pre-filter or after-filter differential pressure should **NOT** exceed 15 psid. ☐
- In service air compressor high discharge pressure (**Unit 1**: greater than or equal to 125 psig, **Unit 2**: greater than or equal to 130 psig) or relief valves lifting could be an indication of air dryer high differential pressure potentially caused by power failures resulting in valves in the flow path failing closed. ☐
- 1(2)SA-PV-5067 [Serv Air Dryer 1(2)A Bypass Pressure Control Valve], is located in the Turbine Building air compressor area. ☐

i. **IF** UA-01 5-3, Air Dryer 1(2)A Trouble, is in alarm, **THEN perform** the following:

(1) **Unit 1 Only**: **Confirm** 1-SA-PV-5067 (Serv Air Dryer 1A Bypass Pressure Control Valve), is OPEN. ☐

CAUTION

The service air dryer provides a low dew point pneumatic source to downstream components. A low dew point is necessary to insure long term reliability of these components. The time the dryer is bypassed should be minimized {7.1.1}. ☐

(2) **Unit 1 Only**: **IF** 1-SA-PV-5067 is **NOT** open, **THEN open** 1-SA-V5089 (Serv Air Dryer Manual Bypass Valve). ☐

(3) **Unit 2 Only**: **Confirm** 2-SA-PV-5067 (Serv Air Dryer 2A Bypass Pressure Control Valve), is OPEN. ☐

(4) **Unit 2 Only**: **IF** 2-SA-PV-5067 is **NOT** open, **THEN open** 2-SA-V5089 (Serv Air Dryer Manual Bypass Valve). ☐

(5) **IF** available, **THEN place** 1B Service Air Dryer in service **AND shutdown** 1(2)A Service Air Dryer in accordance with 0OP-46, Instrument and Service Air System Operating Procedure. ☐

4.2 Supplementary Actions (continued)

c. **IF** air is **NOT** cross-tied,
AND cross-tie operation will **NOT** cause a loss of instrument air on the unaffected unit,
THEN perform the following:

(1) **Obtain** permission from the non-affected unit. ☐

(2) **Ensure** 1-SA-PV-5071 (Cross-Tie Valve), located on Unit 1, Panel XU-2, is OPEN. ☐

(3) **Ensure** 2-SA-PV-5071 (Cross-Tie Valve), located on Unit 2, Panel XU-2, is OPEN. ☐

(4) **IF** opening the cross-tie valve degrades the non-affected unit,
THEN return to Step 1.b(4). ☐

d. **IF** the in service air dryer is in sweep mode,
THEN consider securing sweep mode in accordance with Attachment 1, Setting Service Air Dryer(S) Maximum Sweep Value To Zero. ☐

61. 300000 2
Unit One is in MODE 3 following a seismic event and reactor scram with the following plant conditions:

Reactor level	55 inches
Reactor pressure	500 psig
Drywell pressure	9 psig
Division I PNS header pressure	93 psig
Division II PNS header pressure	98 psig

Which one of the following completes both statements below?

Div I Backup N2 Rack Isol Vlv, RNA-SV-5482 is ____(1)____.

Div II Backup N2 Rack Isol Vlv, RNA-SV-5481 is ____(2)____.

A. (1) open
(2) open

B. (1) open
(2) closed

C. (1) closed
(2) open

D. (1) closed
(2) closed

Answer: B
K/A:
300000 Instrument Air System
K3 Knowledge of the effect that a loss or malfunction of the (INSTRUMENT AIR SYSTEM) will have on the following: (CFR: 41.7 / 45.6)
01 Containment air system

RO/SRO Rating: 2.7/2.9

Tier 2 / Group 1

K/A Match: This meets the KA because it is testing the effect of the low pressure (loss or malfunction) of the air system on containment air (N2 backup).

Pedigree: Last used on 2007 NRC Exam

Objective: LOI-CLS-LP-046-A, Obj. 8
Given plant conditions, determine the effects that the following conditions will have on the Pneumatic System: (LOCT) b. Low Instrument Air/Pneumatic Nitrogen (IAN/RNA/PNS) Header Pressure

Reference: None

Cog Level: High

Explanation: No LOCA signal is present so the Backup N2 valves will not be open on a Core Spray initiation signal. The Backup N2 valves open at 95 psig or lower in the PNS header. This would result in Division I Backup N2 valve (5482) being open and Division II (5481) being closed.

Distractor Analysis:

Choice A: Plausible if the student believes that either division will open both valves.

Choice B: Correct Answer, see explanation.

Choice C: Plausible if the student uses the valves for the division separation.

Choice D: Plausible if the student only checks the LOCA signal and not the low pressure signal.

SRO Basis: N/A

Unit 1
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RB INSTR AIR RECEIVER 1A PRESS LOW

AUTO ACTIONS

1. RNA-SV-5482, High Pressure Bottle Rack Isolation Valve, opens, supplying SRVs and CAC-16 with a pneumatic source.

DEVICE/SETPOINTS

RNA-PSL-3596	95 psig decreasing

Unit 1
APP UA-01 1-2
Page 1 of 2

RB INSTR AIR RECEIVER 1B PRESS LOW

AUTO ACTIONS

1. RNA-SV-5481, High Pressure Bottle Rack Isolation Valve, opens, supplying SRVs and CAC-17 with a pneumatic source.

DEVICE/SETPOINTS

RNA-PSL-3597	95 psig decreasing

62. 400000 1

Unit One is operating at rated power with the following conditions:

CSW Pump 1A trips
Conventional header pressure lowers to 35 psig

Which one of the following completes both statements below?

If CSW header pressure remains at this pressure for ____(1)____ seconds, the SW-V3, SW To TBCCW HXs Otbd Isol Vlv, and SW-V4, SW To TBCCW HXs Inbd Isol Vlv, will close to a throttled position.

IAW 0AOP-19, *Conventional Service Water System Failure*, the SW-V3 and SW-V4 are reopened ____(2)____.

A. (1) 30
   (2) ONLY after a reactor Scram is inserted

B. (1) 30
   (2) if system pressure is restored by starting the standby CSW pump

C. (1) 70
   (2) ONLY after a reactor Scram is inserted

D. (1) 70
   (2) if system pressure is restored by starting the standby CSW pump

Answer: D

K/A:

400000 Component Cooling Water System (CCWS)

A2 Ability to (a) predict the impacts of the following on the CCWS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation: (CFR: 41.5 / 45.6)

01 Loss of CCW pump

RO/SRO Rating: 3.3/3.4

Tier 2 / Group 1

K/A Match: This meets the KA because it is testing the auto start signal/logic for a cooling water system.

Pedigree: Last used on the 2010 NRC exam

Objective: CLS-LP-302-H, Obj. 4 Given plant conditions and any of the following AOPs, determine the required supplementary actions: d. 0AOP-19.0, Conventional Service Water System Failure

Reference: None

Cog Level: High

Explanation: **IF** conventional service water header pressure remains below 40 psig for 70 seconds,
THEN:
- *SW TO TBCCW HXS OTBD ISOL, SW-V3* closes to a throttled position
- *SW TO TBCCW HXS INBD ISOL, SW-V4* closes to a throttled position
The Standby CSW pump should start and restore CSW header pressure to normal prior to the SW valves throttling closed. If the standby CSW pump fails to auto start, manually starting the pump will restore CSW header pressure. AOP-19 provides guidance to re-open the SW valves only after header pressure has been restored and the cause of low pressure is known (pump trip).

Distractor Analysis:

Choice A: Plausible because 30 seconds is when the DG cooling water valves close and a Scram is inserted only after the SW valves have closed to the throttled position AND CSW header pressure cannot be immediately restored above 40 psig - under this condition all CSW pumps would be shutdown.

Choice B: Plausible because 30 seconds is when the DG cooling water valves close and system pressure restored by the STBY pump start is correct.

Choice C: Plausible because 70 seconds is correct and a Scram is inserted only after the SW valves have closed to the throttled position AND CSW header pressure cannot be immediately restored above 40 psig - under this condition all CSW pumps would be shutdown.

Choice D: Correct Answer, see explanation

SRO Basis: N/A

CONVENTIONAL SERVICE WATER SYSTEM FAILURE	0AOP-19.0
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3.0 AUTOMATIC ACTIONS

1. Standby pump selected to the conventional service water header starts at 40 psig ☐

2. <u>IF</u> all conventional service water pumps are tripped,
<u>THEN</u>:
   - SW-V36 (SW To CW Pumps Inbd Vlv), closes ☐
   - SW-V37 (SW To CW Pumps Otbd Vlv), closes ☐
   - CWIPs trip on low bearing lubricating water flow (5 - 6 gpm, time-delayed 15 minutes), resulting in loss of condenser vacuum ☐

3. <u>IF</u> conventional service water header pressure remains less than 40 psig for 70 seconds,
<u>THEN</u>:
   - SW-V3 (SW To TBCCW HXs Otbd Isol), closes to a throttled position ☐
   - SW-V4 (SW To TBCCW HXs Inbd Isol), closes to a throttled position ☐

4.2 Supplementary Actions (continued)

d. **Attempt** to isolate any source of leakage. .... ☐

e. **Ensure** discharge valves are CLOSED on shutdown pump(s). .... ☐

f. **Check** service water traveling screens for excessive build-up **AND wash** if excessive buildup is occurring. .... ☐

g. **Check** service water trash racks for excessive build-up **AND notify** Maintenance to clean if excessive buildup is occurring. .... ☐

h. Locally **monitor** each pump discharge strainer differential pressure. .... ☐

i. **Check** Annunciator Panel UA-01 for lit annunciators. .... ☐

10. **Refer to** Technical Specification 3.7.2, Service Water (SW) System and Ultimate Heat Sink (UHS) for operability requirements. .... ☐

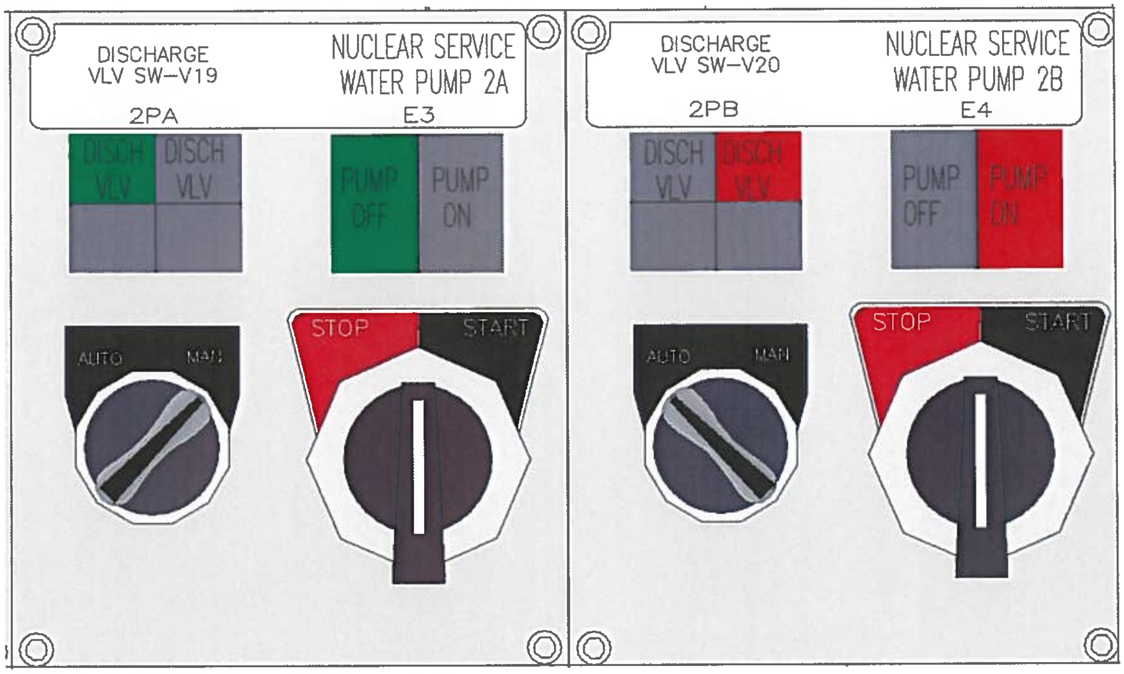
11. **WHEN** conventional service water header pressure is restored to normal
AND the cause of low header pressure has been corrected,
THEN:

a. **Open** SW-V3 (SW To TBCCW HXs Otbd Isol). .... ☐

b. **Open** SW-V4 (SW To TBCCW HXs Inbd Isol). .... ☐

63. 400000 2

Unit Two Nuclear Service Water (NSW) pumps are aligned as follows in preparation for equipment realignment:



Subsequently, Off-site power is lost.

Which one of the following completes the statement below?

___(1)___ NSW pump(s) will auto start ___(2)___ associated E Bus is re-energized.

A. (1) 2A and 2B
   (2) immediately when their

B. (1) 2A and 2B
   (2) five seconds after their

C. (1) 2B ONLY
   (2) immediately when its

D. (1) 2B ONLY
   (2) five seconds after its

Answer: A

K/A:

400000 Component Cooling Water System (CCWS)

K4 Knowledge of CCWS design feature(s) and or interlocks which provide for the following: (CFR: 41.7)

01 Automatic start of standby pump

RO/SRO Rating: 3.4/3.9

Tier 2 / Group 1

K/A Match: This meets the KA because it is testing the auto start signal/logic for a cooling water system.

Pedigree: Bank

Objective: LOI-CLS-043, Objective 8a
State the power supply (bus and voltage) for the following Service Water System components: Nuclear Service Water Pumps.

Reference: N/A

Cog Level: High

Explanation: NSW pumps auto start immediately after LOOP signal regardless of mode selector switch or discharge valve position. 5 second timer applies only on a LOCA.

Distractor Analysis:

Choice A: Correct Answer, see explanation

Choice B: Plausible because this would be the case with a LOCA signal present.

Choice C: Plausible because examinee must know the power supply scheme.

Choice D: Plausible because examinee must know the power supply scheme.

SRO Basis: N/A

Each pump is powered by a 4160 VAC motor supplied from the emergency bus power supplies:

Component	Power Supply
1A CSW pump	E4
1B CSW pump	E1
1C CSW pump	E2
1A NSW pump	E1
1B NSW pump	E2
2A CSW pump	E3
2B CSW pump	E4
2C CSW pump	E1
2A NSW pump	E3
2B NSW pump	E4

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In addition to the low header pressure auto start, the NSW pumps will start five seconds after receipt of a LOCA signal, regardless of mode selector switch or discharge valve position. For example, a Division I LOCA signal from either Unit 1 or Unit 2 will auto start the 1A and 2A NSW pumps; the Division II LOCA logic will auto start 1B and 2B NWS pumps.

The NSW pumps, powered through the 4160 VAC emergency buses, will also automatically start immediately after the start of the diesel generators and reenergization of the emergency buses on loss of off-site power (LOOP), regardless of mode selector switch or discharge valve position. For example, a Division I LOOP signal from either Unit 1 or Unit 2 will auto start the 1A and 2A NSW pumps; the Division II LOOP logic will auto start 1B and 2B NSW pumps. If a LOCA signal exists on the division sensing the LOOP, auto start will occur after five seconds, provided that a LOOP signal is not present on the opposite unit. On a dual unit LOOP the NSW pump(s) of the LOCA (and non-LOCA) unit start immediately after the emergency buses are reenergized by their respective diesel generators without the five second delay.

64. 600000 1

Which one of the following identifies the potential consequence of failing to place backup nitrogen in service by placing RNA keylock switches in LOCAL IAW 0ASSD-02, *Control Building*?

RNA keylock switch noun names:
2-RNA-CS-001, Override Switch For Valve RNA-SV-5482
2-RNA-CS-002, Override Switch For Valve RNA-SV-5253

A. Misoperation of RCIC.

B. Loss of drywell cooling.

C. Inability to operate SRVs.

D. Spurious operation of MSIVs.

Answer: C

K/A:
600000 Plant Fire On Site
AK3 Knowledge of the reasons for the following responses as they apply to PLANT FIRE ON SITE: (CFR: 41.5 / 45.6)
04 Actions contained in the abnormal procedure for plant fire on site

RO/SRO Rating: 2.8/3.4

Tier 1 / Group 1

K/A Match: This matches the KA because it tests the reason a step in the ASSD procedure is performed. The ASSD procedures are the plant fire procedures.

Pedigree: Bank

Objective: LOI-CLS-LP-304, Obj. 25k
Given ASSD procedures and plant conditions, predict the consequences of FAILURE to perform the following actions: Deenergize RNA-SV-5482 and RNA-SV-5253 via keylock switches RNA-CS-001 and RNA-CS-002.

Reference: None

Cog Level: fundamental

Explanation: The Reactor Building MCC Operator places the key lock switches to the LOCAL position to ensure Nitrogen System is lined up to provide reliable operation of the SRVs.

Distractor Analysis:

Choice A: Plausible because actions for the operation of RCIC are contained in the ASSD procedures.

Choice B: Plausible because a loss of pneumatics would cause the DW cooler dampers to close.

Choice C: Correct Answer, see explanation.

Choice D: Plausible because actions to prevent the spurious operation of the MSIV is contained in the ASSD procedure.

SRO Basis: N/A

1.6.5 **WHEN** directed to start RCIC, **THEN PERFORM** the following at MCC 2XDB:

1. **OPEN** *RCIC TURB TR & THR VLV, E51-V8*, at Compt B37 **(Row C1)**. ☐

2. **OPEN** *RCIC TURB STM SPLY VLV, E51-F045*, at Compt B44 **(Row F2)**. ☐

3. **INFORM** Unit 2 CRS RCIC should be running. ☐

1.7 **WHEN** directed, **THEN PERFORM** the following at Unit 2 Reactor Building 50 foot elevation:

1.7.1 **PLACE** keylock switch *2-RNA-CS-001* in *LOCAL* for valve *2-RNA-SV-5482*. ☐

1.7.2 **PLACE** keylock switch *2-RNA-CS-002* in *LOCAL* for valve *2-RNA-SV-5253*. ☐

1.7.3 **INFORM** the Unit 2 CRS that backup nitrogen has been made available for SRV operation. ☐

1.8 **IF** directed, **THEN TRANSFER** RCIC suction from CST to suppression pool at MCC 2XDB as follows:

1.8.1 **OPEN** *RCIC SUPP POOL SUCT VLV, E51-F031*, at Compt B45 **(Row G1)**. ☐

1.8.2 **OPEN** *RCIC SUPP POOL SUCT VLV, E51-F029*, at Compt B46 **(Row G2)**. ☐

1.8.3 **CLOSE** *RCIC CST SUCT VLV, E51-F010*, at Compt B38 **(Row C2)**. ☐

65. 700000 1

A grid disturbance occurs with the following Unit One plant parameters:

Generator Load	980 MWe
Generator Reactive Load	160 MVARs, out
Generator Gas Pressure	50 psig

(REFERENCE PROVIDED)

Which one of the following identifies both available options that will place the Unit within the Estimated Capability Curve?

A. Raise gas pressure to 58 psig or lower power to 940 MWe.

B. Raise gas pressure to 58 psig or raise reactive load to 240 MVARs.

C. Raise gas pressure to 58 psig or lower reactive load to 70 MVARs.

D. Lower power to 940 MWe or raise reactive load to 240 MVARs.

Answer: A

K/A:

700000 Generator Voltage and Electric Grid Disturbances

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

03 Generator current outside the capability curve

RO/SRO Rating: 3.5/3.6

Tier 1 / Group 1

K/A Match: This meets the K/A because the tests the ability to determine action needed to remain within capability curve.

Pedigree: Last used on 2014 NRC exam

Objective:

CLS-LP-27, Obj. 9 - Given the Generator estimated capability curves, hydrogen pressure and either MVARS, MW, or power factor, determine the limit for MW and MVARS.

Reference: 1OP-27 Attachment 2, Estimated Capability Curves

Cog Level: High

Explanation: Based on the conditions the student should plot the current location on the graph. Plot MWe along the bottom and MVARs up the side. Where these two points intersect, based on 50 psig gas pressure line is outside of the safe area. (Must be inside the curve to be safe) Lowering MWe or raising gas pressure are the only options. For this case lowering or raising MVARs would still be outside the curve.

Distractor Analysis:

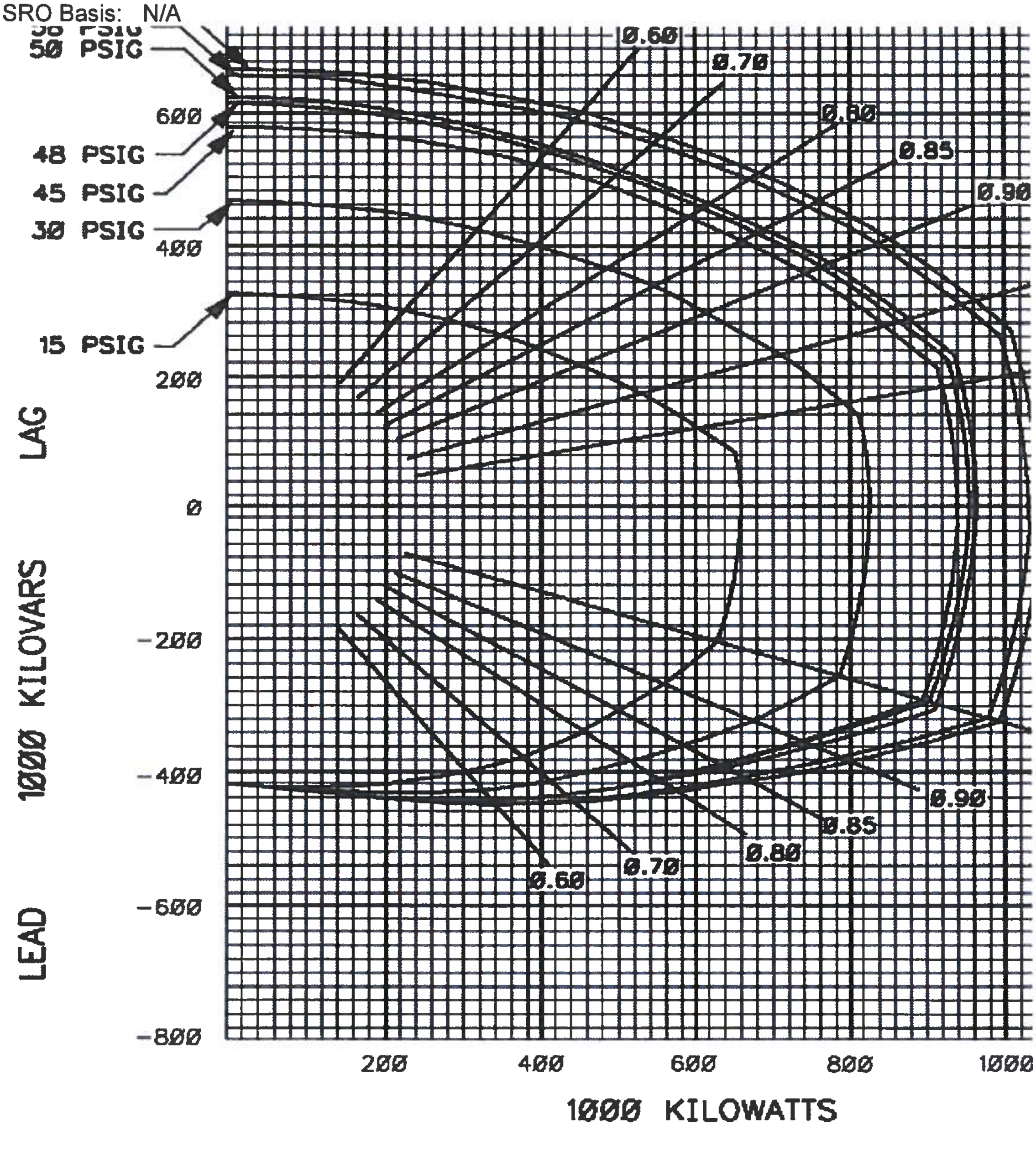
Choice A: Correct Answer, see explanation

Choice B: Plausible because raising pressure will move the plant within the limits of the curve. Raising MVARS will not move the plant within the limits of the curve.

Choice C: Plausible because raising pressure will move the plant within the limits of the curve. Lowering MVARS will not move the plant within the limits of the curve.

Choice D: Plausible because raising MWe will move the plant within the limits of the curve. Raising MVARS will not move the plant within the limits of the curve.

SRO Basis: N/A



66. G2.1.01 1

Which one of the following completes both statements below IAW AD-OP-ALL-1000, *Conduct of Operations*?

With the Unit operating at rated, steady state power, steam flow / feed flow ___(1)___ a key parameter that the OATC must monitor to assure a constant awareness of its value and trend.

An end to end control panel walk down shall be performed every ___(2)___ and documented in the Narrative Logbook.

A. (1) is NOT
   (2) one hour

B. (1) is NOT
   (2) two hours

C. (1) is
   (2) one hour

D. (1) is
   (2) two hours

Answer: D

K/A:

G2.1.01 Knowledge of conduct of operations requirements. (CFR: 41.10 / 45.13)

RO/SRO Rating: 3.8/4.2

Tier 3

K/A Match: This meets the K/A because it is testing knowledge of the Conduct of Operations Manual

Pedigree: New

Objective: LOI-CLS-LP-201-D, Obj. 1j - Explain/describe the following IAW AD-OP-ALL-1000, Conduct of Operations, 0OI-01.01, BNP Conduct of Operations Supplement and OPS-NGGC-1314, Communications: Control Board walkdown and monitoring requirements

Reference: None

Cog Level: Fund

Explanation: IAW the conduct of operations document board walk downs must be completed every two hours and section 5.5.6 lists the key parameters to watch.

Distractor Analysis:

Choice A: Plausible because jet pump flow has a daily surveillance requirement and if a watchstander is relieved for greater than one hour it must be entered in narrative logbook.

Choice B: Plausible because jet pump flow has a daily surveillance requirement and part two is correct.

Choice C: Plausible because part one is correct and if a watchstander is relieved for greater than one hour it must be entered in narrative logbook.

Choice D: Correct Answer, see explanation.

SRO Basis: N/A

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5.5.6 Control Board Monitoring (continued)

k. Unless involved in activities where Reactor Operator involvement is required by the Conduct of Operations (for example reactivity manipulations, peer checks or detailed panel reviews), the operator shall monitor the following key parameters at a frequency to assure a constant awareness of their value and trend:

- Rx Power
- RPV level (BWR)
- Steam generator pressure (PWR)
- RCS temperature
- Steam generator level (PWR)
- RCS pressure
- Steam flow / feed flow
- Pressurizer level (PWR)

5.5.6 Control Board Monitoring (continued)

3. The CRS ensures that a licensed operator performs an end to end control panel walk down every two hours. The walk down shall be documented in the Narrative Logbook.

4. Whenever a watch station is relieved for greater than one hour, this information shall be entered in a Narrative Log Program, a formal turnover and shift turnover sheet will be completed, including the logs signed over.

67. G2.1.32 1
Which one of the following completes the statement below?

1OP-10, *Standby Gas Treatment System Operating Procedure*, prohibits venting the drywell and the suppression pool chamber simultaneously with the reactor at power because this would cause the:

A. unnecessary cycling of reactor building to torus vacuum breakers.

B. unnecessary cycling of torus to drywell vacuum breaker.

C. SBGT Train water seal to blow out of the trough.

D. pressure suppression function to be bypassed.

Answer: D
K/A:
G2.1.32 Ability to explain and apply system limits and precautions. (CFR: 41.10 / 43.2 / 45.12)

RO/SRO Rating: 3.8/4.0

Tier 3

K/A Match: This meets the K/A because it is testing the ability to explain the system precaution.

Pedigree: Last used on 2012 NRC exam

Objective:

Reference: None

Cog Level: Fundamental

Explanation: Per OP-10, torus and drywell cannot be vented at the same time in Modes 1, 2 or 3. per the LER reference, this could result in bypassing pressure suppression function.

Distractor Analysis:

Choice A: Plausible because these vacuum breakers prevent drawing a negative pressure in the suppression pool. Cross connecting the drywell and the suppression pool free air space will not cause a negative pressure in the suppression pool.

Choice B: Plausible because this lineup equalizes pressure between the drywell and the suppression pool free air space since the vacuum breakers operate on a d/p between the spaces this would bypass them, not open them.

Choice C: Plausible because venting containment through large valves with an elevated pressure may blow out the SBGT water seal.

Choice D: Correct Answer, see explanation.

SRO Basis: N/A

1.0 PURPOSE

1. This procedure provides instructional guidance for operation of the Standby Gas Treatment System and its associated deluge system.

2.0 SCOPE

1. This procedure provides the prerequisites, precautions, limitations, and instructional guidance for startup, normal operation, shutdown, and infrequent operation of the Standby Gas Treatment System and its associated deluge system.

3.0 PRECAUTIONS AND LIMITATIONS

1. The Standby Gas Treatment System will **NOT** automatically start if the control switch is in STBY. .... ☐
2. Venting the drywell and suppression pool simultaneously is **NOT** performed when the plant is in MODE 1, 2, or 3. {8.1.1}.............................. ☐

8.0 REFERENCES

8.1 Commitments

1. LER 1-97-011, Drywell and Torus Inerting/Deinerting Lineup Results in Unanalyzed Suppression Pool Bypass Path

68. G2.1.36 1

A core reload is in progress during a refueling outage. The initial loading of fuel bundles around each SRM centered 4-bundle cell was completed with all four SRMs fully inserted and reading 50 cps.

It is now approximately half way through the core loading sequence and SRMs read 80 cps.

Which one of the following completes the statement below IAW 0FH-11, *Refueling*?

Fuel movement must be suspended when any SRM reading **first** rises to ________ upon insertion of the **next** fuel bundle.

A. 100 cps

B. 160 cps

C. 250 cps

D. 400 cps

Answer: B

K/A:

G2.1.36 Knowledge of procedures and limitations involved in core alterations. (CFR: 41.10 / 43.6 / 45.7)

RO/SRO Rating: 3.0/4.1

Tier 3

K/A Match: This meets the K/A because it is testing the fuel movement requirements that an RO would monitor.

Pedigree: New

Objective: LOI-CLS-LP-305, Objectives 18
Given the conditions during a refueling outage state the operator actions required for rising SRM count rates and/or inadvertent criticality.

Reference: None

Cog Level: High

Explanation: An increase in counts by a factor of two during a single bundle insertion is reason to suspend fuel movements. An increase by a factor of five from the baseline is also a reason.

Distractor Analysis:

Choice A: Plausible because this is a doubling of the baseline counts which is used for a different criteria for suspension of fuel movements.

Choice B: Correct Answer, see explanation

Choice C: Plausible because this is an increase of the baseline counts by a factor of five which is a reason to suspend fuel movements.

Choice D: Plausible because this is an increase of the counts by a factor of five which is a reason to suspend fuel movements.

SRO Basis: N/A

FH-11:

24. Suspension of fuel movement and notification of the Reactor Engineer is required if either of the following occur:

- An SRM reading rise by a factor of two upon insertion of any single bundle. During a spiral reload, this restriction applies only after the initial loading of fuel bundles around each SRM is complete. During a Core Shuffle, this restriction does **NOT** apply to the SRM that is having an adjacent fuel bundle inserted or removed. .... ☐
- An SRM rise by a factor of five relative to the SRM baseline count rate recorded on Attachment 6, Documentation for SRM Baseline .... ☐

25. SRM count rate may drop to less than 3 cps during either of the following conditions:

- With less than or equal to four fuel assemblies adjacent to the SRM and **NO** other fuel assemblies in the associated core quadrant .... ☐
- During a core spiral offload .... ☐

69. G2.2.02 1

Unit Two is conducting a routine power reduction for rod pattern improvement. The Reactivity Management Plan contains actions for the RO to insert a group of four rods from position 24 to position 12.

Which one of the following completes the statement below IAW AD-OP-ALL-0203, *Reactivity Management*?

The movement of these rods should be:

A. single notched for the entire movement.

B. continuously inserted to the final intended position.

C. continuously inserted to settle four notches prior to reaching the intended position and then single notched into the final intended position.

D. continuously inserted to settle one notch prior to reaching the intended position and then single notched into the final intended position.

Answer: D

K/A:

G2.2.02 Ability to manipulate the console controls as required to operate the facility between shutdown and designated power levels. (CFR: 41.6 / 41.7 / 45.2)

RO/SRO Rating: 4.6/4.1

Tier 3

K/A Match: This question matches the KA because it tests the generic requirements of control rod movement during any power level.

Pedigree: Bank

Objective: LOI-CLS-LP-201-D, Obj. 22f
Explain the following regarding AD-OP-ALL-0203, Reactivity Management: The procedural requirements for positioning intermediate control rods

Reference: None

Cog Level: High

Explanation: If a rod is to be moved between 46 and 02 three notches or less, it must be single notched the entire move. When moving a control rod four notches or more, the control rod should be stopped one notch prior to reaching the intended position and then single notched into the final intended position.

Distractor Analysis:

Choice A: Plausible because this would apply if the movement was $\leq$ four notches.

Choice B: Plausible because this would apply under emergency conditions.

Choice C: Plausible because the rod does have to be single notched into its final position but the rod can be continuously move if greater than four notches not for four notches.

Choice D: Correct Answer, see explanation.

SRO Basis: N/A

5.2.8 [BWR] Single Recirculation Loop Operation

{7.1.5}

1. Standards

   a. Single-Loop operation for extended periods of time is discouraged.

2. Expectations

   a. Plant procedures that address Single Recirculation Loop Operation will identify applicable limitations and trip criteria.

   b. For operations not covered by an approved procedure the Operational Decision Making process will be used to evaluate continued operation in Single-Loop.

   c. The risk associated with single recirculation loop operations shall be carefully considered and appropriate contingencies will be developed.

   d. Operator JITT shall be conducted for planned Single-Loop operations.

5.2.9 Control Rod Manipulations

1. Standards

   a. Ensure all control rod movements are made in a deliberate, carefully controlled manner while constantly monitoring nuclear instrumentation and redundant indications of reactor power and neutron flux.

2. Expectations

   a. [BWR] To minimize the possibility of mispositioning a control rod when inserting or withdrawing to an intermediate position (notch positions '02' through '46'), the following practices shall be followed:

      (1) When moving a control rod four notches or more, the control rod should be stopped one notch prior to reaching the intended position and then single notched into the final intended position. This guidance does not supersede any other requirement to single notch control rods.

      (2) When moving a control rod three notches or less, the control rod should be single notched for the entire move.

70. G2.2.04 1

Which one of the following identifies the Unit Two "Scram Immediate Operator Action" that utilizes a different criteria for performance than on Unit One?

A. Tripping of the main turbine.

B. Tripping one of the running feed pumps.

C. Master level controller setpoint setdown.

D. Placing the reactor mode switch to Shutdown.

Answer: D

K/A:

G2.2.04 Ability to explain the variations in control board/control room layouts, systems, instrumentation, and procedural actions between units at a facility. (CFR: 41.6 / 41.7 / 41.10 / 45.1 / 45.13)

RO/SRO Rating: 3.6/3.6

Tier 3

K/A Match: This meets the K/A because it is testing the differences between the Units

Pedigree: Bank

Objective: LOI-CLS-LP-300-C, Obj. 2
List the immediate operator actions for a Reactor Scram. (LOCT)

Reference: None

Cog Level: fund

Explanation: On Unit Two the mode switch cannot be placed to shutdown until MSL flow is less than 3 Mlbms. This restriction does not exist on Unit One.

Distractor Analysis:

Choice A: Plausible because this is an immediate operator action.

Choice B: Plausible because this is an immediate operator action.

Choice C: Plausible because this is an immediate operator action.

Choice D: Correct Answer, see explanation.

SRO Basis: N/A

Unit 2 Scram Immediate Actions (0EOP-01-UG)

SCRAM IMMEDIATE ACTIONS

1. **Ensure** SCRAM valves OPEN by manual SCRAM or ARI initiation.
2. <u>**WHEN**</u> steam flow less than 3×10^6 lb/hr,
   <u>**THEN**</u> **place** reactor mode switch in SHUTDOWN.
3. <u>**IF**</u> reactor power below 2% (APRM downscale trip),
   <u>**THEN**</u> **trip** main turbine.
4. **Ensure** master RPV level controller setpoint at +170 inches.
5. <u>**IF:**</u>
   - Two reactor feed pumps running

     <u>**AND**</u>
   - RPV level above +160 inches

     <u>**AND**</u>
   - RPV level rising,

   <u>**THEN**</u> **trip** one.

Unit 1 Scram Immediate Actions (0EOP-01-UG)

SCRAM IMMEDIATE ACTIONS

1. **Ensure** SCRAM valves OPEN by manual SCRAM or ARI initiation.
2. **Place** reactor mode switch in SHUTDOWN.
3. <u>**IF**</u> reactor power below 2% (APRM downscale trip),
   <u>**THEN**</u> **trip** main turbine.
4. **Ensure** master RPV level controller setpoint at +170 inches.
5. <u>**IF:**</u>
   - Two reactor feed pumps running

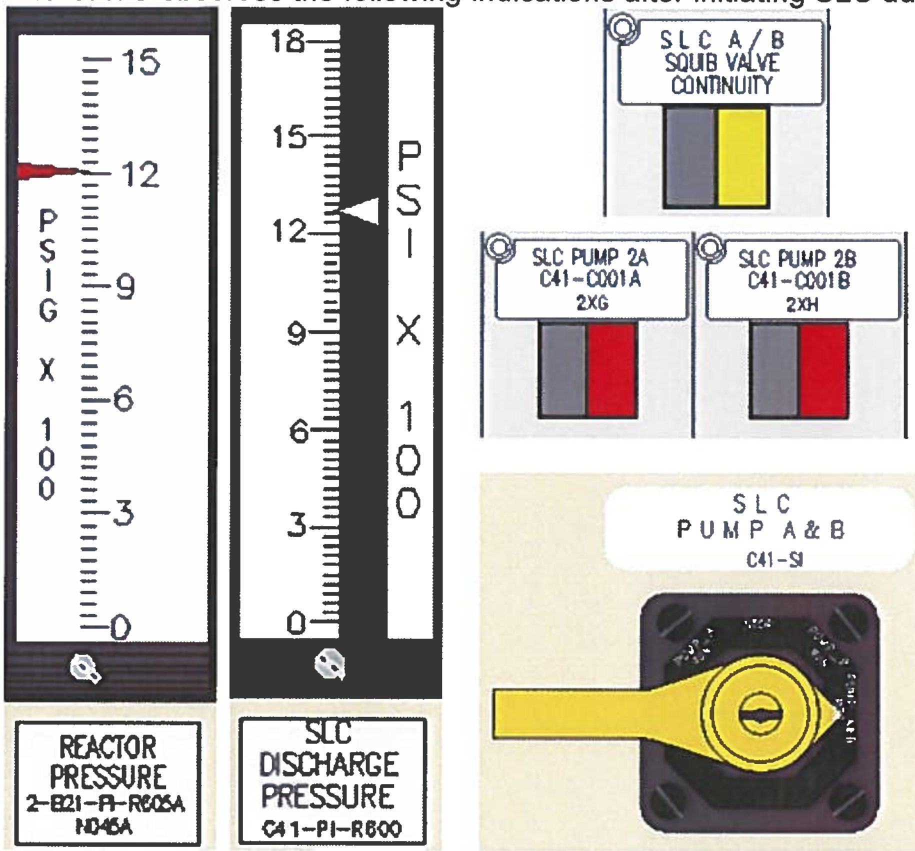
     <u>**AND**</u>
   - RPV level above +160 inches

     <u>**AND**</u>
   - RPV level rising,

   <u>**THEN**</u> **trip** one.

71. G2.2.44 1

The OATC observes the following indications after initiating SLC during an ATWS.



Which one of the following completes both statements below?

Squib valve ___(1)___ has failed to fire.

IAW 2OP-05, *Standby Liquid System Operating Procedure*, the OATC is required to ___(2)___.

A. (1) A
   (2) place the CS-S1, SLC Pump A & B, in the PUMP B RUN position

B. (1) A
   (2) leave the CS-S1, SLC Pump A & B, in the PUMP A/B RUN position

C. (1) B
   (2) place the CS-S1, SLC Pump A & B, in the PUMP A RUN position

D. (1) B
   (2) leave the CS-S1, SLC Pump A & B, in the PUMP A/B RUN position

Answer: C

K/A:
G2.2.44 Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions.
(CFR: 41.5 / 43.5 / 45.12)

RO/SRO Rating: 4.2/4.4

Tier 3

K/A Match: This meets the K/A because it is testing knowledge of the indications and what action is required based on the system lineup.

Pedigree: Previously used on the 2014 NRC exam

Objective: LOI-CLS-LP-005, Obj 13 -
Predict the effect of the following on the Standby Liquid Control System, and based on those predictions use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: a. Failure of one or both squib valves to fire.

Reference: None

Cog Level: Hi

Explanation: The SLC squib valve continuity lights are normally lit and go out when fired on SLC initiation. Per OP-05, if one squib valve fails to fire, two pump SLC operation may still continue provided reactor pressure is below 1184 psig, which it is not.

Distractor Analysis:

Choice A: Plausible because the student may think that the light is illuminated when the squib valve fires and securing 1 pump is correct.

Choice B: Plausible because the student may think that the light is illuminated when the squib valve fires and if reactor pressure was lower this would be correct.

Choice C: Correct Answer, see explanation

Choice D: Plausible because the B squib did not fire and if reactor pressure was lower this would be correct.

SRO Basis: N/A

NOTE:	The SLC pump discharge relief valve should **NOT** actuate with two pumps operating and only one squib valve open unless reactor pressure exceeds 1184 psig, which is possible during an ATWS even with 10 SRVs open.

2. **IF** *SLC A SQUIB VALVE CONTINUITY* **OR** *SLC B SQUIB VALVE CONTINUITY* indicating light on Panel P603 remains on **AND** reactor pressure is greater than or equal to 1184 psig, **THEN PERFORM** the following:

   a. **PLACE** *SLC PUMP A & B* Control Switch, *C41-CS-S1*, to the *SLC PUMP A* **OR** *SLC PUMP B* position. ☐

   b. **ENSURE** the selected SLC pump red indicating light on. ☐

72. G2.3.12 1

Two operators are required to enter a room that is posted as a Locked High Radiation Area (LHRA) to hang a clearance for **scheduled** work.

Which one of the following completes both statements below?

The radiation level at which a LHRA posting is required is ___(1)___ in one hour at 30 centimeters from the radiation source.

The LHRA key is obtained from ___(2)___.

A. (1) >100 mrem
   (2) the Shift Manager

B. (1) >100 mrem
   (2) a RP Technician

C. (1) >1000 mrem
   (2) the Shift Manager

D. (1) >1000 mrem
   (2) a RP Technician

Answer: D

K/A:

G2.3.12 Knowledge of radiological safety principles pertaining to licensed operator duties, such as containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc. (CFR: 41.12 / 45.9 / 45.10)

RO/SRO Rating: 3.2/3.7

Tier 3

K/A Match: This question matches the KA because it is testing the rad requirements for entering a LHRA.

Pedigree: Bank (from Farley)

Objective: LOI-CLS-LP-201-F, Obj. 10
Explain the requirement regarding control of High Radiation Areas per E&RC-0040.

Reference: None

Cog Level: Fundamental

Explanation: Locked High Radiation Area (LHRA) criteria is an area, accessible to individuals, in which radiation levels could result in an individual receiving a dose equivalent in excess of 1.0 rem (1000 mrem) (10 mSv) in one hour at 30 centimeters from the radiation source or from any surface that the radiation penetrates or an area accessible to individuals with dose rates in excess of 1.0 rem per hour at 30 cm from the radiation source or 30 cm from any surface that the radiation penetrates but less than 500 rads in one hour at one meter from the radiation source or from any surface penetrated by the radiation. The Shift Manager has a LHRA key for emergency use.

Distractor Analysis:

Choice A: Plausible because this is the limit for a high radiation area not a LHRA. The Shift manager has a key for LHRA but it is for emergency use, not scheduled work.

Choice B: Plausible because this is the limit for a high radiation area not a LHRA. The second part is correct.

Choice C: Plausible because the first part is correct and the Shift manager has a key for LHRA but it is for emergency use, not scheduled work.

Choice D: Correct Answer, see explanation.

SRO Basis: N/A

14. **High Radiation Area (HRA):** An area, accessible to individuals in which radiation levels from radiation sources external to the body could result in an individual receiving a dose equivalent in excess of 0.1 rem (100 mrem) (1mSv) in one hour at 30 cm from the radiation source or 30 cm from any surface the radiation penetrates.

15. **Hot Spot (HS):** An accessible, localized source of radiation with a contact dose rate of greater than 100 mrem per hour and greater than five times the general area dose rate at 30 cm.

16. **Licensed Material:** Source material, special nuclear material, or byproduct material received, possessed, used, transferred or disposed of under a general or specific license.

17. **Locked High Radiation Area (LHRA):** An area, accessible to individuals, in which radiation levels could result in an individual receiving a dose equivalent in excess of 1.0 rem (1000 mrem) (10 mSv) in one hour at 30 centimeters from the radiation source or from any surface that the radiation penetrates or an area accessible to individuals with dose rates in excess of 1.0 rem per hour at 30 cm from the radiation source or 30 cm from any surface that the radiation penetrates but less than 500 rads in one hour at one meter from the radiation source or from any surface penetrated by the radiation.

ACCESS CONTROLS FOR HIGH, LOCKED HIGH, AND VERY HIGH RADIATION AREAS	AD-RP-ALL-2017
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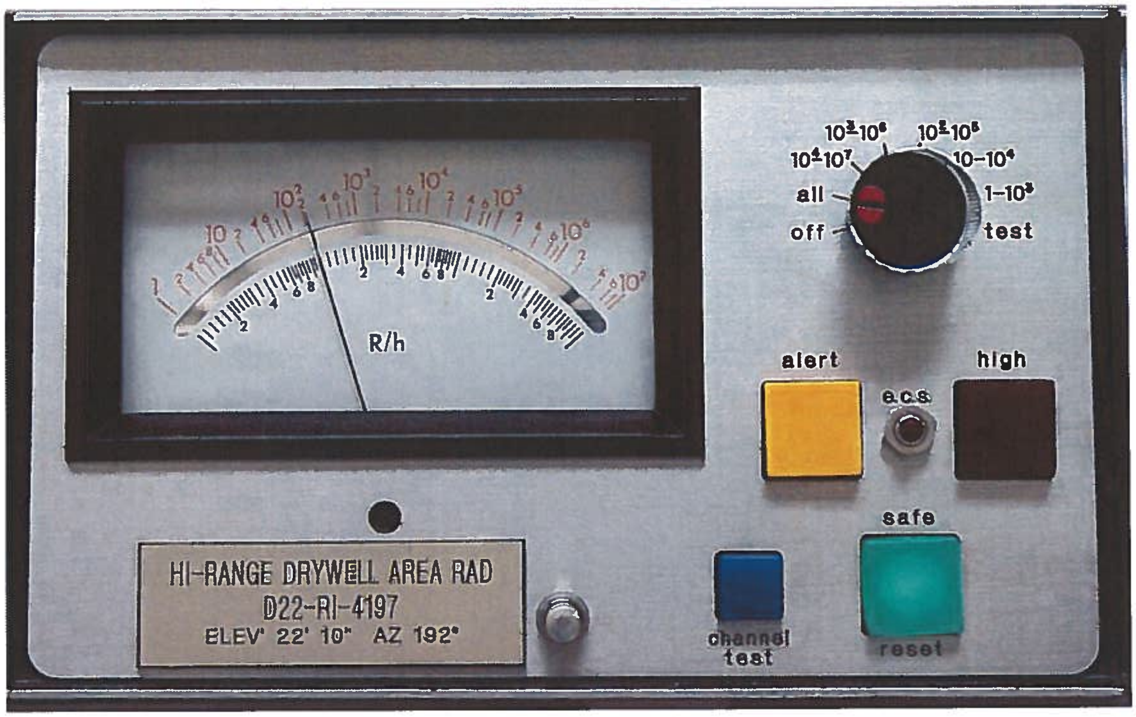
5.1 General Instructions (continued)

12. Entry into HRAs, LHRAs, or VHRAs require a briefing per AD-RP-ALL-2011, Radiation Protection Briefings. {7.1.2}

13. HRA, LHRA less than 10 R/hr, and LHRA greater than or equal to 10 R/hr master keys may be under the control of the Operations Shift Manager for emergency use.

73. G2.3.15 1



Which one of the following identifies the DW radiation value indicated above?

A. ~ 10 R/hr

B. ~ 20 R/hr

C. ~ 100 R/hr

D. ~ 200 R/hr

Answer: D
K/A:
G2.3.15 Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc. (CFR: 41.12 / 43.4 / 45.9)

RO/SRO Rating: 2.9/3.1

Tier 3

K/A Match: This question matches the KA as it requires knowledge of the DW rad monitoring system to answer question.

Pedigree: Bank

Objective: CLS-LP-11.1, Obj. 03a
Describe the function/operation of the following: Drywell High Range Radiation Monitors

Reference: None

Cog Level: Fundamental

Explanation: Drywell high range area monitors provide indications of gross fuel failure and are used to determine emergency plan emergency action level associated with abnormal core conditions. With the function switch in the ALL position the upper (red) scale is used, meter readings are taken from the upper scale between 1 - 1,000,000 R/h. Current indication of 200 R/h

Distractor Analysis:

Choice A: Plausible because this is the reading on the bottom scale.

Choice B: Plausible because if function switch is not taken into account the answer could be 20 R/h.

Choice C: Plausible because if the reading on the bottom scale is adjusted by a factor of 10.

Choice D: Correct Answer, see explanation.

SRO Basis: N/A

74. G2.4.20 1

A transient has occurred on Unit Two with the following plant conditions:

RPV pressure	1000 psig
Drywell ref leg area temp	197°F
Rx Bldg 50' temp	135°F
Wide Range Level	170 inches (N026A/B)
Shutdown Range Level	160 inches (N027A/B)

(REFERENCE PROVIDED)

Which one of the following completes both statements below concerning the level instruments that can be used to determine reactor water level IAW EOP Caution 1?

Wide Range Level instruments N026A/B ___(1)___ be used.

Shutdown Range Level instruments N027A/B ___(2)___ be used.

A. (1) can
   (2) can

B. (1) can
   (2) can NOT

C. (1) can NOT
   (2) can

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

Answer: B

K/A:

G2.4.20 Knowledge of the operational implications of EOP warnings, cautions, and notes.
(CFR: 41.10 / 43.5 / 45.13)

RO/SRO Rating: 3.8/4.3

Tier 3

K/A Match: The question meets the KA because it is testing the knowledge of EOP Caution 1 which deals with the water level instruments availability to determine level.

Pedigree: New

Objective: LOI-CLS-LP-300-B, Objective 16
Given Plant conditions, determine if the RPV water level instrument is providing valid trending information IAW Caution 1.

Reference: Caution 1 (EOP-01-UG, Att. 19, Att. 22 & Att. 31 pages 1 and 2)

Cog Level: High

Explanation: N026s can be used since reading >20" and RB 50' temp is <140 degrees and N027s cannot be used since in unsafe region for minimum indicated level

Distractor Analysis:

Choice A: Plausible because the first part is correct and if Attachment 19 is only looked at then this is plausible.

Choice B: Correct Answer, see explanation.

Choice C: Plausible because if the temperature was a little higher on the RB 50 foot this would be correct and if Attachment 19 only is looked at for the second part this would be correct.

Choice D: Plausible because if the temperature was a little higher on the RB 50 foot this would be correct and the second part is correct.

SRO Basis: N/A

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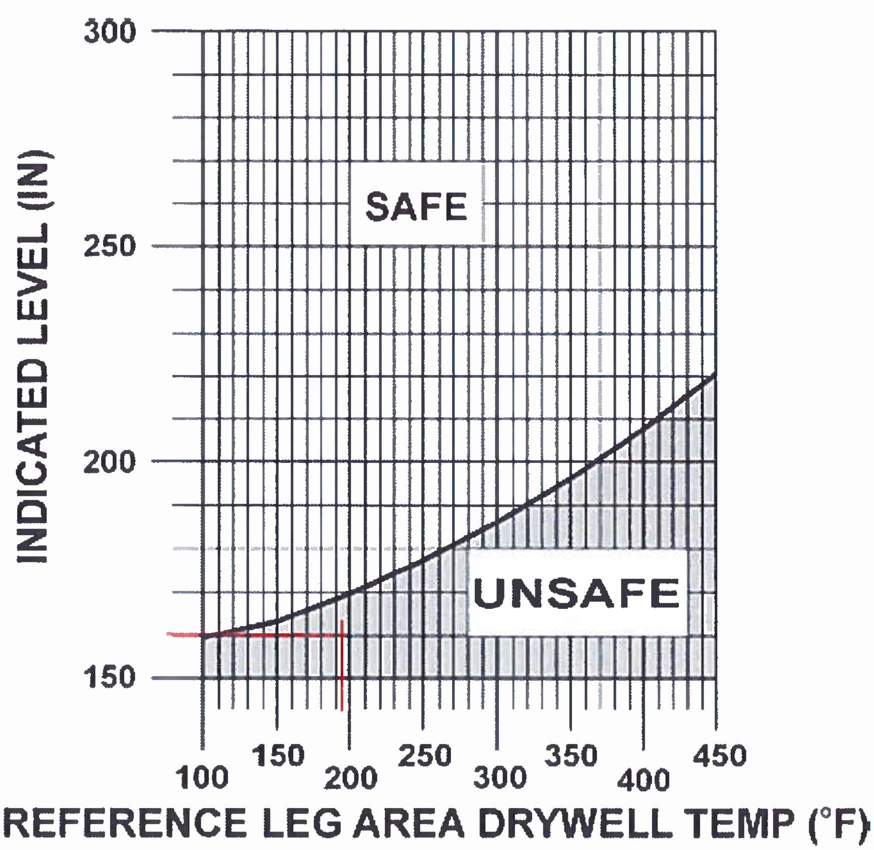
ATTACHMENT 22
Page 1 of 1

<< Shutdown Range Level Instrument (N027A, B) Caution >>



75. G2.4.27 1

A fire has been reported and confirmed in the turbine building breezeway.
A fire hose is being used to control/suppress the fire.

Which one of the following completes both statements below IAW 0PFP-013, *General Fire Plan*?

The RO is required to sound the fire alarm and announce the location of the fire ___(1)___.

A call for offsite assistance to the Brunswick County 911 Center ___(2)___ required.

A. (1) ONLY once
   (2) is

B. (1) ONLY once
   (2) is NOT

C. (1) three times
   (2) is

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

Answer: C

K/A:

G2.4.27 Knowledge of "fire in the plant" procedures. (CFR: 41.10 / 43.5 / 45.13)

RO/SRO Rating: 3.4/3.9

Tier 3

K/A Match: This meets the K/A because it is testing knowledge of the actions contained in the plant fire procedure

Pedigree: New

Objective: FPT-CLS-LP-205
Lesson plan discusses the actions for the control room but no objective is listed.

Reference: None

Cog Level: Fundamental

Explanation: The operator aid (from the General Fire Plan, PFP-013) for the control room operators states to announce the fire location 3 times. The procedure also states to request off site assistance if a fire hose is used for extinguishing the fire.

Distractor Analysis:

Choice A: Plausible because EP announcements are performed once and the second part is correct.

Choice B: Plausible because EP announcements are performed once and the second part because the stem says that the fire is under control.

Choice C: Correct Answer, see explanation

Choice D: Plausible because EP announcements are performed once and the second part because the stem says that the fire is under control.

SRO Basis: N/A

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ATTACHMENT 2
Page 1 of 2

<< (Information Use) - Control Room/Operator Fire Actions >>

1. **Sound** fire alarm, **announce** location of the fire 3 times, then: ☐
   - **Announce**: ☐

     "Fire brigade muster at the fire house."
   - **IF** fire is outside the Protected Area,
     THEN announce:
     "All personnel **NOT** involved in fire fighting or direct support activities are to evacuate the involved area immediately." ☐
   - **IF** fire is inside the Protected Area,
     THEN announce:
     "All personnel in the affected area are to evacuate the involved area immediately and report to your normal work location. If your normal work location is inaccessible, report to the O&M lunch room or TAC auditorium as conditions dictate." ☐
   - **Announce:**
     "Use of the PA and radio is restricted to emergency fire communications, except as directed by the Unit CRS for operational safety concerns." ☐
2. **Announce** the fire over Unit 1 and Unit 2 radio channels ☐

c. **IF** the investigating operator confirms a fire **AND** any of the following conditions exist,
THEN immediately **request** off site assistance by calling 911: ☐
   - Extreme force is necessary to gain entry into fire area ☐
   - A fire hose is required for fire suppression ☐
   - Fire is located outside the Protected Area, but within the Owner-Controlled Area. ☐

76. S209001 1

During a LOCA and LOOP on Unit One, the following plant conditions exist:

An Emergency Depressurization has been performed due to RPV water level

The Reactor Building -17 foot and 20 foot elevations are NOT accessible due to radiation levels.

ALL ECCS pumps are unavailable.

Which one of the following completes the statement below?

The CRS will direct demin water injection to the RPV, IAW 0EOP-01-LEP-01, *Alternate Coolant Injection,* Section:

A. 2.4.3.3a, *Demineralized Water Actions*, Inject demineralized water through Core Spray Loop A

B. 2.4.3.3c, *Demineralized Water Actions*, Inject demineralized water through RHR Loop A

C. 2.4.3.3d, *Demineralized Water Actions*, Inject demineralized water through HPCI

D. 2.4.3.3e, *Demineralized Water Actions,* Inject demineralized water through RCIC

Answer: A

K/A:

209001 Low Pressure Core Spray System

G2.4.35 Knowledge of local auxiliary operator tasks during an emergency and the resultant operational effects. (CFR: 41.10 / 43.5 / 45.13)

RO/SRO Rating: 3.8/4.0

Tier 2 / Group 1

K/A match: This question meets the K/A because an emergency condition exists in the stem (ED followed by no HP injection/building inaccesible). In addition, the SRO is required to have knowledge of the local emergency procedure and that it contains actions the AO must take in the field in order to inject (AO must locally open demin keepfill bypass valves for various injection sources). Contrasting the given conditions, the procedural knowledge of AO field actions, and system knowledge of valve locations will have the operational effect of determining which injection source is viable.

Pedigree: New

Objective: LOI-CLS-LP-300-J Obj 4a Given plant conditions, determine which system should be utilized to restore RPV water level and/or pressure when executing the following:
a. Alternate Cooling Injection Procedure with EOP-01-LEP-01.

Reference: None

Cog Level: High

Explanation: Demineralized water injection requires knowledge from the LEP that the system Keepfill Bypass Valves are to be opened. The only keepfill bypass valve that is not inaccessible is the Core Spray Loop A Keepfill Bypass valve. Therefore, Core Spray Loop A is the section to use in order to inject demin water to the RPV.

Distractor Analysis:

Choice A: Correct answer, see explanation.

Choice B: Plausible because it is a section in 2.4.3, it is incorrect because the keepfill bypass valve for RHR loop A is inaccessible.

Choice C: Plausible because it is a section in 2.4.3, it is incorrect because the keepfill bypass valve for HPCI is inaccessible.

Choice D: Plausible because it is a section in 2.4.3, it is incorrect because the keepfill bypass valve for RCIC is inaccessible.

SRO Basis: Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. [10 CFR 55.43(b)(5)]. Requires the SRO to evaluate the emergency conditions in the stem, and contrast those conditions with the given local emergency procedure sections and then select the appropriate procedure section.

2.4.3 Demineralized Water Actions

NOTE

0E&RC-0040 states that the Shift Manager may grant immediate access to a locked high radiation area to maintain the health and safety of plant personnel or the general public per 10CFR50.54(x) using the keys maintained in the Radiation Protection key locker near the Unit 2 AOG panel. ☐

1. **IF AT ANY TIME** a locked high radiation area is entered, without Radiation Protection support, **THEN** promptly notify RP. ☐ RO

2. Monitor and control MUD tank level greater than 14 feet. ☐ AO

NOTE

Demineralized water transfer pump capacity is 400 gpm. ☐

3. Perform for systems **NOT** operating **AND** available to provide injection to RPV: ☐ RO

   a. Inject demineralized water through Core Spray Loop A.

   (1) Ensure E21-F005A (Inboard Injection Vlv) OPEN ☐ RO

   (2) Ensure E21-F004A (Outboard Injection Vlv) OPEN ☐ RO

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2.4.3 Demineralized Water Actions (continued)

NOTE

E21-F028A is located on Reactor Building 50'. ☐

(3) Open E21-F028A (Core Spray Loop A Keepfill Station Bypass Valve) ☐ AO

c. Inject demineralized water through RHR Loop A.

(1) Ensure E11-F015A (Inboard Injection Vlv) OPEN ☐ RO

(2) Throttle E11-F017A (Outboard Injection Vlv) ☐ RO

NOTE

E11-F082A, E11-F085 and E11-F086A are located on the HPCI mezzanine ☐

(3) Open E11-F082A (RHR Loop A Keepfill Station Bypass Valve) ☐ AO

(4) Open E11-F085 (RHR System Demineralized Water Fill Valve) ☐ AO

d. Inject demineralized water through HPCI.

(1) Ensure E41-F012 (HPCI Min Flow Bypass To Torus Vlv) CLOSED ☐ RO

(2) At MCC XDA, Row H1, Compt B24 (HPCI Min Flo BPV To Supp Chamber Valve E41-F012) place breaker OFF ☐ AO

NOTE

E41-V100 is located in NRHR ☐

(3) Open E41-V100 (HPCI Keepfill Station Bypass Valve) ☐ AO

(4) Ensure E41-F007 (Pump Discharge Vlv) OPEN ☐ RO

e. Inject demineralized water through RCIC.

(1) Ensure E51-F019 (RCIC Min Flow Bypass To Torus Vlv) CLOSED ☐ RO

2.4.3 Demineralized Water Actions (continued)

(2) At MCC XDB, Row H1, Compt B47 (RCIC Min Flo Bypass To Supp Pool Vlv E51-F019), place breaker OFF ☐ AO

NOTE

E51-V70 is located in SRHR ☐

(3) Open E51-V70 (RCIC Keepfill Station Bypass Valve) ☐ AO

(4) Ensure E51-F012 (Pump Discharge Vlv) OPEN ☐ RO

77. S212000 1

Unit One is at rated power performing 0PT-01.1.6, *Reactor Protection System Manual Scram Test.*

The Reactor Scram System A pushbutton has been depressed.

RPS Trip System A Scram Groups light for groups one, two, three, and four are illuminated

(REFERENCE PROVIDED)

Which one of the following completes both statements below?

The scram pilot valve solenoids associated with these lights are __(1)__.

Tech Spec 3.3.1.1, *Reactor Protection System Instrumentation*, Condition B __(2)__ required to be entered.

A. (1) energized
   (2) is

B. (1) energized
   (2) is NOT

C. (1) de-energized
   (2) is

D. (1) de-energized
   (2) is NOT

Answer: B

K/A:

212000 Reactor Protection System

G2.2.44 Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions. (CFR: 41.5 / 43.5 / 45.12)

RO/SRO Rating: 4.2/4.4

Tier 2 / Group 1

K/A match: The applicant must interpret the indications (scram group lights) in comparison with the expected conditions for the given action (Sys A pushbutton depressed). The applicant must then use that knowledge to determine whether the system meets the given limited condition for operability as stated in the TS.

Pedigree: New

Objective: LOI-CLS-LP-003 Rev 3 Obj 27 Given plant conditions, determine whether given plant conditions meet minimum Technical Specification requirements associated with the Reactor Protection System.

Reference: TS 3.3.1.1

Cog Level: High

Explanation: **Part 1** When group lights are OFF that is indication that the solenoid valves are de-energized. All groups remain lit therefore, they are energized. **Part 2** ONLY Condition A and C are required to be entered, due to a failure of the A3 scram channel. Only one required channel is inoperable, but a loss of manual scram function has occurred resulting in RPS trip capability not maintained.

Distractor Analysis:

Choice A: **Part 1** is the correct Answer, see explanation. **Part 2** is plausible because Groups 1 through 4 are also in trip System B (although for the SV-118's), a novice applicant may assume that failure of these solenoids in Trip System A would mean they would not function in Trip System B. In addition, one channel per trip system is required, and since there are only lights for groups 1 through 4 in both trip systems, a novice applicant may assume that all required channels are inop.

Choice B: Correct Answer, see explanation.

Choice C: **Part 1** is plausible because ARI system uses energize to function valves. **Part 2** is plausible because Groups 1 through 4 are also in trip System B (although for the SV-118's), a novice applicant may assume that failure of these solenoids in Trip System A would mean they would not function in Trip System B. In addition, one channel per trip system is required, and since there are only lights for groups 1 through 4 in both trip systems, a novice applicant may assume that all required channels are inop.

Choice D: **Part 1** is plausible because ARI system uses energize to function valves. **Part 2** is the correct Answer, see explanation.

SRO Basis: Facility operating limitations in the TS and their bases. [10 CFR 55.43(b)(2)]. Requires the SRO to evaluate the failure of PT for RPS, and select the appropriate > 1 hour TS condition.

Table 3.3.1.1-1 (page 3 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
7. Scram Discharge Volume Water Level—High	1,2	2	G	SR 3.3.1.1.5 SR 3.3.1.1.9 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 106 gallons
	5[a]	2	H	SR 3.3.1.1.5 SR 3.3.1.1.9 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 106 gallons
8. Turbine Stop Valve—Closure	≥ 26% RTP	4	E	SR 3.3.1.1.5 SR 3.3.1.1.9 SR 3.3.1.1.13 SR 3.3.1.1.15 SR 3.3.1.1.16 SR 3.3.1.1.17	≤ 10% closed
9. Turbine Control Valve Fast Closure, Control Oil Pressure—Low	≥ 26% RTP	2	E	SR 3.3.1.1.5 SR 3.3.1.1.9 SR 3.3.1.1.13 SR 3.3.1.1.15 SR 3.3.1.1.16 SR 3.3.1.1.17	≥ 500 psig
10. Reactor Mode Switch—Shutdown Position	1,2	1	G	SR 3.3.1.1.12 SR 3.3.1.1.15	NA
	5[a]	1	H	SR 3.3.1.1.12 SR 3.3.1.1.15	NA
11. Manual Scram	1,2	1	G	SR 3.3.1.1.9 SR 3.3.1.1.15	NA
	5[a]	1	H	SR 3.3.1.1.9 SR 3.3.1.1.15	NA

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

3.3 INSTRUMENTATION

3.3.1.1 Reactor Protection System (RPS) Instrumentation

LCO 3.3.1.1 The RPS instrumentation for each Function in Table 3.3.1.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.1.1-1.

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required channels inoperable.	A.1 Place channel in trip.	12 hours
	OR	
	A.2 -----NOTE----- Not applicable for Functions 2.a, 2.b, 2.c, 2.d, or 2.f. ----- Place associated trip system in trip.	12 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. ——NOTE—— Not applicable for Functions 2.a, 2.b, 2.c, 2.d, or 2.f. —— One or more Functions with one or more required channels inoperable in both trip systems.	B.1	Place channel in one trip system in trip.	6 hours
	OR		
	B.2	Place one trip system in trip.	6 hours
C. One or more Functions with RPS trip capability not maintained.	C.1	Restore RPS trip capability.	1 hour
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1	Enter the Condition referenced in Table 3.3.1.1-1 for the channel.	Immediately
E. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	E.1	Reduce THERMAL POWER to < 26% RTP.	4 hours

(continued)

C71A-S3A, B; C72A-S3A, B

INSTRUMENT NUMBER: C71A-S3A, B; C72A-S3A, B
INSTRUMENT NAME: Manual Scram

TS REFERENCE: 3.3.1.1; TRM Table 3.3.1.1-1.11

TRIP CHANNEL: A3 - S3A
B3 - S3B

TRIP SYSTEM: A3 - S3A
B3 - S3B

TRIP LOGIC: A3 and B3 = Reactor scram

Place channel in TRIPPED condition by: Pull fuse

(Figure 03-26 and 03-27).

The Reactor Manual Scram relays deenergize the Scram pilot valve solenoids for the RPS Trip System.

- The SV-117 valves are in RPS Trip System A.
- The SV-118 valves are in RPS Trip System B.

Shorting links are normally installed around the auxiliary trip relay contacts in the Manual Scram circuits allowing this Scram signal to be bypassed. These shorting links are located in the back panels and are color coded red for identification.

NOTE:	The shorting links are removed prior to and during the time any control rod is withdrawn (except for control rods removed per Technical Specifications) during operation in refueling mode or during shutdown margin demonstration.

There are two shorting links per RPS Trip System, a total of four for the entire Reactor Protection System.

1.0 PURPOSE

This test is performed to determine the OPERABILITY of the Reactor Protection System Manual Scram function.

2.0 SCOPE

This procedure performs the following:

- A quarterly Channel Functional Test per TS SR 3.3.1.1.9 for Table 3.3.1.1-1 Function 11, Manual Scram.
- Satisfies a portion of the 24 month TS SR 3.3.1.1.15 Logic System Functional Test for Table 3.3.1.1-1 Function 11, Manual Scram.

3.0 PRECAUTIONS AND LIMITATIONS

1. A half-scram signal will exist until RESET .... ☐

4.0 GENERAL INFORMATION

The following annunciators will alarm during the performance of this test:

- A-05, 1-8, Reactor Manual Scram Sys A
- A-05, 2-8, Reactor Manual Scram Sys B

5.0 ACCEPTANCE CRITERIA

1. This test may be considered satisfactory when all of the following criteria are met:

   a. A trip is indicated on RPS A and alarmed on RTGB Panel H12-P603 when C71(C72)-S3A (Manual Reactor Scram System A) push button is depressed.

   b. A trip is indicated on RPS B and alarmed on RTGB Panel H12-P603 when C71(C72)-S3B (Manual Reactor Scram System B) push button is depressed.

   c. The Scram valve solenoids are DE-ENERGIZED when the associated RPS is tripped.

7.0 INSTRUCTIONS

7.1 Test Preparation

1. Obtain Unit CRS permission to perform this test. .... _______

2. Ensure all prerequisites listed in Section 6.0, Prerequisites are met. .... _______

> **NOTE**
>
> The length of time a half-scram is sealed-in is to be minimized. .... ☐

3. **IF** during the performance of this procedure, the expected test results from a half-scram initiation are **NOT** observed,
THEN immediately reset the half-scram and notify the Unit CRS. .... _______

7.2 Manual Scram A Test

1. Depress C71(C72)-S3A (Manual Reactor Scram System A), push button and observe the following actions occur:

   a. Plant Process Computer Event Log displays Manual Scram Channel A Trip (Computer Point D533). .... _______

   b. **IF** the proper Plant Process Computer Event Log (Computer Point D533) was **NOT** received in Step 1.a,
   THEN generate a WO. .... _______

   c. Manual Reactor Scram System A push button light comes ON. ..... _______

   d. A-05, 1-8, Reactor Manual Scram Sys A, ALARMS. .... _______

   e. RPS Trip System A Scram Group lights 1, 2, 3, and 4 located on Panel H12-P603 are OFF, indicating Scram valve solenoids are DE-ENERGIZED. .... _______

   f. RPS Trip System A Scram Group lights 1, 2, 3, and 4 located on RPS A Panel H12-P609 are OFF, indicating Scram valve solenoids are DE-ENERGIZED. .... _______

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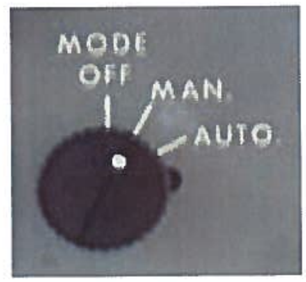
Unit Two is at rated power. A TIP trace is in progress.

TIP D Valve Control Unit and Monitor indications are as follows:



(REFERENCE PROVIDED)

Which one of the following completes both statements below?

Tip Valves are Group __(1)__ PCIVs.

Tech Spec 3.6.1.3, *Primary Containment Isolation Valves (PCIVs)*, Condition(s) __(2)__ is/are required to be entered.

A. (1) 2
   (2) A ONLY

B. (1) 2
   (2) A and B

C. (1) 6
   (2) A ONLY

D. (1) 6
   (2) A and B

Answer: A

K/A:

215001 Traversing In-Core Probe

G2.2.44 Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions.
(CFR: 41.5 / 43.5 / 45.12)

RO/SRO Rating: 4.2/4.4

Tier 2 / Group 2

K/A match: The applicant must interpret the given TIP system indications and controls to verify the status of the system. The applicant must then compare those indications and controls with the required conditions for the given mode as stated in the TS to determine the appropriate condition.

Pedigree: New

Objective: LOI-CLS-LP-009.5 Obj 9 Determine whether given plant conditions meet minimum Technical Specification requirements, including the Bases, associated with the Traversing Incore Probe System.

Reference: TS 3.6.1.3

Cog Level: High

Explanation: Part 1: Tip Valves are Group 2 PCIVs. Part 2: Squib Valve Monitor Light On indicates that the squib valve continuity is lost. Therefore, The Shear Valve in the Ball valve and shear valve assembly is inoperable. The ball valve remains operable. Therefore, Condition A is entered for one PCIV inoperable.

Distractor Analysis:

Choice A: Correct Answer, see explanation.

Choice B: Part 1 is plausible because it is correct, see explanation. Part 2 is plausible because squib Valve Monitor Light On indicates that the squib valve continuity is lost. Therefore, The Shear Valve in the Ball valve and shear valve assembly is inoperable. In addition, a novice applicant may assume that with the MODE switch in MANUAL, the ball valve would be inoperable as well. (However, the only position of the MODE switch that would make the TIP inoperable is OFF.) With Two PCIVS inoperable, condition B would be entered,

Choice C: Part 1 is plausible because group six PCIVs also isolate on LL1 and High DW pressure. Part 2 is plausible because it is correct, see explanation.

Choice D: Part 1 is plausible because group six PCIVs also isolate on LL1 and High DW pressure. Part 2 is plausible because squib Valve Monitor Light On indicates that the squib valve continuity is lost. Therefore, The Shear Valve in the Ball valve and shear valve assembly is inoperable. In addition, a novice applicant may assume that with the MODE switch in MANUAL, the ball valve would be inoperable as well . (However, the only position of the MODE switch that would make the TIP inoperable is OFF.) With Two PCIVS inoperable, condition B would be entered,

SRO Basis: Facility operating limitations in the TS and their bases. [10 CFR 55.43(b)(2)]. The SRO applicant is required to select the appropriate > 1 hour TS condition based on the status of the TIP system indications.

Table 09.5-2 - Valve Control Monitor Indications

Indication	Comment
Squib Monitor Light	ON indicates that the TIP Shear Valve squib circuit continuity has been lost.
Shear Valve Monitor Light	ON indicates that the squib charge in the TIP Shear Valve has been detonated.
TIP Ball Valve OPEN Light	ON indicates that TIP Ball Valve is OPEN.
TIP Ball Valve CLOSED Light	ON indicates that TIP Ball Valve is CLOSED.
Time Delay Light	ON indicates that the TIP Ball Valve was not OPEN within 6 seconds from when the detector left the in-shield position and the TIP Drive Motor should have stopped.
Purge Light	Indicates that the solenoid for the Indexer Purge System should be energized OPEN.
Fuse F5 Continuity Light	Indicates power to PCIS Group 2 Bus in drawer is available (Fuse F5 is not blown).

Table 09.5-3 - TIP Drive Control Unit Indications

Indication	Comment
DETECTOR POSITION (illuminated digits)	Dynamic digital display of detector position. ("0001" - reference point about one foot behind the Indexer; "9750" - "In Shield position)
CORE LIMIT (illuminated digits)	Static digital display of pre-programmed core top or bottom limits of selected channel.
READY Light	Indicates that Indexer is properly aligned to selected channel.
CORE TOP Light	Detector is at top of core.
IN CORE Light	Detector is above core bottom limit.
IN SHIELD Light	Detector is in Shield Chamber.
SCAN Light	Axial Flux Profile is being recorded.
LOW Speed Light	Detector is being driven at 3 inches per second, (15 feet per minute).
REV (Reverse) Light	Detector moving away from top of core.
FWD (Forward) Light	Detector moving towards top of core.
VALVE Light	ON if TIP Ball Valve is CLOSED.

Table 09.5-4 - P601 TIP Indications

Indication	Comment
TIP Valve Status - Green Light	Green Light ON indicates that each TIP Ball Valve is FULL CLOSED.
TIP Valve Status - Red Light	Red Light ON indicates that a TIP Ball Valve is NOT FULL CLOSED.

Low Speed

OFF Makes low-speed drive a function of detector position

and independent of operator control.

ON Initiates continuous low-speed detector drive.

Core Limit

TOP Permits digital display of selected channel

pre-programmed top-core limit which corresponds to top of active fuel.

BOTTOM As above, except pre-programmed core bottom limit

is displayed. (The core top and bottom numbers are different for each TIP channel because of different lengths of guide tube run).

Mode (Switch S-7)

OFF Deenergizes power supplies in Drive Control Unit.

MANUAL Positions detector in conjunction with the FWD and

REV position of manual switch S-3.

AUTO Permits automatic mode of operation when Auto start

S-2 is pressed.

Manual Valve Control (Spring Return to Closed Position)

CLOSED Permits TIP Ball Valve to open automatically when

mechanism is operated.

OPEN Opens TIP Ball Valve without energizing the Drive

in the TIP Drive Mechanism.

X-Y Recorder (Figure 09.5-14)

Alternate plotting capability via "AFORA" software.

Controls on the X-Y recorder drawer are as follows:

4.0 SYSTEM OPERATION

4.1 Normal Operational Relationships

TIP traces are required to be performed periodically. Technical Specifications require calibration of LPRM detectors at least once every effective full power month (EFPM). Traces are done to obtain new Gain Adjustment Factors (GAF) as calculated by the process computer for each LPRM. These GAFs can be used to adjust the current applied to the LPRM detectors. Current adjustment is required due to loss of detector sensitivity which results from exposure to the fission process in the core. The TIP Detector is run through the dry tube within the LPRM assembly containing the LPRM to be calibrated. A comparison is made between the TIP output and the existing LPRM reading and a GAF is calculated. The current applied to the LPRM detector is adjusted, if necessary.

TIP Detector calibration is also periodically required. Capability for calibration of TIP probes is provided by a common channel which can align each TIP to the center LPRM assembly (28-29) (Figure 09.5-15) and TIP Dry Tube. Each TIP is run through the center LPRM assembly. Readings from the TIPs are compared and the gains are adjusted to meet an average value of the four TIP readings and so that the gains for the TIP channels fall within a specified band.

The TIP System can be operated in an Automatic mode or a Manual mode. OP-09.1 in conjunction with 0ENP-24.15 covers precautions, initial conditions, and specific instructions related to the operation of the TIP System.

Regardless of the mode of operation there are some precautions that must be observed. The TIP Machine should never be turned off with the detector inserted past the TIP Ball Valve. This conditions prevents the isolation logic from retracting the detector and closing the TIP Ball Valve should an isolation signal be received. Also, the TIP Machines should not be left unattended if the detectors are in motion.

4.1.1 Automatic Operation Sequence

The mode switch on the selected Drive Control Unit is placed in AUTO and the manual switch and low speed switch remain in OFF. This gives a permissive to the TIP logic to be run automatically.

The auto start pushbutton (Drive Control Unit) is then pressed. The detector will automatically move from Shield Chamber to the entrance of the Indexer at low speed (3"/sec or 15 ft/min) and will then stop.

3.6 CONTAINMENT SYSTEMS

3.6.1.3 Primary Containment Isolation Valves (PCIVs)

LCO 3.6.1.3 Each PCIV, except reactor building-to-suppression chamber vacuum breakers, shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,

When associated instrumentation is required to be OPERABLE per LCO 3.3.6.1, "Primary Containment Isolation Instrumentation."

ACTIONS

------------------------------------NOTES------------------------------------

1. Penetration flow paths may be unisolated intermittently under administrative controls.
2. Separate Condition entry is allowed for each penetration flow path.
3. Enter applicable Conditions and Required Actions for systems made inoperable by PCIVs.
4. Enter applicable Conditions and Required Actions of LCO 3.6.1.1, "Primary Containment," when PCIV leakage results in exceeding overall containment leakage rate acceptance criteria.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. -----NOTE----- Only applicable to penetration flow paths with two PCIVs. ---------- One or more penetration flow paths with one PCIV inoperable except for MSIV leakage not within limit.	A.1	Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured.	8 hours
	AND		

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. -----NOTE----- Only applicable to penetration flow paths with two PCIVs. ---------- One or more penetration flow paths with two PCIVs inoperable except for MSIV leakage not within limit.	B.1	Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.	2 hours
C. -----NOTE-----	C.1	Isolate the affected	8 hours except for

TABLE 12-2
Primary Containment Isolation System Group Isolation Instrumentation Setpoints

ISOLATION GROUP	ISOLATION SIGNAL	TRIP SETPOINT Tech Spec. Allowable	TRIP SETPOINT Actual (Note 1)	NOTES
Group 5	High Steam Flow Low Steam Pressure High Turb Exh Pressure Steam Line Area Hi Temp Steam Line Tunnel High Amb Temp Steam Line Tunnel dT High Equip Area High Temp Equip Area dT High	≤ 275% ≥ 53psig ≤ 6 psig ≤ 175°F ≤ 200°F ≤ 50°F ≤ 175°F ≤ 50°F	220% 70 psig 5 psig 165°F 165°F/190°F 47°F 165°F 47°F	Note 5 Note 4 Note 4 Note 4
Group 6	Low Level #1 High Drywell Pressure Rx Bldg Exhaust Hi Rad Rx Bldg Exhaust Hi Temp High Main Stack Rad	≥ 153" ≤ 1.8 psig ≤ 16 mR/hr N/A ODCM	166" 1.7 psig 4 mR/hr 135°F* ODCM	 Note 6 Note 2
Group 7	Low Steam Pressure AND High Drywell Pressure	≥ 104 psig ≤ 1.8 psig	115 psig 1.7 psig	
Group 8	Low Level #1 High Steam Dome Pressure	≥ 153" ≤ 137 psig	166" 130.8 psig	
Group 9	Low Steam Pressure AND High Drywell Pressure	≥ 53 psig ≤ 1.8 psig	70 psig 1.7 psig	
Group 10	Low Level #3 High Drywell Pressure AND Low Reactor Pressure	N/A	45" 1.7 psig 410 psig	

Note 1: All "Actual" values from TRM
Note 2: Stack radiation high level is calculated in accordance with the Offsite Dose Calculation Manual.
Note 3: After a 28.5 minute time delay
Note 4: After 27 minute time delay
Note 5: After a 5 second time delay
*Note 6: Specific "Actual" values from EOP User's Guide, Attachment 1

TABLE 12-2
Primary Containment Isolation System Group Isolation Instrumentation Setpoints

ISOLATION GROUP	ISOLATION SIGNAL	TRIP SETPOINT Tech Spec. Allowable Value	TRIP SETPOINT Actual (Note 1)	NOTES
Group 1	Low Level #3 Main Steam Line High Temp Turbine Bldg Area Hi Temp Main Steam Line High Flow Not in RUN Low Condenser Vacuum Low Steam Pressure	≥ +13" ≤ 197°F N/A ≤ 138% ≤ 33% ≥ 7.5" Hg ≥ 825 psig	45" 190°F 190°F* 137% 30% 10" Hg 835 psig	 Note 6 - Unit 2 only
Group 2	Low Level #1 High Drywell Pressure	≥ +153" ≤ 1.8 psig	166" 1.7 psig	
Group 3	Low Level #2 High Diff Flow Area High Temp Area Vent dT High HELB Isolation NRHX Outlet Temp High SLC Initiation	≥ 101" ≤ 73 gpm ≤ 150°F ≤ 50°F ≤ 120°F N/A N/A	105" 43 gpm 140°F 47°F 115°F 135°F* N/A	 Note 3 Note 6
Group 4	High Steam Flow Low Steam Pressure High Turb Exh Pressure Steam Line Area Hi Temp Steam Line Tunnel High Amb Temp Steam Line Tunnel dT High Equip Area High Temp	≤ 275% ≥ 104 psig ≤ 9 psig ≤ 200°F ≤ 200°F ≤ 50°F ≤ 175°F	220% 115 psig 7 psig 165°F 165°F/190°F 47°F 165°F	Note 5

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Unit One is operating at rated power.
Unit Two is in MODE 5 with UAT backfeed established.
A main generator backup lockout occurs on Unit One.

(REFERENCE PROVIDED)

Which one of the following completes both statements below?

All four diesels ___(1)___ automatically start.

IAW Unit One Tech Spec 3.8.1, *AC sources Operating*, Condition E ___(2)___ required to be entered.

A. (1) will
   (2) is

B. (1) will
   (2) is NOT

C. (1) will NOT
   (2) is

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

Answer: D

K/A:

262001 A.C. Electrical Distribution

A2 Ability to (a) predict the impacts of the following on the A.C. ELECTRICAL DISTRIBUTION ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5 / 45.6)

09 Turbine/generator trip

RO/SRO Rating: 2.9/3.0

Tier 2 / Group 1

K/A Match: This meets the K/A because the candidate is required to predict the status of the EDGs and determine whether the appropriate TS condition is entered.

Pedigree: New

Objective: LOI-CLS-LP-050 Obj 17 Given plant conditions, determine the required action(s) to be taken in accordance with Technical Specifications associated with the 230 KV Electrical Distribution System. **(LOCT)** (SRO Only)

Reference: T.S. 3.8.1 (blank out the LCO statement and Applicability)

Cog Level: High

Explanation: Part 1: DGs do not auto start on a generator backup lockout. Part 2: Condition E is not entered because a loss of two offsite circuits has not occurred. Only one offsite circuit is lost, the Unit One UAT.

Distractor Analysis:

Choice A: Part 1 is plausible because all four DGs start on a generator primary lockout. Part 2 is plausible because if UAT was not in backfeed on Unit Two this would be the case.

Choice B: Part 1 is plausible because all four DGs start on a generator primary lockout. Part 2 is plausible because it is correct, see explanation.

Choice C: Part 1 is plausible because it is correct, see explanation. Part 2 is plausible because if UAT was not in backfeed on Unit Two this would be the case.

Choice D: Correct Answer, see explanation.

SRO Basis: Facility operating limitations in the TS and their bases. [10 CFR 55.43(b)(2)]. Part two requires knowledge of the TS 3.8.1 bases to determine if Unit Two on UAT backfeed can qualify as an offsite source. In addition it requires the appropriate determination of the TS condition application.

3.2.6 Main Transformer, UAT, and Main Generator Protection

1. The Main Generator, MPT, and UAT are all three protected by the Generator/Transformer Primary (86GP) and Backup (86GB) Lockout Relays. The Main Generator is provided additional protection by the Main Generator Differential Lockout Relays (86G).

   a. Generator/Transformer Primary, Backup, and Main Generator Differential Relay trip actions are as follows:

      (1)Main Generator Output breakers trip and lock out.

      (2)Main turbine trips.

      (3)Main generator exciter field breaker trips and locks out.

      (4)UAT 4160 supply breakers to B, C and D buses trip and lock out.

      (5)SAT feeders to C and D buses auto close.

      (6)Four diesel generators auto start for the Main Generator Differential Lockout or the Generator/Transformer Primary Lockout. They do not auto start for a Generator/Transformer Backup Lockout.

Offsite power is supplied to the 230 kV switchyards from the transmission network by eight transmission lines. From the 230 kV switchyards, two qualified electrically and physically separated circuits provide AC power, through either a startup auxiliary transformer (SAT) or backfeeding via a unit auxiliary transformer (UAT), to 4.16 kV BOP buses. A single circuit path (master/slave breakers and interconnecting cables) from each BOP bus provides offsite power to its associated downstream 4.16 kV emergency bus. A detailed description of the offsite power network and circuits to the onsite Class 1E emergency buses is found in the UFSAR, Sections 8.2 and 8.3 (Ref. 2).

A qualified offsite circuit consists of all breakers, transformers, switches, interrupting devices, cabling, and controls required to transmit power from either 230 kV bus (bus A or B) to the onsite Class 1E emergency buses.

The Unit 1 main generator provides the normal source of power to 4.16 kV emergency buses E1 and E2 via its respective UAT. The Unit 2 main generator provides the normal source of power to 4.16 kV emergency buses E3 and E4 via its respective UAT. In the event of a

(continued)

BACKGROUND (continued)

unit trip, an automatic transfer from the normal circuit (main generator output via the UAT) to the respective unit SAT occurs resulting in the SAT supplying power to two 4.16 kV emergency buses. As such, the Unit 1 SAT provides the preferred source of power to emergency buses E1 and E2 and the Unit 1 UAT (backfeed mode) is the alternate source of power to emergency buses E1 and E2. The Unit 2 SAT provides the preferred source of power to emergency buses E3 and E4 and the Unit 2 UAT (backfeed mode) is the alternate source of power to emergency buses E3 and E4. Each UAT can only be considered a qualified offsite source if it is capable of being powered from the 230 kV switchyard (Ref. 3).

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

b. Four diesel generators (DGs); and

c. Two Unit 2 qualified circuits between the offsite transmission network and the onsite Class 1E AC Electrical Power Distribution System.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

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

E. Two or more offsite circuits inoperable for reasons other than Condition B.	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 all but one offsite circuit to OPERABLE status.	24 hours

(continued)

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Unit One was operating at power when a Group 1 isolation and reactor scram occurred.

Reactor pressure is 950 psig and being manually controlled by SRVs.

An SRV is stuck open with a stuck open **SRV tailpipe** vacuum breaker.

Torus and Drywell sprays have been initiated IAW PCCP

(REFERENCE PROVIDED)

Which one of the following completes both statements below?

The SRV is discharging through the open vacuum breaker directly into the ___(1)___.

The **highest** EAL classification for this event is a(n) ___(2)___.

A. (1) drywell
   (2) Alert

B. (1) drywell
   (2) Site Area Emergency

C. (1) suppression chamber air space
   (2) Alert

D. (1) suppression chamber air space
   (2) Site Area Emergency

Answer: A

K/A:

239002 Safety Relief Valves

A2 Ability to (a) predict the impacts of the following on the RELIEF/SAFETY VALVES ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5 / 45.6)

01 Stuck open vacuum breakers

RO/SRO Rating: 3.0/3.3

Tier 2 / Group 1

K/A match: THIS QUESTION WAS PRE-SUBMITTED FOR APPROVAL.
The applicant must determine that the effect of the stuck open vacuum breaker on the SRV is that it is now bypassing the pressure suppression function and discharging directly into the DW. The applicant must also determine that the ultimate effect of the given conditions is that the RCS barrier has been lost, and that the consequences of this conditions and its potential effects on the health and safety of the public are mitigated by declaring an ALERT IAW with 0PEP-2.1.

Pedigree: New

Objective: LOI-CLS-LP-020 Obj 15e Given plant conditions, predict how ADS/SRVs will be affected by the following: Failure of the SRV tailpipe vacuum breakers.

Reference: 0PEP-02.1

Cog Level: High

Explanation: Part 1: SRV tailpipe vacuum breakers relieve to the drywell, therefore a stuck open tailpipe breaker with a stuck open SRV would discharge steam directly into the drywell. Part 2: With a stuck open relief valve and stuck open vacuum breaker, a LOCA is occurring. With Torus pressure sufficient to warrant drywell sprays (11.5 psig) drywell pressure has more than exceeded 1.7 psig. Therefore the conditions for a loss of the RCS barrier (Primary Containment Pressure >1.7psig due to RCS leakage) are met and an ALERT should be declared based on FA1.1 loss of RCS barrier.

Distractor Analysis:

Choice A: Correct Answer, see explanation.

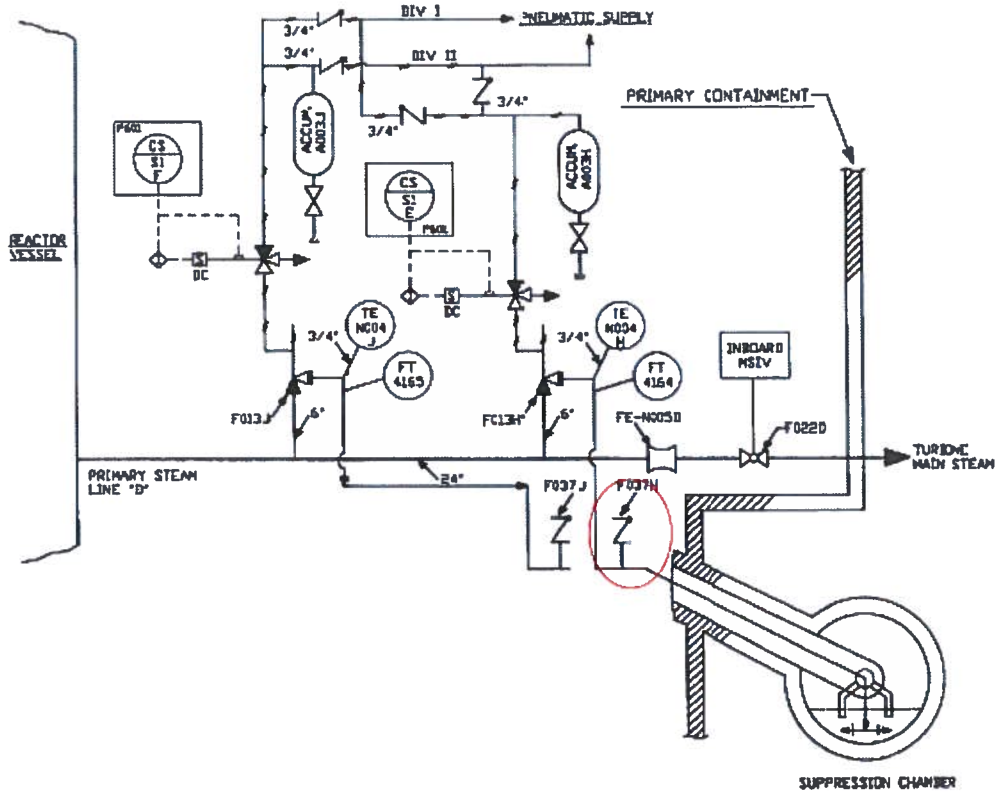
Choice B: Part 1 is plausible because it is correct, see explanation. Part 2 is plausible because a group 1 isolation has occurred which is a primary containment isolation signal, an unisolable LOCA is occurring, and a novice candidate might assume that this meets the conditions for a loss of the containment barrier (Unisolable direct downstream pathway to the environment exists after primary containment isolation signal), this loss coupled with the loss in the explanation would meet the criteria for a loss of two barriers, and a declaration of an SAE IAW FS1.1 They also might think that this condition would not be consistent with a LOCA.

Choice C: Part 1 is Plausible because with a failed open Suppression Chamber to Drywell Vacuum Breaker would allow DW steam to go directly to the suppression Chamber Air Space. in addition, a failed open SRV with a broken tailpipe and broken downcomer would directly pressurize the torus air space. Part 2 is plausible because it is correct, see explanation.

Choice D: Part 1 is Plausible because with a failed open Suppression Chamber to Drywell Vacuum Breaker would allow DW steam to go directly to the suppression Chamber Air Space. in addition, a failed open SRV with a broken tailpipe and broken downcomer would directly pressurize the torus air space. Part 2 is plausible because a group 1 isolation has occurred which is a primary containment isolation signal, an unisolable LOCA is occurring, and a novice candidate might assume that this meets the conditions for a loss of the containment barrier (Unisolable direct downstream pathway to the environment exists after primary containment isolation signal), this loss coupled with the loss in the explanation would meet the criteria for a loss of two barriers, and a declaration of an SAE IAW FS1.1. They also might think that this condition would not be consistent with a LOCA.

SRO Basis: Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. [10 CFR 55.43(b)(5)] The SRO applicant is required to select the appropriate ALERT emergency classification based on the given conditions which equates to a loss of the RCS barrier.

FIGURE 20-2
Typical SRV Piping Arrangement



permissives.

4.2.2 Failure of the SRV Tailpipe Vacuum Breakers

In the event an SRV tailpipe vacuum breaker fails in the open position, the result is a direct path of steam from the reactor to the drywell (i.e., LOCA). In the event an SRV vacuum breaker fails in the closed position, the result is the possible creation of a vacuum in the tailpipe upon closure of an SRV, resulting in the drawing of water into the SRV tailpipe.

FS1.1 1 2 3	FA1.1 1 2 3
Loss or potential loss of any two barriers (Table F-1)	Any loss or any potential loss of either Fuel Clad or RCS (Table F-1)

able F-1 Fission Product Barrier Threshold Matrix

	Reactor Coolant System Barrier		Containment
ss	Loss	Potential Loss	Loss
ied	1. RPV level cannot be restored and maintained > TAF or cannot be determined	None	None
	1. UNISOLABLE break in any of the following: - Main steam line - HPCI steam line - RCIC steam line - RWCU - Feedwater 2. Emergency Depressurization is required	1. UNISOLABLE primary system leakage that results in exceeding EITHER of the following: • One or more Secondary Containment area radiation Maximum Normal Operating Limits (0EOP-03-SCCP Table SC-3) • One or more Secondary Containment area temperature Maximum Normal Operating Limits (0EOP-03-SCCP Table SC-1)	1. UNISOLABLE primary system leakage that results in exceeding one or more Secondary Containment area temperature Maximum Safe Operating Limits (0EOP-03-SCCP Table SC-1)
	1. Primary Containment pressure > 1.7 psig due to RCS leakage	None	1. UNPLANNED rapid drop in Primary Containment pressure following Primary Containment pressure rise 2. Primary Containment pressure response not consistent with LOCA conditions
	1. Drywell radiation > 27 R/hr with reactor shutdown	None	None
	None	None	1. UNISOLABLE direct downstream pathway to the environment exists after Primary Containment isolation signal

and assumptions consistent with the DBA-LOCA analysis.

4.2.7 Containment Response with Vacuum Breaker Failure

1. **Suppression Chamber to Drywell Vacuum Breakers Failed Open**

   Steam flows from drywell to suppression chamber through the open vacuum breakers equalizing pressure between the two immediately. The steam is not forced through the water of the suppression pool; therefore it will operate only as a surface condenser. As a result, the drywell pressure will probably exceed the design pressure. To prevent this occurrence, light indication is provided for each vacuum breaker. It indicates if the valve is off its seat and is displayed in the Control Room.

2. **Suppression Chamber to Drywell Vacuum Breakers Failed Closed**

   The steam in the drywell will condense and drywell pressure will decrease. With vacuum breakers failed shut, pressure cannot equalize between the suppression pool and the drywell. The pressure in the drywell may decrease such that the suppression chamber to drywell differential pressure limit (10 psid) is exceeded. Vent pipe buckling can occur if suppression chamber pressure is 10 psia greater than drywell pressure. To prevent this situation, the vacuum breakers are operationally checked periodically and 133% capacity is provided with ten vacuum breakers.

81. S219000 1
Unit Two is operating at rated power with RHR Loop A operating in suppression pool cooling mode.

A-01 (2-8) *RHR Relay Logic Pwr Failure,* is in alarm due to a blown fuse affecting RHR Logic A ONLY.

(REFERENCE PROVIDED)

Which one of the following completes both statements below?

IAW Tech Spec 3.3.5.1, *ECCS Instrumentation*, ~~the~~ any required channels ___(1)___ required to be placed in trip within 24 hours.

If a LOCA signal were to occur, 2-E11-F015A, Inboard Injection Vlv, ___(2)___ open automatically on low reactor pressure.

A. (1) are
   (2) will

B. (1) are NOT
   (2) will

C. (1) are
   (2) will NOT

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

Answer: A
K/A:
219000 RHR/LPCI: Torus/Suppression Pool Cooling Mode
A2 Ability to (a) predict the impacts of the following on the RHR/LPCI: TORUS/SUPPRESSION POOL COOLING MODE ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5 / 45.6)
12 Valve logic failure

RO/SRO Rating: 3.0/3.1

Tier 2 / Group 2

K/A Match: This meets the K/A because it is testing whether RHR in suppression pool cooling mode will transfer to LPCI injection mode with a failure of divisional valve logic, and whether the LCO action statement for condition B should be applied.

Pedigree: New

Objective: LOI-CLS-LP-017 Obj 7. Given plant conditions, determine if the RHR system should automatically initiate in the LPCI mode. (LOCT)

Reference: T.S. 3.3.5.1

Cog Level: High

Explanation: Part 1: With the plant at rated power and torus cooling in service on loop A, RHR loop A is not in its normal standby lineup. The loss of Div I RHR relay logic power, will result in the failure of the A loops suppression cooling valves (f024/28) to automatically close on a LPCI initiation. However, the F015A will auto open from the div 2 logic. Part 2: The loss of power to the Div I RHR logic will result in the loss of the functions applicable to condition C.

Distractor Analysis:

Choice A: This is the Correct answer, see explanation.

Choice B: Part 1 is plausible because a novice candidate might believe the note that required action B.2. is only applicable to functions 3a and 3b also applies to condition B.3. Part 2 is plausible because it is correct, see explanation.

Choice C: Part 1 is plausible because it is correct, see explanation. Part 2 is plausible because a candidate might believe a loss of Div I logic would prevent the auto function of the F015a, since the F024 and F028 will not automatically close under these conditions.

Choice D: Part 1 is plausible because a novice candidate might believe the note that required action B.2. is only applicable to functions 3a and 3b also applies to condition B.3. Part 2 is plausible because a candidate might believe a loss of Div I logic would prevent the auto function of the F015a, since the F024 and F028 will not automatically close under these conditions.

SRO Basis: Facility operating limitations in the TS and their bases. [10 CFR 55.43(b)(2)]. The SRO applicant is required to select whether an action statement is applicable given a loss of Div I RHR logic.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. (continued)	B.2	-------------NOTE------------- Only applicable for Functions 3.a and 3.b. -------------------------------------	
		Declare High Pressure Coolant Injection (HPCI) System inoperable.	1 hour from discovery of loss of HPCI initiation capability
	AND		
	B.3	Place channel in trip.	24 hours
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.d, 2.c, 2.d, and 2.f. -------------------------------------	
		Declare supported feature(s) inoperable when its redundant feature ECCS initiation capability is inoperable.	1 hour from discovery of loss of initiation capability for feature(s) in both divisions
	AND		
	C.2	Restore channel to OPERABLE status.	24 hours

(continued)

Table 3.3.5.1-1 (page 1 of 4)
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 Level 3	1,2,3, 4(a), 5(a)	4	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.4 SR 3.3.5.1.5	≥ 13 inches
b. Drywell Pressure—High	1,2,3	4	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.4 SR 3.3.5.1.5	≤ 1.8 psig
c. Reactor Steam Dome Pressure—Low	1,2,3	4	C	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.4 SR 3.3.5.1.5	≥ 402 psig and ≤ 425 psig
	4(a), 5(a)	4	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.4 SR 3.3.5.1.5	≥ 402 psig and ≤ 425 psig
d. Core Spray Pump Start—Time Delay Relay	1,2,3, 4(a), 5(a)	2 1 per pump	C	SR 3.3.5.1.4 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 14 seconds and ≤ 16 seconds
2. Low Pressure Coolant Injection (LPCI) System					
a. Reactor Vessel Water Level—Low Level 3	1,2,3, 4(a), 5(a)	4	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.4 SR 3.3.5.1.5	≥ 13 inches
b. Drywell Pressure—High	1,2,3	4	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.4 SR 3.3.5.1.5	≤ 1.8 psig

(continued)

<< Plant Effects from Loss of DC Distribution Panel 3A(4A) >>

RCIC: Will NOT shutdown on reactor high water level, inboard isolation logic INOPERABLE (E51-F007, -F031, and -F062 will NOT auto close). Valves E51-F005 and -F025 fail closed.

ADS: ADS Logic B is INOPERABLE. ADS will initiate from ADS Logic A if Core Spray Pump B or both RHR Loop B pumps are operating.

HPCI: Will NOT auto initiate, outboard isolation logic INOPERABLE (E41-F003, -F041, and -F075 will NOT auto close), HPCI flow controller and EGM INOPERABLE (no flow control or indication), HPCI trip logic INOPERABLE, valves E41-F053, -F054, and -F026 fail closed. HPCI isolation is required in accordance with APP 1(2)-A-01 5-5.

CS Loop A: Will NOT auto initiate (manual operation possible but minimum flow valve will NOT auto open, and injection valves can NOT be opened simultaneously).

RPS Logic A: Will NOT have 10 second time delay prior to reset of full scram, power lost to backup scram valves.

RHR Loop A: Will auto initiate from RHR Logic B, however the following effects exist: Pumps can NOT be restarted if stopped by control switch, pumps will NOT trip on No Suction Path Interlock, LOCA interlocks NOT functional, min flow valve will NOT auto open, Loop A Containment Spray can NOT be initiated. If a loss of DC Distribution Panel 3B(4B) has also occurred, RHR Loop A will NOT auto initiate (manual operation possible).

82. S261000 1

Unit Two is operating at rated power.

Subsequently, a Div I pneumatic leak occurs causing drywell pressure to rise to 1.9 psig.

Which one of the following completes both statements below?

The SBGT trains __(1)__ running.

The Div I pneumatics are required be isolated IAW __(2)__.

A. (1) are NOT
   (2) 0EOP-01-SEP-16, *Drywell Systems Isolation*

B. (1) are NOT
   (2) 0AOP-20.0, *Pneumatic (Air/Nitrogen) System Failures*

C. (1) are
   (2) 0EOP-01-SEP-16, *Drywell Systems Isolation*

D. (1) are
   (2) 0AOP-20.0, *Pneumatic (Air/Nitrogen) System Failures*

Answer: C

K/A:

261000 Standby Gas Treatment System

A2 Ability to (a) predict the impacts of the following on the STANDBY GAS TREATMENT SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5 / 45.6)

09 Plant air system failure

RO/SRO Rating: 2.4/2.6

Tier 2 / Group 1

K/A match: A failure of plant air in the drywell would require that the drywell be vented due to increasing DW pressure. In this case, since the impact of the failure is DW pressure increasing > 1.7 psig, the specific impact on SBGT is that it can not be used to vent containment. The procedure to mitigate the consequences of the failure is the leak is required to be isolated using SEP-16. (NOTE: the conditions are NOT entry conditions for the procedures)

Pedigree: New

Objective: LOI-CLS-LP-046A Obj 13. Predict the effect that a loss or malfunction of the Pneumatic System would have on plant operation.

Reference: None

Cog Level: High

Explanation: With drywell pressure >1.7 psig, 2OP-10 cannot be used to vent the drywell. SBGT has automatically started at 1.7 psig. With drywell pressure > 1.7 psig, PCCP directs the use of SEP-16 to isolate containment leaks, and SEP 16 contains the steps to isolate DIV I pneumatics.

Distractor Analysis:

Choice A: Part 1 is plausible because PCCP directs 2OP-10 to control DW pressure <1.7 psig. Part 2 is plausible because it is correct, see explanation.

Choice B: Part 1 is plausible because PCCP directs 2OP-10 to control DW pressure <1.7 psig. Part 2 is plausible because AOP-20 directs the isolation of various pneumatic sources, but while executing the EOPs the EOPs take precedence.

Choice C: Correct Answer, see explanation.

Choice D: Part 1 is plausible because it is correct, see explanation. Part 2 is plausible because AOP-20 directs the isolation of various pneumatic sources, but while executing the EOPs the EOPs take precedence.

SRO Basis: Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. [10 CFR 55.43(b)(5)]. The applicant is required to select the appropriate procedure for venting the drywell and isolating pneumatics based on the given conditions.

DRYWELL SYSTEMS ISOLATION	0EOP-01-SEP-16
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2.2 Drywell Pneumatics Isolation

2.2.1 Manpower Required

- 1 Reactor Operator

2.2.2 Special Equipment

None

2.2.3 Operator Actions

NOTE
If both divisions are isolated MSIV and SRV operation will be limited to accumulators. ☐

1. IF Division I pneumatic leakage suspected, THEN:

   a. Notify CRS. ☐ RO

   b. Close RNA-SV-5262 (Div I Non-Intrpt RNA). ☐ RO

   c. Close RNA-SV-5253 (Div I Bu N2 Supp To DW Isol Vlv) ☐ RO

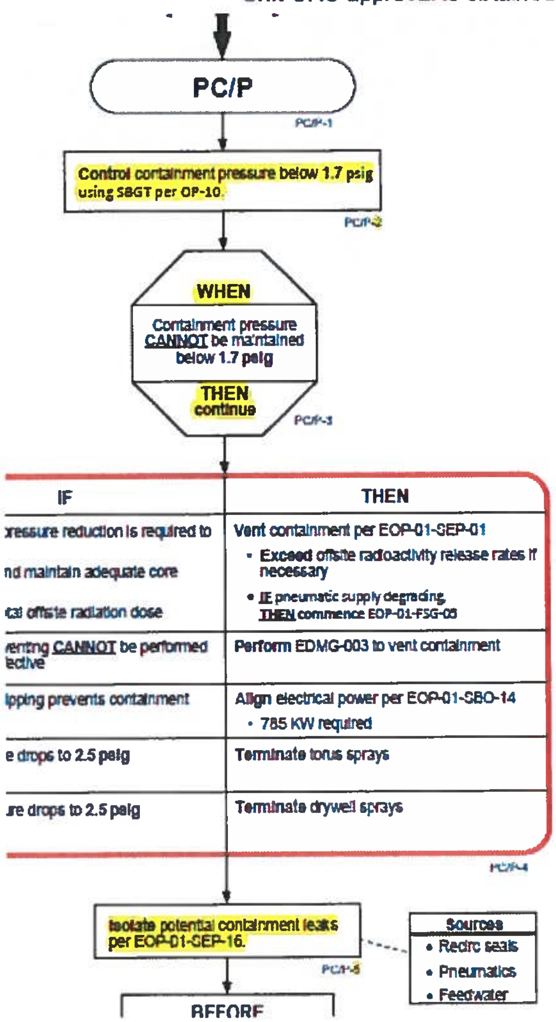
2. IF Division II pneumatic leakage suspected

6.3.2 Venting Containment Via SBGT

Date/Time Started_______________

1. Confirm the following Initial Conditions are met:

- Drywell pressure has risen to greater than 0.15 psig. ....________
- SBGT System is in STANDBY in accordance with Section 6.1.1. ....________
- One of the following:
  - ◊ Plant stack radiation monitor is in service and CAC-CS-5519 (CAC Purge Vent Isol Ovrd) is in OFF......________
  - ◊ E&C has sampled the drywell atmosphere and has determined that it is suitable for release. ....________
- Unit CRS approval is obtained prior to venting............................________



83. S271000 1

Unit Two is operating at rated power.

> UA-48 (5-4) *AOG System Bypass,* has been alarming for 1 minute due to High-High off gas flow

(REFERENCE PROVIDED)

Which one of the following completes both statements below?

AOG-XCV-142, Guard Bed Isolation Valve, ___(1)___ automatically close.

ODCM 7.3.10, *Gaseous Radwaste Treatment System*, Condition A entry ___(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

Answer: C

K/A:

271000 Offgas System

A2 Ability to (a) predict the impacts of the following on the OFFGAS SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5 / 45.6)

10 Offgas system high flow

RO/SRO Rating: 3.1/3.3

Tier 2 / Group 2

K/A match: The applicant is required to predict the status of the AOG system (XCV-142) based on high-high offgas flow, and the required procedural actions (i.e ODCM).

Pedigree: New

Objective: LOI-CLS-LP-030 Obj 9 Given plant conditions and Technical Specifications, including the Bases, TRM, ODCM and COLR, determine the required action(s) to be taken in accordance with Technical Specifications, the TRM or ODCM associated with the Condenser Air Removal/Augmented Offgas System.

Reference: ODCM 7.3.10

Cog Level: High

Explanation: Part 1: With XCV-142 remaining open the probable cause for UA-48 (5-4) is off gas flow high, all other conditions (besides circuit failure) for this alarm would cause a closure of the XCV-142. Part 2: with UA-48 (5-4) in alarm, HCV-102 is open. This would bypass the AOG portion of the Gaseous Radwaste Treatment System, leading to reduced hold up times and increased main stack rad levels. ODCM 7.3.10 requires AOG in operation so the comp measure is required.

Distractor Analysis:

Choice A: Part 1 is plausible because high-high cooler condenser condensate level would also cause UA-48 (5-4) to alarm, but it would also close the XCV-142. Part 2 is plausible because it is correct, see explanation.

Choice B: Part 1 is plausible because high-high cooler condenser condensate level would also cause UA-48 (5-4) to alarm, but it would also close the XCV-142. Part 2 is plausible because with the opening of the HCV-102, an additional flowpath is provided around the charcoal adsorbers and the normal flowpath remains in service.

Choice C: This is the Correct answer, see explanation.

Choice D: Part 1 is plausible because it is correct, see explanation. Part 2 is plausible because with the opening of the HCV-102, an additional flowpath is provided around the charcoal adsorbers and the normal flowpath remains in service.

SRO Basis: Conditions and limitations in the facility license. [10 CFR 55.43(b)(1). Requires the SRO applicant to have basis knowledge to determine whether ODCM compensatory actions are required based on plant status of the AOG system.

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AOG SYSTEM BYPASS

AUTO ACTIONS

1. *AOG SYSTEM BYPASS VALVE, AOG-HCV-102,* opens

CAUSES

1. High hydrogen - Train A
2. High hydrogen - Train B
3. High-high cooler condenser condensate level
4. High-high off-gas flow
5. Circuit failure

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COOLER CNDSR DRN LEVEL HI AOG SYS BYP

AUTO ACTIONS

1. *GUARD BED ISOLATION VALVE, AOG-XCV-142,* closes
2. *AOG SYSTEM BYPASS VALVE, AOG-HCV-102,* opens

DISCHARGE H2 CONC HIGH

AUTO ACTIONS

1. Isolation to AOG System. (Closes XCV-148, 147, 142, 143, and 141 after a 30 second time delay)
2. Open AOG-HCV-102.

CAUSES

B 7.3.10 GASEOUS RADWASTE TREATMENT SYSTEM

BASES

This requirement provides reasonable assurance that the releases of radioactive materials in gaseous effluents will be kept "as low as reasonably achievable." This specification implements the requirements of 10 CFR Part 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50, and the design objectives given in Section II.D of Appendix I to 10 CFR Part 50. The GASEOUS RADWASTE TREATMENT SYSTEM refers to the 30-minute offgas holdup line, stack filter house filtration, and the Augmented Off-Gas-Treatment System.

7.3.10 GASEOUS RADWASTE TREATMENT SYSTEM

ODCMS 7.3.10 The GASEOUS RADWASTE TREATMENT SYSTEM shall be in operation.

APPLICABILITY: Whenever the Main Condenser Air Ejector (evacuation) System is in operation.

COMPENSATORY MEASURES

CONDITION	REQUIRED COMPENSATORY MEASURE	COMPLETION TIME
A. GASEOUS RADWASTE TREATMENT SYSTEM not in operation.	A.1 Place GASEOUS RADWASTE TREATMENT SYSTEM in operation.	7 days
B. NOTE: Required Compensatory Measure B.1 shall be completed if this Condition is entered. Required Compensatory measure and associated Completion Time not met.	B.1 Submit a Special Report to the NRC that identifies the required inoperable equipment and the reasons for the inoperability, corrective actions taken to restore the required inoperable equipment to OPERABLE status, and a summary description of the corrective actions taken to prevent recurrence.	30 days

Condenser Air Inleakage

CONDENSER AIR INLEAKAGE: MODE 1				
DIAGNOSTIC PARAMETER	**SAMPLE FREQUENCY**	**DIAGNOSTIC LIMIT**	**REMARKS**	**REQUIRED ACTIONS IF LIMIT IS EXCEEDED**
Air In-leakage, scfm		< 50	Limit applies above 50% power. At 150 scfm the AOG System Bypass valve (AOG-HCV-102) will open. Compensatory measures are required by ODCM 7.3.10.	Develop and implement an in-leakage test plan.

84. S295001 1

Unit One is operating at 72% power with the following conditions:

Jet Pump Flow Loop A (B21-R611A)	25 Mlbs/hr
Jet Pump Flow Loop B (B21-R611B)	29 Mlbs/hr
Total Core Flow (U1CPWTCF)	54 Mlbs/hr

Which one of the following completes both statements below IAW Tech Spec 3.4.1, *Recirculation Loops Operating,* and Bases? (consider each statement separately)

The current Jet Pump Flow mismatch ____(1)____.

If Jet Pump Flows are **not** matched within limits, then the loop with the ____(2)____ must be considered not in operation.

A. (1) is within limits
   (2) lower flow

B. (1) is within limits
   (2) higher flow

C. (1) is not within limits
   (2) lower flow

D. (1) is not within limits
   (2) higher flow

Answer: A

K/A:

295001 Partial or Complete Loss of Forced Core Flow Circulation

AA2 Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION : (CFR: 41.10 / 43.5 / 45.13)

05 Jet pump operability

RO/SRO Rating: 3.1/3.4

Tier 1 / Group 1

K/A match: This question has the SRO candidate determine jet pump operability based on given core flow conditions.

Pedigree: Bank NRC 10-1

Objective: CLS-LP-002*34
Given plant conditions and Technical Specifications, including the Bases, TRM, ODCM, and COLR determine the required action(s) to be taken in accordance with Technical Specifications associated with the Reactor Recirculation System. (SRO/STA only)

Reference: None

Cog Level: High

Explanation: Two recirculation loops are normally required to be in operation with their flows matched within the limits specified in SR 3.4.1.1 to ensure that that during a LOCA caused by a break of the piping of one recirculation loop the assumptions of the LOCA analysis are satisfied. Jet pump loop flow mismatch should be maintained within the following limits:

- jet pump loop flows within 10% (maximum indicated difference 6.0 x10^6 lbs/hr) with total core flow less than 57.5 x10^6 lbs/hr
- jet pump loop flows within 5% (maximum indicated difference 3.0 x10^6 lbs/hr) with total core flow greater than or equal to 57.5 x10^6 lbs/hr

Distractor Analysis:

Choice A: Correct Answer, see explanation

Choice B: Plausible because flow mismatch is within limits for lower reactor power level and because the belief that the higher flow loop will experience excessive vibration could cause them to select the "higher flow" response.

Choice C: Plausible because flow mismatch is not within limits for a higher reactor power level. Part 2 is correct.

Choice D: Plausible because flow mismatch is not within limits for a higher reactor power level. Part 2 is plausible because the belief that the higher flow loop will experience excessive vibration could cause them to select the "higher flow" response.

SRO Basis: Facility operating limitations in the TS and their bases. [10 CFR 55.43(b)(2)]. The candidate is required to determine whether jet pump flow is within the limits and then use TS bases information to determine which loop is considered not in operation.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.1.1	NOTE Not required to be performed until 24 hours after both recirculation loops are in operation. Verify recirculation loop jet pump flow mismatch with both recirculation loops in operation: a. ≤ 10% of rated core flow when operating at < 75% of rated core flow; and b. ≤ 5% of rated core flow when operating at ≥ 75% of rated core flow.	24 hours

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.1.1

This SR ensures the recirculation loops are within the allowable limits for mismatch. At low core flow (i.e., < 75% of rated core flow), the MCPR requirements provide larger margins to the fuel cladding integrity Safety Limit such that the potential adverse effect of early boiling transition during a LOCA is reduced. A larger flow mismatch can, therefore, be allowed when core flow is < 75% of rated core flow. The recirculation loop jet pump flow, as used in this Surveillance, is the summation of the flows from all of the jet pumps associated with a single recirculation loop.

The mismatch is measured in terms of the percent of rated core flow. If the flow mismatch exceeds the specified limits, the loop with the lower flow is considered not in operation. The SR is not required when both loops are not in operation since the mismatch limits are meaningless during single loop or natural circulation operation. The Surveillance must be performed within 24 hours after both loops are in operation. The 24 hour Frequency is consistent with the Surveillance Frequency for jet pump OPERABILITY verification and has been shown by operating experience to be adequate to detect off normal jet pump loop flows in a timely manner.

REFERENCES

1. UFSAR, Section 5.4.1.3.
2. UFSAR, Chapter 15.
3. 10 CFR 50.36(c)(2)(ii).

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4.2 Supplementary Actions

1. **IF** the yellow Speed Hold light is lit,
THEN confirm the applicable VFD is maintaining stable recirculation pump speed. ☐

NOTE

Jet pump loop flows should be maintained within the following limits: ☐

- Jet pump loop flows within 10% (maximum indicated difference 6.0 x 10^6 lbs/hr) with total core flow less than 57.5 x 10^6 lbs/hr
- Jet pump loop flows within 5% (maximum indicated difference 3.0 x 10^6 lbs/hr) with total core flow greater than or equal to 57.5 x 10^6 lbs/hr

85. S295022 1

Unit One was at full power when all offsite power was lost.
The following is the Emergency Diesel Generator status:

DG1	Locked out on fault
DG2	Running and loaded
DG3	Running and loaded
DG4	Locked out on fault

Which one of the following completes the statements below?

The ___(1)___ CRD pump must be started to re-establish the CRD system.

0AOP-36.1, *Loss Of Any 4160V Buses or 480V E-Buses*, ___(2)___ contain the step for placing the CRD Flow Control, C11-FC-R600, in manual with manual potentiometer at minimum setting following the loss of the CRD pump?

A. (1) 1A
(2) does

B. (1) 1A
(2) does NOT

C. (1) 1B
(2) does

D. (1) 1B
(2) does NOT

Answer: D

K/A:
295022 LOSS OF CRD PUMPS
AA2 Ability to determine and/or interpret the following as they apply to LOSS OF CRD PUMPS:
(CFR: 41.10 / 43.5 / 45.13)
02 CRD system status

RO/SRO Rating: 3.3/3.4

K/A match: The first part can be answered by knowing the power supply to the pump (systems knowledge. The second part of the question (SRO Knowledge) is not systems knowledge, is not an immediate operator action, is not an entry condition for AOP/EOP, and is not purpose or mitigative strategy of the procedure. It assesses plant abnormal conditions and then selects a procedure to recover or with which to proceed.

Pedigree: Bank, last used on the 2014 NRC Exam

Objective: CLS-LP-302G, Obj. 4c.
Given plant conditions and any of the following AOP's, determine the required supplementary actions: AOP-36.1.

Reference: None

Cog Level: High

Explanation: With a loss of all offsite power the E-Buses will strip the loads (CRD Pumps), there are no auto starts for these pumps, so both CRD pumps will be off. DG1 is lost which means E1 is lost and A CRD pump will not be able to be started. The 1B does however have power, and can be restarted. The steps for restart are located in the OP, and AOP2.0, however they are not located in AOP36.1. DG-2 and DG3 provide power to the 1B and 2A CRD Pumps. The DG4 loss is a loss of the 2B CRD pump.

Distractor Analysis:

Choice A: Part 1 is plausible because the 1A CRD pump is tripped but it cannot be re-started due to E1 remaining de-energized (DG1 Locked ouot on fault). In addition, Unit 2 has power for the 2A CRD pump. Part 2 is plausible because an AOP (AOP-2.0) does contain the steps for re-energizing the pump, and AOP36.1 is an AOP we would be in due to the given electrical power loss, however AOP 36.1 does nto contain the steps for restarting the pump.

Choice B: Part 1 is plausible because the 1A CRD pump is tripped but it cannot be re-started due to E1 remaining de-energized (DG1 Locked ouot on fault). In addition, Unit 2 has power for the 2A CRD pump. Part 2 is correct, see explanation.

Choice C: Part 1 is correct, see explanation. Part 2 is plausible because an AOP (AOP-2.0) does contain the steps for re-energizing the pump, and AOP36.1 is an AOP we would be in due to the given electrical power loss, however AOP 36.1 does nto contain the steps for restarting the pump.

Choice D: The answer is correct, see explanation.

SRO Basis: Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations [10 CFR 55.43(b)(5)]

LOSS OF ANY 4160V BUSES OR 480V E-BUSES	0AOP-36.1
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ATTACHMENT 4
Page 1 of 6

4160V and 480V Emergency Bus Loads

Bus E1	**Load (kw)**	**Bus E2**	**Load (kw)**
RHR Pump 1C	750	RHR Pump 1D	750
RHRSW Pump 1C	600	RHRSW Pump 1D	600
CS Pump 1A	940	CS Pump 1B	940
CRD Pump 1A	190	CRD Pump 1B	190
NSW Pump 1A	225	NSW Pump 1B	225
CSW Pump 1B	225	CSW Pump 1C	225
RHR Pump 2C	750	RHR Pump 2D	750
RHRSW Pump 2C	600	RHRSW Pump 2D	600
CSW Pump 2C	225	Fire Pump (normal)	190

CONTROL ROD DRIVE HYDRAULIC SYSTEM OPERATING PROCEDURE	1OP-08
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6.3.20 Restarting CRD Hydraulic System Following Loss Of CRD Pump

1. **Ensure** the following Initial Conditions are met:
   - CRD System was in operation per Section 6.1.1. ...._______
   - The operating CRD pump has STOPPED......................................_______
2. **Close** B32-V22 (Seal Injection Vlv) for Recirc Pump A. ...._______
3. **Close** B32-V30 (Seal Injection Vlv) for Recirc Pump B. ...._______
4. **Place** C11-FC-R600 (CRD Flow Control) in MAN. ...._______
5. **Reduce** C11-FC-R600 (CRD Flow Control) potentiometer to minimum setting. ...._______
6. **Ensure** C11-PCV-F003 (Drive Press Vlv) is OPEN. ...._______
7. **Ensure** RBCCW is in operation to supply cooling water to CRD pumps. ...._______
8. **Start** non-operating (desired) CRD Pump A or B. ...._______

LOSS OF ANY 4160V BUSES OR 480V E-BUSES	0AOP-36.1
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4. **Start** the CRD System in accordance 1OP-08(2OP-08) Control Rod Drive Hydraulic System Operating Procedure, using the section for restarting the system following loss of the operating CRD pump.................☐

86. S295015 1

Unit One is performing the ATWS Procedure with the following conditions:

A-05 (2-6) *Reactor Vess Lo Level Trip,* is illuminated
A-06 (1-6) *Reactor Vess Lo Lo Water Level Sys A,* is NOT illuminated
A-06 (2-6) *Reactor Vess Lo Lo Water Level Sys B,* is NOT illuminated
MSIVs are closed
Reactor pressure peaked at 1141 psig and is now being controlled 800-1000 psig.
Torus water temperature is 105°F and rising
Reactor power is 25%

IAW 0OI-37.5, *ATWS Procedure Basis Document*, which one of the following identifies the action that will have the **highest** priority?

A. SLC initiation.

B. Inhibiting ADS.

C. Trip both Reactor Recirc Pumps.

D. Termination and prevention of RPV injection.

Answer: D

K/A:
295015 Incomplete SCRAM
G2.4.31 Knowledge of annunciator alarms, indications, or response procedures. (CFR: 41.10 / 45.3)

RO/SRO Rating: 4.2/4.1

Tier 1 / Group 2

K/A match: THIS QUESTION WAS PRE-SUBMITTED FOR APPROVAL.
This question requires knowledge of the annunciator response procedure status (where level is at) to determine the appropriate course of action while executing the EOP. Also for prioritization of the action to direct.

Pedigree: New

Objective: 300E-17e Given plant conditions and the Anticipated Transient Without Scram Procedure, determine the following: Priority of execution given to each leg of the procedure.

Reference: None

Cog Level: Hi

Explanation: With Reactor vessel lo lo water level not illuminated, RPV water level is >90". Reactor power is also >23% with the RR pumps tripped. IAW RC/Q-8 and RC/L-2, terminate and prevent is immediately required in order to prevent THI.

Distractor Analysis:

Choice A: Plausible because SLC initiation is required with >2% power and rising torus temperatures in order to not exceed the HCTL. However, it is not the first step required because power is >23%.

Choice B: Plausible because Inhibiting ADS is required, However, it is not the first step required because power is >23%.

Choice C: Plausible because tripping the recirc pumps would be done prior to terminate and prevent but with RPV pressure peaking at > 1138 ARI would have tripped the pumps, Level is not at LL2 yet according to the alarms.

Choice D: Correct Answer, see explanation.

SRO Basis: Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. [10 CFR 55.43(b)(5)]. This SRO applicant is required to direct the appropriate actions out of the ATWS EOP based on plant conditions. The action is contained in a note in the EOP and not a general strategy of the EOP.

DEVICE/SETPOINTS

NS333 Auxiliary Relay A71-K1A (actuated from B21-LTM-N024A-1-1)	De-energized
OR	
NS333 Auxiliary Relay A71-K1C (actuated from B21-LTM-N025A-1-1)	De-energized
Level Transmitter Master Trip Unit B21-LTM-N024A-1-1 and N025A-1-1	105 inches

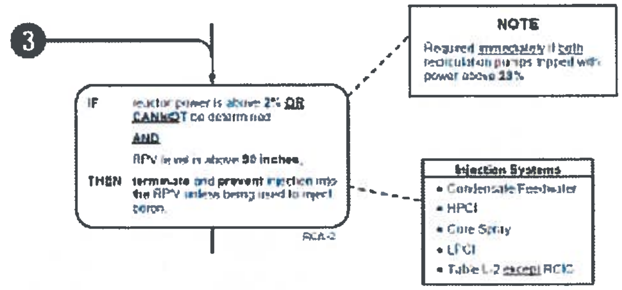
POSSIBLE PLANT EFFECTS

1. Inoperable equipment may result in a Tech Spec LCO.

REFERENCES

1. LL-93064 - 92
2. 0AOP-39.0, Loss of DC Power

5.4 **Step RC/L-2**

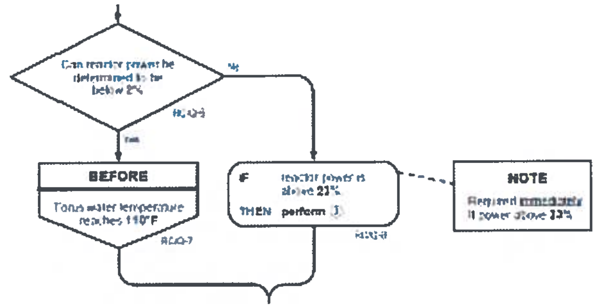


If reactor power is greater than 23% with both reactor recirculation pumps tripped and RPV level above 90 inches, RPV level needs to be promptly reduced below the feedwater nozzles, to avoid thermal hydraulic instabilities. This is accomplished by termination and prevention of injection systems, from identified systems, particularly feedwater, within 120 seconds.

To prevent or mitigate the consequences of any large irregular neutron flux oscillations induced by neutronic/thermal-hydraulic instabilities, RPV level is initially lowered sufficiently below the elevation of the feedwater sparger nozzles. This places the feedwater spargers in the steam space providing effective heating of the relatively cold feedwater and eliminating the potential for high core inlet subcooling. For conditions that are susceptible to oscillations, initiation and growth of oscillations is principally dependent upon subcooling at the core inlet; the greater the subcooling, the more likely oscillations will commence and increase in magnitude.

If reactor power is at or below the APRM downscale trip setpoint (2%), it is highly unlikely that the core bulk boiling boundary would be below that which provides suitable stability margin for operation at high powers and low flows. (A minimum boiling boundary of 4 ft above the bottom of active fuel has been shown to be effective as a stability control because a relatively long two-phase column is required to develop a coupled neutronic/ thermal-hydraulic instability.) Furthermore, flow/density variations would be limited with reactor power this low since the core has a relatively low average void content.

5.33 **Step RC/Q-6 through RC/Q-8**



If reactor power is below 2%, the operator is directed to inject boron before torus water temperature reaches 110°F. This allows sufficient time for Hot Shutdown Boron Weight (HSBW) of boron to be injected.

As long as the core remains submerged (the preferred method of core cooling), fuel integrity and RPV integrity are not directly challenged even under failure-to-scram conditions. A scram failure coupled with an MSIV isolation however, results in rapid heatup of the torus due to the steam discharged from the RPV via SRVs. The challenge to containment thus becomes the limiting factor which defines the requirement for boron injection.

If torus temperature and RPV pressure cannot be maintained below the Heat Capacity Temperature Limit (HCTL), rapid depressurization of the RPV will be required. To avoid depressurizing the RPV with the reactor at power, it is desirable to shut down the reactor prior to reaching HCTL, thus minimizing the quantity of heat rejected to the torus. The Boron Injection Initiation Temperature (BIIT) is defined so as to achieve this when practicable.

5.10 **Step RC/L-8 through RC/L-10**

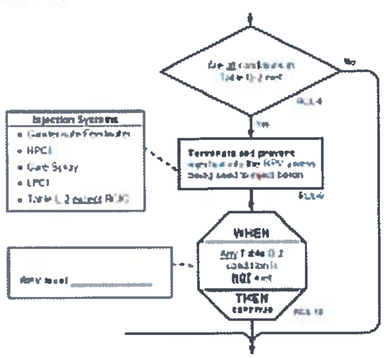


Table Q-2
Terminate and Prevent Determination

	Yes	No
Reactor power above 2% OR CANNOT be determined		
Torus water temperature above 110°F		
RPV level above TAF		
Either: • Any SRV OPEN OR being used to control pressure • Drywell pressure above [illegible] psig		

Based on reactor power being above 2%, Step RC/L-2 initially lowered RPV level, to the feedwater sparger, by terminating and preventing injection from identified systems. If all of the conditions in Table Q-2 are met, Step RC/L-9 will lower RPV level further to suppress reactor power. When any condition in Table Q-2 is no longer met the operator is directed to continue to subsequent steps which will establish a new RPV level band.

5.10 Step RC/L-8 through RC/L-10 (continued)

Terminating and preventing injection from:

- Condensate and Feedwater is addressed in 1OP-32 (2OP-32), Condensate And Feedwater System Operating Procedure, and covers terminating and preventing injection by either tripping both Reactor Feed Pumps (RFPs) or by idling one RFP.

- Core Spray is accomplished by tripping the associated Core Spray loop's operating pump.

- HPCI is addressed in 1OP-19 (2OP-19), High Pressure Coolant Injection System Operating Procedure, and covers terminating and preventing injection when HPCI is either operating or not operating.

- RHR is accomplished by tripping the associated RHR loop's operating pump(s).

The Boron Injection Initiation Temperature (BIIT) is a function of reactor power and is the torus temperature before which boron injection must be initiated if a reactor depressurization, due to exceeding the Heat Capacity Temperature Limit (HCTL), is to be precluded. This temperature is 110°F.

The combination of high reactor power (above the APRM downscale trip), high torus temperature (above BIIT), and an open SRV or high drywell pressure (above the scram setpoint), are symptomatic of heat being rejected to the torus at a rate in excess of that which can be removed by the torus cooling system. Unless mitigated, these conditions ultimately result in loss of NPSH for ECCS pumps taking suction on the torus, containment overpressurization, and (ultimately) loss of Primary Containment integrity, which in turn could lead to a loss of adequate core cooling and uncontrolled release of radioactivity to the environment.

The conditions listed in Table Q-2, combined with the inability to shut down the reactor through control rod insertion, dictate a requirement to promptly further reduce reactor power in order to preserve Primary Containment integrity since, as long as these conditions exist, torus heatup will continue.

Since RPV level is only allowed to drop to TAF before injection is restarted, if RPV level is already below TAF, then the objective of the step has been accomplished. Further lowering of RPV level is not necessary, and the steps which deliberately lower RPV level are bypassed.

87. S295023 1

Which one of the following completes both statements below?

IAW Tech Spec 3.9.6, *Reactor Pressure Vessel (RPV) Water Level*, the minimum water level over the top of irradiated fuel assemblies seated within the RPV during movement of irradiated fuel assemblies in the RPV is ____(1)____.

The Tech Spec bases for the minimum water level is to provide for ____(2)____ during a fuel handling accident.

A. (1) 19 feet 11 inches
   (2) iodine retention

B. (1) 19 feet 11 inches
   (2) shielding of radioactive decay particles

C. (1) 23 feet
   (2) iodine retention

D. (1) 23 feet
   (2) shielding of radioactive decay particles

Answer: C

K/A:

295023 Refueling Accidents

G2.2.25 Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits. (CFR: 41.5 / 41.7 / 43.2)

RO/SRO Rating: 3.2/4.2

Tier 1 / Group 1

K/A Match: THIS QUESTION WAS PRE-SUBMITTED FOR APPROVAL.
This meets the K/A because it is testing the TS bases for the RPV water level during movement of irradiated fuel assemblies in the RPV.

Pedigree: New

Objective: LOI-CLS-LP-200-B Obj 12. Identify conditions and limitations in the facility license. (SRO/STA only)

Reference: None

Cog Level: Fundamental

Explanation: Part 1: LCO 3.9.6 states, "RPV water level shall be = 23 ft above the top of irradiated fuel assemblies seated within the RPV." " Part 2: The minimum water level of 23 ft allows a decontamination factor of 200 to be used in the accident analysis for iodine. This relates to the assumption that 99.5% of the total iodine released from the pellet to cladding gap of all the damaged fuel assembly rods is retained by the water.

Distractor Analysis:

Choice A: Part 1 is plausible because IAW TS 3.7.7 the spent fuel storage pool water level shall be = 19 feet 11 inches over the top of irradiated fuel assemblies seated in the spent fuel storage pool racks. Part 2 is plausible because it is correct, see explanation.

Choice B: Part 1 is plausible because IAW TS 3.7.7 the spent fuel storage pool water level shall be = 19 feet 11 inches over the top of irradiated fuel assemblies seated in the spent fuel storage pool racks. Part 2 is plausible because the water does provide shielding from radioactive decay particles, but that is not the TS bases for the minimum water level.

Choice C: Correct Answer, see explanation

Choice D: Part 2 is plausible because it is correct, see explanation. Part 2 is plausible because the water does provide shielding from radioactive decay particles, but that is not the TS bases for the minimum water level.

SRO Basis: Facility operating limitations in the TS and their bases. [10 CFR 55.43(b)(2)]. Requires the SRO applicant to know the TS bases for RPV water level.

3.9 REFUELING OPERATIONS

3.9.6 Reactor Pressure Vessel (RPV) Water Level

LCO 3.9.6 RPV water level shall be ≥ 23 ft above the top of irradiated fuel assemblies seated within the RPV.

APPLICABILITY: During movement of irradiated fuel assemblies within the RPV,

During movement of new fuel assemblies or handling of control rods within the RPV, when irradiated fuel assemblies are seated within the RPV.

3.7 PLANT SYSTEMS

3.7.7 Spent Fuel Storage Pool Water Level

LCO 3.7.7 The spent fuel storage pool water level shall be ≥ 19 feet 11 inches over the top of irradiated fuel assemblies seated in the spent fuel storage pool racks.

APPLICABILITY: During movement of irradiated fuel assemblies in the spent fuel storage pool.

B 3.9.6 Reactor Pressure Vessel (RPV) Water Level

BASES

BACKGROUND

The movement of fuel assemblies or handling of control rods within the RPV requires a minimum water level of 23 ft above the top of irradiated fuel assemblies seated within the RPV. During refueling, this maintains a sufficient water level in the reactor vessel. Sufficient water is necessary to retain iodine fission product activity in the water in the event of a fuel handling accident (Refs. 1 and 2). Sufficient iodine activity would be retained to limit offsite doses from the accident to well below the 10 CFR 50.67 exposure guidelines (Ref. 3).

APPLICABLE SAFETY ANALYSES

During movement of fuel assemblies or handling of control rods, the water level in the RPV is an initial condition design parameter in the analysis of a fuel handling accident in containment postulated by Regulatory Guide 1.183 (Ref. 1). A minimum water level of 23 ft allows a decontamination factor of 200 to be used in the accident analysis for iodine (Ref. 1). This relates to the assumption that 99.5% of the total iodine released from the pellet to cladding gap of all the damaged fuel assembly rods is retained by the water.

Analysis of the fuel handling accident inside containment is described in Reference 2. With a minimum water level of 23 ft and a minimum decay time of 24 hours prior to fuel handling, the analysis and test programs demonstrate that the iodine release due to a postulated fuel handling accident is adequately captured by the water and that offsite doses are maintained well below the allowable limits of Reference 3.

RPV water level satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii) (Ref. 4).

(continued)

88. S295037 1

Unit Two is in an ATWS executing RXFP, with the following plant conditions:

Injection to the RPV has been terminated and prevented
The **Minimum Number of SRVs Required for Emergency Depressurization** are open.

Table P-3
Minimum Steam Cooling Pressure

Open SRVs	Pressure (psig)
7 or more	120
6	145
5	175
4	220
3	300
2	455
1	915

IAW RXFP, which one of the following completes the statement below?

The CRS should direct injection to the RPV when EITHER:

____(1)____ SRV remains open
OR
when reactor pressure lowers below the Minimum Steam Cooling Pressure of ____(2)____
.

A. (1) NO
(2) 175 psig

B. (1) NO
(2) 455 psig

C. (1) ONLY one
(2) 175 psig

D. (1) ONLY one
(2) 455 psig

Answer: A
K/A:
295037 SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown
G2.4.21 Knowledge of the parameters and logic used to assess the status of safety functions, such as reactivity control, core cooling and heat removal, reactor coolant system integrity, containment conditions, radioactivity release control, etc. (CFR: 41.7 / 43.5 / 45.12)

RO/SRO Rating: 4.0/4.6

Tier 1 / Group 1

K/A Match: This meets the K/A because it is testing the required steam cooling pressure for adequate core cooling.

Pedigree: New

Objective: LOI-CLS-LP-300-F Obj.3 Given the Reactor Flooding Procedure, which steps have been completed and plant parameter values, determine the required operator actions.

Reference: None

Cog Level: High

Explanation: Part 1: IAW RXFP-9/10, injection is reestablished when either no SRVs are open or Part 2: Reactor is below the MSCP, which in this case for 5 SRVs (MNSRED) is 175 psig.

Distractor Analysis:

Choice A: Correct answer, see explanation.

Choice B: Part 1 is plausible because it is correct, see explanation. Part 2 is plausible because this is the minimum steam cooling pressure for having the Minimum Number of SRVs Required for Decay Heat Removal open.

Choice C: Part 1 is plausible because once injection is re-established, RXFP-10 directs injection to continue until at least one SRV is open. In addition, with no SRVs open or below the MSCP adequate core cooling could possibly not exist, so a candidate might assume we would never wait for that condition to reestablish injection. Part 2 is plausible, because it is correct, see explanation.

Choice D: Part 1 is plausible because once injection is re-established, RXFP-10 directs injection to continue until at least one SRV is open. In addition, with no SRVs open or below the MSCP adequate core cooling could possibly not exist, so a candidate might assume we would never wait for that condition to reestablish injection. Part 2 is plausible because this is the minimum steam cooling pressure for having the Minimum Number of SRVs Required for Decay Heat Removal open.

SRO Basis: Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. [10 CFR 55.43(b)(5)]. This requires the SRO to have knowledge of more than the overall sequence of events in the EOPs or mitigating strategy. Since it requires the SRO to know the MNSRED and how to implement Table-P3.

3.0 DEFINITIONS (continued)

37. **Maximum Subcritical Banked Withdrawal Position:** The lowest control rod position to which all controls rods may be withdrawn in bank and the reactor will nonetheless remain shutdown under all conditions. This position is utilized to assure the reactor will remain shutdown irrespective of reactor water temperature.

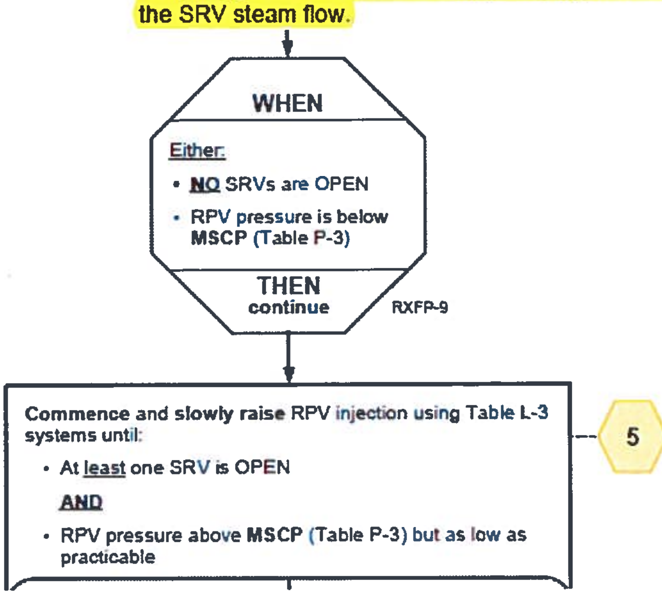
38. **Minimum Core Steam Flow:** The lowest core steam flow which is sufficient to preclude any clad temperature from exceeding 1500°F even if the reactor core is not completely covered.

39. **Minimum Debris Retention Injection Rate:** The lowest RPV injection rate at which it is expected that core debris will be retained in the RPV when RPV level cannot be determined to be above the bottom of active fuel. (Attachment 17)

40. **Minimum Indicated Level:** The highest RPV level instrument indication which results from off-calibration instrument run temperature conditions when RPV level is actually at the elevation of the instrument variable leg tap.

41. **Minimum Number of SRVS Required for Decay Heat Removal:** The least number of SRVs (2) which will remove all decay heat from the core at a pressure sufficiently low that the ECCS with the lowest head will be capable of making up the SRV steam flow.

42. **Minimum Number of SRVS Required for Emergency Depressurization:** The number of SRVs (5) which correspond to a minimum steam cooling pressure sufficiently low that the ECCS with the lowest head will be capable of making up the SRV steam flow.



89. S295026 1

An event on Unit One has resulted in the following plant conditions:

Reactor pressure:	1000 psig
Reactor Water Level	120 inches
Control Rod position	Unknown
APRMs	Downscale
Drywell pressure:	3 psig
Torus pressure:	2 psig
Torus water temp:	152°F
Torus water level:	-36 inches

(REFERENCE PROVIDED)

Which one of the following identifies the required actions for reactor pressure control?

A. Exit the RC/P flowpath of ATWS, and go to 0EOP-01-EDP, *Emergency Depressurization.*

B. Exit the RC/P flowpath of RVCP, and go to 0EOP-01-EDP, *Emergency Depressurization.*

C. Remain in the RC/P flowpath of ATWS, and exceed 100°F/hr cooldown rate if necessary.

D. Remain in the RC/P flowpath of RVCP, and exceed 100°F/hr cooldown rate if necessary.

Answer: C

K/A:

295026 Suppression Pool High Water Temperature

G2.1.23Ability to perform specific system and integrated plant procedures during all modes of plant operation. (CFR: 41.10 / 43.5 / 45.2 / 45.6)

RO/SRO Rating: 4.3/4.4

Tier 1 / Group 1

K/A match: The applicant is required to determine which procedure to implement based on HCTL

Pedigree: New

Objective: CLS-LP-300L, Obj. 5a Given the Primary Containment Control Procedure, determine the appropriate operator actions if any of the following limits are approached or exceeded: Heat Capacity Temperature Limit

Reference: 0EOP-01-UG, Attachment 7, *Heat Capacity Temperature Limit*

Cog Level: Hi

Explanation: With rods at an unknown position, an ATWS has occurred. Since HCTL is close to the unsafe region (but not violating it), exceeding the cooldown rate in the RC/P flowpath of ATWS is warranted.

Distractor Analysis:

Choice A: Plausible because a novice applicant may misinterpret the graph and believe HCTL is in the unsafe region, and an ED is warranted.

Choice B: Plausible because a novice applicant may believe with APRMs downscale an ATWS has not occurred, and may misinterpret the graph believing an ED is warranted.

Choice C: Correct answer, see explanation.

Choice D: Plausible because a novice applicant may believe with APRMs downscale an ATWS has not occurred. Exceeding the cooldown rate is correct because of the operating point on the HCTL graph.

SRO Basis: Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations (43(b)(5) SRO is required to select the appropriate EOP action based on HCTL.

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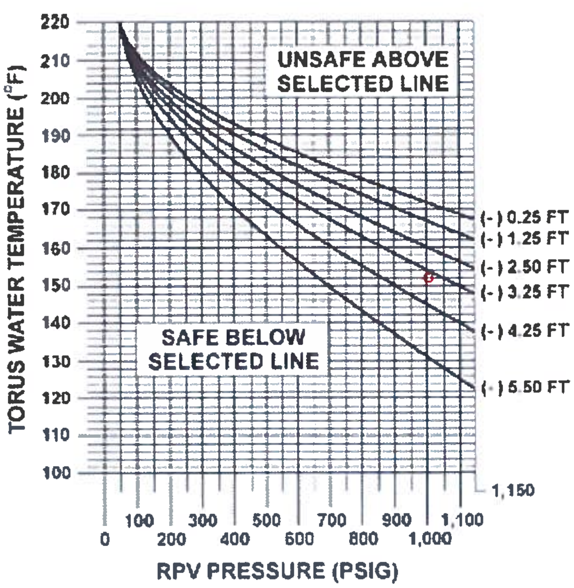
Heat Capacity Temperature Limit



START

ATWS-1

While in this procedure:

IF	THEN
RPV level CANNOT be determined	Exit RC/L AND RC/P flowpaths and go to EOP-01-RXFP
Emergency depressurization is OR has been required	• Proceed to 6 • Exit RC/P flowpath and go to EOP-01-EDP
Reactor is S/D Without Boron under all conditions (Table Q-1)	1. Terminate boron injection NOT required by other EOPs 2. Exit this flowchart and go to EOP-01-RVCP

ATWS-2

ATWS-2

RC/P

RC/P-1

IF	THEN
RPV pressure approaches 440 psig	Control injection from: 1. Condensate 2. Core Spray 3. LPCI
HCTL CANNOT be maintained in safe region, but only if RPV depressurization will NOT result in loss of injection required for adequate core cooling	**Maintain RPV pressure below HCTL** • **Exceed 100°F/hr cooldown rate if necessary**
MSIVs are CLOSED **AND** Boron injection is required **AND** Main condenser is available as a heat sink **AND** **NO** indication of a main steam line break exists	**Equalize pressure and open** MSIVs (OP-25): • Defeat low RPV level Group 1 isolation per EOP-01-SEP-10
A continuous pneumatic supply is **NOT** available to SRVs	• **IF** stabilizing pressure, **THEN** place control switch to AUTO/CLOSE • **IF** depressurizing RPV, **THEN** minimize cycles

RC/P-2

90. S295035 1

Unit Two is operating at rated power.
PCCP has been entered due to high torus water temperature with the following plant conditions:

UA-12 (3-3) *Rx Bldg Diff Press High/Low,* is in alarm.

UA-05 (6-10) *Rx Bldg Isolated,* is in alarm.

Reactor Building Pressure (indication on the left)

Which one of the following completes both statements below?

Reactor Building pressure is ___(1)___.

The CRS will direct Reactor Building HVAC restarted IAW ___(2)___.

A. (1) positive
(2) 2OP-37.1, *Reactor Building Heating and Ventilation System Operating Procedure*

B. (1) positive
(2) 0EOP-01-SEP-04, *Reactor Building HVAC Restart Procedure*

C. (1) negative
(2) 2OP-37.1, *Reactor Building Heating and Ventilation System Operating Procedure*

D. (1) negative
(2) 0EOP-01-SEP-04, *Reactor Building HVAC Restart Procedure*

Answer: C
K/A:
295035 Secondary Containment High Differential Pressure
EA2 Ability to determine and/or interpret the following as they apply to SECONDARY CONTAINMENT HIGH DIFFERENTIAL PRESSURE: (CFR: 41.8 to 41.10)
01 Secondary containment pressure

RO/SRO Rating: 3.8/3.9

Tier 1 / Group 2

K/A match: The applicant is required to interpret secondary containment pressure and select the appropriate procedure based on this interpretation.

Pedigree: New

Objective: LOI-CLS-LP-300-M Obj 11 Given plant conditions involving Reactor Building HVAC system isolation and the Secondary Containment Control Procedure, determine if the Reactor Building HVAC system should be restarted.

Reference: None

Cog Level: Hi

Explanation: Part 1: With indications at upscale > +0.5 inches of h20, reactor building pressure is negative.
Part 2: IAW with UA-5 (6-10) and UA-12 (3-3) RBHVAC is restarted 2OP37.1.

Distractor Analysis:

Choice A: Part 1 is plausible because the reading is +.05 inches of h20 and upscale high, a novice candidate could mistake this for a positive pressure indication. Part 2: is plausible because it is correct, see explanation.

Choice B: Part 1 is plausible because the reading is +0.5 inches of h20 and upscale high, a novice candidate could mistake this for a positive pressure indication. Part 2: is plausible because RBHVAC is restarted using SEP-04, when in SCCP when LL2 and high drywell pressure needs to be defeated. In addition the title is rbhvac restart procedure, and it is a SEP.

Choice C: Correct Answer, see explanation.

Choice D: Part 1 is plausible because it is correct, see explanation. Part 2: is plausible because RBHVAC is restarted using SEP-04, when in SCCP when LL2 and high drywell pressure needs to be defeated. In addition the title is rbhvac restart procedure, and it is a SEP.

SRO Basis: Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. [10 CFR 55.43(b)(5)] The SRO applicant is required to interpret secondary containment pressure and select the appropriate procedure based on this interpretation.

RX BLDG DIFF PRESS HIGH/LOW
(Reactor Building Differential Pressure High/Low)

AUTO ACTIONS

1. Reactor Building supply and exhaust fans trip.

CAUSE

1. High or low differential pressure between the Reactor Building and atmospheric pressure.
2. Circuit malfunction.

OBSERVATIONS

1. Reactor Building Static Pressure Indicator, 2-VA-PI-1297, on RTGB Panel XU-3.

ACTIONS

1. If secondary containment integrity is required and differential pressure is low, enter 0EOP-03-SCCP, Secondary Containment Control, and execute concurrently with this procedure.
2. Inform E&RC Chemistry Reactor Building Ventilation is not in service.
3. Verify that the valve lineup is correct per 2OP-37.1, Reactor Building Heating and Ventilation System.
4. Start up the system per Section 5.1 of 2OP-37.1.
5. If a circuit malfunction is suspected, ensure that a WR/WO is submitted.

ACTIONS

6. If area radiation levels on Table 3 of 0EOP-03-SCCP exceed maximum normal operating values, enter 0EOP-03-SCCP, Secondary Containment Control.
7. If the Reactor Building HVAC has isolated and it is desired to restart ventilation, enter 2OP-37.1 Reactor Building Heating and Ventilation System.
8. Notify E&RC Counting Room that reactor building ventilation has been secured.

While in this procedure:

IF	THEN
Either: • RB ventilation exhaust radiation exceeds 4 mR/hr • RB ventilation temperature exceeds 135°F (UA-03, 6-2)	• Ensure RB HVAC isolated • Ensure SBGT operating
RB HVAC isolates **AND** all conditions exist: • RB ventilation exhaust radiation below 4 mR/hr • RB ventilation exhaust radiation monitor has remained on scale • RB ventilation temperature has **NOT** exceeded 135°F (UA-03, 6-2)	Restart RB HVAC per: • OP-37.1 • EOP-01-SEP-04 if necessary to defeat LL-2 or high drywell pressure isolations

SCCP-2

91. S295038 1
A release on Unit Two is occurring with the following plant conditions:

Main Stack Rad Monitor, D12-RM-23S, is reading 2.3E+08 µCl/sec

Turbine Building Vent Rad Monitor, D12-RM-23, is reading 2.5E+07 µCl/sec

Real-time dose assessment using actual meteorology indicates 0.92 Rem TEDE and 5.1 Rem thyroid CDE at the site boundary

(REFERENCE PROVIDED)

Which one of the following completes both statements below?

IAW RRCP, **Unit One** ___(1)___ override and reset the main stack hi-hi isolation signal.

The **highest** EAL classification for this event is ___(2)___.

A. (1) can
(2) Site Area Emergency

B. (1) can
(2) General Emergency

C. (1) can NOT
(2) Site Area Emergency

D. (1) can NOT
(2) General Emergency

Answer: B
K/A:
295038 High Off-Site Release Rate
EA2 Ability to determine and/or interpret the following as they apply to HIGH OFF-SITE RELEASE RATE : (CFR: 41.10 / 43.5 / 45.13)
01 Off-site

RO/SRO Rating: 3.3/4.3

Tier 1 / Group 1

K/A match: The candidate is required to compare the given radiation release values (including site boundary), and compare those to the EALs for rad effluent. Based on this comparison the candidate must make the correct EAL designation.

Pedigree: NEW

Objective: CLS-LP-301-B Obj 9: Given a hypothetical abnormal event and plant operating mode, use 0PEP-02.1 to properly classify or re-classify the event

Reference: 0PEP-02.1

Cog Level: Hi

Explanation: Part 1: IAW RRCP if the release is not from Unit One they are allowed to override and then reset the isolation signal. Part 2: Due to Site Area Boundary dose >5000 mrem thyroid CDE, a GE is the highest classification.

Distractor Analysis:

Choice A: Part1 is plausible because it is correct, see explanation. Part 2 is plausible because the main stack and turbine building rad monitors are reading > the SAE setpoint. However, the dose assessments results are above a GE classification.

Choice B: Correct Answer, see explanation.

Choice C: Part1 is plausible because it is a common stack for both units. Part 2 is plausible because the main stack and turbine building rad monitors are reading > the SAE setpoint. However, the dose assessments results are above a GE classification.

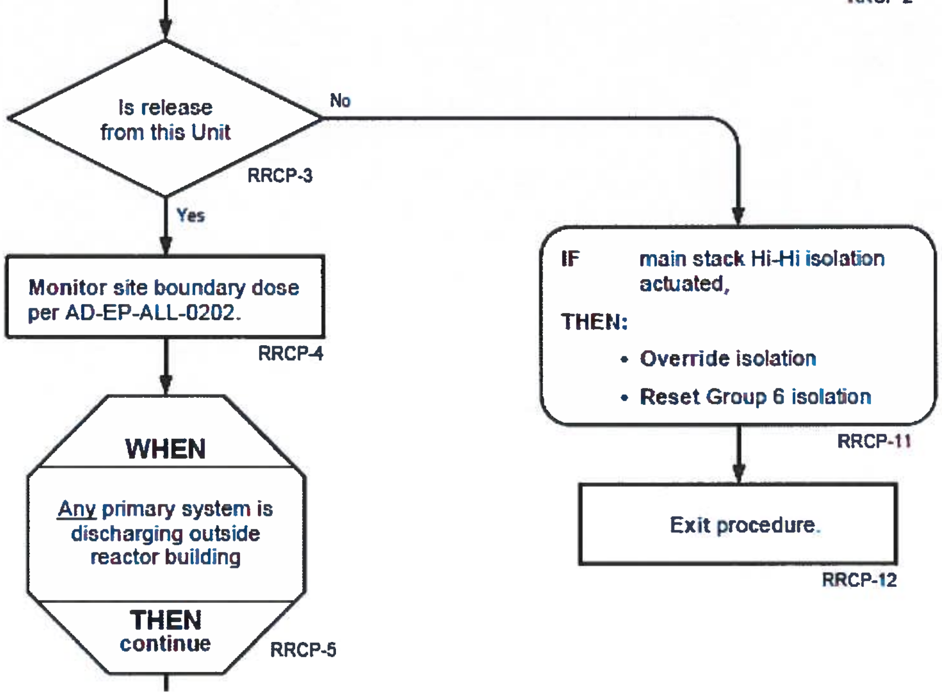
Choice D: Part1 is plausible because it is a common stack for both units. Part 2 is plausible because it is correct, see explanation

SRO Basis: Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. [10 CFR 55.43(b)(5)] The SRO candidate is required to compare the given radiation release values (including site boundary), and compare those to the EALs for rad effluent. Based on this comparison the candidate must make the correct EAL designation.

While in this procedure:

IF	THEN
Either: • TB ventilation is SHUTDOWN • TB ventilation is in Once Through Mode	Perform per OP-37.3 • Place TB HVAC in Recirculation Mode • Ensure both TB Air Filter Exhaust Fans operating
Fuel failure indicated by: • Main Steam Line Rad Hi (UA-23, 2-6) • Process Off-Gas Rad Hi (UA-03, 5-2) • Process OG Vent Pipe Rad Hi (UA-03, 6-4)	Ensure Control Building Emergency Recirculation operating (OP-37)
Reactor building breached	Request ERO evaluate use of mitigating sprays per EDMG-002

RRCP-2



5.2 Step RRCP-2

While in this procedure:	
IF	**THEN**
Either: • TB ventilation is SHUTDOWN • TB ventilation is in Once Through Mode	**Perform** per OP-37.3 • **Place** TB HVAC in Recirculation Mode • **Ensure** both TB Air Filter Exhaust Fans operating
Fuel failure indicated by: • Main Steam Line Rad Hi (UA-23, 2-6) • Process Off-Gas Rad Hi (UA-03, 5-2) • Process OG Vent Pipe Rad Hi (UA-03, 6-4)	**Ensure** Control Building Emergency Recirculation operating (OP-37)
Reactor building breached	**Request** ERO evaluate use of mitigating sprays per EDMG-002

RRCP-2

Step RRCP-2 is a procedure override which applies the entire time RRCP is being executed. Each of the three components specify applicable conditions and direct performance of actions as discussed below.

5.2.1 Step RRCP-2 First Override

Continued personnel access to the turbine building may be essential for responding to emergencies or transients which may degrade into emergencies. The turbine building is not an air tight structure, and radioactivity release inside the turbine building would not only limit personnel access but would eventually lead to an unmonitored ground level release, or release via the turbine building ventilation if operating in the once-through lineup.

Operation of the turbine building ventilation in the recirculation lineup helps to improve turbine building accessibility. In addition, since both units share a common turbine building airspace, if the building is intact, removing turbine building ventilation from once through lineup will terminate a large unfiltered volume discharge flow path for a leak on either unit. Due to normal operational requirements when in once through lineup, at least one Air Filter Exhaust Fan and WRGM will be in service providing a monitored and filtered discharge flowpath.

Table R-1 Effluent Monitor Classification Thresholds

	Release Point	Monitor	GE	SAE	Alert	UE
Gaseous	Main Stack Rad	D12-RM-23S	2.13E+09 µCi/sec	2.13E+08 µCi/sec	2.13E+07 µCi/sec	1.60E+06 µCi/sec
	Reactor Bldg Vent Noble Gas	CAC-AQH-1264-3	—	—	—	6.14E+04 cpm
	Turbine Building Vent Rad	D12-RM-23	1.07E+08 µCi/sec	1.07E+07 µCi/sec	1.07E+06 µCi/sec	1.13E+04 µCi/sec
Liquid	Service Water Effluent Rad	D12-RM-K605	—	—	—	2 X hi alarm
	Radwaste Effluent Rad	D12-RM-K604	—	—	—	2 X hi-hi alarm

GENERAL EMERGENCY	SITE AREA EMERGENCY
RG1 Release of gaseous radioactivity resulting in offsite dose greater than 1,000 mrem TEDE or 5,000 mrem thyroid CDE 1 | 2 | 3 | 4 | 5 | DEF	RS1 Release of gaseous radioactivity resulting in offsite dose greater than 100 mrem TEDE or 500 mrem thyroid CDE 1 | 2 | 3 | 4 | 5 | DEF
RG1.1 In the absence of real-time dose assessment, reading on any Table R-1 effluent radiation monitor > column "GE" for ≥ 15 min. (Notes 1, 2, 3, 4) **RG1.2** Dose assessment using actual meteorology indicates doses > 1000 mrem TEDE or 5000 mrem thyroid CDE at or beyond the SITE BOUNDARY (Note 4) **RG1.3** Field survey results indicate **EITHER** of the following at or beyond the SITE BOUNDARY: - Closed window dose rates > 1000 mR/hr expected to continue for ≥ 60 min. - Analyses of field survey samples indicate thyroid CDE > 5000 mrem for 60 min. of inhalation. (Notes 1, 2)	**RS1.1** In the absence of real-time dose assessment, reading on any Table R-1 effluent radiation monitor > column "SAE" for ≥ 15 min. (Notes 1, 2, 3, 4) **RS1.2** Dose assessment using actual meteorology indicates doses > 100 mrem TEDE or 500 mrem thyroid CDE at or beyond the SITE BOUNDARY (Note 4) **RS1.3** Field survey results indicate **EITHER** of the following at or beyond the SITE BOUNDARY: - Closed window dose rates > 100 mR/hr expected to continue for ≥ 60 min. - Analyses of field survey samples indicate thyroid CDE > 500 mrem for 60 min. of inhalation. (Notes 1, 2)

92. S600000 1

Unit One and Unit Two are executing 0ASSD-01, *Alternative Safe Shutdown Procedure Index*, due to a fire in Main Control Room back panels requiring Main Control Room evacuation. Current plant conditions are:

Unit One and Two have scrammed
All MSIVs are shut

Which one of the following completes both statements below?

The CRS will enter 0ASSD-02, *Control Building,* and ___(1)___ 0ASSD-01.

The CRS will direct actions to achieve a safe shutdown using ___(2)___.

A. (1) exit
(2) HPCI

B. (1) exit
(2) RCIC

C. (1) concurrently perform
(2) HPCI

D. (1) concurrently perform
(2) RCIC

Answer: B

K/A:
600000 Plant Fire On Site
AA2 Ability to determine and interpret the following as they apply to PLANT FIRE ON SITE: (CFR: 41.10 / 43.5 / 45.13)
07 Whether malfunction is due to common-mode electrical failures

RO/SRO Rating: 2.6/3.0

Tier 1 / Group 1

K/A match: THIS QUESTION WAS PRE-SUBMITTED FOR APPROVAL.
The applicant is required to determine that based on a fire in the MCR requiring evacuation, common mode failures of electrical equipment could occur and interpret the procedure steps that require exiting ASSD-01 and entering the standalone ASSD-02. In addition, the applicant will determine which train of ASSD equipment (HPCI/RCIC) will remain unaffected by the potential common mode electrical failure.

Pedigree: Modified 14 NRC

Objective: LOI-CLS-LP-301 Obj 20 Given a fire in an ASSD area, describe the potential impact that the fire may have on Safe Shutdown Equipment

Reference: None

Cog Level: Fundamental

Explanation: Part 1: 0ASSD-02, Control Building, is an outside Control Room shutdown procedure for both units. This procedure is a stand-alone post fire shutdown procedure for a Control Room evacuation and requires the reactors to be in hot shutdown/manually scrammed prior to leaving the Control Room. There are parts of the control building that this procedure is not used for, i.e. battery rooms. Part 2: The safe shutdown strategy for control room evacuation requires the B train of ASSD equipment (RCIC).

Distractor Analysis:

Choice A: Part 1 is plausible because it is correct, see explanation. Part 2 is plausible because the A train of Safe shutdown equipment (HPCI) is used in other ASSDs.

Choice B: Correct Answer, see explanation

Choice C: Part 1 is plausible because for every other ASSD fire 0ASSD-01 is performed concurrently with specific ASSD sub procedures. Part 2 is plausible because the A train (HPCI) of Safe shutdown equipment is used in other ASSDs.

Choice D: Part 1 is plausible because for every other ASSD fire 0ASSD-01 is performed concurrently with specific ASSD sub procedures. Part 2 is plausible because it is correct, see explanation.

SRO Basis: Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. [10 CFR 55.43(b)(5)]. requires the SRO applicant to decide the appropriate transition to an event specific subprocedure based on fire in the control building.

2014 Question:

A fire in the control building fire area requires entry into 0ASSD-01, Alternative Safe Shutdown Procedure Index. The CRS has determined that alternate safe shutdown actions are required. Both Unit One and Unit Two have been manually scrammed.

Which one of the following completes the statements below IAW 0ASSD-01?

The next action that is required is to __(1)__.

Following this action both units will __(2)__.

A. (1) place MSIV control switches in close
(2) perform 0ASSD-01, Alternative Safe Shutdown Procedure Index concurrently with 0ASSD-02, Control Building.

B. (1) trip both Reactor Recirc pumps
(2) perform 0ASSD-01, Alternative Safe Shutdown Procedure Index concurrently with 0ASSD-02, Control Building.

C✓ (1) place MSIV control switches in close
(2) exit 0ASSD-01, Alternative Safe Shutdown Procedure Index and enter 0ASSD-02, Control Building

D. (1) trip both Reactor Recirc pumps
(2) exit 0ASSD-01, Alternative Safe Shutdown Procedure Index and enter 0ASSD-02, Control Building

3.5.2 **IF** the fire is in the Control Building fire area, **AND** control room evacuation is required, **THEN PERFORM** the following:

a. **MANUALLY SCRAM** Unit 1 reactor. ☐

b. **PLACE** Unit 1 MSIV control switches in *CLOSE*. ☐

c. **MANUALLY SCRAM** Unit 2 reactor. ☐

d. **PLACE** Unit 2 MSIV control switches in *CLOSE*. ☐

e. Both units **EXIT** this procedure **AND ENTER** 0ASSD-02, Control Building. ☐

3.5.3 **IF** the fire is **NOT** in the Control Building, **THEN ENTER** the applicable ASSD procedure **AND EXECUTE** concurrently with this procedure. ☐

NOTE:	A loss of drywell cooling can be determined using CAC-TR-4426-1A, CAC-TR-4426-2A, in the Control Room or CAC-TR-778, points 1, 3, and 4, at the Remote Shutdown Panel. The time the loss of drywell cooling occurred can be determined from the recorder display information.

3.5.4 **IF** RCIC or HPCI is injecting **AND** drywell cooling has been lost, **THEN START** reactor vessel cooldown at 100°F/hr or greater within 60 minutes of the loss of drywell cooling. ☐

3.5.5 **IF** drywell temperature control is lost, **THEN PERFORM** the following to preserve containment overpressure for RHR pump net positive suction head:

1. **STOP** A, B, C, and D RBCCW pumps for the affected unit. ☐
2. **CLOSE** the following valves for the affected unit:
   - *DW EQUIP DRAIN INBD ISOL VLV, G16-F019* ☐
   - *DW EQUIP DRAIN OTBD ISOL VLV, G16-F020* ☐
   - *DW FLOOR DRAIN INBD ISOL VLV, G16-F003* ☐
   - *DW FLOOR DRAIN OTBD ISOL VLV, G16-F004* ☐

2.1 This procedure is entered from Alternative Safe Shutdown Procedure Index, 0ASSD-01,

AND

2.2 Unit CRS has determined both reactors are to be brought to safe and stable conditions from outside the Control Room using ASSD Train B.

3.0 OPERATOR ACTIONS

3.1 **CONTINUE** implementation of this procedure by performing the steps in Section A. ☐

3.2 **IF** the fire is extinguished while executing this procedure **AND** the Unit CRS determines no action within this procedure is required, **THEN EXIT** this procedure. ☐

4.0 RESTORATION

4.1 **RETURN** plant to general operating condition as directed by plant management. ☐

93. S295021 1

Unit One is in MODE 4, when a loss of SDC occurs due to RCS leakage.

CU1	UNPLANNED loss of RPV inventory for 15 minutes or longer
	4 | 5

CU1.1

UNPLANNED loss of reactor coolant results in RPV water level less than a required lower limit for ≥ 15 min. (Note 1)

Which one of the following completes both of the statements below?

The **minimum** required RPV water level to support natural circulation is ___(1)___.

IAW 0PEP-02.2.1, *Emergency Action Level Technical Bases*, the Unusual Event required lower limit is defined as RPV water level less than___(2)___.

A. (1) 200 inches
   (2) 105 inches

B. (1) 200 inches
   (2) 166 inches

C. (1) 254 inches
   (2) 105 inches

D. (1) 254 inches
   (2) 166 inches

Answer: B

K/A:

295021 Loss of Shutdown Cooling

AA2 Ability to determine and/or interpret the following as they apply to LOSS OF SHUTDOWN COOLING : (CFR: 41.10 / 43.5 / 45.13)

03 Reactor water level

RO/SRO Rating: 3.5/3.5

Tier 1 / Group 1

K/A Match: This meets the K/A because it is testing the ability of the applicant to determine the water level designated for a UE with a loss of SDC.

Pedigree: Modified from 10-1

Objective: LOI-CLS-LP-301-B Obj. 6 Define the relationship between fission product barrier loss/potential loss and each of the four emergency classifications.

Reference: None

Cog Level: Fundamental

Explanation: Part 1: IAW 0AOP-15.0, the minimum required level for natural circulation is 200 inches. Part 2: IAW 0PEP-02.2.1, the RPV water level lower limit is the low end of any established band.

Distractor Analysis:

Choice A: Part1 is plausible because this is correct, see explanation. Part 2 is plausible because this would be the lower limit if no band was established for SDC in MODE 4.

Choice B: Correct Answer, see explanation

Choice C: Part 1 is plausible because 254 inches is the level of the MSLs and could be confused with Natural Circulation level due to the requirement to be at this level during alternate SDC.

Choice D: Correct Answer, see explanation.

SRO Basis: Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. [10 CFR 55.43(b)(5)]. The applicant is required to have bases knowledge of the EAL procedure to select the appropriate method of implementing the UE criteria for a lowering RPV level in MODE 4. EAL determination is an SRO only task.

Question from NRC 10.1 exam:

While in Mode 4 a loss of Shutdown Cooling (SDC) occurs.

Which one of the following completes both statements?

The minimum required Reactor Water Level to support Natural Circulation is __(1)__ inches.

An Alert declaration is first required after an unplanned RPV pressure increase greater than __(2)__ psig due to a loss of RCS cooling.

A. (1) 200
(2) 135

B✓ (1) 200
(2) 10

C. (1) 254
(2) 135

D. (1) 254
(2) 10

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4.2 Supplementary Actions (continued)

2. IF forced circulation has been lost,
AND natural circulation has NOT been established,
THEN ensure reactor vessel water level is being maintained between 200 inches and 220 inches as read on B21-LI-R605A(B) (RPV Water Level),
OR as directed by the Unit CRS based on plant conditions until forced circulation is restored... ☐

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RPV Level

Initiating Condition: UNPLANNED loss of RPV inventory for 15 minutes or longer

EAL:

CU1.1	**Unusual Event**
UNPLANNED loss of reactor coolant results in RPV water level less than a required lower limit for ≥ 15 min. (Note 1)	

Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

4 - Cold Shutdown, 5 - Refueling

Definition(s):

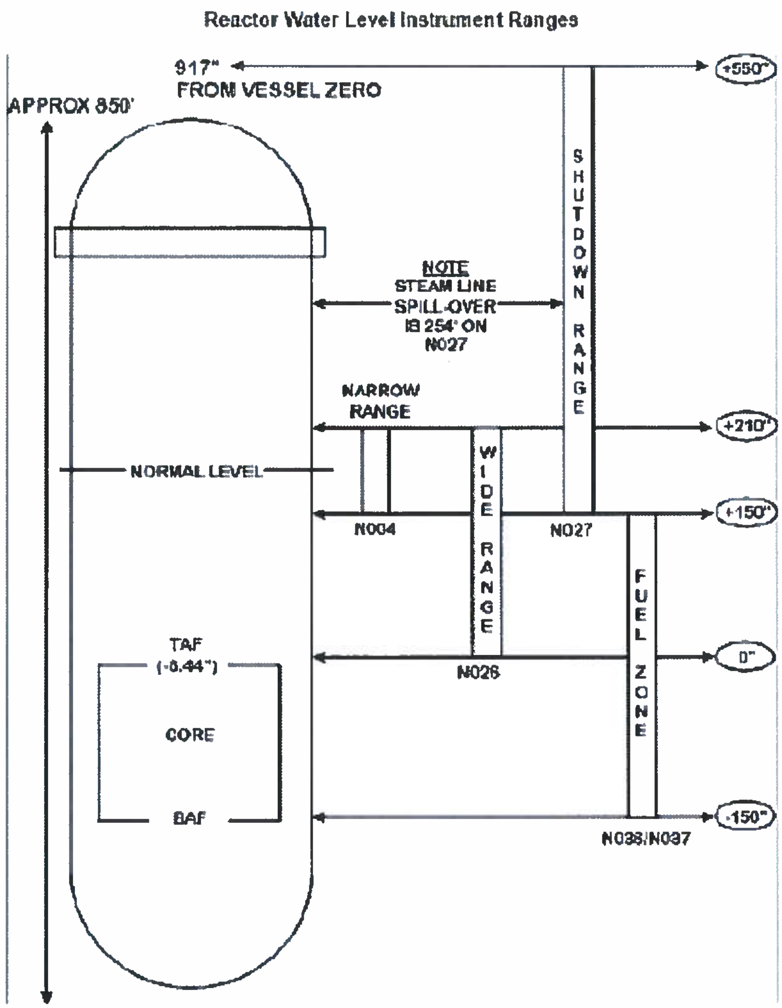
UNPLANNED-. A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

Basis:

Figure C-1 illustrates the elevations of the RPV level instrument ranges (ref. 2).

With the plant in Cold Shutdown, RPV water level is normally maintained above the RPV low level scram setpoint of 166 in. above TAF (ref. 1, 3). However, if RPV level is being controlled below the RPV low level scram setpoint, or if level is being maintained in a designated band in the reactor vessel it is the inability to maintain level above the low end of the designated control band due to a loss of inventory resulting from a leak in the RCS that is the concern.

Figure C-1 RPV Levels (ref. 2)



94. SG2.1.05 1

Which one of the following completes both statements below?
(Consider each statement separately.)

IAW Tech Spec 5.2.2, *Facility Staff*, the shift crew composition may be less than the minimum requirement for a period of time not to exceed ___(1)___ for an unexpected absence of on-duty shift crew members.

IAW 0OI-01.01, *BNP Conduct of Operations Supplement*, the minimum required number of Auxiliary Operators for manning a shift at BNP is ___(2)___.

A. (1) one hour
   (2) three

B. (1) one hour
   (2) nine

C. (1) two hours
   (2) three

D. (1) two hours
   (2) nine

Answer: D

K/A:

G2.1.05 Ability to use procedures related to shift staffing, such as minimum crew complement, overtime limitations, etc. (CFR: 41.10 / 43.5 / 45.12)

RO/SRO Rating: 2.9/3.9

Tier 3

K/A match: knowledge of required tech spec and conduct of ops shift manning requirements.

Pedigree: 2012 BNP NRC

Objective: LOI-CLS-LP-200-B Obj.12.-Identify conditions and limitations in the facility license.

Reference: None

Cog Level: Low

Explanation: IAW the procedure, 9 AO makeup the minimum shift staffing and two hours is the time to find a replacement. One hour is the time on "stepping out" limitation of the control room personnel. The tech Specs 5.2 only address the number of AOs for the Units which is 3, this does not take into account ASSD and Fire Brigade.

Distractor Analysis:

Choice A: Plausible because TS 5.2.2 requires 3 AOs for both Units which does not take into account ASSD and Fire Brigade requirements. One hour is the "stepping out" time limit for control room personnel.

Choice B: Plausible because nine is correct but one hour is the "stepping out" time limit for control room personnel.

Choice C: Plausible because TS 5.2.2 requires 3 AOs for both Units which does not take into account ASSD and Fire Brigade requirements.

Choice D: Correct Answer, see explanation

SRO Basis: Conditions and limitations in the facility license. (10 CFR 55.43(b)(1)). Requires the SRO applicant to know the limitations for shift staffing in the license.

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5.5 Operations Shift Staffing

5.5.1 General

1. In addition to the requirements of AD-OP-ALL-1000, the following requirements apply:

   a. The following table outlines the administrative guideline for the normal Operations shift complement. Any deviation from the normal shift complement must remain in accordance with Section 5.2.2 of Technical Specifications, applicable sections of 0ASSD-00, User's Guide, 0FPP-031, Fire Brigade Staffing Roster and Equipment Requirements, and 0ERP, Radiological Emergency Response Plan (ERP). (Attachment 13, Operations Staffing Roster contains a listing of required ERO Watch Stations and qualifications for each and ASSD positions.)

BNP Watchstations	BNP Shift Complement	License
Shift Manager (SM)	1 Shift Manager	SRO
Control Room Supervisor (CRS)	2 CRSs (1 for each unit)	SRO
Reactor Operator (RO)	4 Reactor Operators (typically, 2 for each unit)	RO/SRO
Auxiliary Operator (AO)	9 (includes 2 in Radwaste)	NA
Operations Center SRO	1 Operations Center SRO	SRO
STA [Note 1]	1 STA	STA Qualified

Notes:

5.2 Organization

5.2.2 Facility Staff (continued)

b. At least one licensed Reactor Operator (RO) shall be present in the control room when fuel is in the reactor. In addition, when either unit is in MODE 1, 2, or 3, at least one licensed Senior Reactor Operator (SRO) shall be present in the control room. With one unit in MODE 1, 2, or 3 and the other unit defueled, the minimum shift crew shall include a total of two SROs and two ROs.

c. Shift crew composition may be less than the minimum requirement of 10 CFR 50.54(m)(2)(i) and Specifications 5.2.2.a and 5.2.2.g for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements.

d. An individual qualified in radiation protection procedures shall be on site when fuel is in the reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.

e. Deleted.

95. SG2.1.43 1
Following the bypass of Unit Two feedwater heaters 4A and 5A, the following plant conditions exist:

Reactor Power is 60%
Feedwater Temperature is 330°F

Final Feedwater Temperature vs Power

RX PWR	Nominal FW Temp	Nominal FW Temp Reduced 10°F	110.3°F Reduced FFWT
65%	394.4	384.4	296.4
64%	393.1	383.1	295.5
63%	391.7	381.7	294.6
62%	390.4	380.4	293.7
61%	389.0	379.0	292.8
60%	387.6	377.6	291.9
[illegible]	[illegible]	[illegible]	[illegible]

(REFERENCE PROVIDED)

IAW 0OI-01.01, *BNP Conduct of Operations Supplement,* which one of the following completes both statements below? (consider each statement separately)

The CRS ____(1)____ required to implement the thermal limit penalties for FHOOS (feedwater heater out of service).

Entry into **Tech Spec 3.0.3** ____(2)____ required if final feedwater temperature is less than the 110.3°F reduced final feedwater temperature value.

A. (1) is
(2) is

B. (1) is
(2) is NOT

C. (1) is NOT
(2) is

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

Answer: B
K/A:
G2.1.43 Ability to use procedures to determine the effects on reactivity of plant changes, such as reactor coolant system temperature, secondary plant, fuel depletion, etc. (CFR: 41.10 / 43.6 / 45.6)

RO/SRO Rating: 4.1/4.3

Tier 3

K/A match: The applicant is required to use the final feedwater temperature reduction attachment to determine if the effect of the feedwater reduction is severe enough on reactivity to require implementation of thermal limit penalties.

Pedigree: New

Objective: CLS-LP-032 obj 27 Given plant conditions and Technical Specifications, including the Bases, TRM, ODCM, and COLR, determine whether given plant conditions meet minimum Technical Specifications requirements associated with the Condensate and Feedwater System.

Reference: T.S. 3.2.1, T.S. 3.2.2, and T.S. 3.2.3

Cog Level: High

Explanation: : Part1: A final fw temp of 385°F is less than the nominal FW temp for 60% power, but >10°F reduced from nominal. therefore the thermal limit penalties for FHOOS do not need to be implemented. Part 2:There are **NO** core operating limits specified in the COLR for operation beyond 110.3°F Final Feedwater Temperature. Thermal limits **CANNOT** be verified to be within the limits specified in the COLR, which requires entry into the Actions of LCO 3.2.1, 3.2.2, and 3.2.3. These LCOs require thermal limits to be restored within 4 hours, LCO 3.0.3 is not entered,

Distractor Analysis:

Choice A: Part 1 is correct, see explanation. Part2 is plausible because a candidate may believe since there are no thermal limits specified in the COLR for this condition, LCO 3.0.3 would be applicable.

Choice B: Correct Answer, see explanation

Choice C: Part 1 is plausible because a final fw temp of 330°F is less than the nominal FW temp reduced by 10°F for 60% power, but greater than the 110.3°F reduced FFWT, a novice applicant may believe thermal limit penalties are only applied at the 110.3°F value. Part2 is plausible because a candidate may believe since there are no thermal limits specified in the COLR for this condition, LCO 3.0.3 would be applicable.

Choice D: Part 1 is plausible because a final fw temp of 330°F is less than the nominal FW temp reduced by 10°F for 60% power, but greater than the 110.3°F reduced FFWT, a novice applicant may believe thermal limit penalties are only applied at the 110.3°F value. Part 2 is correct, see explanation.

SRO Basis: Facility operating limitations in the TS and their bases. [10 CFR 55.43(b)(2)] This question requires that the applicant determines whether the TS thermal limits should incur a penalty. In addition, it also requires that the candidate determines whether LCO 3.0.3 applies for a given condition.

LCO 3.0.3 When an LCO is not met and the associated ACTIONS are not met, an associated ACTION is not provided, or if directed by the associated ACTIONS, the unit shall be placed in a MODE or other specified condition in which the LCO is not applicable. Action shall be initiated within 1 hour to place the unit, as applicable, in:

a. MODE 2 within 7 hours;

b. MODE 3 within 13 hours; and

c. MODE 4 within 37 hours.

Exceptions to this Specification are stated in the individual Specifications.

Where corrective measures are completed that permit operation in accordance with the LCO or ACTIONS, completion of the actions required by LCO 3.0.3 is not required.

LCO 3.0.3 is only applicable in MODES 1, 2, and 3.

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ATTACHMENT 19
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<< Equipment Out Of Service Contingencies >>

EOOS Condition	Power	Required Action (Note 1, 2)
SLO	Any	• Reduce reactor power to ≤ 50% • Implement 0GP-14. • Implement applicable SLO power to flow map. • IF ≥ 23% RTP, THEN implement thermal limit penalty. • Reference TS 3.4.1.
TBVOOS	≥ 23% RTP	• Implement thermal limit penalty. • Reference TS 3.7.6.
FHOOS (FWTR) (FFTR)	≥ 23% RTP	• IF < the value in the "110.3°F Reduced FW Temp" column of 1(2)OP-32, Att 6, THEN enter LCO 3.2.1, 3.2.2 and 3.2.3.
	≥ 23% RTP	• IF > 10°F below nominal FW temperature, THEN: ◊ Implement applicable FWTR power to flow map. ◊ Implement thermal limit penalty.
		• Implement applicable FWTR power to flow map

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Final Feedwater Temperature vs Power

RX PWR	Nominal FW Temp	Nominal FW Temp Reduced 10°F	110.3°F Reduced FFWT
65%	394.4	384.4	296.4
64%	393.1	383.1	295.5
63%	391.7	381.7	294.6
62%	390.4	380.4	293.7
61%	389.0	379.0	292.8
60%	387.6	377.6	291.9
59%	386.2	376.2	290.9

96. SG2.2.15 1
Unit One is operating at rated power.

A-03 (2-2) *Auto Depress Control Pwr Failure*, is in alarm due to Fuse F5 being blown.

(REFERENCE PROVIDED)

Which one of the following completes both statements below?

Fuse F5 (D1 on 1-FP-05887) is located on ADS Logic ___(1)___.

ADS ___(2)___ operable.

A. (1) A
(2) is

B. (1) A
(2) is NOT

C. (1) B
(2) is

D. (1) B
(2) is NOT

Answer: C
K/A:
G2.2.15 Ability to determine the expected plant configuration using design and configuration control documentation, such as drawings, line-ups, tag-outs, etc. (CFR: 41.10 / 43.3 / 45.13)

RO/SRO Rating: 3.9/4.3

Tier 3

K/A match: THIS QUESTION WAS PRE-SUBMITTED FOR APPROVAL.
This question requires the candidate to use a drawing to determine operability of ADS.

Pedigree: New

Objective: LOI-CLS-LP-020 OBJ 15d. Given plant conditions, predict how ADS/SRVs will be affected by the following: Loss of DC power

Reference: 1-FP-05887 (Block out references to which logic string is logic A and B)

Cog Level: High

Explanation: Part 1: Fuse F5 is located on the alternate power source, only logic B has an alternate power source. Therefore, the fuse is on logic B, Part2: Since the drawing is shown in the de-energized state, fuse 5 being blown will have no impact on ADS instrumentation. ADS remains on its normal power source.

Distractor Analysis:

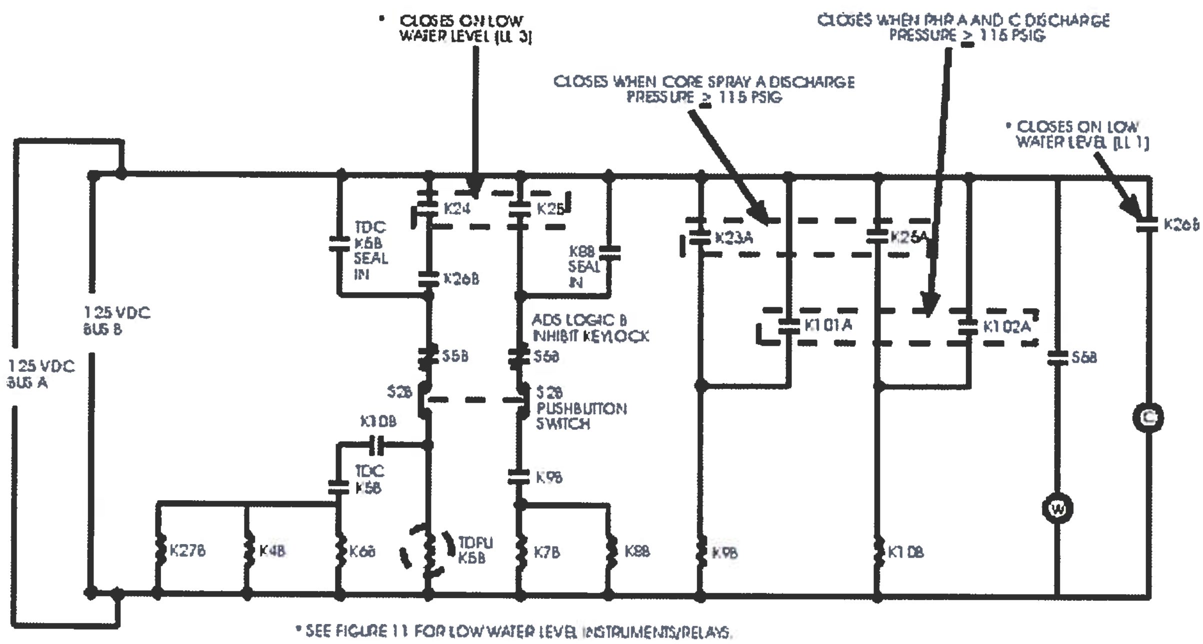
Choice A: Part 1 is plausible the fuse is located on 125V DC 3A power or the operator might forget which train of logic has two power supplies. Part 2 is plausible because it is correct, see explanation.

Choice B: Part 1 is plausible the fuse is located on 125V DC 3A power or the operator might forget which train of logic has two power supplies. Part 2 is plausible because if the drawing was shown in the energized state, this would be correct.

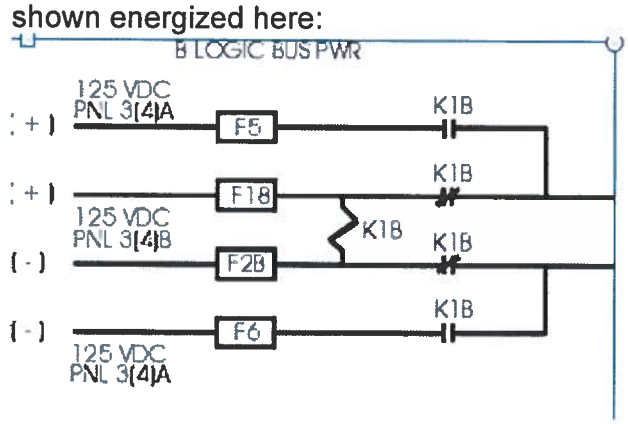
Choice C: Correct Answer, see explanation.

Choice D: Part 1 is plausible because it is correct, see explanation. Part 2 is plausible because if the drawing was shown in the energized state, this would be correct.

SRO Basis: Facility operating limitations in the TS and their bases. [10 CFR 55.43(b)(2)] Requires the SRO candidate to use a drawing to determine the status of ADS power and to know whether that loss effects ADS operability.



shown energized here:



97. SG2.2.22 1

Unit Two is operating at rated power.

While performing 0PT-07.2.4A, *Core Spray Loop A Operability*, Core Spray Room Cooler A fails to start when Core Spray Pump A is started.

The reactor building AO reports that the room cooler tripped on thermal overload.

IAW AD-OP-ALL-1000, *Conduct of Operations*, which one of the following completes both statements below? (consider each statement separately)

Core Spray Loop A is ____(1)____.

A one time reset of the thermal overload ____(2)____ allowed before a Maintenance and Engineering evaluation.

A. (1) OPERABLE
   (2) is

B. (1) OPERABLE
   (2) is NOT

C. (1) INOPERABLE
   (2) is

D. (1) INOPERABLE
   (2) is NOT

Answer: D

K/A:

G2.2.22 Knowledge of limiting conditions for operations and safety limits. (CFR: 41.5 / 43.2 / 45.2)

RO/SRO Rating: 4.0/4.7

Tier 3

K/A match: Requires knowledge of conduct of ops procedure to determine whether Core Spray A meets the conditions for operability in the tech specs based on cooler operation.

Pedigree: Bank NRC 08

Objective: CLS-LP-18, Obj. 18. Given plant conditions and TS, including bases, TRM, ODCM, and COLR, determine the required actions to be taken in accordance the TS associated with the Core Spray System.

Reference: None

Cog Level: Hi

Explanation: Part 1: Per 0OI-01.01, When any ECCS Room Cooler is determined to be INOPERABLE, then the ECCS equipment associated with that room cooler is to be declared INOPERABLE per the applicable Technical Specifications. Part 2: Per AD-OP-ALL-1000, the breaker should only be reset once the condition is identified and corrected, **and** plant conditions dictate the reset before maint and eng personnel are available.

Distractor Analysis:

Choice A: Part 1 is plausible, because the room cooler is not part of the Core Spray system listed in the tech spec bases. Part 2 is plausible because during transient conditions the breaker could be reset, however, the plant is in a stable condition.

Choice B: Part 1 is plausible, because the room cooler is not part of the Core Spray system listed in the tech spec bases. Part 2 is correct, see explanation.

Choice C: Part 1 is correct, see explanation. Part 2 is plausible because during transient conditions the breaker could be reset, however, the plant is in a stable condition.

Choice D: Correct Answer, see explanation.

SRO Basis: Facility operating limitations in the TS and their bases. [10 CFR 55.43(b)(2)] Requires the SRO candidate to have knowledge of conduct of ops procedure to determine whether Core Spray A meets the conditions for operability in the tech specs based on cooler operation.

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5.19 Resetting Protective Devices

{7.1.4}

5.19.1 Standards

1. Protective devices should not be reset without a clear understanding of the reason for the protective device trip.

2. The overriding priority for the operating crew upon the trip of any protective device is to stabilize the plant and restore the systems to the safest possible condition.

5.19.2 Expectations

1. Protection devices which have actuated (breakers, fuses, bistables, MOV thermal overloads, lockouts, etc.) should only be restored with shift supervision approval, under the following conditions. The following conditions do not apply to 120 volt breakers that only supply lighting or receptacles.

   a. The cause of the actuation has been identified and corrected.

   b. Restoring the protective device is not recommended unless plant conditions dictate that the component repositioning must be completed before Maintenance and Engineering personnel are available. Remote operation of the component with no personnel in the immediate area after resetting the protective device is recommended if repositioning is required prior to completion of the evaluation by Maintenance and Engineering.

2. The SM may approve additional protective device resetting after consultation with Engineering.

5.16.2 **Degraded Equipment Controls— System/Component Related Guidance (continued)**

(1) Reference TRM Appendix F, Safety Function Determination Program (SFDP), Attachments 1 and 2 to assist with determination of Technical Specification 3.8.1 and 3.8.7 requirements and to assess the possible impact on supported systems.

(2) If an evaluation of the SFDP is performed, then document the evaluation and the results in the the narrative log or on Attachment 26, if the narrative log is not available.

4. ECCS Room Coolers {7.1.3}

NOTE

- The following step is not required to be performed if the ECCS Room Cooler is INOPERABLE due to the loss of a 4160V or 480V E-Bus. E-Bus INOPERABILITY impacts the OPERABILITY of ECCS subsystems. Technical Specifications and the SFDP will provide Required Actions to be taken for the loss of the E-Bus.
- In Mode 4 and Mode 5, ECCS Room Coolers are not required to be OPERABLE to support OPERABILITY of the associated ECCS Systems.

a. When any ECCS Room Cooler is determined to be INOPERABLE, then the ECCS equipment associated with that room cooler is to be declared INOPERABLE per the applicable Technical Specifications.

EXAMPLE

The RHR Room Coolers are to be considered redundant components required to support the operation of RHR. Therefore, should a room cooler be found or made INOPERABLE, a 7 day Active LCO is required to be established on the RHR system. Likewise, should both room coolers be found INOPERABLE, the action required is the same as if both RHR loops and HPCI were INOPERABLE. Should it be identified that one RHR Room Cooler is INOPERABLE and one RHR Loop is also INOPERABLE (specific combinations do not matter), the action is as if only one RHR Loop is INOPERABLE (7 days).

5 Control Building HVAC Air Compressors

98. SG2.3.11 1

Following a small steam line break in the drywell plant conditions are as follows:

Drywell pressure:	25 psig and rising
Drywell hydrogen:	1.3%
Suppression Chamber hydrogen:	1.2%
Torus level:	42 inches

Which one of the following completes both statements below?

The CRS is required to direct venting containment IAW 0EOP-01-SEP-01, *Primary Containment Venting*, using ____(1)____.

Venting of the ____(2)____ will be directed **first**.

A. (1) Section 2.1, *Containment Pressure Control*
   (2) drywell

B. (1) Section 2.1, *Containment Pressure Control*
   (2) torus

C. (1) Section 2.2, *Containment Hydrogen Control*
   (2) drywell

D. (1) Section 2.2, *Containment Hydrogen Control*
   (2) torus

Answer: D

K/A:

G2.3.11 Ability to control radiation releases. (CFR: 41.11 / 43.4 / 45.10)

RO/SRO Rating: 3.8/4.3

Tier 3

K/A match: Requires the ability to determine the procedure section for venting, and the correct sequence of termination of venting.

Pedigree: New

Objective: CLS-LP-300-L*08d
Given the Primary Containment Control Procedure and plant conditions, determine if the following actions are required: Venting the primary containment IRRESPECTIVE of radioactivity release rate limits

Reference: None

Cog Level: High

Explanation: Part 1: Following the H2 leg of the PCCP with the given conditions will drive you to step PC/G-9 which directs you to "Vent Containment per EOP-01-SEP-01, since H2 is the driving condition for venting, then section 2.2 is the appropriate section to implement. Part 2: IAW SEP-01, the torus is vented first as long as the torus water level is less than 6 feet.

Distractor Analysis:

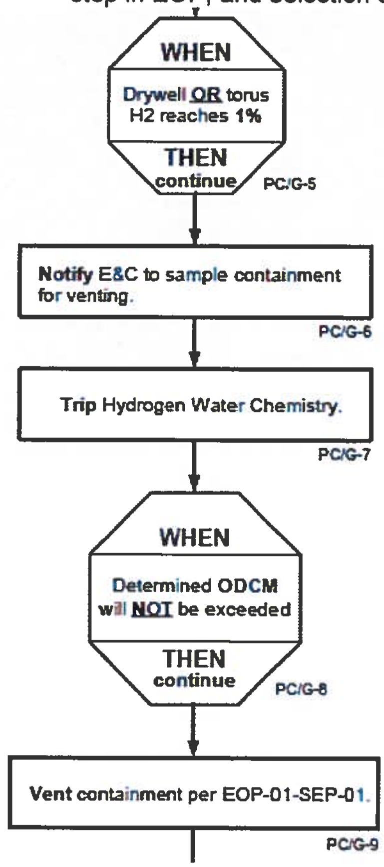
Choice A: Part 1 is plausible because H2 concentration is less than the entry limit into PCCP (entry at 1.5%), and Containment pressure is >11.5 psig (pressure for DW sprays), therefore a novice applicant might believe that the appropriate procedure section required is for containment pressure control. Part 2 is plausible since venting of the drywell is performed first if torus water level is >6 feet.

Choice B: Part 1 is plausible because H2 concentration is less than the entry limit into PCCP, and Containment pressure is >11.5 psig (pressure for DW sprays), therefore a novice applicant might believe that the appropriate procedure section required is for containment pressure control. Part 2 is correct.

Choice C: Part 1 is plausible since it is correct, see explanation. Part 2 is plausible since venting of the drywell is performed first if torus water level is >6 feet.

Choice D: Correct Answer, see explanation..

SRO Basis: Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. [10 CFR 55.43(b)(5)] Requires knowledge of diagnostic step in EOP, and selection of appropriate emergency contingency procedure.



2.2.3 Containment Hydrogen Control Actions (continued)

e. Close VA-D-BFV-RB (SBGT A Isol Damper).. ☐ RO

f. Close VA-H-BFV-RB (SBGT B Isol Damper).. ☐ RO

g. IF venting the torus,
THEN open:

(1) CAC-V7 (Torus Purge Exh Vlv). .... ☐ RO

(2) CAC-V8 (Torus Purge Exh Vlv). .... ☐ RO

h. IF venting the drywell,
THEN open:

(1) CAC-V9 (Drywell Purge Exh Vlv). .... ☐ RO

(2) CAC-V10 (Drywell Purge Exh Vlv).. ☐ RO

i. Open VA-F-BFV-RB (SBGT DW Suct Damper).. ☐ RO

13. IF directed to terminate torus venting,
THEN:

a. Ensure primary containment purging terminated per EOP-01-SEP-05.. ☐ RO

99. SG2.4.30 1
Unit Two is operating at rated power with LPCI A inoperable and the following sequence of events occurs:

0000	7 day completion time for LCO 3.5.1, *ECCS Operating*, Condition A expires and Condition C is entered requiring that the Unit be placed in MODE 3 in 12 hours.
0030	Plant shutdown is commenced per LCO 3.5.1, Condition C.
0050	LPCI A is repaired and declared operable; LCO 3.5.1 Conditions A and C are exited.
0100	Management decides to continue the plant shutdown as planned to complete other maintenance items.
0230	Unit Two in MODE 3

(REFERENCE PROVIDED)

Which one of the following completes both statements below?

IAW OI-01.07, *Notifications*, an Emergency Notification System (ENS) report to the NRC must be submitted no later than ___(1)___.

IAW 10 CFR 50.73, *Licensee Event Reporting System*, an LER ___(2)___ required.

A. (1) 0400
 (2) is

B. (1) 0400
 (2) is NOT

C. (1) 0430
 (2) is

D. (1) 0430
 (2) is NOT

Answer: D
K/A:
G2.4.30 Knowledge of events related to system operation/status that must be reported to internal organizations or external agencies, such as the State, the NRC, or the transmission system operator. (CFR: 41.10 / 43.5 / 45.11)

RO/SRO Rating: 2.7/4.1

Tier 3

K/A match: Applicant required to determine report status for the given condition.

Pedigree: Bank

Objective: LOI-CLS-LP-201-D, Obj 11
Explain the following regarding NRC Reporting requirements per AD-LS-ALL-0006, Notification/Reportability Evaluation: d. Determination of "clock start" time for reportable events (LOCT)

Reference: 0OI-01.07 Attachment 1, NUREG 1022 Table 1

Cog Level: High

Explanation: A TS required shutdown requires a 4 hour NRC report. The time starts when the shutdown is actually started, Completion of the shutdown required by a TS is an LER.

Distractor Analysis:

Choice A: Plausible because this would be 4 hours from when the TS shutdown condition was entered. Part 2 plausible because an LER is required after completing a TS required shutdown, but in this case the shutdown was not TS required.

Choice B: Plausible because this would be 4 hours from when the TS shutdown condition was entered.

Choice C: Plausible because an LER is required after completing a TS required shutdown, but in this case the shutdown was not TS required.

Choice D: Correct Answer, see explanation.

SRO Basis: Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. [10 CFR 55.43(b)(5)]. Requires SRO administration procedure knowledge of reportability requirements based on plant conditions.

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Reportability Evaluation Checklist

NOTE

- If the answer to any of the following questions is YES, the event is reportable within 4 hours.
- If all answers to the following questions are NO, the event is not reportable within 4 hours.

4 HOUR REPORTABILITY			
ITEM #	**YES**	**NO**	**DESCRIPTIVE QUESTION**
2.1			**NOTE** Includes any Safety Limit violation (Tech Spec 2.2). Is plant shutdown required by technical specifications being initiated? [10 CFR 50.72(b)(2)(i)]

Plant Shutdown Required by Technical Specifications (See Section 3.2.1 of this report)	
§ 50.72(b)(2)(i) "The initiation of any nuclear plant shutdown required by the plant's Technical Specifications."	**§ 50.73(a)(2)(i)(A)** "The completion of any nuclear plant shutdown required by the plant's Technical Specifications."

Discussion

The 10 CFR 50.72 reporting requirement is intended to capture those events for which TS require the initiation of reactor shutdown to provide the NRC with early warning of safety-significant conditions serious enough to warrant that the plant be shut down. For 10 CFR 50.72 reporting purposes, the phrase "initiation of any nuclear plant shutdown" includes action to start reducing reactor power; i.e., adding negative reactivity to achieve a nuclear plant shutdown required by TS. This includes initiation of any shutdown due to expected inability to restore equipment prior to exceeding the LCO action time. As a practical matter, in order to meet the time limits for reporting under 10 CFR 50.72, the reporting decision should sometimes be based on such expectations. (See Example 4.)

The "initiation of any nuclear plant shutdown" does not include mode changes required by TS if they are initiated after the plant is already in a shutdown condition.

A reduction in power for some other purpose, not constituting initiation of a shutdown required by TS, is not reportable under this criterion.

For 10 CFR 50.73 reporting purposes, the phrase "completion of any nuclear plant shutdown" is defined as the point in time during a TS-required shutdown when the plant enters the first shutdown condition required by an LCO (e.g., hot standby (Mode 3) for PWRs with the Standard Technical Specifications (STS)). For example, if at 0200 hours a plant enters an LCO action statement that states, "restore the inoperable channel to operable status within 12 hours or be in at least Hot Standby within the next 6 hours," the plant must be shut down (i.e., at least in hot standby) by 2000 hours. An LER is required if the inoperable channel is not returned to operable status by 2000 hours and the plant enters hot standby.

An LER is not required if a failure was or could have been corrected before a plant has completed shutdown (as discussed above) and no other criteria in 10 CFR 50.73 apply.

5.4 Making Emergency Notification System and LER Reports (continued)

Table 1. Emergency Notification System Reporting Overview

Event or Condition	ENS notification within 1 hour	ENS notification within 4 hours	ENS notification within 8 hours	60-day LER	Job Aid Section
Plant shutdown (S/D) required by Tech Specs		Initiation of S/D required by Tech Specs [50.72 (b)(2)(i)]		Completion of a S/D required by Tech Specs [50.73 (a)(2)(i)(A)]	A.3, A.4

100. SG2.4.35 1
Unit One and Unit Two have entered SBO procedures at time 1300 due to a loss of all onsite and offsite power.

Which one of the following completes both statements below?

IAW 1EOP-01-SBO, *Station Blackout*, opening the reactor building roof hatch is required to be performed **no later than** ___(1)___.

IAW 0OI-37.14, *Station Blackout Procedure Basis Document*, the reactor building doors and roof hatch are opened to ensure ___(2)___.

A. (1) 1330
(2) equipment availability

B. (1) 1330
(2) habitability

C. (1) 1500
(2) equipment availability

D. (1) 1500
(2) habitability

Answer: D
K/A:
G2.4.35 Knowledge of local auxiliary operator tasks during an emergency and the resultant operational effects. (CFR: 41.10 / 43.5 / 45.13)

RO/SRO Rating: 3.8/4.0

Tier 3

K/A match: Requires the applicant to have knowledge of when to implement the AO task in a EOP sub procedure (opening RB roof hatch during SBO), and the operational effects if not completed (jeopardized habitability).

Pedigree: New

Objective: LOI-CLS-LP-303-B Obj 2 Given plant conditions, EOP-01-SBO Flowchart, and SBO Support Procedures, determine the required operator actions. Temperature analysis states that access would be prohibited in the RB building due to 117' elevation ceiling temperature if the hatch was not opened.

Reference: None

Cog Level: Fundamental

Explanation: This is a **time sensitive action** from the SBO procedure that directs the RB roof hatch to be opened within 2 hours of the start of the SBO.

Distractor Analysis:

Choice A: Part 1 is plausible because this is the time critical action time limit in SBO procedure for opening control panel doors. Part 2 is plausible because high temperatures could be thought to jeopardize equipment availability, however the hatch and doors are opened to ensure habitability.

Choice B: Part 1 is plausible because this is the time critical action time limit in SBO procedure for opening control panel doors. Part 2 is correct, see explanation.

Choice C: Part 1 is correct, see explanation. Part 2 is plausible because high temperatures could be thought to jeopardize equipment availability, however the hatch and doors are opened to ensure habitability.

Choice D: Correct, see explanation.

SRO Basis: Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. [10 CFR 55.43(b)(5)] Knowledge of when to implement attachments and the basis for the step.

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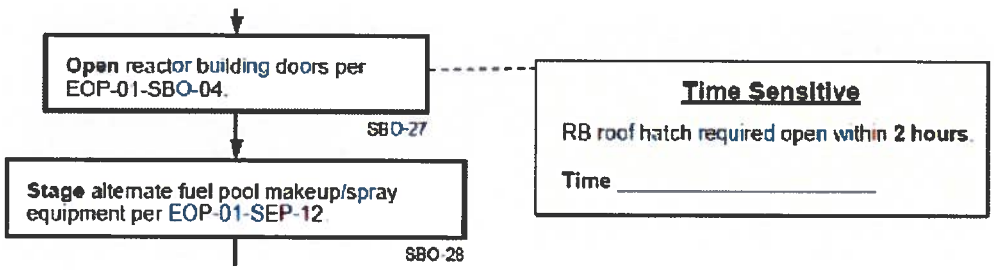
5.25 Steps SBO-27 and SBO-28



If ELAP conditions exist, reactor building temperatures will rise rapidly due to the loss of building ventilation. The refuel floor roof hatch and 20' elevation personnel access doors are blocked opened to provide alternate ventilation. The refuel floor roof hatch should be opened as soon as resources are available and is required to be open within 2 hours of the SBO start time recorded at Step SBO-1. The 20' elevation personnel access doors should be opened as soon as resources are available and are required to be open within 6 hours of the SBO start time recorded at Step SBO-1 by the text procedure. The reactor building temperature analysis (BNP-MECH-FLEX-0001) shows 117' elevation ceiling temperature will reach 114°F at time 2 hours, which is approaching the temperature that access would be prohibited. Alternate ventilation should be established as early as possible based on priorities and available resources.