U. S. Nuclear Regulatory Commission

Site-Specific SRO Written Examination

Applicant Information	
Name:	
Date: 11-24-15	**Facility / Unit: Browns Ferry Units 1,2,3**
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**	_____ / _____ / _____

Q 1

Unit 2 is in mode 5 with Refueling in progress, with the following system alignments:

- 2B RHR pump is in shutdown cooling.
- Both 2A and 2B Reactor Recirc pumps are tagged out of service.
- 2A RPS is powered from its alternate source.

Subsequently:

- Reactor Water Level drops to 0 inches and then recovers to + 75 inches.
- The only action taken by the crew was to reset any actuation(s) that may have occurred.

NOTE: RHR Sys II LPCI Inboard Injection Valve, 2-FCV-74-67
RHR Sys II LPCI Outboard Injection Valve, 2-FCV-74-66
RHR Shutdown Cooling Suction Outboard Isolation Valve, 2-FCV-74-47
RHR Shutdown Cooling Suction Inboard Isolation Valve, 2-FCV-74-48

Which ONE of the following describes the **minimum** actions required, in accordance with 2-AOI-74-1, Loss of Shutdown Cooling, prior to restarting the 2B RHR pump to restore Shutdown cooling?

A. CLOSE the 2-FCV-74-67, OPEN the 2-FCV-74-66, then OPEN 2-FCV-74-47 and 2-FCV-74-48.

B. CLOSE the 2-FCV-74-66, OPEN the 2-FCV-74-67, then OPEN 2-FCV-74-47 and 2-FCV-74-48.

C. CLOSE the 2-FCV-74-67, OPEN 2-FCV-74-66, then OPEN 2-FCV-74-48 only.

D. CLOSE the 2-FCV-74-66, OPEN 2-FCV-74-67, then OPEN 2-FCV-74-47 only.

Q 2

Units 1, 2, and 3 are operating at 100% power.

Subsequently:

A loss of all off site power occurs.

The following conditions exist:

- The C Diesel Generator is supplying the C 4KV shutdown board.
- The 3EB Diesel Generator is supplying the 3EB 4KV shutdown board.
- All other Diesel Generators failed to start.

Assume No Operator Actions Have Been Taken

Which ONE of the following completes the statement below?

Unit(s) _____ is (are) in a station black out.

A. 1 only

B. 2 only

C. 1 and 3 only

D. 2 and 3 only

Q 3

Which ONE of the following completes the statements below concerning the 250 VDC Unit batteries and battery chargers?

The Class 1E Unit Batteries have the capacity to compensate for a __ (1) __ Station Blackout event during multi-unit operations without operator action.

In accordance with 1/2-AOI-57-1D, 480V Load Shed, if the load shed logic can **Not** be reset the 2A 250V Battery charger may be returned to service by placing the charger select switch in __ (2) __.

A. (1) 4 hour
   (2) OFF then back to ON

B. (1) 12 hour
   (2) OFF then back to ON

C. (1) 4 hour
   (2) EMERG

D. (1) 12 hour
   (2) EMERG

Q 4

Which ONE of the following completes the statement below?

In the event of a Main Turbine trip without bypass valves from full power, a Reactor Scram is initiated to anticipate the __ (1) __ AND to prevent exceeding the __ (2) __ safety limit.

A. (1) rapid reduction in Reactor water level
   (2) Reactor water level

B. (1) rapid reduction in Reactor water level
   (2) MCPR

C. (1) rapid increase in Reactor pressure
   (2) Reactor water level

D. (1) rapid increase in Reactor pressure
   (2) MCPR

Q 5

Unit 1 is operating at 100% power when the B RPS MG set output breaker trips open.

Which ONE of the following describes required actions to place 1B RPS on alternate power in accordance with 1-AOI-99-1, Loss of Power to One RPS Bus?

A. Verify Circuit Protector 1B1 and 1B2 are Reset,
   Place the RPS bus 1B normal/alt transfer switch to ALT in Battery Board Rm 1

B. Verify Circuit Protector 1B1 and 1B2 are Reset,
   Place the RPS bus 1B normal/alt transfer switch to ALT in Battery Board Rm 2

C. Verify Circuit Protector 1C1 and 1C2 are Reset,
   Place the RPS bus 1B normal/alt transfer switch to ALT in Battery Board Rm 1

D. Verify Circuit Protector 1C1 and 1C2 are Reset,
   Place the RPS bus 1B normal/alt transfer switch to ALT in Battery Board Rm 2

Q 6

The Shift Manager has directed entering 3-AOI-100-2, Control Room Abandonment, due to heavy smoke in the U3 MCR.

Which ONE of the following completes the statements below?

The manual reactor SCRAM performed during 3-AOI-100-2 __ (1) __based on allowing time for operators to prepare for a plant cooldown.

If the Reactor fails to scram, 3-AOI-100-2 will direct __ (2) __.

A. (1) is
   (2) initiating ARI

B. (1) is
   (2) pulling RPS Scram Solenoid Fuses

C. (1) is NOT
   (2) initiating ARI

D. (1) is NOT
   (2) pulling RPS Scram Solenoid Fuses

Q 7

Unit 1 is operating at 100% power.

An RBCCW leak develops causing the 1-FCV-70-48, RBCCW Sectionalizing valve, to close, isolating the leak.

Which ONE of the following components has **Not** lost cooling water?

A. Drywell equipment drain sump

B. Fuel pool cooling heat exchangers

C. Reactor water cleanup pump seal coolers

D. RWCU Non-regenerative heat exchangers

Q 8

A rupture in the control air header has occurred.

- Control air pressure indicates 25 psig and lowering in the Unit 3 Control Room.
- 3-AOI-32-2, Loss of Control Air has been entered.
- Several U3 Control Rods failed to insert during the transient.
- The US directs inserting Control Rods in accordance with 3-EOI Appendix-1D

Which ONE of the following completes the statement below?

In order to insert Control Rods the Unit Operator is required to dispatch personnel to manually _____.

A. **open** the 3-FCV-85-11A, CRD Flow Control Valve, and **open** the 3-PCV-85-27, CRD Cooling Water Pressure Control Valve

B. **close** the 3-FCV-85-11A, CRD Flow Control Valve, and **open** the 3-PCV-85-27, CRD Cooling Water Pressure Control

C. **open** the 3-FCV-85-11A, CRD Flow Control Valve, and **close** the 3-HCV-85-586, Charging Water SOV

D. **close** the 3-FCV-85-11A, CRD Flow Control Valve, and **close** the 3-HCV-85-586, Charging Water SOV

Q 9

Unit 2 is in day 2 of a forced outage with the following conditions:

- Currently in Mode 4
- Moderator Temperature band is 150° F to 180° F
- Both Reactor Recirc pumps are OFF with suction valves open and discharge valves closed
- RHR pump 2A is in Shutdown Cooling

Subsequently:

The 2B Recirc Pump discharge valve is inadvertently opened.

With no other operator actions taken, which ONE of the following completes the statements below 4 hours after the 2B Recirc Pump discharge valve opened?

The RHR outlet temperature from the 2A RHR Heat Exchanger will __ (1) __ and actual moderator temperature will __ (2) __.

A. (1) Remain the same
   (2) Remain the same

B. (1) Lower
   (2) Lower

C. (1) Rise
   (2) Rise

D. (1) Lower
   (2) Rise

Q 10

Unit 1 is loading fuel into the core when the following occurs:

- SRM period lights illuminated.
- Rising count rate on SRM meters.
- Rising power level on IRM recorders.

What **Immediate Operator Actions** are required in accordance with 1-AOI-79-2, Inadvertent Criticality During Incore Fuel Movements?

A. Return fuel bundle to previous Spent Fuel Pool location and evacuate all personnel from Refuel Floor.

B. Remove the fuel bundle from the core and traverse the Refueling Bridge away from the Reactor core.

C. Stop all fuel handling and evacuate all personnel from Refuel Floor.

D. Remove the fuel bundle from the core and if still critical initiate SLC.

Q 11

All three Units are operating at 100% power when a small steam leak develops in Unit 2 Drywell.

The Unit Supervisor directs the Unit Operator to begin venting in accordance with the AOI.

(1) How many SGT train(s) are required to be verified running prior to venting?

(2) When venting is complete, in which Unit's SR-2 should the SGT run time be recorded?

A. (1) 1
   (2) 1

B. (1) 1
   (2) 2

C. (1) 2
   (2) 1

D. (1) 2
   (2) 2

Q 12

Which ONE of the following describes a basis for Alternate Rod Insertion (ARI) due to high Reactor pressure?

A. ARI limits fuel damage due to pellet expansion to less than 1%.

B. ARI reduces the challenge to the integrity of the Reactor Coolant Pressure Boundary.

C. ARI reduces unnecessary safety relief valve operation that challenges SRV and SRV piping integrity.

D. ARI reduces unnecessary safety relief valve operation that results in undesired heatup of the Suppression Pool.

Q 13

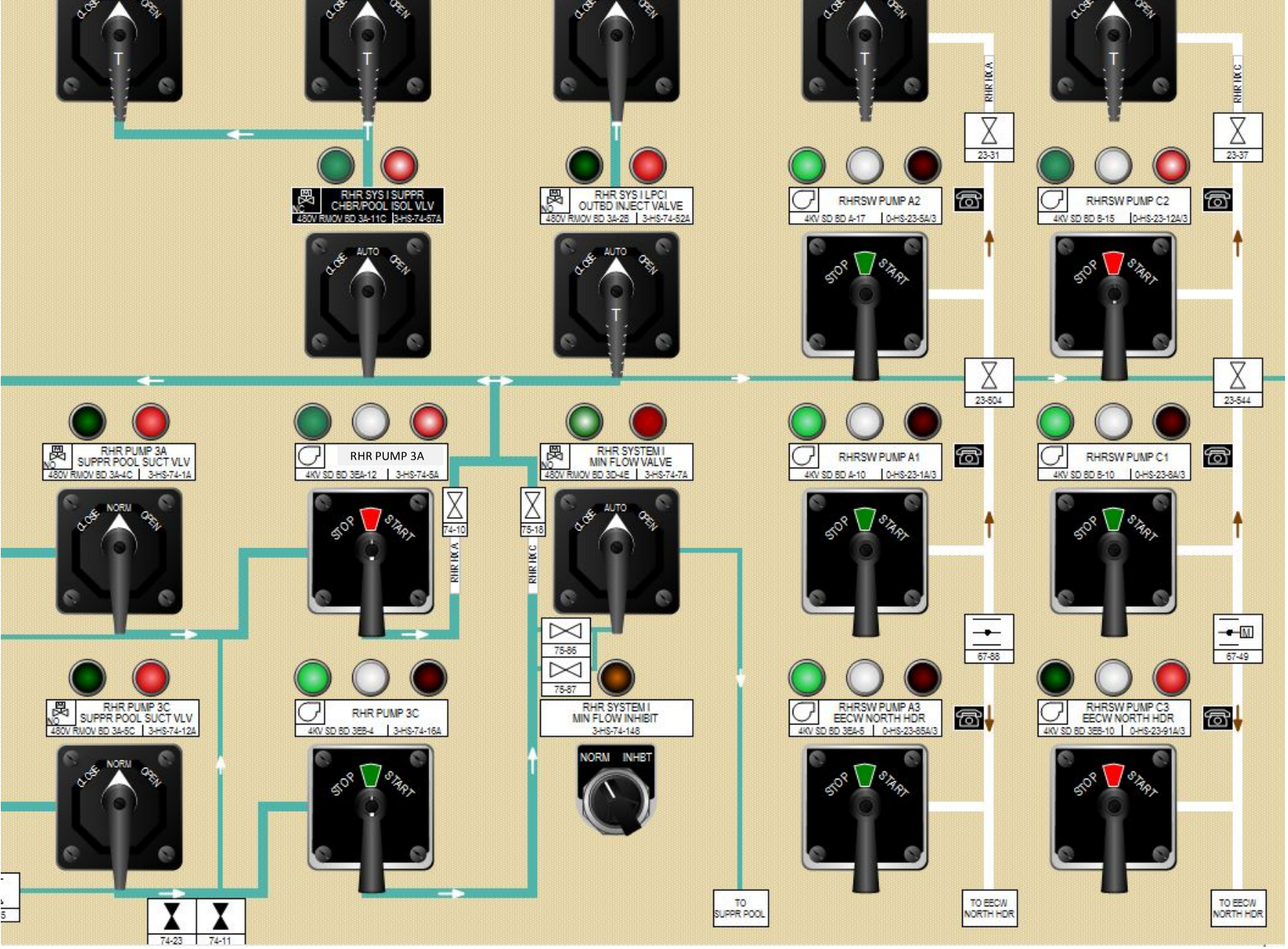
Based on the attached panel display, what changes (if any) are needed to place RHR System in Suppression Pool Cooling in accordance with 3-OI-74, Residual Heat Removal System?

A. No changes required

B. Start RHRSW Pump A2 and secure RHRSW Pump C2

C. Start RHRSW Pump A1 and open 3-FCV-23-34, 3A RHR Hx Outlet Valve

D. Fully open 3-FCV-74-59, RHR System I Suppression Pool Cooling/Test Valve


RHR SYS I SUPPR CHBR/POOL ISOL VLV
480V RMOV BD 3A-11C | 3-HS-74-57A
RHR SYS I LPCI OUTBD INJECT VALVE
480V RMOV BD 3A-2B | 3-HS-74-52A
RHRSW PUMP A2
4KV SD BD A-17 | 0-HS-23-5A/3
RHRSW PUMP C2
4KV SD BD B-15 | 0-HS-23-12A/3
RHR PUMP 3A SUPPR POOL SUCT VLV
480V RMOV BD 3A-4C | 3-HS-74-1A
RHR PUMP 3A
4KV SD BD 3EA-12 | 3-HS-74-5A
RHR SYSTEM I MIN FLOW VALVE
480V RMOV BD 3D-4E | 3-HS-74-7A
RHRSW PUMP A1
4KV SD BD A-10 | 0-HS-23-1A/3
RHRSW PUMP C1
4KV SD BD B-10 | 0-HS-23-8A/3
RHR PUMP 3C SUPPR POOL SUCT VLV
480V RMOV BD 3A-5C | 3-HS-74-12A
RHR PUMP 3C
4KV SD BD 3EB-4 | 3-HS-74-16A
RHR SYSTEM I MIN FLOW INHIBIT
3-HS-74-148
RHRSW PUMP A3 EECW NORTH HDR
4KV SD BD 3EA-5 | 0-HS-23-85A/3
RHRSW PUMP C3 EECW NORTH HDR
4KV SD BD 3EB-10 | 0-HS-23-91A/3
AUTO
NORM
INHBT
STOP
START
OPEN
CLOSE
T
23-31
23-37
23-504
23-544
74-10
75-18
75-86
75-87
67-88
67-49
74-23
74-11
RHR 10C A
RHR 10C C
TO SUPPR POOL
TO EECW NORTH HDR
TO EECW NORTH HDR

Q 14

In accordance with EOI Caution 1, LI-3-55 Reactor Water Level Flood-up Range, indicates above the minimum indicated level (MIL).

DW Temp and Reactor Pressure are in the Action Required region of Curve 8, RPV Saturation Curve.

Which ONE of the following completes the statement below?

LI-3-55 __ (1) __ due to __ (2) __.

A. (1) may **Not** be used
   (2) boiling in the instrument run

B. (1) may be unreliable
   (2) boiling in the instrument run

C. (1) may **Not** be used
   (2) being calibrated for cold conditions

D. (1) may be unreliable
   (2) being calibrated for cold conditions

Q 15

Annunciator 3-9-3B window 15, SUPPR CHAMBER WATER LEVEL ABNORMAL is currently in alarm due to Low SP Water Level.

Which ONE of the following answer choices provides indication supporting a lowering Suppression Pool Level trend?

A. Drywell-to-Suppression Chamber Differential pressure is lowering.

B. 3-9-3C window 9 RCIC PUMP SUCTION PRESS LOW 3-PA-71-21A alarms.

C. 3-9-3E window 11 SUPPR POOL DISCH HDR PRESS LOW 3-PA-74-94 alarms.

D. HPCI Pump Suction valves automatically realign.

Q 16

An ATWS has occurred on Unit 2.

- ATWS actions are complete
- Reactor water level currently indicates +40 inches
- Reactor Power is 46%
- SLC is injecting

Which ONE of the following completes the statements below?

EOI-1A requires operators to stop and prevent all injection except __ (1) __ to mitigate the consequences of the failure-to-scram.

Intentionally lowering Reactor Water Level mitigates the failure to scram by __ (2) __.

A. (1) CRD, and SLC only
(2) reducing natural circulation resulting in increased void fraction

B. (1) CRD, and SLC only
(2) increasing natural circulation resulting in mixing of injected boron

C. (1) RCIC, CRD, and SLC
(2) reducing natural circulation resulting in increased void fraction

D. (1) RCIC, CRD, and SLC
(2) increasing natural circulation resulting in mixing of injected boron

Q 17

In accordance with step ARC-1 and NOTE 1 of EOI-1A, ATWS RPV Control, which ONE of the following conditions would allow exiting EOI-1A and entering EOI-1, RPV Control?

A. All Control Rods inserted to 00 except 18 at notch 02

B. All Control Rods inserted to 00 except 2 at notch 18

C. SLC injected into RPV to a tank level of 60%

D. SLC injected into RPV to a tank level of 40%

Q 18

All three units are operating at 100% power.

A transient occurs on Unit 2 and the following alarms are received:

- 2-9-4C window 27 OG AVG ANNUAL RELEASE LIMIT EXCEEDED
- 2-9-3A window 13 STACK GAS RADIATION HIGH
- 2-9-7A window 3 STACK GAS DILUTION AIR FLOW LOW

The Unit 2 UO reports the following:

- Stack dilution fan 2A tripped and 2B failed to start
- 0-FI-90-271 Stack Gas Flow on Panel 1-9-53 indicates 14,000 scfm

Based on the information provided which ONE of the following identifies the Stack Gas Radiation Monitor(s) with valid indications.

A. None of the Stack Gas Radiation Monitor indications are valid.

B. Only 0-RM-90-306 WRGERMS indication is valid.

C. Only 0-RM-90-147/148 Stack Gas Monitor indications are valid.

D. 0-RM-90-306 WRGERMS and 0-RM-90-147/148 Stack Gas Monitor indications are valid.

Q 19

A fire has occurred in the Unit 3 Reactor Building.

In accordance with 0-AOI-26-1, Fire Response, the reason AUOs are assembled in the Control Room is to…

A. perform required SSI manual actions.

B. complete personnel accountability.

C. retrieve the Control Room Appendix R radios.

D. retrieve necessary SCBA Kits.

Q 20

In accordance with 0-AOI-57-1E, Grid Instability, what is the **maximum** MVAR outgoing limit to maintain the offsite qualification of both 500-Kv and161-Kv offsite power sources?

A. + 50

B. + 100

C. + 150

D. + 300

Q 21

Unit 3 is operating at 100% power when all three Reactor Feed Pump Turbines trip.

The Reactor is manually scrammed and 3-EOI-1 is entered.

As Reactor water level lowers, receipt of which alarm below corresponds to the level at which a Reactor Recirc Pump trip is required?

A. 3-9-5A window 8, REACTOR WATER LEVEL ABNORMAL

B. 3-9-3F window 29, RX WTR LVL LOW LOW HPCI/RCIC INIT

C. 3-9-3C window 28, RX WTR LVL LOW LOW LOW ECCS/ESF INIT

D. 3-9-3C window 3, REACTOR LEVEL LOW ADS BLOWDOWN PERMISSIVE

Q 22

Unit 2 is operating at 100% Power.

Subsequently:
A transient results in the following:

- 2-9-4C window 35 OG POST TRTMT RAD MONITOR Hi-Hi-Hi/INOP alarms and will **NOT** reset.

- The Automatic and Immediate actions of 2-AOI-66-2, Offgas Post-Treatment Radiation HI-HI-HI were completed.

ASSUME NO OTHER OPERATOR ACTIONS ARE PERFORMED.

Which ONE of the following describes the expected system response?

The SJAE in service prior to the event will __ (1) __.

The indication on 0-RM-90-147/148 Stack Gas RAD Monitors one hour after the immediate actions of 2-AOI-66-2 are complete will __ (2) __ they indicated prior to the transient.

A. (1) remain in service
   (2) be lower than

B. (1) remain in service
   (2) remain the same as

C. (1) shutdown
   (2) be lower than

D. (1) shutdown
   (2) remain the same as

Q 23

Due to an error while performing surveillance testing on Unit 2, a Secondary Containment isolation was initiated.

In accordance with 0-OI-65, Standby Gas Treatment System, which ONE of the following completes the statement below?

The SGT Relative Humidity heaters will ___(1)___.

The Refuel Zone Exhaust to SGT dampers 1-FC0-064-0044 & 0045, __ (2) __ auto open.

A. (1) energize
   (2) will

B. (1) energize
   (2) will Not

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

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

Q 24

Unit 1 Suppression Pool Level is + 5.5 inches.

Which ONE of the following completes the statements below?

HPCI Suction __ (1) __ automatically transfer to the Suppression Pool.
RCIC Suction __ (2) __ automatically transfer to the Suppression Pool.

A. (1) will
(2) will

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

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

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

Q 25

Unit 2 is operating at 100% Power when 2-9-3D window 24 MAIN STEAM LINE LEAK DETECTION TEMP HIGH alarms.

The BOP Operator reports that 2-TIS-1-60A; MN STEAM TUNNEL TEMP indicates170 °F and rising.

Which ONE of the following completes the statements below?

An EOI-3 entry condition __ (1) __ been met.

The MSIV closure setpoint for the Steam Tunnel temperature is __ (2) __ °F.

A. (1) has
   (2) 189

B. (1) has
   (2) 315

C. (1) has Not
   (2) 189

D. (1) has Not
   (2) 315

Q 26

Unit 1 is in Mode 5, Units 2 and 3 are in Mode 1.

A Refueling accident occurs on Unit 1 resulting in the following readings:

1-RM-90-140/142

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

1-RM-90-141/143

- Reactor Zone 1-RM-90-143A indicates 68mr/hr
- Reactor Zone 1-RM-90-143B indicates down scale
- Refuel Zone 1-RM-90-141A indicates 70mr/hr
- Refuel Zone 1-RM-90-141B indicates 69mr/hr

Which ONE of the following identifies the ventilation response?

A. Refuel Zone isolation only

B. Reactor and Refuel Zone isolation

C. Reactor Zone isolation and CREV auto initiation

D. Refuel Zone isolation and CREV auto initiation

Q 27

The Radwaste Operator reports that Unit 1 Reactor Building Floor Drain Sump B level is 50 inches and rising with the B Sump pump running.

Which ONE of the following completes the statements below?

An EOI-3, Secondary Containment Control, entry condition is first met when level rises an additional __ (1) __ inches.

In accordance with the EOI Program Manual Section 0-V-E, EOI-3 Secondary Containment Control Bases, the reason for isolating a system that is discharging into Secondary Containment is to __ (2) __.

A. (1) 16
   (2) terminate the radioactivity release

B. (1) 16
   (2) maintain Reactor Building pressure negative

C. (1) 25
   (2) terminate the radioactivity release

D. (1) 25
   (2) maintain Reactor Building pressure negative

Q 28

A loss of coolant accident occurred on Unit 3
The following conditions exist:

- 3A RHR pump is running in LPCI mode
- Reactor water level has been stabilized at +15 inches
- Drywell spray was initiated using 3B RHR pump
- Drywell Temperature is 250 °F and slowly lowering
- Drywell Pressure is 13 psig and slowly lowering

Subsequently:

- 3-9-3D window 29 RHR/CS DIV I TEMP HIGH alarms
- The Reactor Building AUO reports that the 3A RHR Room Cooler is **NOT** running and would **NOT** start using the local pushbutton

Which ONE of the following completes the statements below?

Place RHR pump __ (1) __ in service in LPCI mode and secure 3A RHR pump.

The lowest temperature at which the RHR Room Cooler will auto start is__ (2) __ if it is not already running.

A. (1) 3C
   (2) 95 °F

B. (1) 3D
   (2) 95 °F

C. (1) 3C
   (2) 148 °F

D. (1) 3D
   (2) 148 °F

Q 29

What is the power supply for RHR SYS II INBD INJECTION VLV, 2-FCV-74-67?

480 V RMOV Board…

A. 2A

B. 2B

C. 2D

D. 2E

Q 30

HPCI is running in pressure control in accordance with 2-EOI Appendix-11C, Alternate RPV Pressure Control Systems HPCI Test Mode when the following event occurs:

- Condensate Storage Tank (CST) level dropped below 6800 gallons.

What is the current status of the HPCI system?

HPCI is…

A. operating in pressure control with suction from the CST.

B. operating in pressure control with suction from the Suppression Pool.

C. operating at shutoff head with suction from the Suppression Pool.

D. tripped on low suction pressure.

Q 31

Unit 1 Core Spray is being shut down following an automatic actuation in accordance with 1-OI-75, Core Spray, section 7.1 Core Spray System Shutdown.

At what flow is the Minimum Flow Valve, 1-FCV-75-9(37) expected to open when the Inboard Injection Valve, 1-FCV-75-25(53) is throttled closed?

A. 900 gpm

B. 1350 gpm

C. 2200 gpm

D. 2600 gpm

Q 32

Given the following conditions:

- A Unit 1 ATWS occurred
- During the performance of 1-EOI-Appendix 3A, SLC INJECTION, the Standby Liquid Control (SLC) pump control switch was placed in the START- A position
- SQUIB VALVE A CONTINUITY blue light is illuminated
- SQUIB VALVE B CONTINUITY blue light is extinguished
- SLC INJECTION FLOW TO REACTOR (Panel 1-9-5B, Window 14) is in alarm

Which ONE of the following completes the statements below?

The SLC Squib valve ___ (1) ___ is OPEN.
The time to inject Hot Shutdown Boron Weight is ___ (2) ___ compared to the time with both squib valves open.

A. (1) A
   (2) the same

B. (1) A
   (2) longer

C. (1) B
   (2) the same

D. (1) B
   (2) longer

Q 33

2-SR-3.1.4.1, SCRAM Insertion Times, is in progress.

- At Panel 2-9-16 the UO performs the actions to test Control Rod 26-43, and returns the SCRAM TEST switch back to the NORMAL position.

What is the status of the scram blue light for control rod 26-43 on the full core display?

A. Illuminated while the SCRAM TEST switch is in the TEST position, but extinguishes **immediately** when placed back in NORMAL (power restored).

B. Illuminated while the SCRAM TEST switch is in the TEST position, but extinguishes when **either** scram valves reclose (limit switch).

C. Illuminated until the SCRAM RESET switch on panel 9-5 is placed in **either** the GRP 2/3 or 1/4 position.

D. Illuminated until the SCRAM RESET switch on panel 9-5 is placed in **both** the GRP 2/3 and GRP 1/4 positions.

Q 34

A reactor plant startup is being conducted on Unit 2 in accordance with GOI-100-1A, Unit Startup and Power Operation.

- The reactor is critical and SRM/IRM overlap data has just been completed.
- All SRMs are reading between 5.0 X 10^3 and 1.0 X 10^4 cps.
- All IRMs are mid scale on range 1.
- The operator has inadvertently selected both the SRMs and the IRMs for withdraw.

Which ONE of the following Control Rod Blocks will be the **first** automatic action to occur as the detectors are withdrawn?

A. SRM Detector Wrong position

B. IRM Detector Wrong position

C. SRM Downscale

D. IRM Downscale

Q 35

What are the power supplies to the SRM Channels/detectors?

SRM Channels/Detectors _____.

A. A & B are powered from the A channel ± 24VDC System and C & D are powered from the B channel ± 24VDC System

B. A & C are powered from the A channel ± 24VDC System and B & D are powered from the B channel ± 24VDC System

C. A & B are powered from Division I, 250 VDC System and C & D are powered from Division II, 250 VDC System

D. A & C are powered from Division I, 250 VDC System and B & D are powered from Division II, 250 VDC System

Q 36

Unit 2 is operating at power.

Given that Core Flow is 65%, APRM 1 will display which approximate Rod Block setpoint?

A. 102

B. 108

C. 113

D. 119

Q 37

How many LPRM detectors are assigned to each APRM channel and how many LPRM detectors are in each LPRM string?

A. 21; 3

B. 21; 4

C. 43; 3

D. 43; 4

Q 38

What Reactor Core Isolation Cooling (RCIC) design feature provides for the prevention of water hammer?

A. Suction head pressure provided by the CST

B. Minimum flow valve automatic operation

C. System snubbers

D. Low pressure isolation

Q 39

During a Unit 1 ATWS, the UO places ADS LOGIC INHIBIT switches 1-XS-1-159A and 1-XS-1-161A in inhibit then reports:

- 1-9-5 window 18 ADS LOGIC BUS A INHIBITED failed to alarm.
- 1-9-5 window 31 ADS LOGIC BUS B INHIBITED is in alarm.

Which ONE of the following completes the statement below?

In accordance with 1-ARP-9-3C window 18 the UO will direct an AUO to _____.

A. open the ADS System Logic Bus A breaker on 250V RMOV board 1B

B. pull 250V Logic A fuses on Panel 1-9-30 in the Auxiliary Instrument room

C. place all ADS transfer switches in emergency at Panel 1-25-32, Backup Control Panel

D. pull all ADS Solenoid power fuses at Panel 1-25-32, Backup Control Panel

Q 40

Which ONE of the following completes the statements below?

Reactor Water Level Instruments,__ (1) __ provide the Reactor Vessel water level Low-Low-Low initiation signal to ADS logic.

RHR or Core Spray pumps __ (2) __ required to be running to initiate the ADS timers.

NOTE: LIS-3-184 is Reactor Water Level A
LIS-3-185 is Reactor Water Level B
LIS-3-58A through D is Reactor Water Level A through D

A. (1) LIS-3-58A through D
(2) are

B. (1) LIS-3-58A through D
(2) are Not

C. (1) LIS-3-184 and LIS-3-185
(2) are

D. (1) LIS-3-184 and LIS-3-185
(2) are Not

Q 41

What design feature allows testing of MSIV Reactor Water Level Instrumentation associated with Primary Containment Isolation System (PCIS) without causing a device actuation?

A. 1 out of 2 taken twice logic

B. 2 out of 3 logic

C. 2 out of 4 voter logic

D. 2 out of 2 taken once logic

Q 42

How are the ADS MSRVs affected by a loss of Drywell Control Air (DWCA)?

ADS MSRVs will…

A. Not operate in the Manual mode.

B. operate a **minimum** of three times in the Manual mode.

C. operate a **minimum** of five times in the Manual mode.

D. operate indefinitely in the Manual mode.

Q 43

Unit 3 is operating at 100% power, with the following feedwater alignment:

- Reactor Water Level Master Controller in **MAN**
- A RFPT Speed Controller in **AUTO** at 5000 RPM
- B RFPT Speed Controller in **AUTO** at 4995 RPM
- C RFPT Speed Controller in **MAN** at 5005 RPM

How will Reactor Feed Pumps respond when the Reactor Water Level Master Controller raise pushbutton is depressed?

A. A, B and C RFPT speeds increase.

B. A, B and C RFPT speeds remain the same.

C. A and B RFPT speeds increase; RFPT C speed remains the same.

D. A and B RFPT speeds remain the same; RFPT C speed increases.

Q 44

Which ONE of the following (if any) identifies the suction source(s) that can be aligned for the Standby Gas Treatment Fans with respect to the Primary Containment System?

A. None

B. Drywell ONLY

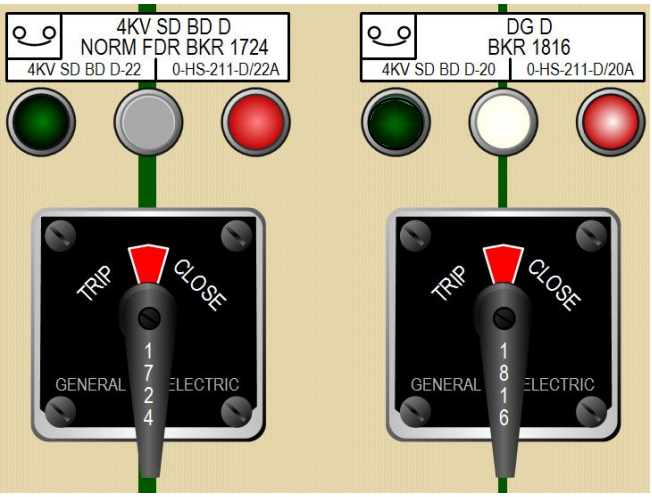
C. Suppression Chamber ONLY

D. Drywell and Suppression Chamber

Q 45

Unit 2 is performing 0-SR-3.8.1.1(D), Diesel Generator D Monthly Operability; the Diesel has been loaded for 30 minutes.

The following indications have **just occurred**.



Which ONE of the following completes the statements below?

The white light above BKR 1816 is a __ (1) __.

Based on these conditions the first expected response is __ (2) __.

A. (1) disagreement indication
(2) DG D Breaker 1816 will trip open

B. (1) disagreement indication
(2) 4KV SD D Normal FDR BKR 1724 will trip open

C. (1) Diesel Generator Overload indication
(2) DG D Breaker 1816 will trip open

D. (1) Diesel Generator Overload indication
(2) 4KV SD D Normal FDR BKR 1724 will trip open

Q 46

All three Units are operating at 100% power.

- 240V Lighting Board 2A is tagged out of service for scheduled work.
- An electrical fault causes 240 V Lighting Board 3B to deenergize.

Which ONE of the following completes the statements below?

The Plant Preferred MG will start __ (1) __ and energize __ (2) __on all 3 units.

A. (1) immediately
   (2) Panel 9-9 cabinet 4

B. (1) immediately
   (2) Panel 9-9 cabinet 5

C. (1) after a 6 second time delay
   (2) Panel 9-9 cabinet 4

D. (1) after a 6 second time delay
   (2) Panel 9-9 cabinet 5

Q 47

Unit 1 is operating at 100% Power.

1-9-8B window 35 UNIT PFD SUPPLY ABNORMAL alarms

The Control Bay AUO reports the following lights illuminated at the Unit 1 Unit Preferred System Inverter:

- 1-IL-252-0001L (Red Lamp) Inverter Fuse Blown
- 1-IL-252-0001D (Red Lamp) Alternate Source Supplying Load

ASSUME NO OPERATOR ACTIONS HAVE BEEN PERFORMED.

Which ONE of the following completes the statement below?

The Unit 1 Unit Preferred loads are being supplied through the_____.

A. Unit Preferred Inverter Static Switch

B. Alternate supply to 1-PNL-9-9 cabinet 6 only

C. UNIT PFD XFMR1 TO BATTERY BD 1 ALT FDR, 0-BKR-280-001/1002

D. UNIT PFD MMG SET 2 TO BATT BD 1 EMERG FDR, 0-BKR-001/1003

Q 48

All three Units are operating at 100% power when the following alarm is received:

- Panel 1/2-9-23B, Window 17 DIESEL GEN B BAT CHGR OR EXH FAN ABN

What local indication does the AUO have to diagnose that the cause of the alarm is the 125 VDC Battery Charger?

A. Local relay targets on the front of the charger.

B. B Diesel Generator room Emergency lights illuminated.

C. Voltage meter on the front of the battery charger.

D. Central Diesel Information Center Alarm Panels.

Q 49

All three Units are operating at 100% power.

Subsequently,

4KV SD BUS 2 de-energizes.

Which ONE of the following completes the statements below?

The “D” and __ (1) __ Diesel Generators will auto start.

In accordance with 0-OI-82, Standby Diesel Generator System, the Diesel Generator Maximum Continuous steady-state active power output (KW) is limited to __ (2) __.

A. (1) B
   (2) 2600 kW

B. (1) B
   (2) 2860 kW

C. (1) C
   (2) 2600 kW

D. (1) C
   (2) 2860 kW

Q 50

Which ONE of the following completes the statement below?

When Control Air is lost, the Drywell Control Air System…

A. loses its only backup source of pneumatics.

B. loses one of two backup sources of pneumatics.

C. slowly depressurizes.

D. remains pressurized due to installed accumulators.

Q 51

The G Control Air Compressor is in service with the other Control Air Compressors A, B, C, and D in Standby in accordance with 0-OI-32, Control Air System.

Subsequently, the G Control Air compressor trips.

How do the other Control Air Compressors respond as pressure lowers to 90 psig?

A. Only those compressors which are selected as lead start.

B. All compressors start on a common setpoint simultaneously.

C. The compressors which are selected as lead start followed by the lag compressors with a 2 psig offset between them.

D. All control air compressors start one at a time with a 2 psig offset between them.

Q 52

All three units are operating at 100% power.
The A3 RHRSW pump is tagged for motor replacement.

Subsequently:

The C-3 EECW pump trips and the AUO reports that the pump is hot to the touch.

In accordance with 0-OI-67, Emergency Equipment Cooling Water System, which ONE of the following completes the statements below?

RHRSW pump __ (1) __ can be aligned to EECW in place of the C-3 RHRSW pump.

This pump __ (2) __ the same AUTO start signals as the C-3 RHRSW pump.

A. (1) C-1
   (2) has

B. (1) C-1
   (2) does Not have

C. (1) C-2
   (2) has

D. (1) C-2
   (2) does Not have

Q 53

Which way does 1-FCV-70-1, RBCCW Surge Tank fill valve, fail and where do you send someone to control level in the RBCCW Surge Tank?

A. Open; Reactor Building EL 639 foot.

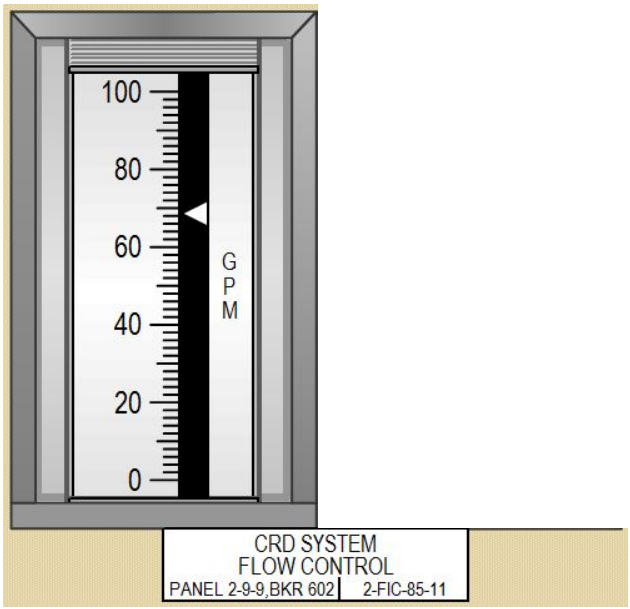
B. Open; Reactor Building EL 593 foot.

C. Closed; Reactor Building EL 639 foot.

D. Closed; Reactor Building EL 593 foot.

Q 54

Unit 2 is operating at 100 % power with the following indication:



Which ONE of the following completes the statements below?

In accordance with 2-OI-85, Control Rod Drive System, The Control Rod Drive system flow __ (1) __ in the normal band.

The next time the UO adjusts CRD system flow, Calculated Thermal Power __ (2) __ be affected.

A. (1) is
(2) will

B. (1) is
(2) will Not

C. (1) is Not
(2) will

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

Q 55

How much flow is provided to the CRD to get the collet fingers released from the notch in the index tube during a normal control rod withdrawal?

Which direction does the Unit Operator throttle the PCV-85-23, CRD Drive Water pressure control valve to raise Drive Water differential pressure?

A. (1) 2 gpm
   (2) open

B. (1) 2 gpm
   (2) closed

C. (1) 4 gpm
   (2) open

D. (1) 4 gpm
   (2) closed

Q 56

Unit 3 is operating at 100% power when the following occurs:

- The 3C RFPT tripped
- Reactor Water level lowered to + 25 inches on the Normal Range instruments.

What is the expected response of the Reactor Recirc System?

Recirc Pumps' speeds lower to achieve…

A. 480 rpm.

B. 1130 rpm.

C. a core flow of 60 Mlbm/hr.

D. a steam flow of 10.9 Mlbm/hr.

Q 57

As Reactor Power rises past 25%, what provides a signal to the Rod Block Monitor (RBM) to begin enforcing Control Rod Blocks?

A. Local Power Range Monitor

B. Average Power Range Monitor

C. Total Steam Flow

D. Total Feedwater Flow

Q 58

Unit 3 is operating at 100% Rx Power with the following indications and alarms present:

- DRYWELL DP AIR COMP DISCH AIR TEMP HIGH, Panel 9-3B window 33
- DRYWELL TO SUPPR CHAMBER DIFF PRESS ABNORMAL, Panel 9-3B window 26
- DRYWELL TO SUPPR CHAMBER DIFF PRESS, 3-PDS-64-137C is reading 1.41 psid
- DRYWELL TEMPERATURE, 3-TE-64-52C is reading 135 °F
- Drywell DP Compressor is running

Which ONE of the actions below describes the highest priority?

A. Stop the DP Air Compressor

B. Bypass the DP Air compressor TCV

C. Open the DP Air Compressor Bypass Valve

D. Blow down RCW to the DP Air Compressor after cooler

Q 59

During Refueling Operations with the Reactor mode switch in the refuel position, the following events occur:

- A fuel bundle is pulled to full up from its spent fuel pool location.
- The bridge is then driven over the core to its new location and the Refueling Bridge operator starts lowering the fuel bundle into the core.
- **NO** Rod Block alarm is received during this evolution.

Based on the events that just occurred what action is **immediately required** by Tech Specs?

A. Suspend in-vessel fuel movement.

B. Insert a control rod withdrawal block only.

C. Verify all control rods are fully inserted only.

D. Place the reactor mode switch in the shutdown position.

Q 60

The following plant conditions exist on Unit 2:

- Main Turbine Shell Warming is in progress
- The UO is pulling Control Rods in accordance with 3-GOI-100-1, Unit Startup

3-OI-47, Turbine-Generator System section 5.2 Turbine Shell Warming cautions the Operators that a Reactor Scram may result when Main Turbine First stage pressure exceeds _____ psig.

A. 105

B. 115

C. 147

D. 165

Q 61

The Unit 1 Main Generator synchronization is in the progress IAW 1-GOI-100-A, Unit Startup.

The following indications are observed on panel 1-9-8:

- VOLTAGE REGULATOR MAN/AUTO in MAN
- GEN SYNC REF VOLTAGE, 1-E-57-54 is reading 27 V
- SYSTEM SYNC REF VOLTAGE is reading 28 V
- SYNCHROSCOPE 1-XI-57-55 is stopped at the 6:00 position

In accordance with 1-OI-47 before the Generator PCB 214 is closed, the operator must go to __ (1) __ on the Voltage Regulator Lower/Raise Adjust Switch to match voltages.

The operator must also go to __ (2) __ on the Turbine Generator Synch Speed INC/DEC Adjust Switch until the Synchroscope is moving slowly in the clockwise direction.

A. (1) raise
   (2) INC

B. (1) raise
   (2) DEC

C. (1) lower
   (2) INC

D. (1) lower
   (2) DEC

Q 62

What is the effect on the Reactor Feedwater System from a loss of **120V I&C Bus A**?

A. RPFT 2B Woodward Governor loses power.

B. RFP 2C Minimum Flow Valve fails open.

C. RFW Start-up Level Control PDS controls are rendered inoperative.

D. RFPT/RFP 2A Vibration Monitoring Equipment loses indication.

Q 63

The Waste Collector Tank **normally** receives discharge from which system drains?

A. Floor

B. Laundry

C. Laboratory

D. Equipment

Q 64

What is the power supply to the Stack-Gas Radiation Monitor (0-RM-90-147 & 148) scintillation detectors?

A. Unit 1 (±) 24 VDC Neutron Monitoring Battery System

B. Unit 2 (±) 24 VDC Neutron Monitoring Battery System

C. Unit 1 120 VAC Reactor Protection System

D. Unit 2 120 VAC Reactor Protection System

Q 65

A fire has been reported in Unit 2 Auxiliary Instrument Room and the CO_2 System failed to automatically or manually initiate.

The Unit Supervisor has ordered the AUO to manually initiate CO_2 using the Pilot Control Valve Station(s).

How will the CO_2 System respond when the pilot valve lever is placed in the OPEN position?

CO_2 will be dispensed ___ (1) ___ and the evacuation alarm ___ (2) ___ sound.

A. (1) immediately
   (2) will

B. (1) immediately
   (2) will Not

C. (1) after 60 sec time delay
   (2) will

D. (1) after 60 sec time delay
   (2) will Not

Q 66

What is the frequency of panel walk downs in accordance with OPDP-1, Conduct of Operations?

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

A. per hour

B. every 2 hours

C. every 4 hours

D. every 6 hours

Q 67

In accordance with 2-GOI-100-1A, Unit Startup and Power Operation, there are Initial Steps that contain an (R) before that step, what requirement is imposed?

The step…

A. requires Radiation Protection notification only.

B. requires holding for Radiation Protection, (RP Hold Point).

C. indicates a restriction on Reactor Power, prior to proceeding.

D. is required and shall not be omitted, unless permitted in the step.

Q 68

In accordance with ODM-4.5, Operator Aids and Operator Information System, how does the Unit Operator determine during the panel walk down, that a system is aligned correctly?

The normally running pumps shall have a ___ (1) ___red lens cover.

The normally closed valves shall have a ___ (2) ___ green lens cover.

A. (1) clear
   (2) clear

B. (1) diffused
   (2) diffused

C. (1) diffused
   (2) clear

D. (1) clear
   (2) diffused

Q 69

Unit 1 is performing a startup in accordance with 1-GOI-100-1A, Unit Startup.

During control rod notch withdrawal, prior to criticality, SRM PERIOD, (1-9-5A, Window 20), alarms and **seals in**.

What action(s) is/are required by 1-GOI-100-1A?

A. **PAUSE** Control Rod withdrawal until a stable period of greater than 100 seconds is observed.

B. **REINSERT** the last Control Rod withdrawn to obtain a stable period greater than 60 seconds.

C. **INSERT** Control Rods and **ENSURE** the Reactor is brought subcritical.

D. **SHUTDOWN** the Reactor until a thorough assessment has been performed.

Q 70

Which ONE of the following completes the statement below?

In accordance with NPG-SPP-07.3.4, Protected Equipment, when unscheduled work requires protecting equipment, it is required to be posted as Protected Equipment unless the expected unavailability time is less than __ (1) __.

A. the duration of the current shift

B. the duration of the following shift

C. 12 hours

D. 24 hours

Q 71

Which ONE of the following meets the requirements to be considered a "Complex Infrequently Performed Test or Evolution" (CIPTE) per NPG-SPP-06.9.1, Conduct of Testing?

A. Switching Order to remove the West Point 500KV line

B. 0-SR-3.8.1.9(A) Diesel Generator A Emergency Unit 1 Load Acceptance Test

C. 1-SR-3.5.1.7(COMP), HPCI Comprehensive Pump Test (IST Data)

D. 1-SR-3.5.1.6(RHR I) Quarterly RHR System Rated Flow Test Loop I

Q 72

Which ONE of the following completes the statement below?

The Wide Range Gaseous Effluent Radiation Monitor System (WRGERMS) consists of__ (1) __ ranges, AND can be monitored remotely from__ (2) __.

A. (1) two
   (2) all three Units Control Room

B. (1) two
   (2) the Unit 2 Control Room

C. (1) three
   (2) all three Units Control Room

D. (1) three
   (2) the Unit 2 Control Room

Q 73

Which ONE of the following completes the statements below in accordance with RCI-9.1, Radiation Work Permits?

During an emergency situation, the Shift Manager has authorized immediate entry for Maintenance personnel into a High Radiation Area for which an RWP is not current.

Radiation Protection __ (1) __ be required to escort personnel entering the area.

When the area has been exited and the emergency situation is over, an RWP __ (2) __ required to be completed for this entry.

A. (1) will
   (2) is

B. (1) will
   (2) is Not

C. (1) will Not
   (2) is

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

Q 74

Which of the following is an **ENTRY CONDITION** into the Emergency Operating Instructions (EOI) and what is the overall mitigating strategy as directed by that EOI?

A. RPV Pressure above 1050 psig; Maintain adequate core cooling.

B. Suppression Pool Level above (-) 1 inch; Maintain the integrity of Primary Containment.

C. Rx power >5% or unknown; Expedite plant cooldown to place the reactor in the lowest energy state.

D. Spent Fuel Pool Temperature above 100 ºF; Maintain the continued operability of equipment needed to carry out the EOIs.

Q 75

Unit 1 is operating at 100% power.

Which ONE of the following completes the statement below?

When assessing the EOI Exclusion Plot Status Boxes on the Safety Parameter Display System (SPDS):

__ (1) __ is expected to be colored “red” because current plant operation __ (2) __ within the “Safe” region of the curve.

A. (1) Curve 6, Press Suppr Press
   (2) is

B. (1) Curve 6, Press Suppr Press
   (2) is Not

C. (1) Curve 5, DW Spray Init Limit
   (2) is

D. (1) Curve 5, DW Spray Init Limit
   (2) is Not

Q 76

In accordance with Tech Spec bases section 3.8.1, AC Sources – Operating, which ONE of the following completes the statements below?

When the A 4kv Shutdown board **normal** feeder breaker trips, the A Diesel Generator is required to energize the A 4KV Shutdown board within __ (1) __ seconds.

Offsite power to the A 4KV Shutdown board, when aligned through the **alternate** feeder breaker, __ (2) __ be credited in accordance with Technical Specification Bases.

A. (1) 5
   (2) Can

B. (1) 5
   (2) Cannot

C. (1) 10
   (2) Can

D. (1) 10
   (2) Cannot

Q 77

Unit 3 is operating at 100% power.

- CSST A is tagged on a SWLD Hold Order.
- Start bus 1A and 2A have been transferred to **alternate**.

Which ONE of the following completes the statements below?

Based on the conditions above, multiple Units __ (1) __ claim a 161KV offsite power circuit simultaneously.

Subsequently,

The Unit 3 Main Turbine trips due to Main Xfmr/USST Differential (386TX) relay operation.

Tech Spec 3.8.1, AC Sources-Operating requires restoring one offsite circuit to operable within __ (2) __.

[REFERENCE PROVIDED]

A. (1) can
   (2) 24 hours

B. (1) can
   (2) 7 days

C. (1) can Not
   (2) 24 hours

D. (1) can Not
   (2) 7 days

Q 78

Unit 1 was operating at 100% Reactor Power when the following series of events occurred:

- 02:05 Unit 1 Control Room Evacuation is initiated due to smoke in the Control Room
- 02:09 the Backup Control Panel, 1-25-32 is manned
- 02:13 the Operator has control of Reactor Pressure
- 02:17 AUO reports HPCI is running
- 02:19 Reactor Water Level is currently (-) 30 inches and rising
- 02:28 RCIC is initiated from Panel 1-25-32

Which ONE of the following completes the statements below?

In accordance with EPIP-1, “Emergency Plan Implementing Procedure”, the HIGHEST Emergency Action Level Classification that is required for these conditions is a (an) __ (1) __.

Drywell temperature and pressure are being controlled by__ (2) __.

[REFERENCE PROVIDED]

A. (1) Alert
   (2) containment venting

B. (1) Alert
   (2) operation of DW Blowers

C. (1) Site Area Emergency
   (2) containment venting

D. (1) Site Area Emergency
   (2) operation of DW Blowers

Q 79

All three Units are operating at 100% power.

The following are received in the Unit 1 Control Room:

- Report of a leak from the South EECW header in the DG building hallway.
- All EECW pumps are operating at their required flow rate.
- The EECW South Header is depressurized.

In accordance with Tech Spec bases section 3.7.2, EECW System and UHS which ONE of the following completes the statements below?

The EECW system __ (1) __ operable.

The EECW system __ (2) __ supply adequate cooling for continuous operation of the Unit 1/2 Diesel Generators.

A. (1) is
   (2) can

B. (1) is
   (2) can Not

C. (1) is Not
   (2) can

D. (1) is Not
   (2) can Not

Q 80

All three Units are operating at 100% Rx Power.
The “G” Control Air Compressor is tagged for scheduled work.

The following are received in the Unit 1 Main Control Room:

- AIR COMPRESSOR ABNORMAL 1-9-20B window 29
- SERVICE AIR XTIE VLV OPEN 1-9-20B window 30
- CONTROL AIR DRYER DISCH PRESSURE LOW 1-9-20B window 32

The Turbine Building AUO reports that a fork lift has ruptured one of the Control Air Receivers.

Assume no Operator actions are taken.

NOTE: 0-AOI-32-1, Loss of Control and Service Air Compressors
1-ARP-9-5B window 28 SCRAM PILOT AIR HEADER PRESS LOW

Which ONE of the following completes the statements below?

The Reactor is required to be scrammed when control air pressure first lowers below __ (1) __ in accordance with __ (2) __.

A. (1) 66 psig
(2) 1-ARP-9-5B window 28

B. (1) 66 psig
(2) 0-AOI-32-1

C. (1) 55 psig
(2) 1-ARP-9-5B window 28

D. (1) 55 psig
(2) 0-AOI-32-1

Q 81

The Unit 1 Reactor Steam Dome Pressure spikes to 1350 psig and the Reactor Scrams.

What is the **earliest** notification required In accordance with NPG-SPP-03.5 Regulatory Reporting Requirements?

[REFERENCE PROVIDED]

A. 1 hour report

B. 4 hour report

C. 8 hour report

D. 24 hour report

Q 82

A LOCA is in progress. The following conditions exist on Unit 2:

- Rx Pressure is 625 psig
- Rx Water Level on 2-LI-3-52 is (-) 242 inches
- No injection sources are available

Which ONE of the following completes the statements below?

(SEE GRAPH BELOW)

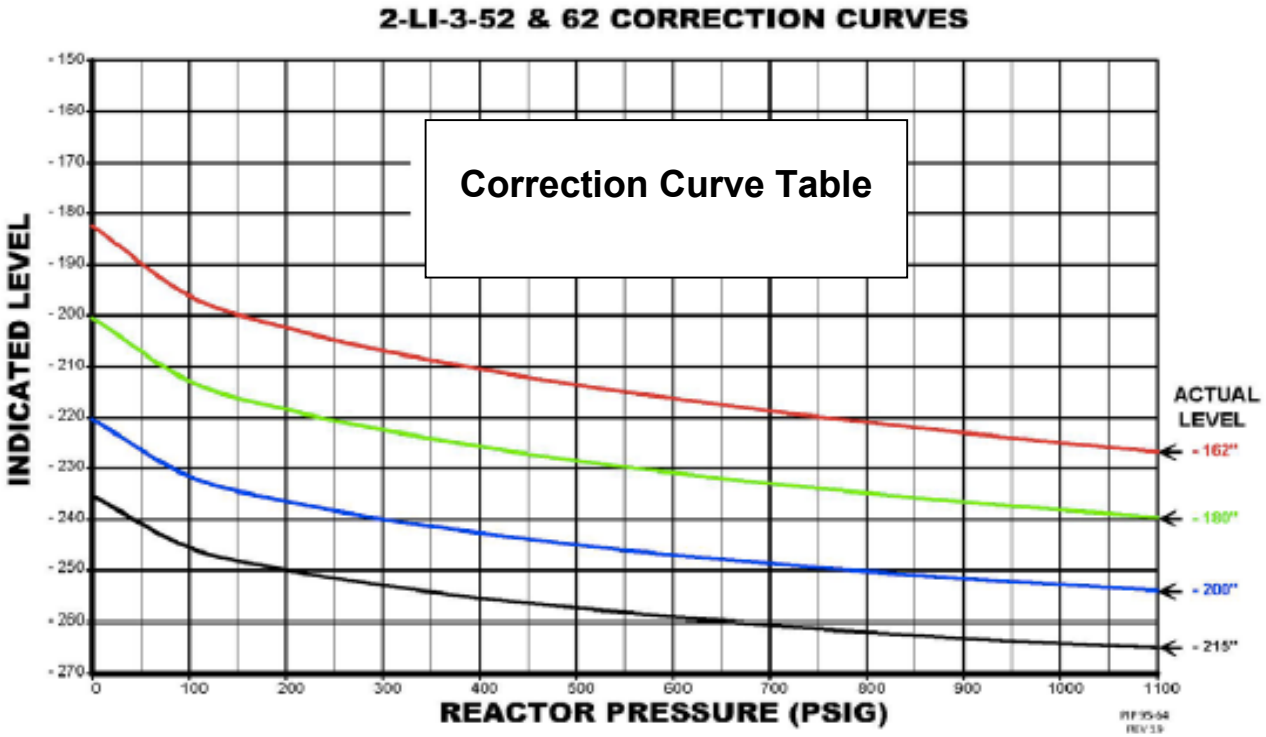
Actual water level is __ (1) __ and 2-EOI-1, RPV Control, requires entering __ (2) __.

A. (1) greater than (-) 200 inches
(2) Steam Cooling

B. (1) greater than (-) 200 inches
(2) Emergency Depressurization

C. (1) less than (-) 200 inches
(2) Steam Cooling

D. (1) less than (-) 200 inches
(2) Emergency Depressurization



Q 83

Unit 1 is operating at 20% power with two Condenser Circulating Water Pumps in service, when the following occurs:

- 1-XR-2-26, CONDENSATE recorder, indicates (-) 24.8 inches of Hg and is slowly degrading.

NOTE: 1-AOI-47-3, Loss of Condenser Vacuum
1-AOI-47-1, Unplanned Turbine Trip Below 30% Reactor Power
1-AOI-100-1, Reactor Scram

Which ONE of the following actions (if any) is/are required to be performed?

A. No action required at this time.

B. Enter 1-AOI-47-3 and TRIP the Main Turbine ONLY.

C. Enter 1-AOI-47-1 and VERIFY Main Turbine TRIPPED.

D. Enter 1-AOI-47-3 and 1-AOI-100-1, SCRAM the Reactor, THEN TRIP the Main Turbine.

Q 84

Unit 2 is operating at 100% power.

The Unit Operator reports:

- Drywell pressure is 2.20 psig
- Drywell Temperature is 162 °F
- Drywell pressure is slowly rising

NOTE: 2-AOI-64-1, Drywell Pressure and/or Temperature High or Excessive Leakage into Drywell
2-EOI Appendix-13, Emergency Venting Primary Containment

Which ONE of the following completes the statement below?

Vent the…

A. Drywell in accordance with 2-AOI-64-1.

B. Suppression Chamber in accordance with 2-AOI-64-1.

C. Drywell in accordance with 2-EOI Appendix-13.

D. Suppression Chamber in accordance with 2-EOI Appendix-13.

Q 85

Unit 2 is operating at 70% power performing a sequence exchange when a transient results in the following conditions:

Unit 2 is manually scrammed
2-9-3A window 27 MAIN STEAM LINE RADIATION HIGH-HIGH is in alarm
2-9-3A window 22 RX BLDG AREA RADIATION HIGH is in alarm
2-9-3F window 10 HPCI LEAK DETECTION TEMP HIGH is in alarm

At 1000 the UO reports:
- HPCI Room Temp 73-55A reading 160 °F
- HPCI Area Radiation 90-24A reading 500 mR/hr

At 1010 the UO reports:
- HPCI failed to isolate
- HPCI Room Temp 73-55A reading 180 °F
- HPCI Area Radiation 90-24A reading 700 mR/hr
- Drywell Radiation 2-RE-90-272A reading 200 R/hr

At 1020 the UO reports:
- HPCI Room Temp 73-55A reading 200 °F
- HPCI Area Radiation 90-24A reading 900 mR/hr
- Drywell Radiation 2-RE-90-272A reading 210 R/hr

ASSUME that the given parameter trends continue at the same rate.

At 1030, what will be the **highest** event classification required to be declared based on expected values?

[REFERENCE PROVIDED]

A. Notification of Unusual Event

B. Alert

C. Site Area Emergency

D. General Emergency

Q 86

All three Units are operating at 100% Power.

- The B Diesel Generator was tagged **at 0100** on 11/23/2015.
- An electrical fault occurred **at 0900** on 11/23/2015.
- See attached Unit 2 ICS screen shot

IAW the Conditions above and Technical Specifications, the most limiting completion time required for Standby Liquid Control (SLC) is _____.

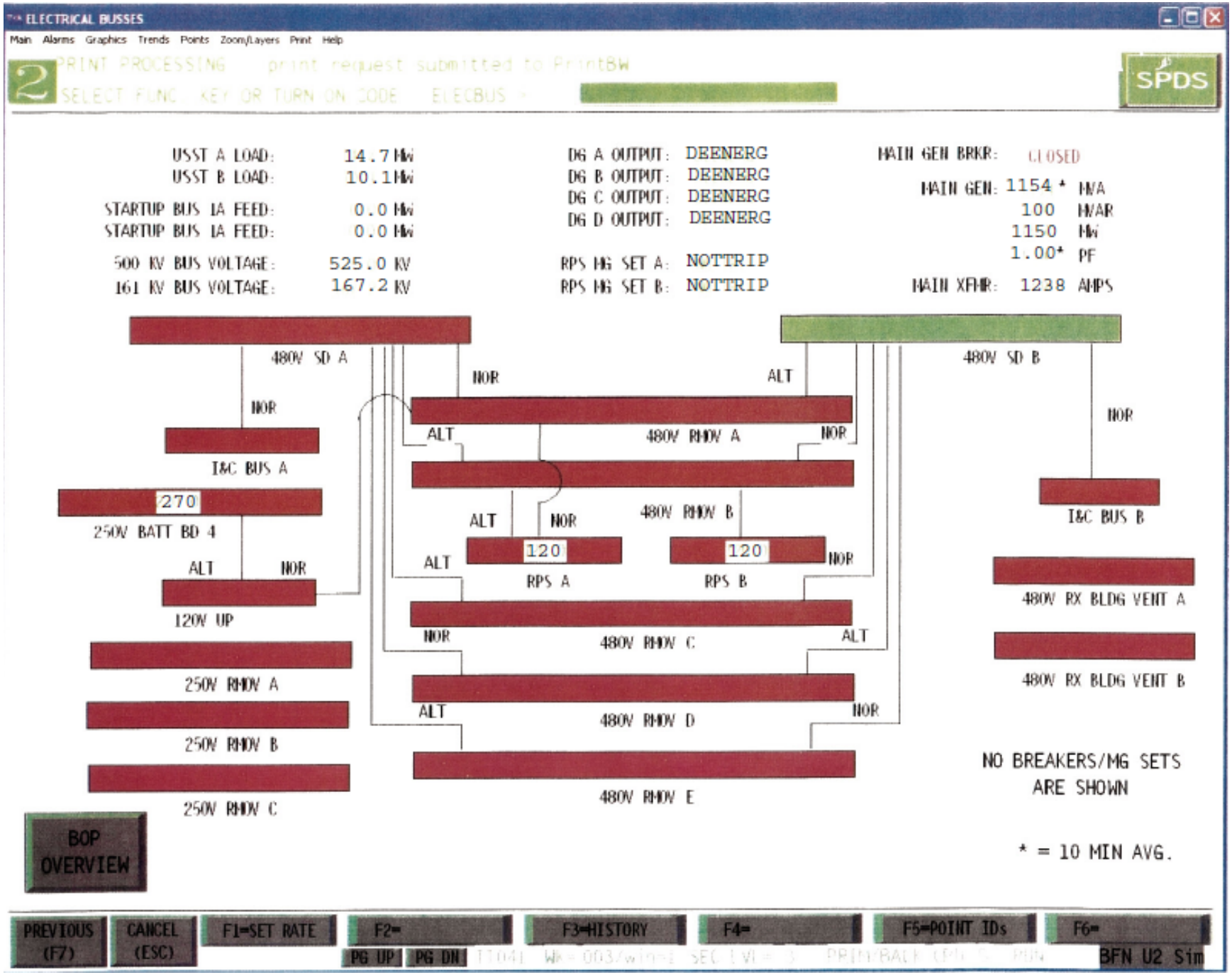
[REFERENCE PROVIDED]

A. 1700 on 11/23/2015

B. 2100 on 11/23/2015

C. 0500 on 11/24/2015

D. 0900 on 11/30/2015


ELECTRICAL BUSSES
Main Alarms Graphics Trends Points Zoom/Layers Print Help
PRINT PROCESSING print request submitted to PrintBW
SELECT FUNC KEY OR TURN ON CODE ELECBUS >
SPDS
USST A LOAD: 14.7 MW
USST B LOAD: 10.1 MW
STARTUP BUS 1A FEED: 0.0 MW
STARTUP BUS 1A FEED: 0.0 MW
500 KV BUS VOLTAGE: 525.0 KV
161 KV BUS VOLTAGE: 167.2 KV
DG A OUTPUT: DEENERG
DG B OUTPUT: DEENERG
DG C OUTPUT: DEENERG
DG D OUTPUT: DEENERG
RPS MG SET A: NOTTRIP
RPS MG SET B: NOTTRIP
MAIN GEN BRKR: CLOSED
MAIN GEN: 1154 * MVA
100 MVAR
1150 MW
1.00* PF
MAIN XFMR: 1238 AMPS
480V SD A
NOR
I&C BUS A
270
250V BATT BD 4
ALT
NOR
120V UP
250V RMOV A
250V RMOV B
250V RMOV C
NOR
ALT
480V RMOV A
ALT
NOR
480V RMOV B
120
RPS A
120
RPS B
ALT
NOR
480V RMOV C
ALT
NOR
480V RMOV D
ALT
NOR
480V RMOV E
480V SD B
NOR
I&C BUS B
480V RX BLDG VENT A
480V RX BLDG VENT B
NO BREAKERS/MG SETS
ARE SHOWN
* = 10 MIN AVG.
BOP
OVERVIEW
PREVIOUS (F7)
CANCEL (ESC)
F1=SET RATE
F2=
F3=HISTORY
F4=
F5=POINT IDs
F6=
PG UP
PG DN
BFN U2 Sim

Q 87

Unit 2 is performing a startup with the following conditions:

All IRMs are on Range 4 reading between 50 and 75 on the 125 scale except:

- IRM E is Bypassed
- IRM D is Bypassed

Subsequently:

At 0800, the IM Foreman reports that the following Acceptance Criteria was recorded during the most recent performance of 2-SR-3.3.1.1.9 (IRM C), IRM Channel C Calibration.

[8] **RECORD** As-Found high voltage trip point as read on DMM in Table 6.10-2.

Table 6.10-2

Desired Vdc	Min Vdc	As-Found Vdc	As-Left Vdc	Max Vdc
90.0	88.0	93.2	93.2	92.0

Which ONE of the following is the required action (if any) in accordance with Tech Spec 3.3.1.1?

[REFERENCE PROVIDED]

A. No action is required

B. Place the Channel or the Trip system in Trip by 1400

C. Place the Channel or the Trip system in Trip by 2000

D. Be in MODE 3 by 2000

Q 88

Unit 2 is in Mode 5 with a Core shuffle in progress.

- 0800 - SRM A was returned to service but remains INOP pending RTO of a Design Change.
- 0900 - SRM C started and continued to exhibit erratic operation causing intermittent Rod blocks and was bypassed.
- 0920 - SRM C **begins** indicating a steady reading consistent with SRM B and D.

At 0925 which ONE of the following completes the statements below?

In accordance with 2-OI-92, Source Range Monitors, SRM C __ (1) __.

In accordance with Tech Spec Bases and 0-GOI-100-3A, Refueling Operations (In-Vessel Operations), fuel loading can __ (2) __.

A. (1) shall remain bypassed
   (2) Not continue

B. (1) shall remain bypassed
   (2) continue in B and D quadrants

C. (1) can be returned to service
   (2) Not continue

D. (1) can be returned to service
   (2) continue in B and D quadrants

Q 89

All three units are operating at 100% power.

0-SR-3.8.1.1(A) Diesel Generator A Monthly Operability test is currently in progress With the Diesel Generator loaded.

Subsequently:

All off-site power is lost.

NOTE: Appendix 17A RHR System Operation Suppression Pool Cooling
0-AOI-57-1A Loss of Offsite Power (161 and 500KV)/Station Blackout

Which ONE of the following completes the statements below?

When off-site power is lost the A Diesel Generator output breaker will __ (1) __.

If Suppression Pool Cooling is required by EOI-2, __ (2) __ dictates the order in which pumps are to be started to place Suppression Pool Cooling in service.

A. (1) remain closed
(2) Appendix 17A

B. (1) trip and then reclose
(2) Appendix 17A

C. (1) remain closed
(2) 0-AOI-57-1A

D. (1) trip and then reclose
(2) 0-AOI-57-1A

Q 90

Unit 2 is operating at 100% power.

IMs are performing RPS And Rod Block High Water Level in Scram Discharge Tank Functional Test (2-LS-85-45A & 2-LS-85-45L)

- 0130 The IMs removed 2-LS-85-45A, West CRD SCRAM Discharge volume SCRAM Trip, from service in accordance with the surveillance.

- 0330 The IMs report that 2-LS-85-45A did Not change states during testing.

[REFERENCE PROVIDED]

Based on information provided which ONE of the following completes the statements below?

A valid high level in only the West CRD SCRAM Discharge volume __ (1) __ cause a full Reactor Scram.

In accordance with Tech Spec 3.3.1.1 the most limiting completion time is __ (2) __.

A. (1) will
   (2) 1530

B. (1) will
   (2) 1930

C. (1) will Not
   (2) 1530

D. (1) will Not
   (2) 1930

Q 91

Unit 1 Rx Startup following a refueling outage is in progress with the following conditions:

- Reactor Mode Switch is in the Startup position
- RCIC System Rated Flow at Low RPV Pressure surveillance is in progress
- Control Rod 18-23 has been declared SLOW
- Control Rod 38-23 has been declared SLOW
- Control Rod 22-23 is currently at position 48

Subsequently:

The UO reports receipt of the Control Rod 22-23 SCRAM Accumulator Light.

The Reactor building AUO reports:
The Nitrogen Charging Connection Cap cannot be re-installed and RT VLV TO PI-85-34, 2-RTV-085-229A (star valve) is currently open.

Which ONE of the following completes the statement below?

The Tech Spec required action(s) is (are) to _____.

[REFERENCE PROVIDED]

A. enter MODE 3 within 12 hours

B. declare Control Rod 22-23 SLOW or INOP only

C. fully insert Control Rod 22-23 and declare it INOP only

D. fully insert Control Rod 22-23, declare it INOP, and disarm it

Q 92

Unit 2 is at 100% power.

2-9-5A window 17 CONTROL ROD DRIVE UNIT TEMP HIGH is in alarm.

Which ONE of the following completes the statements below?

The setpoint for the CONTROL ROD DRIVE UNIT TEMP HIGH is __ (1) __.

If the Control Rod Drive temperature remains above the alarm setpoint after completion of the ARP actions the US is required to determine if Tech Spec section __ (2) __ is met.

A. (1) 240 °F
   (2) 3.1.3 Control Rod Operability

B. (1) 240 °F
   (2) 3.1.4 Control Rod Scram Time

C. (1) 350 °F
   (2) 3.1.3 Control Rod Operability

D. (1) 350 °F
   (2) 3.1.4 Control Rod Scram Time

Q 93

2-SR-3.6.1.3.5(94), TIP System PCIV Operability Test was last performed on 08/01/2015. The Frequency of this test is once every 92 days.

The TIP System PCIV Operability Test was scheduled on 10/24/2015 however the test could not be performed as scheduled.

In accordance with Tech Spec section SR 3.0.2, on what date will the TIP Ball valves become INOP if the surveillance is not performed?

A. 11/02/2015

B. 11/16/2015

C. 11/25/2015

D. 01/25/2016

Q 94

To maintain an active SRO license, an SRO must actively perform a minimum of _____ per calendar quarter in a position credited for watch-standing proficiency.

A. 40 hours only

B. a complete tour of the plant and 40 hours

C. 5-12 hour shifts only

D. a complete tour of the plant and 5-12 hour shifts

Q 95

Unit 2 is operating at 100% power.

- RHR SHUTDOWN COOLING SUCT OUTBD ISOL VLV, 2-FCV-74-47 is INOP
- At 0900: A Tech Spec LCO 3.0.3 is entered.
- At 0920: 2-GOI-100-12A, Unit Shutdown from Power Operation to Cold Shutdown and Reductions in Power During Power Operations, has been entered.
- At 0930: The OATC begins lowering power in accordance with the Urgent Load Reduction Reactivity Control Plan.

Which ONE of the following completes the statements below?

2-FCV-74-47 __ (1) __ required to be returned to operable status in accordance with Tech Spec 3.4.7 Residual Heat Removal (RHR) Shutdown Cooling System - Hot Shutdown prior to entering Mode 3.

In accordance with NPG-SPP-03.5 the NRC is required to be notified by __ (2) __.

A. (1) is
(2) 1320

B. (1) is
(2) 1330

C. (1) is Not
(2) 1320

D. (1) is Not
(2) 1330

Q 96

Which procedure authorizes the use of the eSOMs Off-Normal Equipment Alignment tracker and what is the specified time allowed before making an entry in the Off-Normal Equipment Alignment tracker?

A. NPG-SPP-10.1 System Status Control
   Before the end of the current shift

B. OPDP-1 Conduct of Operations
   Before the end of the current shift

C. NPG-SPP-10.1 System Status Control
   Within 72 hours

D. OPDP-1 Conduct of Operations
   Within 72 hours

Q 97

In accordance with LCO 3.0.4, Tech Spec Bases, what condition allows entry into a MODE or other specified condition in the Applicability with the LCO not met?

When the associated ACTIONS…

A. to be entered can be completed within the specified completion time.

B. permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time.

C. allow continued operation of equipment under administrative control.

D. permit not entering the required actions for supported equipment while the associated support equipment is inoperable.

Q 98

All three Units are operating at 100% power.

The Control Bay AUO reports that there is no Unit 1 & 2 Control Bay Supply Fan running. Attempts to start the 1A & 1B Control Bay Supply Fan has failed.

What action(s) if any are required by Tech Spec?

[REFERENCE PROVIDED]

A. No action is required.

B. Perform 0-SR-3.3.7.1.2 on 0-RM-90-259B once per 24 hours.

C. Verify alternate monitoring capability once per 12 hours.

D. Restore the affected CREV subsystem to Operable within 7 days.

Q 99

The Unit 1 Control Room receives a Turbine Building smoke alarm on the Fire Protection Display Panel.

A member of the plant security force calls the control room and reports smoke in the Turbine Building.

The Shift Manager has evaluated 0-SSI-001, Safe Shutdown Instructions.

NOTE: EPIP-17, Fire Emergency Procedure
0-SSI-26, Turbine Bldg, Radwaste Bldg
0-AOI-26-1, Fire Response

Based on the above conditions, which ONE of the following describes the actions required of the operating crew?

Enter 0-AOI-26-1 and __ (1) __.

In accordance with 0-AOI-26-1, announce the fire location over the PA and __ (2) __.

A. (1) 0-SSI-26
(2) **NOTIFY** the Clements Volunteer Fire Department by calling the Limestone County 911 Center

B. (1) 0-SSI-26
(2) **MONITOR** Control board indications for equipment failures or spurious operation

C. (1) EPIP-17
(2) **NOTIFY** the Clements Volunteer Fire Department by calling the Limestone County 911 Center

D. (1) EPIP-17
(2) **MONITOR** Control board indications for equipment failures or spurious operation

Q 100

Which ONE of the following completes both statements in accordance with EPIP-1 Emergency Classification Procedure?

IF an Emergency Action Level (EAL) for a higher classification was exceeded, but the present situation indicates a lower classification, THEN the higher classification __ (1) __ be **declared**.

IF an Emergency Action Level (EAL) was exceeded but has now been totally resolved (prior to event declaration), THEN the event __ (2) __ required to be **reported** to the NRC.

A. (1) should still
   (2) is

B. (1) should still
   (2) is Not

C. (1) should Not
   (2) is

D. (1) should Not
   (2) is Not

BROWNS FERRY NUCLEAR PLANT

Unit 0

Emergency Plan Implementing Procedure

EPIP-1

EMERGENCY CLASSIFICATION PROCEDURE

Revision 0051

Quality Related

Level of Use: Reference Use

Effective Date: 04/06/2015

Responsible Organization: Radiological Emergency Preparedness

PREPARED BY: Sally C. Taubuki

APPROVED BY: Steven M. Bono

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REACTOR
1.0

NOTES

1.1-U1/1.1-A1 Applicable when the Reactor Head is removed and the Reactor Cavity is flooded.

1.1-S1 Applicable in Mode 5 when the Reactor Head is installed.

1.1-G2 The reactor will remain subcritical under all conditions without boron when:

- Any 19 control rods are inserted to position 02, with all other control rods fully inserted.
- All control rods except one are inserted to or beyond position 00.
- Determined by Reactor Engineering.

CURVES/TABLES:

TABLE 1.1 - G2
MINIMUM STEAM COOLING PRESS (MSCP)

NUMBER OF OPEN MSRVs	MSCP (PSIG)
6 or More	190
5	230
4	290

WATER LEVEL

Description	Description	
1.1-U1 **NOTE**	**1.1-U2**	**UNUSUAL EVENT**
Uncontrolled water level decrease in Reactor Cavity with irradiated fuel assemblies expected to remain covered by water. OPERATING CONDITION: Mode 5	Uncontrolled water level decrease in Spent Fuel Pool with irradiated fuel assemblies expected to remain covered by water. OPERATING CONDITION ALL	
1.1-A1 **NOTE**	**1.1-A2**	**ALERT**
Uncontrolled water level decrease in Reactor Cavity expected to result in irradiated fuel assemblies being uncovered. OPERATING CONDITION: Mode 5	Uncontrolled water level decrease in Spent Fuel Storage Pool expected to result in irradiated fuel assemblies being uncovered. OPERATING CONDITION: ALL	
1.1-S1 **NOTE**	**1.1-S2**	**SITE EMERGENCY**
Reactor water level can NOT be maintained above -162 inches. (TAF) OPERATING CONDITION: ALL	Reactor water level can NOT be determined. OPERATING CONDITION: Mode 1 or 2 or 3	
1.1-G1	**1.1-G2** **NOTE** **TABLE**	**GENERAL EMERGENCY**
Reactor water level can NOT be restored and maintained above -180 inches. OPERATING CONDITION: Mode 1 or 2 or 3	Reactor water level can NOT be determined **AND** Either of the following exists: • The reactor will remain subcritical without boron under all conditions, and ➢ Less than 4 MSRVs can be opened, or ➢ Reactor pressure can NOT be restored and maintained above Suppression Chamber pressure by at least 70 psi. • It has NOT been determined that the reactor will remain subcritical without boron under all conditions and unable to restore and maintain MSCP in Table 1.1-G2. OPERATING CONDITION: Mode 1 or 2 or 3	

NOTES

1.2 Subcritical is defined as reactor power below the heating range and not trending upward.

CURVES/TABLES:

SCRAM FAILURE	REACTOR COOLANT ACTIVITY	
Description	**Description**	
	1.3-U	**UNUSUAL EVENT**
	Reactor coolant activity exceeds 26 μCi/gm dose equivalent I-131 (Technical Specification Limits) as determined by chemistry sample. OPERATING CONDITION ALL	
1.2-A **NOTE** Failure of RPS automatic scram functions to bring the reactor subcritical **AND** Manual scram or ARI (automatic or manual) was successful. OPERATING CONDITION: Mode 1 or 2	**1.3-A** Reactor coolant activity exceeds 300 μCi/gm dose equivalent Iodine-131 as determined by chemistry sample. OPERATING CONDITION: Mode 1 or 2 or 3	**ALERT**
1.2-S **NOTE** Failure of automatic scram, manual scram, and ARI to bring the reactor subcritical. OPERATING CONDITION: Mode 1 or 2		**SITE EMERGENCY**
1.2-G **CURVE** Failure of automatic scram, manual scram, and ARI. Reactor power is above 3% **AND** Either of the following conditions exists: • Suppression Pool temp exceeds HCTL. Refer to Curve 1.2-G. • Reactor water level can NOT be restored and maintained at or above -180 inches. OPERATING CONDITION: Mode 1 or 2		**GENERAL EMERGENCY**

NOTES

CURVES/TABLES:

MSL / OFFGAS RADIATION	LOSS OF DECAY HEAT REMOVAL	
Description	**Description**	
1.4-U Valid MAIN STEAM LINE RADIATION HIGH-HIGH alarm, 1, 2, or 3-RA-90-135C **OR** Valid OG PRETREATMENT RADIATION HIGH alarm, 1, 2, or 3-RA-90-157A. OPERATING CONDITION: Mode 1 or 2 or 3		UNUSUAL EVENT
	1.5-A Reactor moderator temperature can NOT be maintained below 212^0 F whenever Technical Specifications require Mode 4 conditions or during operations in Mode 5. OPERATING CONDITION: Mode 4 or 5	ALERT
	1.5-S **CURVE** Suppression Pool temperature, level and RPV pressure can NOT be maintained in the safe area of Curve 1.5-S. OPERATING CONDITION: Mode 1 or 2 or 3	SITE EMERGENCY
		GENERAL EMERGENCY

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PRIMARY CONTAINMENT 2.0

NOTES

CURVES/TABLES:

TABLE 2.1-A INDICATIONS OF PRIMARY SYSTEM LEAKAGE INTO PRIMARY CONTAINMENT
Primary Containment Pressure High Alarm
Drywell Floor Drain Sump Pump Excessive Operation
Drywell CAM Activity Increasing
Drywell Temperature High Alarm
Chemistry Sample Radionuclide Comparison To Reactor Water

PRIMARY CONTAINMENT PRESSURE	PRIMARY CONTAINMENT HYDROGEN	
Description	**Description**	
		UNUSUAL EVENT
2.1-A **TABLE** Drywell pressure at or above 2.45 psig **AND** Indication of Primary System leakage into Primary Containment. Refer to Table 2.1-A. OPERATING CONDITION: Mode 1 or 2 or 3		ALERT
2.1-S **CURVE** Suppression Chamber pressure can NOT be maintained in the safe area of Curve 2.1-S. OPERATING CONDITION: Mode 1 or 2 or 3	**2.2-S** Drywell or Suppression Chamber hydrogen concentration at or above 4% **AND** Drywell or Suppression Chamber oxygen concentration at or above 5%. OPERATING CONDITION: Mode 1 or 2 or 3	SITE EMERGENCY
2.1-G **CURVE** Drywell pressure can NOT be maintained in the safe area of Curve 2.1-G. OPERATING CONDITION: Mode 1 or 2 or 3	**2.2-G** Drywell or Suppression Chamber hydrogen concentration at or above 6% **AND** Drywell or Suppression Chamber oxygen concentration at or above 5%. OPERATING CONDITION: Mode 1 or 2 or 3	GENERAL EMERGENCY

NOTES

CURVES/TABLES:

TABLE 2.3-A/2.3-S2
DRYWELL RADIATION LEVELS
WITH RCS BARRIER INTACT INSIDE PRIMARY CONTAINMENT

UNIT 1		UNIT 2		UNIT 3	
RAD MONITOR	**R/HR**	**RAD MONITOR**	**R/HR**	**RAD MONITOR**	**R/HR**
1-RE-90-272A	196	2-RE-90-272A	642	3-RE-90-272A	196
1-RE-90-273A	297	2-RE-90-273A	297	3-RE-90-273A	297

TABLE 2.3-S1/2.3-G2
DRYWELL RADIATION LEVELS
WITH RCS BARRIER NOT INTACT INSIDE PRIMARY CONTAINMENT

UNIT 1		UNIT 2		UNIT 3	
RAD MONITOR	**R/HR**	**RAD MONITOR**	**R/HR**	**RAD MONITOR**	**R/HR**
1-RE-90-272A	2981	2-RE-90-272A	2263	3-RE-90-272A	2981
1-RE-90-273A	2960	2-RE-90-273A	2960	3-RE-90-273A	2960

TABLE 2.3-G1
DRYWELL RADIATION LEVELS
WITH RCS BARRIER NOT INTACT INSIDE PRIMARY CONTAINMENT

UNIT 1		UNIT 2		UNIT 3	
RAD MONITOR	**R/HR**	**RAD MONITOR**	**R/HR**	**RAD MONITOR**	**R/HR**
1-RE-90-272A	90091	2-RE-90-272A	68405	3-RE-90-272A	90091
1-RE-90-273A	89450	2-RE-90-273A	89450	3-RE-90-273A	89450

TABLE 2.3/2.5-U **INDICATIONS OF LOSS OF PRIMARY CONTAINMENT**
Unexplained Loss Of Containment Pressure
Exceeding 1, 2, or 3-SI-4.7.A.2.a Limits
Inability To Isolate Any Line Exiting Containment When Isolation Is Required
Venting Irrespective Of Offsite Release Rates Per EOIs/SAMGs

DRYWELL RADIATION

Description	Description	
		UNUSUAL EVENT
2.3-A TABLE US Drywell radiation levels at or above the values listed in Table 2.3-A/2.3-S2, with the RCS barrier intact inside Primary Containment. OPERATING CONDITION: Mode 1 or 2 or 3		ALERT
2.3-S1 TABLE US Drywell radiation levels at or above the values listed in Table 2.3-S1/2.3-G2 with the RCS barrier NOT intact inside Primary Containment. OPERATING CONDITION: Mode 1 or 2 or 3	**2.3-S2** TABLE US Drywell radiation levels at or above the values listed in Table 2.3-A/2.3-S2, with the RCS barrier intact inside Primary Containment, **AND** Either of the following exists: • Indications of loss of Primary Containment. Refer to Table 2.3/2.5-U. • Primary Containment integrity can NOT be maintained. OPERATING CONDITION: Mode 1 or 2 or 3	SITE EMERGENCY
2.3-G1 TABLE US Drywell radiation levels at or above the values listed in Table 2.3-G1 with the RCS barrier NOT intact inside Primary Containment. OPERATING CONDITION: Mode 1 or 2 or 3	**2.3-G2** TABLE US Drywell radiation levels at or above the values listed in Table 2.3-S1/2.3-G2 with the RCS barrier NOT intact inside Primary Containment, **AND** Either of the following exists: • Indications of loss of Primary Containment. Refer to Table 2.3/2.5-U. • Primary Containment integrity can NOT be maintained. OPERATING CONDITION: Mode 1 or 2 or 3	GENERAL EMERGENCY

NOTES

CURVES/TABLES:

TABLE 2.3/2.5-U INDICATIONS OF LOSS OF PRIMARY CONTAINMENT
Unexplained Loss Of Containment Pressure
Exceeding 1, 2, or 3-SI-4.7.A.2.a Limits
Inability To Isolate Any Line Exiting Containment When Isolation Is Required
Venting Irrespective Of Offsite Release Rates Per EOIs/SAMGs

DRYWELL INTERNAL LEAKAGE	LOSS OF PRIMARY CONTAINMENT	
Description	**Description**	
2.4-U Drywell unidentified leakage exceeds 10 gpm **OR** Drywell identified leakage exceeds 40 gpm. OPERATING CONDITION: Mode 1 or 2 or 3	**2.5-U** **TABLE** Inability to maintain Primary Containment pressure boundary. Refer to Table 2.3/2.5-U. OPERATING CONDITION: Mode 1 or 2 or 3	UNUSUAL EVENT
2.4-A Drywell unidentified leakage exceeds 50 gpm. OPERATING CONDITION: Mode 1 or 2 or 3		ALERT
		SITE EMERGENCY
		GENERAL EMERGENCY

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SECONDARY CONTAINMENT 3.0

NOTES

CURVES/TABLES:

TABLE 3.1
MAXIMUM SAFE OPERATING AREA TEMPERATURE LIMITS

AREA	APPLICABLE PANEL 9-21 TEMPERATURE ELEMENTS (UNLESS OTHERWISE NOTED)	MAX SAFE OPERATING VALUE ^{0}F		
		UNIT 1	UNIT 2	UNIT 3
RHR A/C Pump Room	74-95A	215	150	155
RHR B/D Pump Room	74-95B	150	210	215
HPCI Turbine Area	73-55A	275	270	270
CS A/C Pump and RCIC Turbine Area	71-41A	190	190	190
RCIC Steam Supply Area	71-41B, 41C, 41D	195	200	250
HPCI Steam Supply Area	73-55B, 55C, 55D	245	240	240
RHR A/C Pump Supply Area	74-95H	245	240	240
RHR B/D Pump Supply Area	74-95G	190	240	240
Main Steam Line Leak Detection High	(XA-55-3D-24) Panel 9-3 TIS-1-60A	315	315	315
RHR Valve Room	74-95E	175	170	175
RWCU Isol Logic Channel A/B Temp High	(XA-55-5B-32/33) Panel 9-5 69-835A, B, C, D Aux Inst Room	175	170	175
RWCU Outbd Isol Vlv Area	69-29F	220	220	220
RWCU Hx Area	69-29G	220	220	220
RWCU Hx Exh Duct	69-29H	220	220	220
RWCU Recirc Pump A Area	69-29D	215	215	215
RWCU Recirc Pump B Area	69-29E	215	215	215
RHR A/C Hx Room	74-95C	210	195	200
RHR B/D Hx Room	74-95D	210	195	200
FPC Hx Area	74-95F	160	155	155

TABLE 3.1-G/3.2-G
INDICATIONS OF POTENTIAL OR SIGNIFICANT FUEL CLADDING FAILURE
WITH RCS BARRIER INTACT INSIDE PRIMARY CONTAINMENT

UNIT 1 DRYWELL RADIATION		UNIT 2 DRYWELL RADIATION		UNIT 3 DRYWELL RADIATION	
1-RE-90-272A	$\geq$ 196 R/HR	2-RE-90-272A	$\geq$ 642 R/HR	3-RE-90-272A	$\geq$ 196 R/HR
1-RE-90-273A	$\geq$ 297 R/HR	2-RE-90-273A	$\geq$ 297 R/HR	3-RE-90-273A	$\geq$ 297 R/HR
Reactor Coolant Activity $\geq$ 300 µCi/gm Dose Equivalent Iodine 131		Reactor Coolant Activity $\geq$ 300 µCi/gm Dose Equivalent Iodine 131		Reactor Coolant Activity $\geq$ 300 µCi/gm Dose Equivalent Iodine 131	

SECONDARY CONTAINMENT TEMPERATURE

Description	
	UNUSUAL EVENT
	ALERT
3.1-S **TABLE** **US** An unisolable Primary System leak is discharging into Secondary Containment **AND** Any area temperature exceeds the Maximum Safe Operating Temperature limit listed in Table 3.1. OPERATING CONDITION: Mode 1 or 2 or 3	SITE EMERGENCY
3.1-G **TABLE** **US** An unisolable Primary System leak is discharging into Secondary Containment **AND** Any area temperature exceeds the Maximum Safe Operating Temperature limit listed in Table 3.1 **AND** Any indication of potential or significant fuel cladding failure exists. Refer to Table 3.1-G/3.2-G with RCS Barrier intact inside Primary Containment. OPERATING CONDITION Mode 1 or 2 or 3	GENERAL EMERGENCY

NOTES

CURVES/TABLES:

TABLE 3.2
MAXIMUM SAFE OPERATING AREA RADIATION LIMITS

AREA	RAD MONITOR	MAX SAFE VALUE MR/HR		
		UNIT 1	UNIT 2	UNIT 3
RHR West Room	90-25A	1000	1000	1000
RHR East Room	90-28A	1000	1000	1000
HPCI Room	90-24A	1000	1000	1000
CS/RCIC Room	90-26A	1000	1000	1000
Core Spray Room	90-27A	1000	1000	1000
Suppr Pool Area	90-29A	1000	1000	1000
CRD-HCU West Area	90-20A	1000	1000	1000
CRD-HCU East Area	90-21A	1000	1000	1000
TIP Drive Area	90-23A	1000	1000	1000
North RWCU System Area	90-13A	1000	1000	1000
South RWCU System Area	90-14A	1000	1000	1000
RWCU System Area	90-9A	1000	1000	1000
MG Set Area	90-4A	1000	1000	1000
Fuel Pool Area	90-1A	1000	1000	1000
Service Flr Area	90-2A	1000	1000	1000
New Fuel Storage	90-3A	1000	N/A	N/A

TABLE 3.1-G/3.2-G
INDICATIONS OF POTENTIAL OR SIGNIFICANT FUEL CLADDING FAILURE
WITH RCS BARRIER INTACT INSIDE PRIMARY CONTAINMENT

UNIT 1 DRYWELL RADIATION		UNIT 2 DRYWELL RADIATION		UNIT 3 DRYWELL RADIATION	
1-RE-90-272A	$\geq$ 196 R/HR	2-RE-90-272A	$\geq$ 642 R/HR	3-RE-90-272A	$\geq$ 196 R/HR
1-RE-90-273A	$\geq$ 297 R/HR	2-RE-90-273A	$\geq$ 297 R/HR	3-RE-90-273A	$\geq$ 297 R/HR
Reactor Coolant Activity $\geq$ 300 µCi/gm Dose Equivalent Iodine 131		Reactor Coolant Activity $\geq$ 300 µCi/gm Dose Equivalent Iodine 131		Reactor Coolant Activity $\geq$ 300 µCi/gm Dose Equivalent Iodine 131	

SECONDARY CONTAINMENT RADIATION

Description	
	UNUSUAL EVENT
3.2-A Any of the following high radiation alarms on Panel 9-3: • 1, 2, or 3-RA-90-1A, Fuel Pool Floor Alarm • 1, 2, or 3-RA-90-250A, Reactor, Turbine, Refuel Exhaust • 1, 2, or 3-RA-90-142A, Reactor Refuel Exhaust • 1, 2, or 3-RA-90-140A, Refueling Zone Exhaust **AND** Confirmation by Refuel Floor personnel that irradiated fuel damage may have occurred. OPERATING CONDITION: ALL	ALERT
3.2-S **TABLE US** An unisolable Primary System leak is discharging into Secondary Containment **AND** Any area radiation level at or above the Maximum Safe Operating Area radiation limit listed in Table 3.2. OPERATING CONDITION: Mode 1 or 2 or 3	SITE EMERGENCY
3.2-G **TABLE US** An unisolable Primary System leak is discharging into Secondary Containment **AND** Any area radiation level at or above the Maximum Safe Operating Area radiation limit listed in Table 3.2. **AND** Any indication of potential or significant fuel cladding failure exists. Refer to Table 3.1-G/3.2-G with RCS Barrier intact inside Primary Containment. OPERATING CONDITION Mode 1 or 2 or 3	GENERAL EMERGENCY

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RADIOACTIVITY RELEASES 4.0

NOTES

4.1-U Prior to making this emergency classification based upon the WRGERMS indication, assess the release by either of the following:

1. Actual field measurements exceed the limits in table 4.1-U
2. 0-SI 4.8.B.1.a.1 release fraction exceeds 2.0

If neither assessment can be conducted within 60 minutes then the declaration must be made on the valid WRGERMS reading.

4.1-A Prior to making this emergency classification based upon the WRGERMS indication, assess the release by either of the following:

1. Actual field measurements exceed the limits in table 4.1-A
2. 0-SI 4.8.B.1.a.1 release fraction exceeds 200

If neither assessment can be conducted within 15 minutes then the declaration must be made on the valid WRGERMS reading.

4.1-S Prior to making this emergency classification based upon the gaseous release rate indication, assess the release by either of the following methods:

1. Actual field measurements exceed the limits in table 4.1-S.
2. Projected or actual dose assessments exceed 100 mrem TEDE or 500 mrem CDE.

If neither assessment can be conducted within 15 minutes then the declaration must be made based on the valid WRGERMS reading.

4.1-G Prior to making this emergency classification based upon the gaseous release rate indication, assess the release by either of the following methods:

1. Actual field measurements exceed the limits in table 4.1-G.
2. Projected or actual dose assessments exceed 1000 mrem TEDE or 5000 mrem CDE.

If neither assessment can be conducted within 15 minutes then the declaration must be made based on the valid WRGERMS reading.

CURVES/TABLES:

Table 4.1-U
RELEASE LIMITS FOR UNUSUAL EVENT

TYPE	MONITORING METHOD	LIMIT	DURATION
Gaseous Release Rate	Stack Noble Gas (WRGERMS)	2.88 X 10^{7} µCi/sec	1 Hour
Gaseous Release Rate	0-SI 4.8.B.1.a.1	Release Fraction 2.0	1 Hour
Site Boundary Radiation Reading	Field Assessment Team	0.10 MREM/HR Gamma	1 Hour

Table 4.1-A
RELEASE LIMITS FOR ALERT

TYPE	MONITORING METHOD	LIMIT	DURATION
Gaseous Release Rate	Stack Noble Gas (WRGERMS)	2.88 X 10^{9} µCi/sec	15 Minutes
Gaseous Release Rate	0-SI 4.8.B.1.a.1	Release Fraction 200	15 Minutes
Site Boundary Radiation Reading	Field Assessment Team	10 MREM/HR Gamma	15 Minutes

Table 4.1-S
RELEASE LIMITS FOR SITE AREA EMERGENCY

TYPE	MONITORING METHOD	LIMIT	DURATION
Gaseous Release Rate	Stack Noble Gas (WRGERMS)	5.9 X 10^{9} µCi/sec	15 Minutes
Site Boundary Radiation Reading	Field Assessment Team	100 MREM/HR Gamma	1 Hour
Site Boundary Iodine-131	Field Assessment Team	3.9 X 10^{-7} µCI /cm^{3}	1 Hour

Table 4.1-G
RELEASE LIMITS FOR GENERAL EMERGENCY

TYPE	MONITORING METHOD	LIMIT	DURATION
Gaseous Release Rate	Stack Noble Gas (WRGERMS)	5.9 X 10^{10} µCi/sec	15 Minutes
Site Boundary Radiation Reading	Field Assessment Team	1000 MREM/HR Gamma	1 Hour
Site Boundary Iodine-131	Field Assessment Team	3.9 X 10^{-6} µCI / cm^{3}	1 Hour

GASEOUS EFFLUENT

Description	
4.1-U **NOTE** **TABLE** Gaseous release exceeds ANY limit and duration in Table 4.1-U. OPERATING CONDITION: ALL	UNUSUAL EVENT
4.1-A **NOTE** **TABLE** Gaseous release exceeds ANY limit and duration in Table 4.1-A. OPERATING CONDITION: ALL	ALERT
4.1-S **NOTE** **TABLE** EITHER of the following conditions exists: • Gaseous release exceeds or is expected to exceed ANY limit and duration in Table 4.1-S. • Dose assessment indicates actual or projected dose consequences above 100 mrem TEDE or 500 mrem thyroid CDE. OPERATING CONDITION: ALL	SITE EMERGENCY
4.1-G **NOTE** **TABLE** EITHER of the following conditions exists: • Gaseous release exceeds or is expected to exceed ANY limit and duration in Table 4.1-G. • Dose assessment indicates actual or projected dose consequences above 1000 mrem TEDE or 5000 mrem thyroid CDE. OPERATING CONDITION ALL	GENERAL EMERGENCY

NOTES

CURVES/TABLES:

MAIN STEAM LINE BREAK	LIQUID EFFLUENT	
Description	**Description**	
4.2-U Main Steam Line break outside Primary Containment with isolation. OPERATING CONDITION: Mode 1 or 2 or 3	**4.3-U** Liquid release rate exceeds 20 times ECL as determined by chemistry sample **AND** Release duration exceeds or will exceed 60 minutes. OPERATING CONDITION: ALL	**UNUSUAL EVENT**
	4.3-A Liquid release rate exceeds 2000 times ECL as determined by chemistry sample **AND** Release duration exceeds or will exceed 15 minutes. OPERATING CONDITION: ALL	**ALERT**
4.2-S Unisolable Main Steam Line break outside Primary Containment. OPERATING CONDITION: Mode 1 or 2 or 3		**SITE EMERGENCY**
		GENERAL EMERGENCY

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LOSS OF POWER
5.0

NOTES

5.1-U Loss of normal and alternate supply voltage implies inability to restore voltage from any qualified source to normal or alternate feeder for at least one of the unit specific boards within 15 minutes. At least two boards must be energized from Diesel power to meet this classification. If only one board can be energized and that board has only one source of power then refer to EAL 5.1-A1 or 5.1-A2.

5.1-A1 Only one source of power (Diesel or Offsite) is available to any one of the listed unit specific 4KV Shutdown Boards. No power is available to the three remaining boards.

5.1-A2 Loss of voltage to all unit specific 4KV Shutdown Boards applies to those boards which normally supply emergency AC power to the affected unit only. Determination of the event classification depends on the affected unit operating mode. For units in operation 5.1-S would apply.

5.1-S Loss of voltage to all unit specific 4KV Shutdown Boards applies to those boards which normally supply emergency AC power to the affected unit only. Determination of the event classification depends on the affected unit operating mode. For units in Shutdown or Refuel 5.1-A2 would apply.

5.1-G Loss of voltage to all unit specific 4KV Shutdown Boards applies to those boards which normally supply emergency AC power to the affected unit only.

CURVES/TABLES:

Table 5.1
UNIT 4KV SHUTDOWN BOARD APPLICABILITY

APPLICABLE UNIT	APPLICABLE 4KV SHUTDOWN BOARDS
UNIT 1	A, B, C, and D
UNIT 2	A, B, C, and D
UNIT 3	3A, 3B, 3C, and 3D

LOSS OF AC POWER

Description	Description	
5.1-U **NOTE** **TABLE** **US** Loss of normal and alternate supply voltage to ALL unit specific 4KV shutdown boards from Table 5.1 for greater than 15 minutes **AND** At least two Diesel Generators supplying power to unit specific 4KV shutdown boards listing in Table 5.1. OPERATING CONDITION: ALL		**UNUSUAL EVENT**
5.1-A1 **NOTE** **TABLE** **US** Loss of voltage to ANY THREE unit specific 4KV shutdown boards from Table 5.1 for greater than 15 minutes **AND** Only ONE source of power available to the remaining board. OPERATING CONDITION: Mode 1 or 2 or 3	**5.1-A2** **NOTE** **TABLE** **US** Loss of voltage to ALL unit specific 4KV shutdown boards from Table 5.1 for greater than 15 minutes. OPERATING CONDITION: Mode 4 or 5 or Defueled	**ALERT**
5.1-S **NOTE** **TABLE** **US** Loss of voltage to ALL unit specific 4KV shutdown boards from Table 5.1 for greater than 15 minutes. OPERATING CONDITION: Mode 1 or 2 or 3		**SITE EMERGENCY**
5.1-G **NOTE** **TABLE** **US** Loss of voltage to ALL unit specific 4KV shutdown boards from Table 5.1 **AND** Either of the following conditions exists; • Restoration of at least one 4KV shutdown board is NOT likely within three hours. • Adequate core cooling can NOT be assured. OPERATING CONDITION: Mode 1 or 2 or 3		**GENERAL EMERGENCY**

NOTES

5.2 250V DC power voltage below 248 volts constitutes a loss of DC power to the affected board. The voltage readings may be obtained at the 250V Shutdown Battery Board (or the 250V Plant Battery Board) that is feeding the affected board.

CURVES/TABLES:

Table 5.2-U
UNIT 4KV SHUTDOWN BOARD APPLICABILITY

APPLICABLE UNIT	APPLICABLE 4KV SHUTDOWN BOARDS
UNIT 1	A, B, C, AND D
UNIT 2	A, B, C, AND D
UNIT 3	3A, 3B, 3C, AND 3D

Table 5.2-S
CRITICAL DC POWER AND ESSENTIAL SYSTEMS

COMBINATION	LOSS OF CRITICAL 250V DC POWER (Unit Specific Unless Otherwise Noted)	POTENTIALLY RESULTS IN
I	Control Power for 4KV Unit Boards A, B, and C **AND** Control Power for 480V Unit Boards A and B **AND** Power for Panel 9-9 Cabinet 1	Loss of Main Condenser **AND** Loss of Both EHC Pumps **AND** Loss of All Reactor Feed Pumps
II	Power for 250V DC RMOV Board A	Loss of HPCI
III	Power for 250V DC RMOV Board C	Loss of RCIC
IV	Power for 250V DC RMOV Boards A, B, and C **AND** Control Power for 4KV Shutdown Boards A, B, C, and D (4KV Shutdown Boards 3A, 3B, 3C, and 3D for Unit 3)	Less than 4 MSRVs **AND** Loss of All RHR Pumps And Core Spray Pumps

LOSS OF 250V DC POWER

Description	Description	
5.2-U **NOTE** **TABLE** **US**		UNUSUAL EVENT
Unplanned loss of 250V DC control power to ALL unit specific 4KV shutdown boards from Table 5.2-U for greater than 15 minutes **OR** Unplanned loss of 250V DC control power to unit specific 480V shutdown boards A and B for greater than 15 minutes. OPERATING CONDITION: Modes 4 or 5		
		ALERT
5.2-S **NOTE** **TABLE** **US**		SITE EMERGENCY
Loss of 250V DC power to ALL combinations (I, II, III, and IV) of essential systems from Table 5.2-S for greater than 15 minutes. OPERATING CONDITION: Mode 1 or 2 or 3		
		GENERAL EMERGENCY

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HAZARDS
6.0

NOTES

CURVES/TABLES:

RADIOLOGICAL

Description	Description	
6.1-U Valid, unexpected increase of ANY in-plant ARM reading to 1000 mrem/hr (except TIP room). OPERATING CONDITION: ALL		UNUSUAL EVENT
6.1-A1 Valid, unexpected increase of ANY in-plant ARM reading to 1000 mrem/hr (except TIP room). **AND** Personnel required in the affected area(s). OPERATING CONDITION: ALL	**6.1-A2** Control Room radiation levels greater than 15 mrem/hr. OPERATING CONDITION: ALL	ALERT
		SITE EMERGENCY
		GENERAL EMERGENCY

NOTES

CURVES/TABLES:

CONTROL ROOM EVACUATION	TURBINE FAILURE	
Description	**Description**	
	6.3-U	UNUSUAL EVENT
	Turbine failure resulting in casing penetration **OR** Significant damage to turbine or generator seals during operation. OPERATING CONDITION: Mode 1, or 2	
6.2-A	**6.3-A**	ALERT
Control Room Abandonment from entry into 1, 2, or 3-AOI-100-2 or 0-SSI-16 for ANY Unit Control Room. OPERATING CONDITION: ALL	Turbine failure resulting in visible structural damage to or visible penetration of ANY of the following structures from missles: ◆Reactor Building ◆Diesel Generator Building ◆Intake Structure ◆Control Bay OPERATING CONDITION: Mode 1 or 2	
6.2-S		SITE EMERGENCY
Control Room Abandonment from entry into 1, 2, or 3-AOI-100-2 or 0-SSI-16 for ANY Unit Control Room **AND** Control of reactor water level, reactor pressure, and reactor power (for Modes 1, or 2, or 3) or decay heat removal (for Modes 4, or 5) per 1, 2, or 3-AOI-100-2 or 0-SSI-16 as applicable, can NOT be established within 20 minutes after evacuation is initiated. OPERATING CONDITION: ALL		
		GENERAL EMERGENCY

NOTES

CURVES/TABLES:

Table 6.4-U1 APPLICABLE PLANT AREA
Reactor Building
Refuel Floor
4KV Shutdown Board Rooms
4KV Shutdown Battery Board Rooms
480V Shutdown Board Rooms
RMOV Board 3A and 3B Rooms
4KV Bus Tie Board Room
Control Bay Elevation 593', 606', And 617'
Diesel Generator Buildings (All Elevations)
Turbine Building (All Elevations)
Intake Pumping Station (All Elevations)
Radwaste Building (All Elevations)
Cable Tunnel (Intake To Turbine Building)
Standby Gas Treatment Building

Table 6.4-A APPLICABLE PLANT AREA
Reactor Building
Refuel Floor
4KV Shutdown Board Rooms
4KV Shutdown Battery Board Rooms
480V Shutdown Board Rooms
RMOV Board 3A and 3B Rooms
4KV Bus Tie Board Room
Control Bay Elevation 593', 606', And 617'
Diesel Generator Buildings (All Elevations)
Intake Pumping Station (All Elevations)
Cable Tunnel (Intake To Turbine Building)
Standby Gas Treatment Building

FIRE / EXPLOSION

Description	Description	
6.4-U1 **TABLE**	**6.4-U2**	**UNUSUAL EVENT**
Confirmed fire in ANY plant area listed in Table 6.4-U1 **AND** NOT extinguished within 15 minutes. OPERATING CONDITION: ALL	Unanticipated explosion within the protected area resulting in visible damage to ANY permanent structure or equipment. OPERATING CONDITION: ALL	
6.4-A **TABLE**		**ALERT**
Fire or explosion in ANY plant area listed in Table 6.4-A affecting safety system performance **OR** Fire or explosion causing visible damage to permanent structure of safety systems in ANY plant area listed in Table 6.4-A. OPERATING CONDITION: ALL		
		SITE EMERGENCY
		GENERAL EMERGENCY

NOTES

CURVES/TABLES:

Table 6.5/6.6 APPLICABLE PLANT AREA
Reactor Building
Refuel Floor
Control Bay
Diesel Generator Buildings
Turbine Building
Intake Pumping Station
Radwaste Building
Cable Tunnel (Intake To Turbine Building)
Standby Gas Treatment Building

TOXIC GASES

Description

6.5-U **TABLE**

EITHER of the following conditions exists:

- Normal operations impeded due to access restrictions caused by toxic gas concentrations within any building or structure listed in Table 6.5/6.6.
- Confirmed report by local, county, or state officials that a large offsite toxic gas release has occurred within one mile of the site with potential to enter the site boundary in concentrations at or above the Permissible Exposure Limit (PEL) causing an evacuation of any site personnel.

OPERATING CONDITION:
ALL

UNUSUAL EVENT

6.5-A **TABLE**

ALL of the following conditions exist:

- Plant personnel report toxic gas within any building or structure listed in Table 6.5/6.6.
- Plant personnel report severe adverse health reactions due to toxic gas (i.e., burning eyes, throat, or dizziness), or sampling results by Fire Protection or Industrial Safety personnel indicate levels above the Permissible Exposure Limit (PEL).
- Determination by the Site Emergency Director that plant personnel would be unable to perform actions necessary to establish and maintain cold shutdown conditions while utilizing appropriate personnel protective equipment.

OPERATING CONDITION:
ALL

ALERT

SITE EMERGENCY

GENERAL EMERGENCY

NOTES

CURVES/TABLES:

Table 6.5/6.6 APPLICABLE PLANT AREA
Reactor Building
Refuel Floor
Control Bay
Diesel Generator Buildings
Turbine Building
Intake Pumping Station
Radwaste Building
Cable Tunnel (Intake To Turbine Building)
Standby Gas Treatment Building

FLAMMABLE GASES

Description	
6.6-U **TABLE** EITHER of the following conditions exists: • Release of flammable gas within the site boundary in concentrations at or above 25% of the Lower Explosive Limit (LEL) for any three readings obtained in a 10 ft. triangular area as indicated by Fire Protection or Industrial Safety personnel using appropriate monitoring instrumentation. • Confirmed report by local, county, or state officials that a large offsite flammable gas release has occurred within one mile of the site with potential to enter the site boundary in concentrations at or above 25% of the Lower Explosive Limit (LEL). OPERATING CONDITION: ALL	**UNUSUAL EVENT**
6.6-A **TABLE** Release of flammable gases within any building or structure listed in Table 6.5/6.6 in concentrations at or above 25% of the Lower Explosive Limit (LEL) for any three readings obtained in a 10 ft. triangular area as indicated by Fire Protection or Industrial Safety personnel using appropriate monitoring instrumentation. OPERATING CONDITION: ALL	**ALERT**
	SITE EMERGENCY
	GENERAL EMERGENCY

NOTES

CURVES/TABLES:

SECURITY

	Description	Description	
6.7-U	1. A SECURITY CONDITION that does NOT involve a HOSTILE ACTION as reported by the Security Shift Supervisor. **OR** 2. A credible Browns Ferry threat notification **OR** 3. A validated notification from NRC providing information of an aircraft threat. OPERATING CONDITION: ALL		**UNUSUAL EVENT**
6.7-A	1. A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLED AREA as reported by the Security Shift Supervisor. **OR** 2. A validated notification from NRC of an airliner attack threat within 30 minutes of the site. OPERATING CONDITION: ALL		**ALERT**
6.7-S	A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the Security Shift Supervisor OPERATING CONDITION: ALL		**SITE EMERGENCY**
6.7-G	**1.** A HOSTILE ACTION has occurred such that plant personnel are unable to operate equipment required to maintain safety functions. **OR** 2. A HOSTILE ACTION has caused failure of Spent Fuel Cooling Systems and IMMINENT fuel damage is likely for a freshly off-loaded reactor core in pool. OPERATING CONDITION: ALL		**GENERAL EMERGENCY**

NOTES

CURVES/TABLES:

VEHICLE CRASH

Description	
6.8-U Vehicle crash (for example; aircraft or barge) into plant structures or systems within the protected area boundary. OPERATING CONDITION: ALL	UNUSUAL EVENT
6.8-A Vehicle crash (for example; aircraft or barge) into ANY plant vital area. . OPERATING CONDITION: ALL	ALERT
	SITE EMERGENCY
	GENERAL EMERGENCY

NOTES

CURVES/TABLES:

SPENT FUEL STORAGE

Description	
6.9-U Damage to a loaded cask CONFINEMENT BOUNDARY from ANY of the following: • Natural phenomena (e.g., seismic event, tornado, flood, lightning, snow/ice accumulation, etc.) • Accident (e.g., dropped cask, tipped over cask, explosion, missile damage, fire damage, burial under debris, etc.). • Judgement of the Site Emergency Director that the CONFINEMENT BOUNDARY damage is a degradation in the level of safety of the ISFSI. OPERATING CONDITION: ALL	UNUSUAL EVENT
	ALERT
	SITE EMERGENCY
	GENERAL EMERGENCY

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NATURAL EVENTS
7.0

NOTES

CURVES/TABLES:

EARTHQUAKE

Description	
7.1-U Valid annunciation in Unit 1 Control Room, Panel 1-XA-55-22C, Window 5, START OF STRONG MOTION ACCELEROGRAPH **AND** Assessment by Unit One and Two Control Room personnel that an earthquake has occurred. OPERATING CONDITION: ALL	UNUSUAL EVENT
7.1-A Valid annunciation in the Unit 1 Control Room, Panel 1-XA-55-22C, Window 6, $^{1}/_{2}$ SSE RESPONSE SPECTRUM EXCEEDED **AND** Assessment by Unit One and Two Control Room personnel that an earthquake has occurred. OPERATING CONDITION: ALL	ALERT
	SITE EMERGENCY
	GENERAL EMERGENCY

NOTES

CURVES/TABLES:

TORNADO / HIGH WINDS

Description	
7.2-U Report by plant personnel of tornado striking within the protected area boundary. OPERATING CONDITION: ALL	UNUSUAL EVENT
7.2-A Tornado striking plant vital area **OR** Onsite wind speed above 90 MPH as indicated using the meteorological data screen of the Integrated Computer System (ICS). OPERATING CONDITION: ALL	ALERT
	SITE EMERGENCY
	GENERAL EMERGENCY

NOTES

CURVES/TABLES:

FLOOD

	Description	
7.3-U	Wheeler Lake level exceeds or is predicted to exceed elevation 565 feet. **AND** Water entering permanent plant structures due to flooding. OPERATING CONDITION: ALL	UNUSUAL EVENT
7.3-A	Wheeler Lake level exceeds or is predicted to exceed elevation 565 feet. **AND** EITHER of the following conditions exists: • Breech or failure of any water-tight structure is causing flooding of the structure • Equipment required for safe shutdown is affected. OPERATING CONDITION: ALL	ALERT
		SITE EMERGENCY
		GENERAL EMERGENCY

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EMERGENCY DIRECTOR JUDGMENT 8.0

NOTES

CURVES/TABLES:

TECHNICAL SPECIFICATIONS

Description	
8.1-U Inability to reach required shutdown condition (Mode 3 or Mode 4) within Technical Specification Limiting Conditions for Operation (LCO) limits. OPERATING CONDITION: Mode 1 or 2 or 3	UNUSUAL EVENT
	ALERT
	SITE EMERGENCY
	GENERAL EMERGENCY

NOTES

CURVES/TABLES:

Table 8.2-U
LOSS OF COMMUNICATIONS

Onsite Communications	Offsite Communication
Plant Phone System Node 1	Bell Lines
Two-Way Radio System (NSS 1, NSS 2, OPS F2, and OPS F4)	Digital Microwave
Sound Power Phones	NRC Emergency Telecommunication System
Nextel Communication System	Cellular Phones (If Available)
	Health Physics Radio Network

LOSS OF COMMUNICATION

Description

8.2-U **TABLE**

Unplanned loss of onsite communication listed in Table 8.2-U that defeats the Plant Operations Staff's ability to perform routine operations

OR

Unplanned loss of ALL off-site communication listed in Table 8.2-U.

OPERATING CONDITOIN:
ALL

UNUSUAL EVENT

ALERT

SITE EMERGENCY

GENERAL EMERGENCY

NOTES

8.3 Significant Transient is an unplanned event involving one or more of the following:
(1) Automatic turbine runback greater than 25% thermal reactor power, or
(2) Electrical load reduction greater than 25% full electrical load, or
(3) Thermal power oscillations greater than 10%, or
(4) Reactor scram, or
(5) Valid ECCS initiation.

CURVES/TABLES:

Table 8.3-S APPLICABLE SAFETY FUNCTIONS
Reactor Power
Reactor Pressure
Reactor Level
Subcriticality
Drywell Temperature
Drywell Pressure
Suppression Chamber Pressure
Suppression Pool Temperature
Suppression Pool Level

LOSS OF ASSESSMENT CAPABILITY

Description

8.3-U

Unplanned loss of most or all safety system annunciators or indicators which causes a significant loss of plant assessment capability for greater than 15 minutes

AND

Compensatory non-alarming safety system indications are available (SPDS, ICS)

AND

In the opinion of the Shift Manager, increased surveillance is required to safely operate the plant.

OPERATING CONDITION:
MODE 1, or 2, or 3

UNUSUAL EVENT

8.3-A **NOTE**

Unplanned loss of most or all safety system annunciators or indicators which causes a significant loss of plant assessment capability for greater than 15 minutes

AND

In the opinion of the Shift Manager, increased surveillance is required to safely operate the plant

AND

EITHER of the following conditions exists:

- Compensatory non-alarming safety system indications are NOT available (SPDS, ICS)
- A significant transient is in progress.

OPERATING CONDITION:
MODE 1, or 2, or 3

ALERT

8.3-S **NOTE** **TABLE**

Loss of most or all annunciators associated with safety systems

AND

Compensatory non-alarming safety system indications are NOT available (SPDS, ICS)

AND

Indications needed to monitor safety functions are NOT available (Refer to Table 8.3-S)

AND

A significant transient is in progress.

OPERATING CONDITION:
MODE 1, or 2, or 3

SITE EMERGENCY

GENERAL EMERGENCY

NOTES

8.4-U Table 8.4-U contains only example events that may justify Unusual Event classification. This event classification is intended to address unanticipated conditions not explicitly addressed elsewhere, but warrant declaration of an emergency because conditions exists which the Emergency Director believes to fall under the Unusual Event Classification. Additionally this EAL should be considered in making emergency classifications regarding challenges to fission product barriers not specifically address elsewhere in the EAL matrix.

8.4-A This event classification is intended to address unanticipated conditions not explicitly addressed elsewhere, but that warrant declaration of an emergency because conditions exist which the Site Emergency Director believes to fall under the Alert classification. Additionally this EAL should be considered in making emergency classifications regarding challenges to fission product barriers not specifically address elsewhere in the EAL matrix.

8.4-S This event classification is intended to address unanticipated conditions not explicitly addressed elsewhere, but that warrant declaration of an emergency because conditions exist which the Site Emergency Director believes to fall under the Site Area Emergency classification. Additionally this EAL should be considered in making emergency classifications regarding challenges to fission product barriers not specifically address elsewhere in the EAL matrix.

8.4-G This event classification is intended to address unanticipated conditions not explicitly addressed elsewhere, but that warrant declaration of an emergency because conditions exist which the Site Emergency Director believes to fall under the General Emergency classification. Additionally this EAL should be considered in making emergency classifications regarding challenges to fission product barriers not specifically address elsewhere in the EAL matrix.

CURVES/TABLES:

Table 8.4-U OTHER EXAMPLE UNUSUAL EVENTS
Plant Transient Response Unexpected Or Not Understood
Unanalyzed Safety System Configuration Affecting, Threatening Safe Shutdown
Inadequate Personnel To Achieve Or Maintain Safe Shutdown
Degraded Plant Conditions Beyond License Basis Threatening Safe Operation Or Safe Shutdown
Emergency Procedures Not Adequate To Maintain Safe Operation Or Achieve Safe Shutdown

OTHER

Description	
8.4-U **NOTE** **TABLE** Events are in process or have occurred which indicate a potential degradation in the level of safety of the plant or indicate a security threat to facility protection has been initiated. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs. Refer to Table 8.4-U for examples. **OR** Any loss or any potential loss of containment. OPERATING CONDITION: ALL	**UNUSUAL EVENT**
8.4-A **NOTE** Events are in process or have occurred which involve an actual or potential substantial degradation in the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels. **OR** Any loss or potential loss of fuel cladding or RCS pressure boundary. OPERATING CONDITION: ALL	**ALERT**
8.4-S **NOTE** Events are in process or have occurred which involve actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts (1) toward site personnel or equipment that could lead to the likely failure thereof or, (2) prevent effective access to equipment needed for protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the site boundary. **OR** Any loss or potential loss of both fuel cladding and RCS pressure boundary. **OR** Potential loss of either fuel cladding or RCS pressure boundary and loss of any additional barrier. OPERATING CONDITION: ALL	**SITE EMERGENCY**
8.4-G **NOTE** Events are in process or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area. **OR** Loss of any two barriers and potential loss of third barrier. OPERATING CONDITION: ALL	**GENERAL EMERGENCY**

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NPG Standard Programs and Processes	**Regulatory Reporting Requirements**	**NPG-SPP-03.5** **Rev. 0011** **Page 1 of 97**
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Corporate Functional Area Manager — Date

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Reporting of Events or Conditions Affecting Licensed Nuclear Power Plants

1.0 PURPOSE

This Attachment identifies reporting requirements; and instructions for determining reportability, preparation, and transmittal of LERs; and notification to NRC for events occurring at TVA's licensed nuclear plants.

2.0 SCOPE

TVA is required by §50.72 and §50.73 to promptly report various types of conditions or events and provide written follow-up reports, as appropriate. This Attachment provides reporting guidance applicable to licensed power reactors.

NOTES

1) Attachment 2, provides additional reporting criteria found in §Part 20, 30, 40, and 70 that may be applicable to events involving byproduct, source or special nuclear material possessed by the licensed nuclear plant. Site Licensing and Site RadCon are responsible for making the reportability determinations for §Part 20, 30, 40, or 70 events associated with their site. Corporate Licensing and Corporate RadChem are responsible for making the reportability determinations for §Part 20, 30, 40, or 70 events associated with all other TVA licensed activities. Depending on the location of the licensed activity, either Site or Corporate Licensing is responsible for developing (with input from affected organizations) and submitting the immediate notification and written reports to NRC in accordance with §Part 20, 30, 40, or 70 requirements. Reporting requirements for personnel exposure required by §Part 20 are contained in RCTP-105, Personnel Inprocessing and Dosimetry Administrative Processes.

2) Attachment 3 contains the criteria for reporting if events or conditions affecting ISFSI. TVA, as the general licensee of the ISFSI, is required by §72.216 to make initial and written reports in accordance with §72.74 and §72.75. Operations is responsible for making the reportability determinations for §72.74 and §72.75 reports. For any event, condition, or issue having the potential for being reportable, contact Site Licensing for consultation and concurrence on the reportability determination. In no event shall the lack of licensing concurrence result in a failure to meet specified reporting timeframes. Operations is responsible for making the immediate notification to NRC in accordance with §72.74. Operations is responsible for making the immediate, 4-hour, and 24-hour notifications to NRC in accordance with §72.75. Site Licensing is responsible for developing (with input from affected organizations) and submitting the written reports required by §72.75.

3) Reporting requirements for events or conditions affecting the physical protection of the licensed nuclear plant specified in §73.71 are contained in NSDP-1, Safeguards Event Reporting Guidelines. Responsibilities for reportability determinations and immediate notification requirements are assigned to Site Nuclear Security and Corporate Nuclear Security. Site Licensing is responsible for developing (with input from affected organizations) and submitting the written reports required by §73.71.

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Reporting of Events or Conditions Affecting Licensed Nuclear Power Plants

3.0 REQUIREMENTS

NOTES

1) Internal management notification requirements for reportable plant events and conditions are found in Procedure NPG-SPP-01.12, TVA Nuclear Event Response Process. The Operations Shift Manager is responsible for notifying Site Operations Management and the Site Duty Plant Manager. The Site Duty Plant Manager is responsible for making the remaining internal management notifications to the NDO who communicates to the fleet executives, in accordance with NPG-SPP-01.6, Nuclear Duty Officer. Internal management notification was previously described in Appendix D of NPG-SPP-03.5. Internal management notifications of emergent issues is described in Procedure NPG-SPP-01.12, TVA Nuclear Event Response Process.

2) NRC NUREG-1022, Revision 3 and subsequent revisions should be used, in its entirety, as guidance for determining reportability of plant events pursuant to §50.72 and §50.73. A text searchable copy of NUREG-1022 is maintained on the TVA NPG Nuclear Licensing Webpage.

3) In addition to reviewing the clarifying discussion and examples associated with specific reporting criteria [e.g., discussion of utilization of engineering judgment when evaluating Unanalyzed Conditions in NUREG -1022, Section 3.2.4(B)], NUREG-1022, Section 2, Reporting Areas Warranting Special Mention, should also be reviewed. [R.1]

3.1 Immediate Notification - NRC

TVA is required by §50.72 and §73.71 to notify NRC immediately if certain types of events occur. This Attachment contains the types of events and the allotted time in which NRC must be notified. (Refer to NRC Form 361 at www.nrc.gov). Operations is responsible for making the reportability determinations for §50.72 and §50.73 reports. Site Nuclear Security and Corporate Nuclear Security are responsible for making the reportability determinations for 73.71 reports. For any §50.72, §50.73, or §73.71 event, condition, or issue having the potential for being reportable, contact Site Licensing for consultation and concurrence on the reportability determination. In no event shall the lack of licensing concurrence result in a failure to meet specified reporting timeframes. Operations is responsible for making the immediate notification to NRC in accordance with §50.72. The Site Security Manager will request the Plant Shift Manager to call the NRC Operations Center, when appropriate.

Notification is via the Emergency Notification System. If the Emergency Notification System is not operative, use a telephone, telegraph, mailgram, or facsimile.

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Reporting of Events or Conditions Affecting Licensed Nuclear Power Plants

3.1 Immediate Notification - NRC (continued)

NOTE

The NRC Event Notification Worksheet may be used in preparing for notifying the NRC. This Worksheet may be obtained directly from the NRC website (www.nrc.gov) by performing a "Form 361" search. Attachment 12 provides guidance for completing NRC Form 361.

A. The Immediate Notification Criteria of §50.72 is divided into 1-hour, 4-hour, and 8- hour phone calls. Notify the NRC Operations Center within the applicable time limit for any item which is identified in the Immediate Notification Criteria.

B. The following criteria require 1-hour notification:

1. 10 CFR 50.36(c)(1)(i)(A), (Technical Specifications) - Safety Limits as defined by the Technical Specifications which have been exceeded (violated)

2. §50.72 (a)(1)(i) - The declaration of any of the Emergency classes specified in the licensee's approved Emergency Plan.

NOTE

If it is discovered that a condition existed which met the Emergency Plan criteria but no emergency was declared and the basis for the emergency class no longer exists at the time of discovery, an ENS notification (and notification of the Operations Duty Specialist), within one hour of discovery of the undeclared (or misclassified) event, shall be made. However, actual declaration of the emergency class is not necessary in these circumstances.

3. §50.72(b)(1) - Any deviation from the plant's Technical Specifications authorized pursuant to §50.54(x).

4. 10 CFR 73, Appendix G, paragraph I - Safeguards Events. The requirements of §73.71, Reporting of Safeguard Events, are also applicable. Refer to NSDP-1, "Safeguards Event Reporting Guidelines," for additional information.

a. Any event in which there is reason to believe that a person has committed or caused, or attempted to commit or cause, or has made a credible threat to commit or cause:

(1) A theft or unlawful diversion of special nuclear material; or

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Reporting of Events or Conditions Affecting Licensed Nuclear Power Plants

3.1 Immediate Notification - NRC (continued)

(2) Significant physical damage to a power reactor or any facility possessing SSNM or its equipment or carrier equipment transporting nuclear fuel or spent nuclear fuel, or to the nuclear fuel or spent nuclear fuel a facility or carrier possesses; or

(3) Interruption of normal operation of a licensed nuclear power reactor through the unauthorized use of or tampering with its machinery, components, or controls including the security system. [Note: a Confirmed Cyber Attack at any NPG site is reported to the NRC in accordance with the requirements of 10 CFR 73, Appendix G. Review the 'Incident Categorization' section in NPG-SPP-12.8.8.]

b. An actual entry of an unauthorized person into a protected area, material access area, controlled access area, vital area, or transport.

c. Any failure, degradation, or the discovered vulnerability in a safeguard system that could allow unauthorized or undetected access to a protected area, material access area, controlled access area, vital area, or transport for which compensatory measures have not been employed.

d. The actual or attempted introduction of contraband into a protected area, material access area, vital area, or transport.

C. The following criteria require 4-hour notification:

1. §50.72(b)(2)(i) - The initiation of any nuclear plant shutdown required by the plant's Technical Specifications.

2. §50.72(b)(2)(iv)(A) - Any event that results or should have resulted in Emergency Core Cooling System (ECCS) discharge into the reactor coolant system as a result of a valid signal except when the actuation results from and is part of a pre-planned sequence during testing or reactor operation.

3. §50.72(b)(2)(iv)(B) - Any event or condition that results in actuation of the reactor protection system (RPS) when the reactor is critical except when the actuation results from and is part of a pre-planned sequence during testing or reactor operation.

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Reporting of Events or Conditions Affecting Licensed Nuclear Power Plants

3.1 Immediate Notification - NRC (continued)

NOTES

1) NPG-SPP-05.14 provides additional instructions regarding addressing and informally communicating events to outside agencies involving radiological spills and leaks.

2) Routine or day-to-day communications between TVA organizations and state agencies typically do not constitute a formal notification to other government agencies that would require a report in accordance with §50.72(b)(2)(xi).

4. §50.72(b)(2)(xi) - Any event or situation, related to the health and safety of the public or onsite personnel, or protection of the environment, for which a news release is planned or notification to other government agencies has been or will be made. Such an event may include an onsite fatality or inadvertent release of radioactive contaminated materials.

D. The following criteria require 8-hour notification:

NOTE

With the exception of “Events or Conditions that Could Have Prevented Fulfillment of a Safety Function," ENS notifications are required for any event that occurred within three years of discovery, even if the event was not ongoing at the time of discovery.

1. §50.72(b)(3)(ii)(A) - Any event or condition that results in the condition of the nuclear power plant, including its principal safety barriers, being seriously degraded.

2. §50.72(b)(3)(ii)(B) - Any event or condition that results in the nuclear power plant being in an unanalyzed condition that significantly degrades plant safety.

3. §50.72(b)(3)(iv)(A) - Any event or condition that results in valid actuation of any of the systems listed in paragraph (b)(3)(iv)(B) [see list below], except when the actuation results from and is part of a pre-planned sequence during testing or reactor operation.

   a. The systems to which the requirements of paragraph §50.72(b)(3)(iv)(A) apply are:

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Reporting of Events or Conditions Affecting Licensed Nuclear Power Plants

3.1 Immediate Notification - NRC (continued)

NOTE

Actuation of the RPS when the reactor is critical is also reportable under §50.72(b)(2)(iv)(B) above.

(1) Reactor protection system (RPS) including: reactor scram or reactor trip.

(2) General containment isolation signals affecting containment isolation valves in more than one system or multiple main steam isolation valves (MSIVs).

(3) Emergency core cooling systems (ECCS) for pressurized water reactors (PWRs) including: high-head, intermediate-head, and low-head injection systems and the low pressure injection function of residual (decay) heat removal systems.

(4) ECCS for boiling water reactors (BWRs) including: core spray systems; high-pressure coolant injection system; low pressure injection function of the residual heat removal system.

(5) BWR reactor core isolation cooling system; isolation condenser system; and feedwater coolant injection system.

(6) PWR auxiliary or emergency feedwater system.

(7) Containment heat removal and depressurization systems, including containment spray and fan cooler systems.

(8) Emergency ac electrical power systems, including: Emergency diesel generators (EDGs).

NOTE

For systems within scope, the inadvertent TS inoperability of a system in a required mode of applicability constitutes an event or condition for which there is no longer reasonable expectation that equipment can fulfill its safety function. Therefore, such events or conditions are reportable as an "Event or Condition that Could Have Prevented Fulfillment of a Safety Function."

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Reporting of Events or Conditions Affecting Licensed Nuclear Power Plants

3.1 Immediate Notification - NRC (continued)

4. §50.72(b)(3)(v) - Any event or condition that at the time of discovery could have prevented the fulfillment of the safety function of structures or systems that are needed to:

   (A) Shut down the reactor and maintain it in a safe shutdown condition;

   (B) Remove residual heat;

   (C) Control the release of radioactive material; or

   (D) Mitigate the consequences of an accident.

NOTE

According to §50.72 (b)(3)(vi) events covered by §50.72(b)(3)(v) may include one or more procedural errors, equipment failures, and/or discovery of design, analysis, fabrication, construction, and/or procedural inadequacies. However, individual component failures need not be reported pursuant this paragraph if redundant equipment in the same system was operable and available to perform the required safety function.

5. §50.72(b)(3)(xii) - Any event requiring the transport of a radioactively contaminated person to an offsite medical facility for treatment.

6. §50.72(b)(3)(xiii) - Any event that results in a major loss of emergency assessment capability, offsite response capability, or offsite communications capability (e.g., significant portion of control room indication, emergency notification system, or offsite notification system).

E. Follow-up Notification (§50.72(c))

With respect to the telephone notifications made under paragraphs (a) and (b) [§50.72 (a) and §50.72 (b), respectively] of this section [§50.72], in addition to making the required initial notification, during the course of the event:

1. Immediately report:

   (i) Any further degradation in the level of safety of the plant or other worsening plant conditions including those that require the declaration of the Emergency Classes, if such a declaration has

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Reporting of Events or Conditions Affecting Licensed Nuclear Power Plants

3.1 Immediate Notification - NRC (continued)

not been previously made; or

(ii) Any change from one Emergency Class to another, or

(iii) A termination of the Emergency Class.

(1) Immediately report:

(i) The results of ensuing evaluations or assessments of plant conditions,

(ii) The effectiveness of response or protective measures taken, and

(iii) Information related to plant behavior that is not understood.

(2) Maintain an open, continuous communication channel with the NRC Operations Center upon request by the NRC.

3.2 Twenty-Four Hour Notification - NRC

Any violation of the requirement contained in specific operating license conditions, shall be reported to NRC in accordance with the license condition.

3.3 Two-Day Notification - NRC

§50.9(b) - The NRC shall be notified of incomplete or inaccurate information which contains significant implications for the public health and safety or common defense and security. Notification shall be provided to the administrator of the appropriate regional office within two working days of identifying the information. Depending on where the information originates, either Corporate or Site Licensing is responsible for determining reportability (with input from affected organizations) and notifying NRC in accordance with §50.9.

3.4 Sixty-Day Verbal Report

§50.73(a)(2)(iv)(A) requires that any event or condition that resulted in manual or automatic actuation of the specified systems be reported as a Licensee Event Report (LER [Refer to Attachment 1, Section 3.5]). This CFR section also allows that in the case of an invalid actuation, other than actuation of the reactor protection system when the reactor is critical, an optional telephone notification may be placed to the NRC Operations Center within 60 days after discovery of the event instead of submitting a written LER.

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Reporting of Events or Conditions Affecting Licensed Nuclear Power Plants

3.4 Sixty-Day Verbal Report (continued)

A. Telephone Report Required Content:

If the telephone notification option is selected (NUREG 1022, Revision 3, Section 3.2.6., System Actuation), instead of an LER, the verbal report:

1. Is not considered an LER.

2. Should identify that the report is being made under §50.73(a)(2)(iv)(A).

3. Should provide the following information:

   a. The specific train(s) and system(s) that were actuated.

   b. Whether each train actuation was complete or partial.

   c. Whether or not the system started and functioned successfully.

NOTE

Licensing will ensure that the information that is provided to NRC during the Sixty-Day telephone report is verified in accordance with NPG-SPP-03.10.

B. Telephone Report Development and Review

Licensing will:

1. Develop (with input from responsible organization) the response (i.e., report summary) to address the required input.

2. Ensure that the reporting details are approved by site vice president or his designee prior to making the verbal report.

C. Telephone Report Timeliness

Operations will make the 60-day telephone report promptly after the response is approved by the site vice president or his designee.

3.5 Written Report - NRC

A. For events in which safety limits or limiting safety system settings are exceeded, reports are made as required by 10 CFR 50.72 and 50.73.

B. Any violation of the requirements contained in the Operating license conditions in lieu of other reporting requirements requires a written follow-up report if specified in the license.

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Reporting of Events or Conditions Affecting Licensed Nuclear Power Plants

3.5 Written Report - NRC (continued)

C. Reporting Radiation Injuries

1. §140.6(a) requires, as promptly as practicable, submittal of a written notice [e.g., report] in the event of:

   a. Bodily injury or property damage arising out of or in connection with the possession or use of the radioactive material at the licensee's facility [location]; or

   b. In the course of transportation; or

   c. In the event any radiation exposure claim is made. (Refer to RCDP-9, Radiological and Chemistry Control Radiological Exposure Inquiries)

2. The written notice shall contain particulars sufficient to identify the licensee and reasonably obtainable information with respect to time, place, and circumstances thereof, or the nature of the claim.

D. Licensee Event Reports

A written report shall be prepared in accordance with §50.73(a)(1) for items in the 60-day report criteria or Technical Specifications. The report shall be complete and accurate in accordance with the methods outlined in this procedure. The completed forms shall be submitted to the USNRC, Document Control Desk, Washington, DC 20555. NUREG 1022, Revision 3, contains the instructions for completion of the LER form. Licensing is responsible for developing (with input from affected organizations) and submitting the written reports (or optional telephone reports [refer to Attachment 1, Section 3.4]) required by §50.73.

NOTE

Unless otherwise specified in the reporting criteria below, an event shall be reported if it occurred within three years of the date of discovery regardless of the plant mode or power level, and regardless of the significance of the structure, system, or component that initiated the event.

E. Report Criteria

1. §50.73(a)(2)(i)(A) - The completion of any nuclear plant shutdown required by the plant's Technical Specifications.

2. §50.73(a)(2)(i)(B) - Any operation or condition which was prohibited by the plant's Technical Specifications, except when:

   a. The Technical Specification is administrative in nature;

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Reporting of Events or Conditions Affecting Licensed Nuclear Power Plants

3.5 Written Report - NRC (continued)

b. The event consisted solely of a case of a late surveillance test where the oversight was corrected, the test was performed, and the equipment was found to be capable of performing its specified safety functions; or

c. The Technical Specification was revised prior to discovery of the event such that the operation or condition was no longer prohibited at the time of discovery of the event.

3. §50.73(a)(2)(i)(C) - Any deviation from the plant's Technical Specifications authorized pursuant to §50.54(x).

4. §50.73(a)(2)(ii)(A) - Any event or condition that resulted in the condition of the nuclear power plant, including its principal safety barriers, being seriously degraded.

5. §50.73(a)(2)(ii)(B) - Any event or condition that resulted in the nuclear power plant being in an unanalyzed condition that significantly degraded plant safety.

6. §50.73(a)(2)(iii) - Any natural phenomenon or other external condition that posed an actual threat to the safety of the nuclear power plant or significantly hampered site personnel in the performance of duties necessary for the safe operation of the nuclear power plant.

7. §50.73(a)(2)(iv)(A) - Any event or condition that resulted in manual or automatic actuation of any of the systems listed in paragraph (a)(2)(iv)(B) [see list under Item 8 below], except when

a. The actuation resulted from and was part of a pre-planned sequence during testing or reactor operation; or

b. The actuation was invalid and

(i) Occurred while the system was properly removed from service or

(ii) Occurred after the safety function had been already completed.

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Reporting of Events or Conditions Affecting Licensed Nuclear Power Plants

3.5 Written Report - NRC (continued)

NOTE

In the case of an invalid actuation, other than actuation of the reactor protection system (RPS) when the reactor is critical, a telephone notification to the NRC Operations Center within 60 days after discovery of the event may be provided instead of submitting a written LER (§50.73(a)). [Refer to , Attachment 1, Section 3.4]

8. §50.73(a)(2)(iv)(B) - The systems to which the requirements to paragraph (a)(2)(iv)(A) of this section apply are:

   a. Reactor protection system (RPS) including: reactor scram or reactor trip.

   b. General containment isolation signals affecting containment isolation valves in more than one system or multiple main steam isolation valves (MSIVs).

   c. Emergency core cooling systems (ECCS) for pressurized water reactors (PWRs) including: high-head, intermediate-head, and low-head injection systems and the low pressure injection function of residual (decay) heat removal systems.

   d. ECCS for boiling water reactors (BWRs) including: core spray systems; high-pressure coolant injection system; low pressure injection function of the residual heat removal system.

   e. BWR reactor core isolation cooling system; isolation condenser system; and feedwater coolant injection system.

   f. PWR auxiliary or emergency feedwater system.

   g. Containment heat removal and depressurization systems, including containment spray and fan cooler systems.

   h. Emergency ac electrical power systems, including: emergency diesel generators (EDGs).

   i. Emergency service water systems that do not normally run and that serve as ultimate heat sinks.

9. §50.73(a)(2)(v) - Any event or condition that could have prevented the fulfillment of the safety function of structures or systems that are needed to:

   (A) Shut down the reactor and maintain it in a safe shutdown condition;

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Reporting of Events or Conditions Affecting Licensed Nuclear Power Plants

3.5 Written Report - NRC (continued)

(B) Remove residual heat;

(C) Control the release of radioactive material; or

(D) Mitigate the consequences of an accident.

NOTE

Events reported above may include one or more procedural errors, equipment failures, and/or discovery of design, analysis, fabrication, construction, and/or procedural inadequacies. However, individual component failures need not be reported pursuant to this criterion if redundant equipment in the same system was operable and available to perform the required safety function [§50.73(a)(2)(vi)].

10. §50.73(a)(2)(vii) - Any event where a single cause or condition caused at least one independent train or channel to become inoperable in multiple systems or two independent trains or channels to become inoperable in a single system designed to:

    (A) Shut down the reactor and maintain it in a safe shutdown condition;

    (B) Remove residual heat;

    (C) Control the release of radioactive material; or

    (D) Mitigate the consequences of an accident.

11. §50.73(a)(2)(viii)(A) - Any airborne radioactivity release that, when averaged over a time period of 1 hour, resulted in airborne radionuclide concentrations in an unrestricted area that exceeded 20 times the applicable concentration limits specified in Appendix B to Part 20, table 2, column 1.

12. §50.73(a)(2)(viii)(B) - Any liquid effluent release that, when averaged over a time period of 1 hour, exceeds 20 times the applicable concentrations specified in Appendix B to Part 20, table 2, column 2, at the point of entry into the receiving waters (i.e., unrestricted area) for all radionuclides except tritium and dissolved noble gases.

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Reporting of Events or Conditions Affecting Licensed Nuclear Power Plants

3.5 Written Report - NRC (continued)

13. §50.73(a)(2)(ix)(A) - Any event or condition that as a result of a single cause could have prevented the fulfillment of a safety function for two or more trains or channels in different systems that are needed to:

    a. Shut down the reactor and maintain it in a safe shutdown condition;

    b. Remove residual heat;

    c. Control the release of radioactive material; or

    d. Mitigate the consequences of an accident.

    NOTE

    Events covered above may include cases of procedural error, equipment failure, and/or discovery of a design, analysis, fabrication, construction, and/or procedural inadequacy. However, licensees are not required to report an event pursuant to this criterion if the event results from a shared dependency among trains or channels that is a natural or expected consequence of the approved plant design or normal and expected wear or degradation [§50.73(a)(2)(ix)(B)].

14. §50.73(a)(2)(x) - Any event that posed an actual threat to the safety of the nuclear power plant or significantly hampered site personnel in the performance of duties necessary for the safe operation of the nuclear power plant including fires, toxic gas releases, or radioactive releases.

15. 10 CFR 73, Appendix G, paragraph I - If a one hour notification is made in Attachment 1, section 3.1.B.4 of this procedure, then a written notification to the NRC is required within 60 days.

16. For reporting a defect found installed in the Plant's Safety Related Equipment, Radioactive Wastes System, and Special Nuclear Material within an LER, §Part 21 NRC Reporting of Defects and Noncompliance, see Attachment 5 in this procedure.

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Reporting of Events or Conditions Affecting Licensed Nuclear Power Plants

3.5 Written Report - NRC (continued)

17. **SQN and WBN only** (Non-radiological environmental reporting requirements to the NRC, as required from SQN and WBN Operating License (OL), Appendix B.)

    a. WBN or SQN shall record any occurrence of unusual or important environmental events. Unusual or important events are those that potentially could cause or indicate environmental impact causally related with station operation. The following are examples:

        (1) Excessive bird impaction events;

        (2) Onsite plant or animal disease outbreaks;

        (3) Unusual mortality of any species protected by the Endangered Species Act of 1973;

        (4) Fish kills near the plant site;

        (5) Unanticipated or emergency discharges of waste water or chemical substances that exceeds the limits of, or is not authorized by, the NPDES permit and requires 24-hour notification to the County or State of Tennessee;

        WBN only

        (6) Identification of any threatened or endangered species for which the NRC has not initiated consultation with the Federal Wildlife Service (FWS).

        (7) Increase in nuisance organisms or conditions in excess of levels anticipated in station environmental impact appraisals.

    b. SQN OL Appendix B compliance guidance is provided in the flowchart in NPG-SPP-05.5, Environmental Control, Appendix B.

    c. WBN OL Appendix B compliance is met through the procedures referenced in NPG-SPP-05.5.

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Reporting of Events or Conditions Affecting Licensed Nuclear Power Plants

3.5 Written Report - NRC (continued)

d. Once an unusual or important event has occurred, the required actions are:

(1) Refer to NPG-SPP-05.5, Environmental Control, Section Compliance with the NRC Appendix B to the Facility Operating License, for additional guidance.

(2) If required, SQN or WBN Site Licensing shall make a written report to the NRC in accordance with the NRC Non-routine Report, OL Appendix B, Subsections 5.4.2, within 30 days, in the event of a reportable occurrence in which a limit specified in a relevant permit or certificate issued by another Federal, State, or local agency is exceeded.

3.6 Retraction or Cancellation of Event Reports

An ENS notification may be retracted via a follow-up telephone call. If an ENS notification is make and its later determined that the event or condition was not reportable, Plant Operations should call the NRC Operations Center on the ENS telephone to retract the notification and explain the rational for that decision. There is no set time limit for ENS telephone retractions. However, because most retractions occur following completion of engineering and/or management review, it is expected that retractions would occur shortly after such review.

Cancellation of LERs submitted should be made by letter. The letter should state that the LER is being cancelled (i.e., formally withdrawn). The bases for the cancellation should be explained so that the staff can review and understand the reasons supporting the determination.

3.1 REACTIVITY CONTROL SYSTEMS

3.1.3 Control Rod OPERABILITY

LCO 3.1.3 Each control rod shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

----------NOTE----------
Separate Condition entry is allowed for each control rod.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One withdrawn control rod stuck.	----------NOTE---------- Rod worth minimizer (RWM) may be bypassed as allowed by LCO 3.3.2.1, "Control Rod Block Instrumentation," if required, to allow continued operation. --- A.1 Verify stuck control rod separation criteria are met. AND A.2 Disarm the associated control rod drive (CRD). AND	Immediately 2 hours

(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.3	Perform SR 3.1.3.3 for each withdrawn OPERABLE control rod.	24 hours from discovery of Condition A concurrent with THERMAL POWER greater than the low power setpoint (LPSP) of the RWM
	AND		
	A.4	Perform SR 3.1.1.1.	72 hours
B. Two or more withdrawn control rods stuck.	B.1	Be in MODE 3.	12 hours
C. One or more control rods inoperable for reasons other than Condition A or B.	C.1	----------NOTE---------- RWM may be bypassed as allowed by LCO 3.3.2.1, if required, to allow insertion of inoperable control rod and continued operation. ----------	
		Fully insert inoperable control rod.	3 hours
	AND		
	C.2	Disarm the associated CRD.	4 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. ----------NOTE---------- Not applicable when THERMAL POWER > 10% RTP. ---------- Two or more inoperable control rods not in compliance with banked position withdrawal sequence (BPWS) and not separated by two or more OPERABLE control rods.	D.1 Restore compliance with BPWS. <u>OR</u> D.2 Restore control rod to OPERABLE status.	4 hours 4 hours
E. Required Action and associated Completion Time of Condition A, C, or D not met. <u>OR</u> Nine or more control rods inoperable.	E.1 Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.3.1	Determine the position of each control rod.	24 hours
SR 3.1.3.2	(Deleted).	
SR 3.1.3.3	----------NOTE---------- Not required to be performed until 31 days after the control rod is withdrawn and THERMAL POWER is greater than the LPSP of the RWM. ---------- Insert each withdrawn control rod at least one notch.	31 days
SR 3.1.3.4	Verify each control rod scram time from fully withdrawn to notch position 06 is $\leq$ 7 seconds.	In accordance with SR 3.1.4.1, SR 3.1.4.2, SR 3.1.4.3, and SR 3.1.4.4

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.3.5	Verify each control rod does not go to the withdrawn overtravel position.	Each time the control rod is withdrawn to "full out" position AND Prior to declaring control rod OPERABLE after work on control rod or CRD System that could affect coupling

3.1 REACTIVITY CONTROL SYSTEMS

3.1.4 Control Rod Scram Times

LCO 3.1.4

a. No more than 13 OPERABLE control rods shall be "slow," in accordance with Table 3.1.4-1; and

b. No more than 2 OPERABLE control rods that are "slow" shall occupy adjacent locations.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

----------NOTE----------
During single control rod scram time Surveillances, the control rod drive (CRD) pumps shall be isolated from the associated scram accumulator.

SURVEILLANCE		FREQUENCY
SR 3.1.4.1	Verify each control rod scram time is within the limits of Table 3.1.4-1 with reactor steam dome pressure [redacted] psig.	Prior to exceeding 40% RTP after each reactor shutdown ≥ 120 days
SR 3.1.4.2	Verify, for a representative sample, each tested control rod scram time is within the limits of Table 3.1.4-1 with reactor steam dome pressure [redacted] psig.	120 days cumulative operation in MODE 1

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.4.3	Verify for each affected control rod scram time is within the limits of Table 3.1.4-1 with any reactor steam dome pressure.	Prior to declaring control rod OPERABLE after work on control rod or CRD System that could affect scram time
SR 3.1.4.4	Verify each affected control rod scram time is within the limits of Table 3.1.4-1 with reactor steam dome pressure $\geq$ 800 psig.	Prior to exceeding 40% RTP after fuel movement within the affected core cell AND Prior to exceeding 40% RTP after work on control rod or CRD System that could affect scram time

Table 3.1.4-1 (page 1 of 1)
Control Rod Scram Times

----------NOTES----------

1. OPERABLE control rods with scram times not within the limits of this Table are considered "slow."

2. Enter applicable Conditions and Required Actions of LCO 3.1.3, "Control Rod OPERABILITY," for control rods with scram times > 7 seconds to notch position 06. These control rods are inoperable, in accordance with SR 3.1.3.4, and are not considered "slow."

NOTCH POSITION	SCRAM TIMES[a][b] (seconds)
	REACTOR STEAM DOME PRESSURE ≥ 800 psig
46	0.45
36	1.08
26	1.84
06	3.36

(a) Maximum scram time from fully withdrawn position, based on de-energization of scram pilot valve solenoids at time zero.

(b) Scram times as a function of reactor steam dome pressure, when < 800 psig are within established limits.

3.1 REACTIVITY CONTROL SYSTEMS

3.1.5 Control Rod Scram Accumulators

LCO 3.1.5 Each control rod scram accumulator shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

----------NOTE----------
Separate Condition entry is allowed for each control rod scram accumulator.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One control rod scram accumulator inoperable with reactor steam dome pressure ≥ 900 psig.	A.1	----------NOTE---------- Only applicable if the associated control rod scram time was within the limits of Table 3.1.4-1 during the last scram time Surveillance. ---------- Declare the associated control rod scram time "slow."	8 hours
	OR		
	A.2	Declare the associated control rod inoperable.	8 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Two or more control rod scram accumulators inoperable with reactor steam dome pressure ≥ 900 psig.	B.1 Restore charging water header pressure to ≥ 940 psig.	20 minutes from discovery of Condition B concurrent with charging water header pressure < 940 psig
	AND	
	B.2.1 ----------NOTE---------- Only applicable if the associated control rod scram time was within the limits of Table 3.1.4-1 during the last scram time Surveillance. ---------- Declare the associated control rod scram time "slow."	1 hour
	OR	
	B.2.2 Declare the associated control rod inoperable.	1 hour

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One or more control rod scram accumulators inoperable with reactor steam dome pressure < 900 psig.	C.1 Verify all control rods associated with inoperable accumulators are fully inserted. AND	Immediately upon discovery of charging water header pressure < 940 psig
	C.2 Declare the associated control rod inoperable.	1 hour
D. Required Action and associated Completion Time of Required Action B.1 or C.1 not met.	D.1 ----------NOTE---------- Not applicable if all inoperable control rod scram accumulators are associated with fully inserted control rods. ---------- Place the reactor mode switch in the shutdown position.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.5.1	Verify each control rod scram accumulator pressure is [redacted] psig.	7 days

3.1 REACTIVITY CONTROL SYSTEMS

3.1.7 Standby Liquid Control (SLC) System

LCO 3.1.7 Two SLC subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SLC subsystem inoperable.	A.1 Restore SLC subsystem to OPERABLE status.	7 days
B. Two SLC subsystems inoperable.	B.1 Restore one SLC subsystem to OPERABLE status.	8 hours
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.7.1	Verify available volume of sodium pentaborate solution (SPB) is [redacted] gallons.	24 hours
SR 3.1.7.2	Verify continuity of explosive charge.	31 days
SR 3.1.7.3	Verify the SPB concentration is [redacted]% by weight.	31 days AND Once within 24 hours after water or boron is added to solution
SR 3.1.7.4	Verify the SPB concentration is [redacted] by weight. OR	31 days AND Once within 24 hours after water or boron is added to solution

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
	Verify the concentration and temperature of boron in solution are within the limits of Figure 3.1.7-1.	Once within 8 hours after discovery that SPB concentration is > 9.2% by weight AND 12 hours thereafter
SR 3.1.7.5	Verify the minimum quantity of Boron-10 in the SLC solution tank and available for injection is $\geq$ 186 pounds.	31 days
SR 3.1.7.6	Verify the SLC conditions satisfy the following equation: $\frac{(\ C\ )(\ Q\ )(\ E\ )}{(13\text{ wt. }\%)(86\text{ gpm})(19.8\text{ atom}\%)} \geq 1$ where, C = sodium pentaborate solution concentration (weight percent) Q = pump flow rate (gpm) E = Boron-10 enrichment (atom percent Boron-10)	31 days AND Once within 24 hours after water or boron is added to the solution
SR 3.1.7.7	Verify each pump develops a flow rate [redacted] gpm at a discharge pressure [redacted] psig.	24 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.7.8	Verify flow through one SLC subsystem from pump into reactor pressure vessel.	24 months on a STAGGERED TEST BASIS
SR 3.1.7.9	Verify all piping between storage tank and pump suction is unblocked.	24 months
SR 3.1.7.10	Verify sodium pentaborate enrichment is within the limits established by SR 3.1.7.6 by calculating within 24 hours and verifying by analysis within 30 days.	24 months AND After addition to SLC tank
SR 3.1.7.11	Verify each SLC subsystem manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position, or can be aligned to the correct position.	31 days

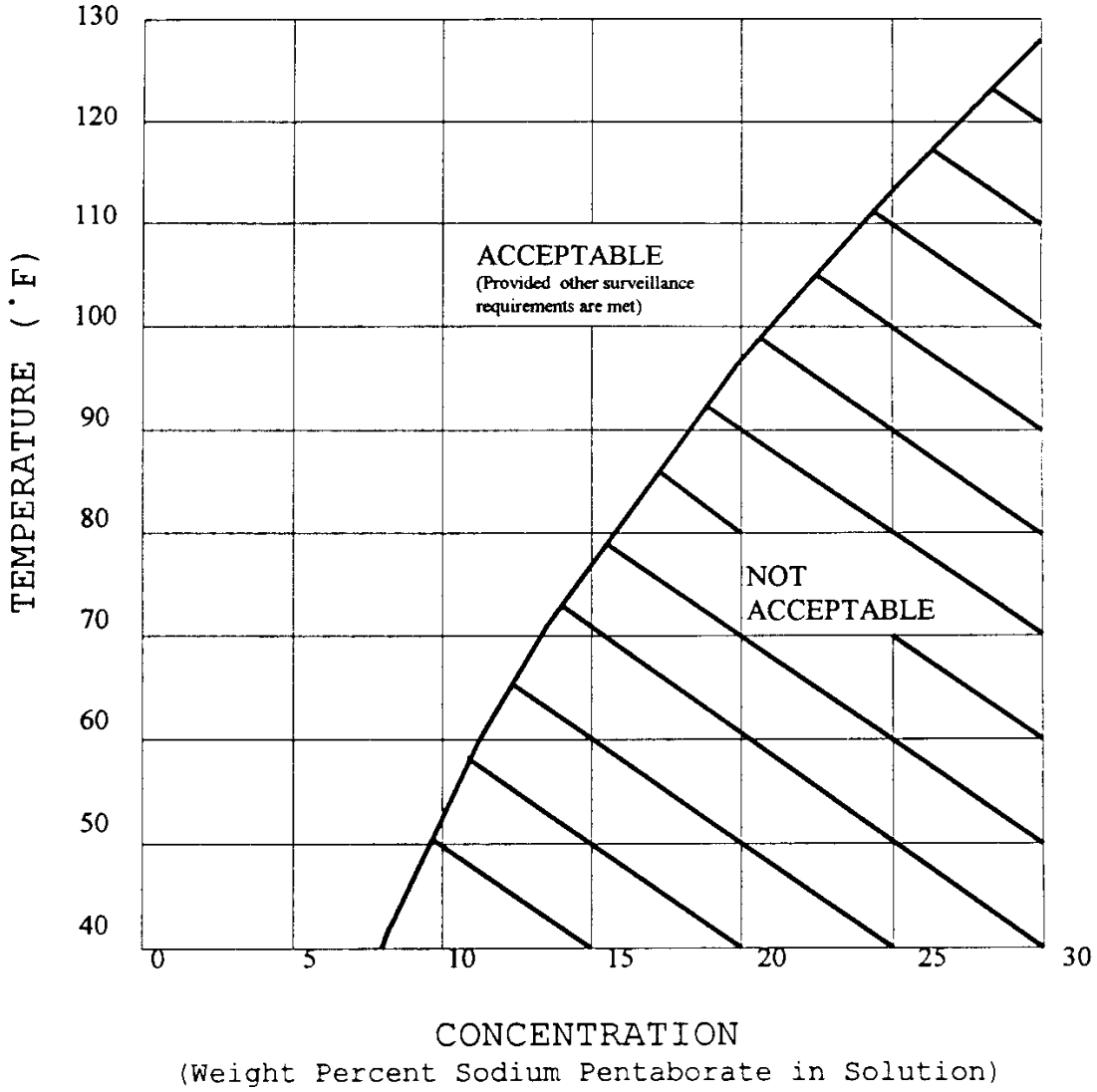

Figure 3.1.7-1
Sodium Pentaborate Solution Temperature Versus Concentration Requirements

3.3 INSTRUMENTATION

3.3.7.1 Control Room Emergency Ventilation (CREV) System Instrumentation

LCO 3.3.7.1 The CREV System instrumentation for each Function in Table 3.3.7.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.7.1-1.

ACTIONS

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

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more required channels inoperable.	A.1	Enter the Condition referenced in Table 3.3.7.1-1 for the channel.	Immediately
B. As required by Required Action A.1 and referenced in Table 3.3.7.1-1.	B.1	Declare associated CREV subsystem inoperable.	1 hour from discovery of loss of CREV initiation capability
	AND		
	B.2	Place channel in trip.	12 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. As required by Required Action A.1 and referenced in Table 3.3.7.1-1.	C.1	Declare associated CREV subsystem inoperable.	1 hour from discovery of loss of CREV initiation capability
	AND		
	C.2	Place channel in trip.	24 hours
D. As required by Required Action A.1 and referenced in Table 3.3.7.1-1.	D.1	Perform SR 3.3.7.1.2 on the OPERABLE channel.	Once per 24 hours
	AND		
	D.2	Verify alternate monitoring capability.	Once per 12 hours from discovery of both channels inoperable
	AND		
	D.3	Restore one channel to OPERABLE status.	30 days from discovery of both channels inoperable

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
E. Required Action and associated Completion Time of Condition B, C, or D not met.	E.1	Place the associated CREV subsystem(s) in the pressurization mode of operation.	1 hour
	OR		
	E.2	Declare associated CREV subsystem inoperable.	1 hour

SURVEILLANCE REQUIREMENTS

----------NOTES----------

1. Refer to Table 3.3.7.1-1 to determine which SRs apply for each CREV 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 CREV initiation capability.
3. For Functions 3 and 4, when a channel is placed in an inoperable status solely for the performance of a CHANNEL CALIBRATION or maintenance, entry into the associated Conditions and Required Actions may be delayed for up to 24 hours provided the downscale trip of the inoperable channel is placed in the trip condition.

SURVEILLANCE		FREQUENCY
SR 3.3.7.1.1	Perform CHANNEL CHECK.	24 hours
SR 3.3.7.1.2	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.7.1.3	Perform CHANNEL CALIBRATION.	92 days
SR 3.3.7.1.4	Perform LOGIC SYSTEM FUNCTIONAL TEST.	184 days
SR 3.3.7.1.5	Perform CHANNEL CALIBRATION.	24 months
SR 3.3.7.1.6	Perform LOGIC SYSTEM FUNCTIONAL TEST.	24 months

Table 3.3.7.1-1 (page 1 of 1)
Control Room Emergency Ventilation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Reactor Vessel Water Level - Low, Level 3	1,2,3,(a)	2	B	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.5 SR 3.3.7.1.6	[redacted]
2. Drywell Pressure - High	1,2,3	2	B	SR 3.3.7.1.2 SR 3.3.7.1.5 SR 3.3.7.1.6	[redacted]
3. Reactor Zone Exhaust Radiation - High	1,2,3 (a)	1	C	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.5 SR 3.3.7.1.6	[redacted]
4. Refueling Floor Exhaust Radiation - High	1,2,3, (a)	1	C	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.5 SR 3.3.7.1.6	[redacted]
5. Control Room Air Supply Duct Radiation - High	1,2,3, (a)	1	D	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.3 SR 3.3.7.1.4	[redacted]

(a) During operations with a potential for draining the reactor vessel.

3.3 INSTRUMENTATION

3.3.1.1 Reactor Protection System (RPS) Instrumentation

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

APPLICABILITY: According to Table 3.3.1.1-1.

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required channels inoperable.	A.1 Place channel in trip. OR	12 hours
	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. OR B.2 Place one trip system in trip.	6 hours 6 hours
C. One or more Functions with RPS trip capability not maintained.	C.1 Restore RPS trip capability.	1 hour
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 [redacted] RTP.	4 hours
F. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	F.1 Be in MODE 2.	6 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
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 Be in Mode 2.	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	Perform CHANNEL CHECK.	24 hours
SR 3.3.1.1.2	----------NOTE---------- Not required to be performed until 12 hours after THERMAL POWER[redacted]RTP. ---------- Verify the absolute difference between the average power range monitor (APRM) channels and the calculated power is [redacted]RTP while operating at[redacted]RTP.	 7 days
SR 3.3.1.1.3	----------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 3.3.1.1.4	Perform CHANNEL FUNCTIONAL TEST.	7 days
SR 3.3.1.1.5	Verify the source range monitor (SRM) and intermediate range monitor (IRM) channels overlap.	Prior to withdrawing SRMs from the fully inserted position
SR 3.3.1.1.6	----------NOTE---------- Only required to be met during entry into MODE 2 from MODE 1. ---------- Verify the IRM and APRM channels overlap.	7 days
SR 3.3.1.1.7	Calibrate the local power range monitors.	1000 MWD/T average core exposure
SR 3.3.1.1.8	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.1.1.9	----------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. ---------- Perform CHANNEL CALIBRATION.	92 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.10	Perform CHANNEL CALIBRATION.	184 days
SR 3.3.1.1.11	(Deleted)	
SR 3.3.1.1.12	Perform CHANNEL FUNCTIONAL TEST.	24 months
SR 3.3.1.1.13	----------NOTE---------- Neutron detectors are excluded. ---------- Perform CHANNEL CALIBRATION.	24 months
SR 3.3.1.1.14	Perform LOGIC SYSTEM FUNCTIONAL TEST.	24 months
SR 3.3.1.1.15	Verify Turbine Stop Valve - Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure - Low Functions are not bypassed when THERMAL POWER is [redacted] RTP.	24 months
SR 3.3.1.1.16	----------NOTE---------- For Function 2.a, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. ---------- Perform CHANNEL FUNCTIONAL TEST.	184 days
SR 3.3.1.1.17	Verify OPRM is not bypassed when APRM Simulated Thermal Power is [redacted] and recirculation drive flow is [redacted] of rated recirculation drive flow.	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 3.3.1.1.1 SR 3.3.1.1.3 SR 3.3.1.1.5 SR 3.3.1.1.6 SR 3.3.1.1.9 SR 3.3.1.1.14	[redacted]
	5(a)	3	H	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.9 SR 3.3.1.1.14	[redacted]
b. Inop	2	3	G	SR 3.3.1.1.3 SR 3.3.1.1.14	NA
	5(a)	3	H	SR 3.3.1.1.4 SR 3.3.1.1.14	NA
2. Average Power Range Monitors					
a. Neutron Flux - High, (Setdown)	2	3(b)	G	SR 3.3.1.1.1 SR 3.3.1.1.6 SR 3.3.1.1.7 SR 3.3.1.1.13 SR 3.3.1.1.16	[redacted]
b. Flow Biased Simulated Thermal Power - High	1	3(b)	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.7 SR 3.3.1.1.13 SR 3.3.1.1.16	[redacted]
c. Neutron Flux - High	1	3(b)	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.7 SR 3.3.1.1.13 SR 3.3.1.1.16	[redacted]RTP

(continued)

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

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

(c) [redacted]RTP when reset for single loop operation per LCO 3.4.1, “Recirculation Loops Operating.

NOTE: This page is applicable after commencing Cycle 11 operation.

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

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. Average Power Range Monitors (continued)					
d. Inop	1,2	3(b)	G	SR 3.3.1.1.16	NA
e. 2-Out-Of-4 Voter	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.14 SR 3.3.1.1.16	NA
f. OPRM Upscale	1	3(b)	I	SR 3.3.1.1.1 SR 3.3.1.1.7 SR 3.3.1.1.13 SR 3.3.1.1.16 SR 3.3.1.1.17	NA(e)
3. Reactor Vessel Steam Dome Pressure - High(d)	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.14	[redacted] psig
4. Reactor Vessel Water Level - Low, Level 3(d)	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	[redacted] inches above vessel zero
5. Main Steam Isolation Valve - Closure	1	8	F	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	[redacted] closed
6. Drywell Pressure - High	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	[redacted] psig
7. Scram Discharge Volume Water Level - High					
a. Resistance Temperature Detector	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	[redacted] gallons
	5(a)	2	H	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	[redacted] gallons

(continued)

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.
(b) Each APRM channel provides inputs to both trip systems.
(d) During instrument calibrations, if the As Found channel setpoint is conservative with respect to the Allowable Value but outside its acceptable As Found band as defined by its associated Surveillance Requirement procedure, then there shall be an initial determination to ensure confidence that the channel can perform as required before returning the channel to service in accordance with the Surveillance. If the As Found instrument channel setpoint is not conservative with respect to the Allowable Value, the channel shall be declared inoperable.

Prior to returning a channel to service, the instrument channel setpoint shall be calibrated to a value that is within the acceptable As Left tolerance of the setpoint; otherwise, the channel shall be declared inoperable.

The nominal Trip Setpoint shall be specified on design output documentation which is incorporated by reference in the Updated Final Safety Analysis Report. The methodology used to determine the nominal Trip Setpoint, the predefined As Found Tolerance, and the As Left Tolerance

band, and a listing of the setpoint design output documentation shall be specified in Chapter 7 of the Updated Final Safety Analysis Report.

(e) Refer to 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 (continued)					
b. Float Switch	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	[redacted] gallons
	5(a)	2	H	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	[redacted] gallons
8. Turbine Stop Valve - Closure	[redacted] RTP	4	E	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14 SR 3.3.1.1.15	[redacted] closed
9. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low(d)	[redacted] RTP	2	E	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14 SR 3.3.1.1.15	[redacted] psig
10. Reactor Mode Switch - Shutdown Position	1,2	1	G	SR 3.3.1.1.12 SR 3.3.1.1.14	NA
	5(a)	1	H	SR 3.3.1.1.12 SR 3.3.1.1.14	NA
11. Manual Scram	1,2	1	G	SR 3.3.1.1.8 SR 3.3.1.1.14	NA
	5(a)	1	H	SR 3.3.1.1.8 SR 3.3.1.1.14	NA
12. RPS Channel Test Switches	1,2	2	G	SR 3.3.1.1.4	NA
	5(a)	2	H	SR 3.3.1.1.4	NA
13.Deleted					

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

(d) During instrument calibrations, if the As Found channel setpoint is conservative with respect to the Allowable Value but outside its acceptable As Found band as defined by its associated Surveillance Requirement procedure, then there shall be an initial determination to ensure confidence that the channel can perform as required before returning the channel to service in accordance with the Surveillance. If the As Found instrument channel setpoint is not conservative with respect to the Allowable Value, the channel shall be declared inoperable.

Prior to returning a channel to service, the instrument channel setpoint shall be calibrated to a value that is within the acceptable As Left tolerance of the setpoint; otherwise, the channel shall be declared inoperable.

The nominal Trip Setpoint shall be specified on design output documentation which is incorporated by reference in the Updated Final Safety Analysis Report. The methodology used to determine the nominal Trip Setpoint, the predefined As Found Tolerance, and the As Left Tolerance band, and a listing of the setpoint design output documentation shall be specified in Chapter 7 of the Updated Final Safety Analysis Report.

3.7 PLANT SYSTEMS

3.7.3 Control Room Emergency Ventilation (CREV) System

LCO 3.7.3 Two CREV subsystems shall be OPERABLE.

----------NOTE----------
The main control room envelope (CRE) boundary may be opened intermittently under administrative control.

APPLICABILITY: MODES 1, 2, and 3, during operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CREV subsystem inoperable for reasons other than Condition B, C, or D.	A.1 Restore CREV subsystem to OPERABLE status.	7 days
B. One or more CREV subsystems inoperable due to inoperable CRE boundary in MODE 1, 2, or 3.	B.1 Initiate action to implement mitigating actions AND	Immediately
	B.2 Verify mitigating actions ensure CRE occupant exposures to radiological hazards will not exceed limits, and verify the CRE occupants are protected from smoke and chemical hazards. AND	24 hours
	B.3 Restore CRE boundary to OPERABLE status.	90 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. Two CREV subsystems inoperable due to inoperable High Efficiency Particulate Air (HEPA) filter or charcoal adsorbers which do not impact ability of CREV subsystems to meet flowrate requirements specified in the Ventilation Filter Testing Program (VFTP).	C.1	Restore HEPA filter and one charcoal adsorber to OPERABLE status.	7 days
D. One CREV subsystem inoperable due to inoperable charcoal adsorber which does not impact the ability of CREV subsystem to meet flowrate requirements specified in the VFTP.	D.1	Restore charcoal adsorber to OPERABLE status.	14 days
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.	12 hours
	AND		
	E.2	Be in MODE 4.	36 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. Required Action and associated Completion Time of Condition A or D not met during OPDRVs.	F.1 Place OPERABLE CREV subsystem in pressurization mode. OR F.2 Initiate action to suspend OPDRVs.	Immediately Immediately
G. Two CREV subsystems inoperable in MODE 1, 2, or 3 for reasons other than Condition B or C.	G.1 Enter LCO 3.0.3.	Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
H. Required Action and associated Completion Time of Condition C not met during OPDRVs. OR Two CREV subsystems inoperable during OPDRVs for reasons other than Condition C. OR One or more CREV subsystems inoperable due to an inoperable CRE Boundary during OPDRVs.	H.1 Initiate action to suspend OPDRVs.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.3.1	Operate each CREV subsystem for ≥ 10 continuous hours with the heaters operating.	31 days
SR 3.7.3.2	Perform required CREV filter testing in accordance with the VFTP.	In accordance with the VFTP
SR 3.7.3.3	Verify each CREV subsystem actuates on an actual or simulated initiation signal.	24 months
SR 3.7.3.4	Perform required CRE unfiltered air inleakage testing in accordance with the Control Room Envelope Habitability Program.	In accordance with the Control Room Envelope Habitability Program

3.8 ELECTRICAL POWER SYSTEMS

3.8.1 AC Sources - Operating

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

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

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

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

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

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

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

ACTIONS

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

(continued)

ACTIONS (continued)

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

(continued)

ACTIONS

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

(continued)

ACTIONS (continued)

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

(continued)

ACTIONS (continued)

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

(continued)

ACTIONS (continued)

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

(continued)

ACTIONS (continued)

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

SURVEILLANCE REQUIREMENTS

----------NOTE----------
SR 3.8.1.1 through SR 3.8.1.9 are applicable to the Unit 1 and 2 AC sources.
SR 3.8.1.10 is applicable only to Unit 3 AC sources.

SURVEILLANCE		FREQUENCY
SR 3.8.1.1	----------NOTES---------- 1. Performance of SR 3.8.1.4 satisfies this SR. 2. All DG starts may be preceded by an engine prelube period and followed by a warmup period prior to loading. 3. A modified DG start involving idling and gradual acceleration to synchronous speed may be used for this SR as recommended by the manufacturer. When modified start procedures are not used, the time, voltage, and frequency tolerances of SR 3.8.1.4 must be met. --- Verify each DG starts from standby conditions and achieves steady state voltage [redacted] V and [redacted] V and frequency [redacted] Hz and [redacted] Hz.	31 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.2	----------NOTES---------- 1. DG loadings may include gradual loading as recommended by the manufacturer. 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 3.8.1.1 or SR 3.8.1.4. ---------- Verify each DG is synchronized and loaded and operates for $\geq$ 60 minutes at a load [redacted] kW and [redacted] kW.	31 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.3	Verify the fuel oil transfer system operates to automatically transfer fuel oil from 7-day storage tank to the day tank.	31 days
SR 3.8.1.4	----------NOTE---------- All DG starts may be preceded by an engine prelube period. ---------- Verify each DG starts from standby condition and achieves, in [redacted] seconds, voltage [redacted] V and frequency [redacted] Hz. Verify after DG fast start from standby conditions that the DG achieves steady state voltage [redacted] V and [redacted] V and frequency [redacted] Hz and ≤ [redacted] Hz.	184 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.5	----------NOTE---------- If performed with the DG synchronized with offsite power, it shall be performed at a power factor [redacted] ---------- Verify each DG rejects a load greater than or equal to its associated single largest post-accident load, and: a. Following load rejection, the frequency is [redacted] Hz; and b. Following load rejection, the steady state voltage recovers to [redacted] V and [redacted] V. c. Following load rejection, the steady state frequency recovers to [redacted] Hz and [redacted] Hz.	24 months
SR 3.8.1.6	----------NOTE---------- All DG starts may be preceded by an engine prelube period followed by a warmup period. ---------- Verify on an actual or simulated accident signal each DG auto-starts from standby condition.	24 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.7	----------NOTE---------- Momentary transients outside the load and power factor ranges do not invalidate this test. ---------- Verify each DG operating at a power factor [redacted] operates for ≥ 24 hours: a. For ≥ 2 hours loaded [redacted] kW and [redacted] kW; and b. For the remaining hours of the test loaded [redacted] kW and [redacted] kW.	24 months
SR 3.8.1.8	Verify interval between each timed load block is within the allowable values for each individual timer.	24 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.9	----------NOTE---------- All DG starts may be preceded by an engine prelube period. ---------- Verify, on an 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 [redacted] seconds, 2. energizes auto-connected emergency loads through individual timers, 3. achieves steady state voltage [redacted] V and [redacted] V, 4. achieves steady state frequency [redacted] Hz and [redacted] and 5. supplies permanently connected and auto-connected emergency loads for [redacted] minutes.	24 months
SR 3.8.1.10	For required Unit 3 DGs, the SRs of Unit 3 Technical Specifications are applicable.	In accordance with applicable SRs

3.8 ELECTRICAL POWER SYSTEMS

3.8.1 AC Sources - Operating

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

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

b. Unit 3 diesel generators (DGs) with two divisions of 480 V load shed logic and common accident signal logic OPERABLE; and

c. Unit 1 and 2 DG(s) capable of supplying the Unit 1 and 2 4.16 kV shutdown board(s) required by LCO 3.8.7, "Distribution Systems - Operating."

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

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

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

ACTIONS

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

(continued)

ACTIONS

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

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.5 Restore Unit 3 DG to OPERABLE status.	7 days from discovery of unavailability of TDG(s) AND 24 hours from discovery of Condition B entry $\geq$ 6 days concurrent with unavailability of TDG(s) AND 14 days AND 21 days from discovery of failure to meet LCO

(continued)

ACTIONS (continued)

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

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
----------NOTE---------- Only applicable when more than one 4.16 kV shutdown board is affected. ---------- F. One required offsite circuit inoperable. <u>AND</u> One Unit 3 DG inoperable.	----------NOTE---------- Enter applicable Conditions and Required Actions of LCO 3.8.7, "Distribution Systems - Operating," when Condition F is entered with no AC power source to any 4.16 kV shutdown board. ---------- F.1 Restore required offsite circuit to OPERABLE status. <u>OR</u> F.2 Restore Unit 3 DG to OPERABLE status.	12 hours 12 hours
----------NOTE---------- Applicable when only one 4.16 kV shutdown board is affected. ---------- G. One required offsite circuit inoperable. <u>AND</u> One Unit 3 DG inoperable.	G.1 Declare the affected 4.16 kV shutdown board inoperable.	Immediately

(continued)

ACTIONS (continued)

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

(continued)

ACTIONS (continued)

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

SURVEILLANCE REQUIREMENTS

----------NOTE----------
SR 3.8.1.1 through SR 3.8.1.9 are applicable to the Unit 3 AC sources. SR 3.8.1.10 is applicable only to Unit 1 and 2 AC sources.

SURVEILLANCE		FREQUENCY
SR 3.8.1.1	----------NOTES---------- 1. Performance of SR 3.8.1.4 satisfies this SR. 2. All DG starts may be preceded by an engine prelube period and followed by a warmup period prior to loading. 3. A modified DG start involving idling and gradual acceleration to synchronous speed may be used for this SR as recommended by the manufacturer. When modified start procedures are not used, the time, voltage, and frequency tolerances of SR 3.8.1.4 must be met. --- Verify each DG starts from standby conditions and achieves steady state voltage [redacted] and frequency [redacted]	31 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.2	----------NOTES---------- 1. DG loadings may include gradual loading as recommended by the manufacturer. 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 3.8.1.1 or SR 3.8.1.4. ---------- Verify each DG is synchronized and loaded and operates for $\geq$ 60 minutes at a load [redacted] kW and [redacted] kW.	31 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.3	Verify the fuel oil transfer system operates to automatically transfer fuel oil from 7-day storage tank to the day tank.	31 days
SR 3.8.1.4	----------NOTE---------- All DG starts may be preceded by an engine prelube period. ---------- Verify each DG starts from standby condition and achieves, in [redacted] seconds, voltage [redacted] V and frequency [redacted] Hz. Verify after DG fast start from standby conditions that the DG achieves steady state voltage [redacted] V and [redacted] V and frequency [redacted] Hz and [redacted] Hz.	184 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.5	----------NOTE---------- If performed with the DG synchronized with offsite power, it shall be performed at a power factor [redacted] ---------- Verify each DG rejects a load greater than or equal to its associated single largest post-accident load, and: a. Following load rejection, the frequency is [redacted] Hz; and b. Following load rejection, the steady state voltage recovers to [redacted] V and [redacted] V. c. Following load rejection, the steady state frequency recovers to [redacted] and [redacted] Hz.	24 months
SR 3.8.1.6	----------NOTE---------- All DG starts may be preceded by an engine prelube period followed by a warmup period. ---------- Verify on an actual or simulated accident signal each DG auto-starts from standby condition.	24 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.7	----------NOTE---------- Momentary transients outside the load and power factor ranges do not invalidate this test. ---------- Verify each DG operating at a power factor [redacted] operates for ≥ 24 hours: a. For ≥ 2 hours loaded [redacted] kW and [redacted] kW; and b. For the remaining hours of the test loaded [redacted] kW and [redacted] kW.	24 months
SR 3.8.1.8	Verify interval between each timed load block is within the allowable values for each individual timer.	24 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.9	----------NOTE---------- All DG starts may be preceded by an engine prelube period. ---------- Verify, on an 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 [redacted] seconds, 2. energizes auto-connected emergency loads through individual timers, 3. achieves steady state voltage [redacted] V and [redacted] V, 4. achieves steady state frequency [redacted] Hz and [redacted] Hz, and 5. supplies permanently connected and auto-connected emergency loads for ≥ 5 minutes.	24 months
SR 3.8.1.10	For required Unit 1 and 2 DGs, the SRs of Unit 1 and 2 Technical Specifications are applicable.	In accordance with applicable SRs

Test: _____

Class: _____

Instructor: _____

LXR•TEST™ Response Form LXR-20020 Side 1

Name: _____

Signature: _____

Date: _____

READ CAREFULLY!

OK NOT OK

- Use black ink only.
- Mark responses darkly and fill completely.
- Erase unwanted marks clearly.
- Do NOT make any stray marks on the page.
- No credit will be given for improper marks.
- If Side 2 is used, fill in ID on both sides.

DOCKET NUMBER

Version

C 1 C 2 C 3 C 4 V

SECTION 1

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

SECTION 2

1 2 3

4 5 6

7 8 9

10 11 12

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