FINAL AS GIVEN (SRO)

2009B ONS SRO NRC Examination - Q 1

Given the following Unit 1 conditions:

Initial conditions:

- Reactor power = 100%

Current conditions:

- RCS pressure = 2078 psig decreasing
- Pzr level = 350 inches increasing
- 1SA-18/A-1 (Pressurizer Relief Valve Flow) actuated
- ALL 1RC-66 flow monitor red lights are illuminated

Immediate Manual Actions of AP/44 (Abnormal Pressurizer Pressure Control) directs the operator to __(1)__ AND the actual setpoint (psig) for the Low RCS Pressure RPS trip is __(2)__.

Which ONE of the following completes the above statement?

A. 1. Close 1RC-4
   2. 1800

B. 1. Close 1RC-4
   2. 1810

C. 1. Manually trip the reactor
   2. 1800

D. 1. Manually trip the reactor
   2. 1810

2009B ONS SRO NRC Examination - Q 2

Given the following Unit 1 conditions:

- Time = 03:00
- Reactor power = 100%
- 1A and 1C RBCUs operating in HIGH speed
- 1B RBCU is operable and OFF
- Large Break LOCA occurs

Which ONE of the following describes RBCU status at 03:04?

A. 1A and 1C RBCUs operating in HIGH speed and 1B RBCU is OFF

B. 1A and 1C RBCUs operating in LOW speed and 1B RBCU is OFF

C. ALL RBCUs operating in LOW speed

D. ALL RBCUs will be OFF

2009B ONS SRO NRC Examination - Q 3

Given the following Unit 1 conditions:

- Reactor power = 100%

	1B1 RCP	1B2 RCP
UPPER Cavity Pressure	1426 psig stable	714 psig stable
LOWER Cavity Pressure	2140 psig stable	1412 psig stable
Motor Stand Velocity Vibration	0.8 mil stable	3.2 mils stable
Spool Piece Displacement	6.5 mils stable	14.5 mils stable

Which ONE of the following describes the required actions in accordance with AP/16 (Abnormal Reactor Coolant Pump Operation)?

A. Immediately trip the reactor and then trip the 1B1 RCP

B. Immediately trip the reactor and then trip the 1B2 RCP

C. Reduce reactor power to < 70% and then trip 1B1 RCP

D. Reduce reactor power to < 70% and then trip 1B2 RCP

2009B ONS SRO NRC Examination - Q 4

Given the following Unit 1 conditions:

Initial conditions:

- Reactor power = 100%

Current conditions:

- 1HP-120 (RC VOLUME CONTROL) air line is severed

Pzr level will INITIALLY __(1)__ AND __(2)__ will be used FIRST to control Pzr level in accordance with plant procedures.

Which ONE of the following completes the statement above?

A. 1. increase
   2. 1HP-26

B. 1. increase
   2. 1HP-122 (RC Volume Control Bypass)

C. 1. decrease
   2. 1HP-26

D. 1. decrease
   2. 1HP-122 (RC Volume Control Bypass)

2009B ONS SRO NRC Examination - Q 5

Given the following two sets of plant conditions:

1. Reactor trips from 100% power. Pressurizer level decreases off scale low during the initial cooldown then returns on scale. Lowest Subcooling margin indication during the transient = 18° F. Pressurizer level is returned to 100 inches, RCS pressure = 2100 psig and Pressurizer temperature = 635° F.

2. From an initial Reactor power of 100%, ICS MAX Runback is used to decrease power to 80% to stop the 1D2 HDP. Pressurizer level remains approximately 220 inches during the runback and when the runback is stopped, RCS pressure = 2100 psig and Pressurizer temperature = 643° F.

Which ONE of the following describes the response of the two scenarios if attempting to increase RCS pressure to 2200 psig with ALL Pressurizer heaters energized and Pressurizer level maintained constant AND the reason for the response?

A. Scenario # 1 will reach 2200 psig first because Pressurizer level is lower and therefore less heat is required to raise the temperature of the water.

B. Scenario # 2 will reach 2200 psig first because the Pressurizer in Scenario #1 is subcooled.

C. Both scenarios would reach 2200 psig at approximately the same time because starting pressure is equal in both scenarios.

D. Neither scenario would reach 2200 psig because the spray valve will overcome the RCS pressure increase even with ALL Pressurizer heaters energized.

2009B ONS SRO NRC Examination - Q 6

Given the following Unit 1 conditions:

- Reactor in MODE 3
- RCS temperature = 555 °F slowly decreasing
- SGTR tab in progress due to SGTR in 1A SG

Which ONE of the following;

1) is the range (minimum - maximum) of RCS temperature (degrees F) where the EOP directs isolating the 1A SG?

2) would require steaming the 1A SG after isolation in accordance with the EOP?

A. 1. 535 - 552
   2. When the 1A SG approaches overfill conditions

B. 1. 535 - 552
   2. When the 1A SG tube-to-shell delta T = (-)76 degrees F

C. 1. 525 - 532
   2. When the 1A SG approaches overfill conditions

D. 1. 525 - 532
   2. When the 1A SG tube-to-shell delta T = (-)76 degrees F

2009B ONS SRO NRC Examination - Q 7

Given the following conditions:

Initial conditions:

- All three units Reactor power = 100%

Current conditions:

- All Unit's 4160v Main Feeder Busses are de-energized
- Unit 1, 2, and 3 EOP Blackout tabs in progress

Which ONE of the following describes the required status of Unit 1 Essential Inverters in accordance with the EOP Enclosure 5.38 (Restoration of Power) AND the reason for the status?

A. remain energized to provide power to ES channels

B. remain energized to provide control power to 4160v switchgear

C. de-energized to prevent inverter damage

D. de-energized to extend available battery life

2009B ONS SRO NRC Examination - Q 8

Given the following Unit 1 conditions:

Initial conditions:

- Reactor power = 100%
- Loss of offsite power occurs

Current conditions:

- Main Feeder Buses remain de-energized

1) The position of 1MS-112 (SSRH Control) is __(1)__.

2) 1MS-77 (MS to MSRH) __(2)__ be operated from the control room switch.

Which ONE of the following completes the statements above?

A. 1. open
   2. can

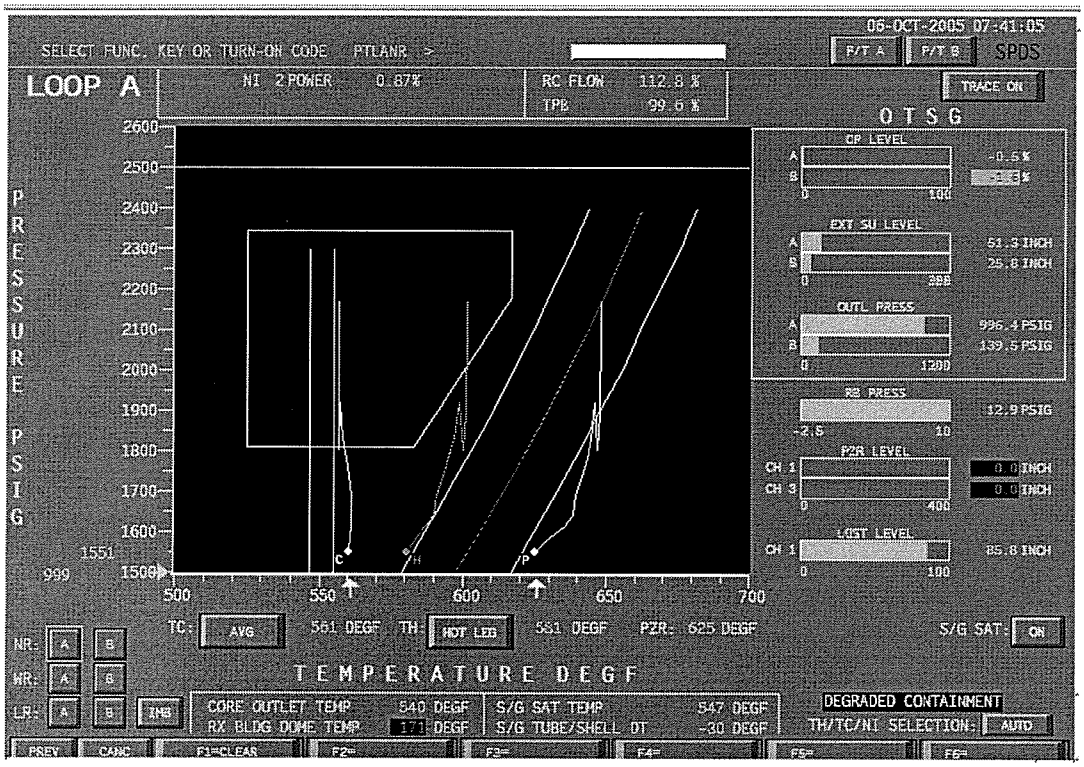
B. 1. closed
   2. can

C. 1. open
   2. can NOT

D. 1. closed
   2. can NOT

2009B ONS SRO NRC Examination - Q 9

Given the following Unit 1 conditions:



- Main Steam Line Break occurred 20 seconds ago
- Plant response as indicated above

1) The EOP will direct the operator to __(1)__.

2) The __(2)__ MDEFWP(s) will automatically start.

Which ONE of the following completes the statement above?

A. 1. Secure LPI pumps
   2. 1A ONLY

B. 1. Throttle HPI
   2. 1A ONLY

C. 1. Secure LPI pumps
   2. 1A <u>and</u> 1B

D. 1. Throttle HPI
   2. 1A <u>and</u> 1B

2009B ONS SRO NRC Examination - Q 10

Given the following Unit 3 conditions:

Initial conditions:

- Reactor power = 100%
- Reactor trip

Current conditions:

- ALL TBV's are in AUTO and closed
- Main Steam pressure cycling between 1050 psig and 1070 psig

1) The current conditions are a result of __(1)__.

2) The __(2)__ will be used to restore control of MS pressure in accordance with the EOP.

Which ONE of the following completes the statements above?

A. 1. Condenser vacuum = 6" hg
   2. TBV's in MANUAL

B. 1. Condenser vacuum = 6" hg
   2. Atmospheric Dump Valves

C. 1. Selected Turbine Header Pressure failed low
   2. TBV's in MANUAL

D. 1. Selected Turbine Header Pressure failed low
   2. Atmospheric Dump Valves

2009B ONS SRO NRC Examination - Q 11

Given the following Unit 1 conditions:

- SBLOCA has occurred
- ALL SCM's = 0°F stable
- RCS pressure = 745 psig slowly decreasing
- LOCA Cooldown Tab in progress

Which ONE of the following describes:

1) the HIGHEST Rx vessel head level (inches) that would indicate voiding in the Rx vessel head?

2) conditions that would make the RV head level NOT valid?

A. 1. 523
   2. Starting 1A LPI pump

B. 1. 523
   2. Open Reactor vessel head high point vents

C. 1. 161
   2. Starting 1A LPI pump

D. 1. 161
   2. Open Reactor vessel head high point vents

2009B ONS SRO NRC Examination - Q 12

Given the following Unit 1 conditions:

Initial conditions:

- Reactor in MODE 5
- Reactor vessel level = 84 inches stable
- Normal LPI decay heat removal in service
- 1C LPI pump operating

Current conditions:

- Reactor vessel level = 84 inches stable
- 1C LPI pump amps are cycling 5 – 55 amps
- 1C LPI pump discharge pressure is fluctuating

Which ONE of the following describes:

1) the Immediate Manual Action(s) required by AP/26 (Loss of Decay heat Removal)?

2) how RCS temperature will be controlled once normal decay heat removal has been restored?

A. 1. Secure the 1C LPI pump ONLY
   2. Adjusting LPSW to LPI coolers

B. 1. Secure the 1C LPI pump ONLY
   2. Adjusting LPI flow through LPI coolers

C. 1. Secure the 1C LPI pump and start the 1A LPI pump
   2. Adjusting LPSW to LPI coolers

D. 1. Secure the 1C LPI pump and start the 1B LPI pump
   2. Adjusting LPI flow through LPI coolers

2009B ONS SRO NRC Examination - Q 13

Given the following Unit 1 conditions:

Initial conditions:

- Reactor power = 100%
- Feedwater and Reactor in MANUAL

Current conditions:

- BOTH Main Feedwater pumps trip
- Reactor power = 35% decreasing
- RCS pressure = 2378 psig increasing
- Rule 1 (ATWS/Unanticipated Nuclear Power Production) in progress

__(1)__ will be the FIRST to add negative reactivity to the reactor AND the position of the PORV is __(2)__.

Which ONE of the following completes the statement above?

A. 1. Moderator temperature coefficient
   2. open

B. 1. Moderator temperature coefficient
   2. closed

C. 1. Control Rods
   2. open

D. 1. Control Rods
   2. closed

2009B ONS SRO NRC Examination - Q 14

Given the following Unit 2 conditions:

Initial conditions:

- Reactor power = 100%
- Both Main FDW pumps trip
- 2A and 2B MDEFDW pumps did NOT start
- TDEFWP did NOT start

Current conditions:

- Tave = 566°F stable
- Recovery from CBP feed with the TDEFDW pump is in progress
- TDEFWP is running and flow has been verified

1) Tave will INITIALLY be controlled by throttling __(1)__.

2) INITIALLY a SG level __(2)__ be established.

Which ONE of the following completes the statements above?

A. 1. EFDW flow
   2. will NOT

B. 1. the TBVs
   2. will NOT

C. 1. EFDW flow
   2. will

D. 1. the TBVs
   2. will

2009B ONS SRO NRC Examination - Q 15

Given the following Unit 1 conditions:

- Reactor power = 100%
- OAC Computer Out of Service
- 1KVIC deenergized

Which ONE of the following describes the instrumentation used to determine axial imbalance in accordance with OP/1/A/1105/014 (Control Room Instrumentation Operation And Information)?

A. Incore Detectors

B. Backup Incore Detectors

C. Imbalance readout indications in RPS cabinets

D. Control Room Power Range Dixon gauge indications

2009B ONS SRO NRC Examination - Q 16

Given the following Unit 1 conditions:

- Reactor power = 100%
- SA-141 (SA to IA Controller) failed closed

Which ONE of the following would result in the Auxiliary Instrument Air system being required to maintain 1HP-5 OPEN assuming NO operator actions are taken?

A. Unit 1 blackout occurs

B. Mechanical failure of the Primary IA compressor

C. 2 inch IA header rupture in Unit 1 East Penetration room

D. 230KV Red Bus lockout occurs in the 230KV switchyard

2009B ONS SRO NRC Examination - Q 17

Given the following Unit 3 conditions:

- A voltage disturbance is occurring
- AP/34 (Degraded Grid) initiated
- Power Factor is leading
- Generator Mwe = 900
- Generator Hydrogen pressure = 60 psig
- Generator output voltage = 18.3 kV

Which ONE of the following is the limit on MVARs in accordance with the Generator Capacity Curve?

REFERENCE PROVIDED

A. 250

B. 290

C. 350

D. 475

2009B ONS SRO NRC Examination - Q 18

Given the following Unit 1 conditions:

Initial conditions:

- Reactor power = 100%

Current conditions:

- Both Main Feedwater pumps tripped
- EFDW NOT available
- 1TD de-energized
- RCS pressure = 2217 psig slowly increasing

1) The __(1)__ will be aligned FIRST to provide decay heat removal in accordance with the EOP.

2) AP/11 (Recovery from Loss of Power) __(2)__ be used to restore power to 1TD.

Which ONE of the following completes the statements above?

A. 1. HPI Pumps
   2. will

B. 1. HPI Pumps
   2. will NOT

C. 1. Condensate Booster Pumps
   2. will

D. 1. Condensate Booster Pumps
   2. will NOT

2009B ONS SRO NRC Examination - Q 19

Given the following Unit 3 conditions:

- Reactor power = 100%

Which ONE of the following would result in an AUTOMATIC injection of HPI AND would require performing EOP Encl. 5.1 (ES Actuation)?

A reactor trip _____.

A. followed by an inadvertent ES channel 1 actuation

B. due to a Steam Line break on piping between 3B2 MSR outlet and inlet to C LP Turbine

C. with main and startup feedwater control valves in HAND

D. concurrent with a failure of the TBV 125 psi bias to be applied following the trip

2009B ONS SRO NRC Examination - Q 20

Given the following Unit 1 conditions:

Initial conditions:

- Reactor power = 68% stable
- 1B2 RCP secured
- Control Rod Group 7 position = 65% withdrawn

Current conditions

- Control Rod Group 7 Rod 1 drops to 40% withdrawn

1) The CRD system __(1)__ generate a runback fault.

2) The MAXIMUM final power level (Core Thermal Power) directed by AP/1 (Unit Runback) is __(2)__.

Which ONE of the following completes the statements above?

A. 1. will
   2. $\leq$ 60%

B. 1. will
   2. $\leq$ 45%

C. 1. will NOT
   2. $\leq$ 60%

D. 1. will NOT
   2. $\leq$ 45%

2009B ONS SRO NRC Examination - Q 21

Given the following Unit 1 conditions:

Initial conditions:

- Mode 6
- Fuel assemblies are being loaded into the core
- All four SR NIs in service
- SR 1NI-1 and SR 1NI-3 are the designated NIs for Fuel Handling

Current conditions:

- Power supply to SR 1NI-1 fails (0 vdc)

Which ONE of the following describes the impact on fuel movement in accordance with OP/1/A/1502/007 (Operations Defueling/Refueling Responsibilities)?

A. Allowed to continue because two other SR NIs remain in service

B. Allowed to continue because SR NI-3 is still in service

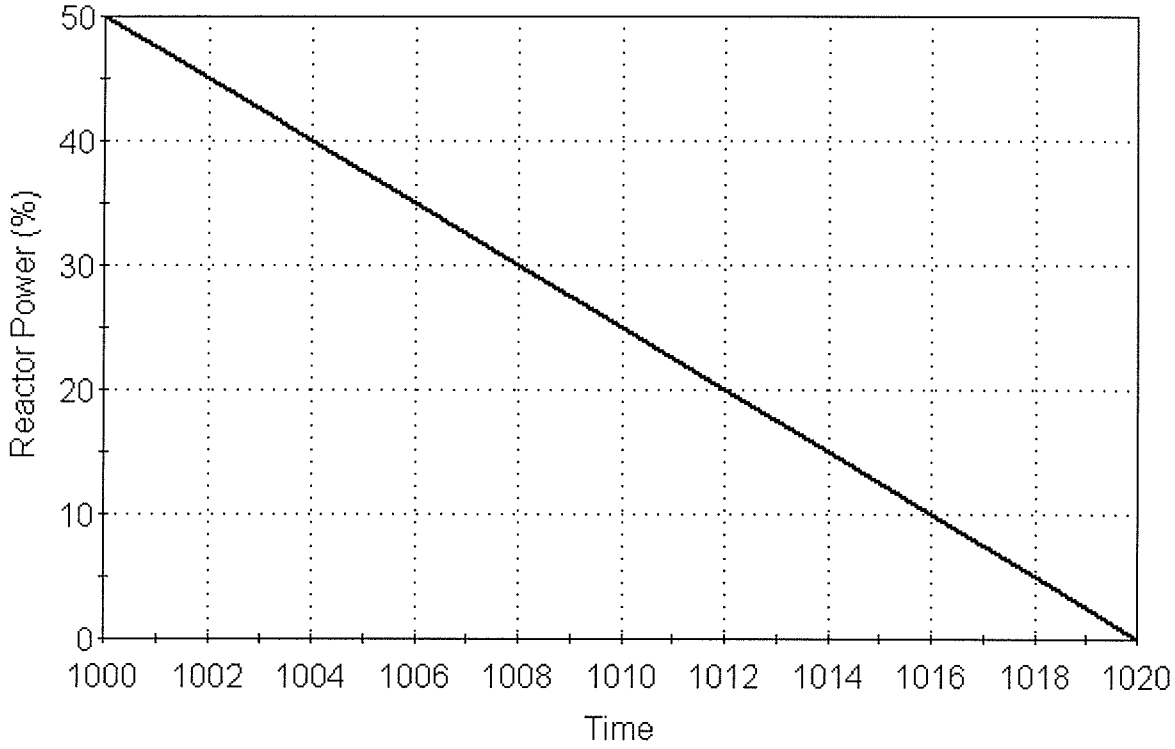
C. Required to be stopped until another SR NI is designated because other NIs are procedurally allowed to be designated

D. Required to be stopped and cannot be resumed until SR 1NI-1 is returned to service because other NIs are NOT procedurally allowed to be designated

2009B ONS SRO NRC Examination - Q 22

Given the following graph:

REACTOR POWER vs. TIME



Which ONE of the following states the LATEST time that 1RIA-59 and 1RIA-60 (Main Steam Line N16 Detectors) will be used to provide a valid indication of Steam Generator tube leakage in accordance with the EOP?

A. 1004

B. 1012

C. 1014

D. 1020

2009B ONS SRO NRC Examination - Q 23

Given the following Unit 1 conditions:

- Reactor in MODE 3
- Unit shutdown in progress
- Containment declared NOT OPERABLE

The MAXIMUM Completion Time allowed by Tech Spec 3.6.1 (Containment) to restore Containment to OPERABLE is __(1)__ AND the HIGHER RCS temperature (degrees F) that would result in being in MODE 4 is __(2)__.

Which ONE of the following completes the statement above?

A. 1. One hour
   2. 245

B. 1. Immediately
   2. 245

C. 1. One hour
   2. 255

D. 1. Immediately
   2. 255

2009B ONS SRO NRC Examination - Q 24

Given the following Unit 1 conditions:

Initial conditions:

- Time = 1200
- Reactor power = 100%
- 1A steam generator tube leak = 2.1 gpd stable
- RCS activity = 0.25 µCi/ml DEI increasing

Current conditions:

- Time = 1400
- NO change in 1A SG tube leak rate
- RCS activity = 0.65 µCi/ml DEI increasing

Which ONE of the following describes the response of the radiation monitors between 1200 and 1400?

A. 1RIA-59 (N-16 monitor) and 1RIA-40 (CSAE Off-gas) increased.

B. 1RIA-16 (Main Steam Line Monitor) and 1RIA-40 increased.

C. 1RIA-59 increased while1RIA-40 remained constant.

D. 1RIA-16 increased while 1RIA-40 remained constant.

2009B ONS SRO NRC Examination - Q 25

Given the following Unit 1 conditions:

Initial conditions:

- Reactor power = 100%
- Turbine Building Flood occurs

Current conditions:

- AP/10 (Turbine Building Flood) in progress

1) CCW pumps are tripped __(1)__ in accordance with AP/10.

2) Core decay heat is removed using __(2)__ in accordance with the EOP.

Which ONE of the following completes the statements above?

A. 1. to establish gravity flow through CCW-8
   2. SSF-ASW

B. 1. to establish gravity flow through CCW-8
   2. HPI Forced Cooling

C. 1. as part of isolating the most probable source of the flooding
   2. SSF-ASW

D. 1. as part of isolating the most probable source of the flooding
   2. HPI Forced Cooling

2009B ONS SRO NRC Examination - Q 26

Given the following Unit 1 conditions:

- Reactor tripped from 100% power due to SBLOCA
- Subcooling Margin = 0°F stable

Which ONE of the following is the reason the EOP directs increasing SG levels to the Loss of Subcooling Margin Setpoint level?

A. Establish a large secondary side inventory in support of a rapid RCS cooldown.

B. Establish a large secondary side inventory to ensure that a loss of coupling will NOT occur if a momentary loss of EFDW occurs.

C. Ensure a secondary water level higher than the primary water level inside the SG tubes to establish boiler condenser mode heat transfer

D. Ensure a secondary side inventory sufficient to minimize the consequences of a total loss of feedwater during boiler condenser mode heat transfer

2009B ONS SRO NRC Examination - Q 27

Given the following Unit 1 conditions:

Initial conditions:

- Time = 1200
- Reactor power = 100%
- 1TA and 1TB lockout occurs

Current condition:

- Time = 1300
- Plant cooldown in progress

1) The source of feedwater to the steam generators is __(1)__.

2) Steam generator levels are __(2)__.

Which ONE of the following completes the statements above?

A. 1. Main Feedwater
   2. 25 inches SUR

B. 1. Main Feedwater
   2. 50% OR

C. 1. Emergency Feedwater
   2. 30 inches XSUR

D. 1. Emergency Feedwater
   2. 240 inches XSUR

2009B ONS SRO NRC Examination - Q 28

Given the following Unit 1 conditions:

- EOP in progress
- RCP starting interlock jumpers installed on 1A1 RCP

Which ONE of the following will PREVENT the start of the 1A1 RCP?

A. CC-7 and CC-8 are closed

B. Total Seal Injection flow = 20 gpm

C. RCP Motor Oil Lift system pressure = 575 psig

D. RCP start switch NOT being placed in the BYPASS position

2009B ONS SRO NRC Examination - Q 29

Given the following Unit 1 conditions:

- Reactor power = 100%
- 1HP-31 (RCP Seal Flow Control) is closed

1) The cooling medium for the RCP MOTORS is __(1)__.

2) ALL cooling for the RCP SEALS __(2)__ been lost.

Which ONE of the following completes the statements above?

A. 1. LPSW
   2. has

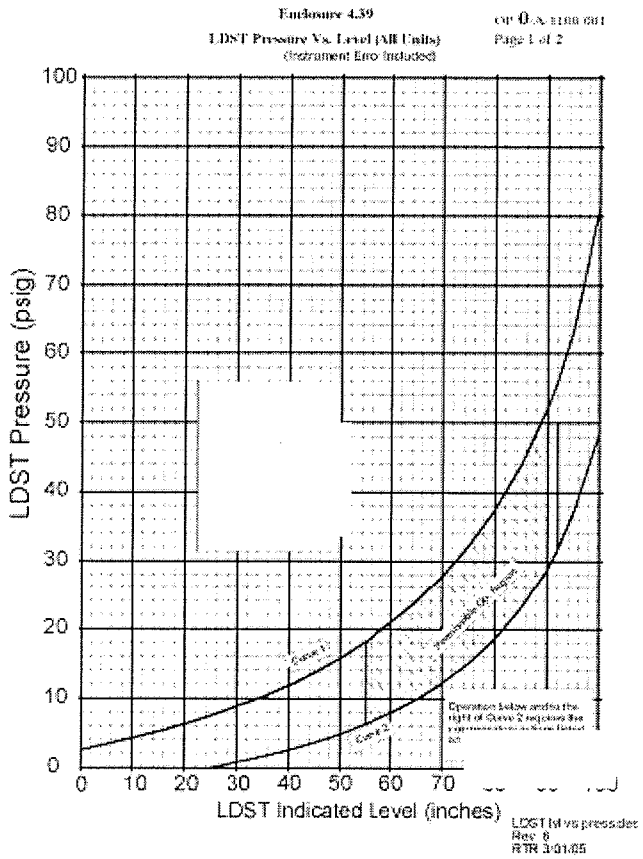
B. 1. LPSW
   2. has NOT

C. 1. CC
   2. has

D. 1. CC
   2. has NOT

2009B ONS SRO NRC Examination - Q 30

Given the following chart:



Which ONE of the following sets of LDST parameters would require declaring BOTH HPI trains inoperable when in MODE 1?

	Level (inches)	/	Pressure (psig)
A.	95	/	10
B.	90	/	35
C.	60	/	10
D.	50	/	30

2009B ONS SRO NRC Examination - Q 31

Given the following Unit 2 conditions:

- Reactor in MODE 3
- RCS Cooldown in progress
- 2A LPI cooler isolated due to cooler leak

Which ONE of the following states the LPI Decay Heat Removal mode that will be used for the INITIAL transition to LPI cooling in accordance with OP/2/A/1104/004 (Low Pressure Injection System)?

A. Series

B. Normal

C. Switchover

D. High Pressure

2009B ONS SRO NRC Examination - Q 32

Given the following Unit 1 conditions:

Initial conditions:

- Reactor power = 100%
- Emergency Feedwater NOT available
- 1C LPI pump NOT available
- 1HP-27 is failed closed

Current conditions:

- Reactor Trips on loss of Main Feedwater
- 1HP-409 will NOT open

1) A __(1)__ would be a precursor to fuel temperatures exceeding design limits.

2) __(2)__ would indicate fuel uncovery has occurred or is imminent.

Which ONE of the following completes the above statements?

A. 1. SBLOCA
   2. Core SCM = (-)2°F stable

B. 1. LBLOCA
   2. Core SCM = (-)2°F stable

C. 1. SBLOCA
   2. Reactor Vessel head level = 12 inches stable

D. 1. LBLOCA
   2. Reactor Vessel head level = 12 inches stable

2009B ONS SRO NRC Examination - Q 33

Given the following Unit 1 conditions:

- Reactor power = 100%
- Loss of ALL feedwater occurs

1) The MINIMUM RCS pressure (psig) at which Quench Tank level would begin to INCREASE is __(1)__.

2) The MINIMUM Quench Tank Pressure (psig) at which Containment pressure would begin to INCREASE is __(2)__.

Which ONE of the following completes the statements above?

A. 1. 2450
   2. 49

B. 1. 2450
   2. 55

C. 1. 2500
   2. 49

D. 1. 2500
   2. 55

2009B ONS SRO NRC Examination - Q 34

1) __(1)__ would result in an increase in CC Cooler outlet temperature.

2) CC Cooler Outlet temperature can be monitored from the Unit 1 Control Room using the __(2)__.

Which ONE of the following completes the statements above?

A. 1. Throttling open 1HP-7 (Letdown Control)
   2. OAC indication ONLY

B. 1. Throttling open 1HP-7 (Letdown Control)
   2. OAC AND temperature gage in Control Room

C. 1. Placing 1HP-14 (LDST Bypass) in "BLEED"
   2. OAC indication ONLY

D. 1. Placing 1HP-14 (LDST Bypass) in "BLEED"
   2. OAC AND temperature gage in Control Room

2009B ONS SRO NRC Examination - Q 35

1) A __(1)__ will result in the CC surge tank level DECREASING.

2) The level that is maintained on the CC Surge Tank Level gage on 1AB3 in accordance with AP/20 (Loss of Component Cooling) is __(2)__ inches.

Which ONE of the following completes the statements above?

A. 1. Letdown cooler leak
   2. 12-18

B. 1. Letdown cooler leak
   2. 18-30

C. 1. CRD Cooling Coil leak
   2. 12-18

D. 1. CRD Cooling Coil leak
   2. 18-30

2009B ONS SRO NRC Examination - Q 36

Given the following Unit 2 conditions:

Initial conditions:

- Reactor power = 100%
- 2A MSLB occurs inside containment

Current conditions:

- RB Pressure peaked at 2.6 psig and is slowly decreasing
- Pressurizer Level decreased to 32" and is now 82" increasing
- RCS pressure decreased to 1540 psig and is now 1920 psig slowly increasing

1) Pressurizer heater banks 2 thru 4 are __(1)__.

2) ES digital channels __(2)__ have actuated.

Which ONE of the following completes the statements above?

A. 1. Off
   2. 1 and 2 ONLY

B. 1. On
   2. 1 and 2 ONLY

C. 1. Off
   2. 1 thru 6

D. 1. On
   2. 1 thru 6

2009B ONS SRO NRC Examination - Q 37

Given the following Unit 1 conditions:

Initial conditions:

- Reactor power = 100%

Current conditions:

- 1B1 Reactor coolant pump trips

Which ONE of the following describes the RPS trip that will prevent exceeding the DNBR safety limit?

A. High flux

B. Flux/Pump

C. Flux/Flow/Imbalance

D. High RCS Temperature

2009B ONS SRO NRC Examination - Q 38

Given the following Unit 1 conditions:

- Reactor power = 100%
- 1A CC pump operating

Which ONE of the following contains conditions, either of which will result in a trip of the 1A CC pump independently of the other?

A. 1. CC Surge Tank level = 10 inches decreasing
   2. ES 1 thru 6 actuation

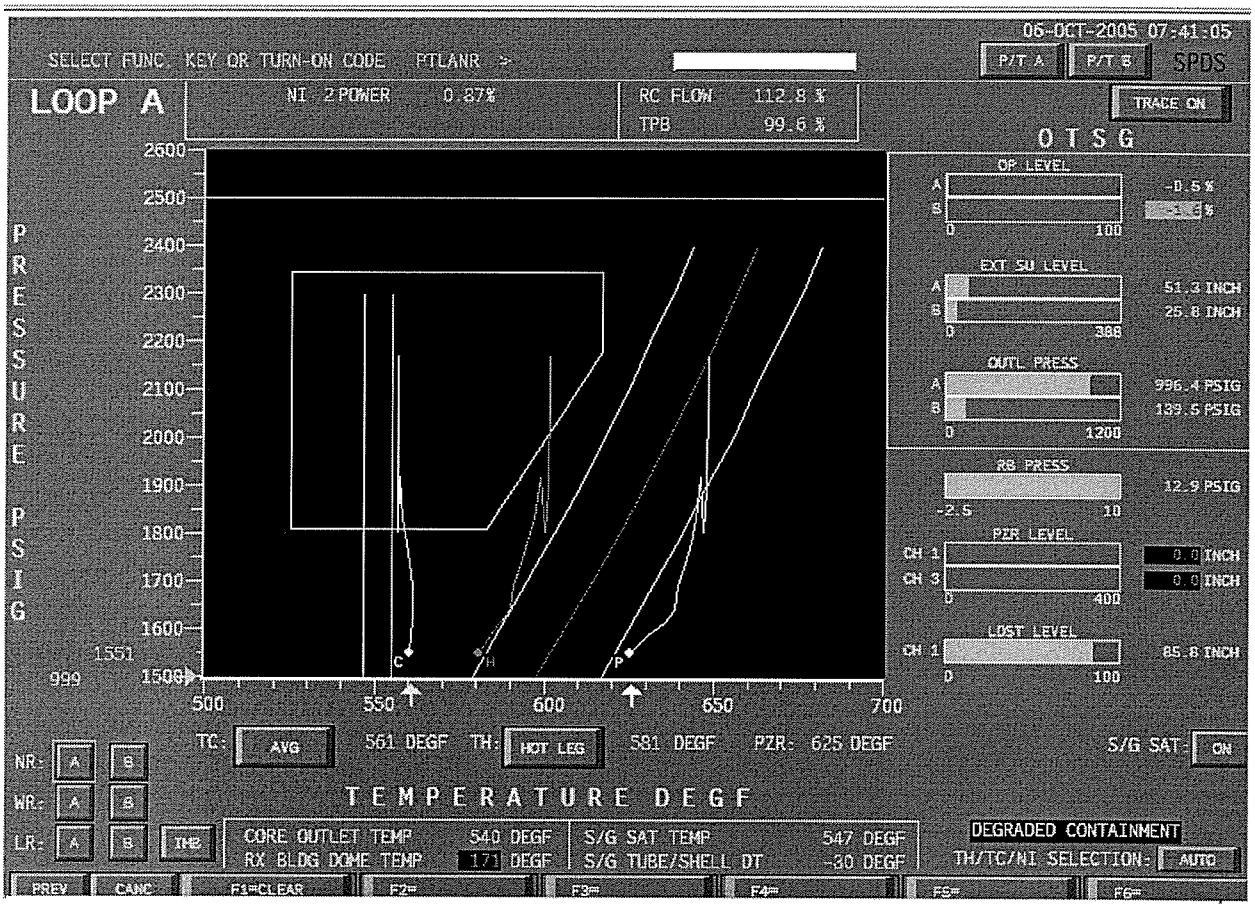
B. 1. CC Surge Tank level = 10 inches decreasing
   2. 1XN de-energized

C. 1. Primary IA compressor trips
   2. Closure of 1CC-7

D. 1. ES 1 thru 6 actuation
   2. Closure of 1CC-7

2009B ONS SRO NRC Examination - Q 39

Given the following Unit 1 conditions:



- Main Steam Line Break occurs from 100% power
- Plant responds as indicated above:

Which ONE of the following states the MINIMUM Pressurizer level (inches) that is <u>required</u> by Rule 5 (Main Steam Line Break) once Pressurizer level returns on scale?

A. > 80

B. > 100

C. > 180

D. > 220

2009B ONS SRO NRC Examination - Q 40

Given the following Unit 1 conditions:

Initial conditions:

- Reactor power = 100%

Current conditions:

- Large Break LOCA occurs
- 1A RBS pump did NOT start

1) The RB Spray system __(1)__ perform its safety function.

2) EOP Enclosure 5.1 (ES Actuation) directs the RO to __(2)__.

Which ONE of the following completes the statements above?

A. 1. can
   2. immediately start the 1A RBS pump

B. 1. can
   2. notify the SRO to evaluate starting the 1A RBS pump

C. 1. can NOT
   2. immediately start the 1A RBS pump

D. 1. can NOT
   2. notify the SRO to evaluate starting the 1A RBS pump

2009B ONS SRO NRC Examination - Q 41

Given the following Unit 1 conditions:

1SA-1

	1	2	3	4	5	6	7	8	9	10	11	12
A	RP CHANNEL A TRIP	RP CHANNEL A LOW PRESS TRIP	RP CHANNEL A FLUX/IMB/FLOW TRIP	RP CHANNEL A HIGH TEMP TRIP	RP CHANNEL A PRESS/TEMP TRIP	RP CHANNEL A HIGH PRESS TRIP	RP CHANNEL A RCP / FLUX TRIP	RP CHANNEL A HIGH FLUX TRIP	RP CHANNEL A R.B. HIGH PRESS TRIP	ES CHANNEL 1 TRIP	ES CHANNEL 5 TRIP	ICS LOSS OF ACS POWER FUSE BLOWN
B	RP CHANNEL B TRIP	RP CHANNEL B LOW PRESS TRIP	RP CHANNEL B FLUX/IMB/FLOW TRIP	RP CHANNEL B HIGH TEMP TRIP	RP CHANNEL B PRESS/TEMP TRIP	RP CHANNEL B HIGH PRESS TRIP	RP CHANNEL B RCP / FLUX TRIP	RP CHANNEL B HIGH FLUX TRIP	RP CHANNEL B R.B. HIGH PRESS TRIP	ES CHANNEL 2 TRIP	ES CHANNEL 6 TRIP	ICS AUTO/HAND POWER FUSE BLOWN
C	RP CHANNEL C TRIP	RP CHANNEL C LOW PRESS TRIP	RP CHANNEL C FLUX/IMB/FLOW TRIP	RP CHANNEL C HIGH TEMP TRIP	RP CHANNEL C PRESS/TEMP TRIP	RP CHANNEL C HIGH PRESS TRIP	RP CHANNEL C RCP / FLUX TRIP	RP CHANNEL C HIGH FLUX TRIP	RP CHANNEL C R.B. HIGH PRESS TRIP	ES CHANNEL 3 TRIP	ES CHANNEL 7 TRIP	LP INJECTION PUMP "A" DIFF. PRESS LOW
D	RP CHANNEL D TRIP	RP CHANNEL D LOW PRESS TRIP	RP CHANNEL D FLUX/IMB/FLOW TRIP	RP CHANNEL D HIGH TEMP TRIP	RP CHANNEL D PRESS/TEMP TRIP	RP CHANNEL D HIGH PRESS TRIP	RP CHANNEL D RCP / FLUX TRIP	RP CHANNEL D HIGH FLUX TRIP	RP CHANNEL D R.B. HIGH PRESS TRIP	ES CHANNEL 4 TRIP	ES CHANNEL 8 TRIP	LP INJECTION PUMP "B" DIFF. PRESS LOW
E	CRD SEQUENCE FAULT	CRD TRIP BKR A TRIP	CRD TRIP BKR B TRIP	CRD TRIP BKR C TRIP	CRD TRIP BKR D TRIP	CRD ELECTRONIC TRIP E	CRD ELECTRONIC TRIP F	RC PUMP 1A1 OIL TANK LEVEL HIGH	RC PUMP 1A2 OIL TANK LEVEL HIGH	RC PUMP 1B1 OIL TANK LEVEL HIGH	RC PUMP 1B2 OIL TANK LEVEL HIGH	LP INJECTION PUMP "C" DIFF. PRESS LOW

Initial conditions:

- Reactor power = 45% stable

Current conditions:

- Reactor power = <1% WR decreasing
- Core SCM = 0°F stable
- RCS pressure = 140 psig decreasing
- Reactor Building pressure = 16.4 psig increasing
- 1SA-1 alarms as indicated above

Which ONE of the following describes actions <u>required</u> by the EOP?

A. Secure running LPI pumps ONLY (Encl. 5.1 ES Actuation)

B. Manually actuate ES Digital Channels 7 & 8 ONLY (Encl. 5.1 ES Actuation)

C. Secure running LPI pumps <u>and</u> Feed to LOSCM setpoint with Emergency Feedwater (Encl. 5.1 ES Actuation and Rule 2 Loss of SCM)

D. Manually actuate ES Digital Channels 7 & 8 <u>and</u> Feed to LOSCM setpoint with Emergency Feedwater (Encl. 5.1 ES Actuation and Rule 2 Loss of SCM)

2009B ONS SRO NRC Examination - Q 42

Which ONE of the following describes the reason that the allowed RCS cooldown rate is more restrictive when performing a natural circulation cooldown?

A. Minimize the possibility of flashing in the RCS Hot Legs

B. Minimize the possibility of drawing a steam bubble in the reactor vessel head area

C. Ensure thermal stresses on reactor vessel inner wall remain within design limits

D. Ensure thermal stresses on the Auxiliary Pressurizer spray line remain within design limits

2009B ONS SRO NRC Examination - Q 43

Given the following Unit 1 conditions:

Initial Conditions:

- Reactor power = 100%
- Two AFIS pressure transmitters on 1A SG fail high

Current Conditions:

- 1A Main Steam Line break occurs
- 1A SG pressure = 480 psig rapidly DECREASING

1) The status of AFIS is __(1)__.

2) Rule 5 (Main Steam Line Break) will give direction to __(2)__.

Which ONE of the following completes the statements above?

A. 1. actuated
   2. open 1AS-40 while closing 1MS-47

B. 1. actuated
   2. select OFF for 1A MD EFDW Pump

C. 1. NOT actuated
   2. open 1AS-40 while closing 1MS-47

D. 1. NOT actuated
   2. select OFF for 1A MD EFDW Pump

2009B ONS SRO NRC Examination - Q 44

1) The power supply to the 3B MDEFWP is __(1)__.

2) The 3B MDEFWP motor is cooled by __(2)__.

Which ONE of the following completes the statements above?

A. 1. 3TD
   2. LPSW

B. 1. 3TD
   2. Air

C. 1. 3TE
   2. LPSW

D. 1. 3TE
   2. Air

2009B ONS SRO NRC Examination - Q 45

Given the following Unit 1 conditions:

Initial conditions:

- 450 EFPD
- Reactor trip from 45%
- HPI Forced Cooling in progress

Current conditions:

- TDEFWP available
- Enclosure 5.40 (Recovery From HPI Forced Cooling) in progress

Which ONE of the following would result in a DECREASE in the amount of feedwater flow indicated by Enclosure 5.13 (Total Feedwater Flow Required to Match NSSS Heat) when determining the INITIAL feed rate required to remove decay heat?

A. Recovery delayed 1 hour due to proximity to 3 psig RB pressure

B. Event occurred at 25 EFPD instead of 450 EFPD

C. Initial trip was from 25% power instead of 45%

D. Three RCP's operating prior to initiating HPI FC

2009B ONS SRO NRC Examination - Q 46

Given the following Unit 2 conditions:

- Reactor power = 100%
- 2TD de-energizes

Which ONE of the following remains available?

A. C LPSW Pump

B. 2C HPI Pump

C. 2B LPI Pump

D. 2C RBCU

2009B ONS SRO NRC Examination - Q 47

Which ONE of the following are loads that are BOTH powered by the Power Batteries?

A. Mulsifyer systems and TDEFWP Auxiliary Oil Pump

B. PCB-9 Control Power and CCW-8 (CCW Emergency Discharge to the tailrace)

C. Main FWPT Auxiliary Oil Pump and PCB-9 Control Power

D. TDEFWP Auxiliary Oil Pump and CCW-8

2009B ONS SRO NRC Examination - Q 48

Given the following conditions:

- ACB-3 closed
- Switchyard isolation has occurred

1) A __(1)__ will PREVENT KHU-1 from Emergency starting.

2) The LOWER KHU-2 output voltage that will allow ACB-2 to close is __(2)__.

Which ONE of the following completes the statements above?

A. 1. STARTUP INHIBIT
   2. 12.6KV

B. 1. NORMAL LOCKOUT
   2. 12.6KV

C. 1 STARTUP INHIBIT
   2. 12.3KV

D. 1. NORMAL LOCKOUT
   2. 12.3KV

2009B ONS SRO NRC Examination - Q 49

Given the following conditions:

- All three Oconee units are in Mode 1
- Keowee Hydro Unit 2 is generating to the grid
- ALL KHU 2 Forebay level sensors are NOT operable

Which ONE of the following describes the action(s) required in accordance with SLC's?

A. Manually input Forebay level into the digital governor AND suspend commercial operation of the KHU's

B. Manually input Forebay level into the digital governor ONLY

C. Suspend commercial operation of the KHU's ONLY

D. Declare BOTH KHU's NOT Operable

2009B ONS SRO NRC Examination - Q 50

Given the following Unit 1 conditions:

Initial conditions:

- Reactor power = 100%
- 1A GWD tank release in progress
- 1RIA-38 OOS

Current conditions:

- Loss of power to RM-80 skid of 1RIA-37
- 1SA8/B9 RM PROCESS MONITOR RADIATION HIGH in alarm
- 1SA8/B10 RM PROCESS MONITOR FAULT in alarm

1) 1GWD-4 (A GWD TANK DISCHARGE) will __(1)__.

2) The GWD tank release may __(2)__ in accordance with OP/1-2/A/1104/018 (GWD System).

Which ONE of the following completes the statements above?

A. 1. remain open
   2. continue as long as 1RIA-37 is re-energized within one hour

B. 1. automatically close
   2. be re-initiated as long as 1RIA-37 is re-energized within one hour

C. 1. remain open
   2. continue as long as two independent samples agree prior to restarting the release

D. 1. automatically close
   2. be re-initiated as long as two independent samples agree prior to restarting the release

2009B ONS SRO NRC Examination - Q 51

Given the following Unit 1 conditions:

Initial conditions:

- Reactor power = 100%
- A and B LPSW pumps operating

Current conditions:

- LPSW line break in Turbine Building Basement
- LPSW Pressure = 65 psig decreasing

The LPSW header pressure setpoint at which LPSW to the RBCU's will automatically isolate is __(1)__ psig AND Unit 1&2 LPSW system AND Unit 3 LPSW system will be cross-tied __(2)__ in accordance with AP/24 (Loss of LPSW).

Which ONE of the following completes the statement above?

A. 1. 18
   2. ANYTIME normal system pressure can NOT be restored

B. 1. 18
   2. ONLY if NO Unit 1 and 2 LPSW pumps are available

C. 1. 25
   2. ANYTIME normal system pressure can NOT be restored

D. 1. 25
   2. ONLY if NO Unit 1 and 2 LPSW pumps are available

2009B ONS SRO NRC Examination - Q 52

Given the following Unit 2 conditions:

Initial conditions:

- Reactor power = 100%
- 2FDW-35 and 44 (Startup FDW Control Valves) are 100% open
- 2FDW-32 and 41 (Main FDW Control Valves) are 60% open

Current conditions:

- IA pressure = 60 psig slowly decreasing

1) 2FDW-35 and 2FDW-44 have failed __(1)__.

2) 2FDW-32 and 2FDW-41 __(2)__.

Which ONE of the following completes the statements above?

A. 1. closed
   2. are controlling Main FDW flow.

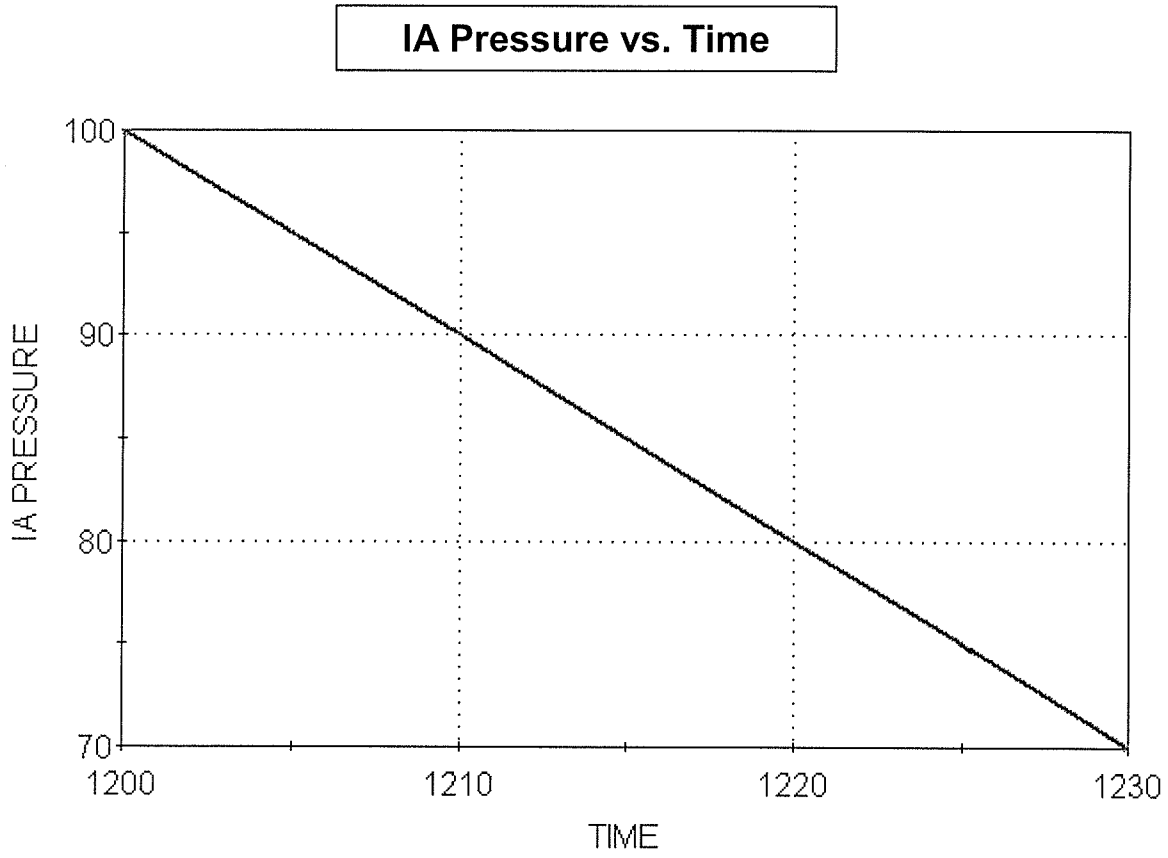
B. 1. closed
   2. have failed closed

C. 1. open
   2. have failed open

D. 1. "as is"
   2. have failed "as is"

2009B ONS SRO NRC Examination - Q 53

Given the following conditions:



Which ONE of the following describes the EARLIEST time at which SA-141 (SA to IA Controller) will automatically open?

A. 1215

B. 1212

C. 1210

D. 1207

2009B ONS SRO NRC Examination - Q 54

Which ONE of the following will receive a signal to close if ES Digital Channel 2 was inadvertently actuated on Unit 1?

A. 1LPSW-21

B. 1LPSW-6

C. 1GWD-12

D. 1LWD-2

2009B ONS SRO NRC Examination - Q 55

Given the following Unit 3 conditions:

- Reactor power = 100%
- Personnel Hatch Interlock Mechanism NOT OPERABLE

Which ONE of the following describes the MINIMUM actions required in accordance with Tech Spec 3.6.2 (Containment Air Locks)?

A. Declare Containment NOT Operable immediately

B. Declare BOTH doors in the air lock NOT Operable immediately

C. Verify ALL OPERABLE doors in the air lock are closed within one hour

D. Verify at least ONE OPERABLE door in the air lock is closed within one hour

2009B ONS SRO NRC Examination - Q 56

Which ONE of the following conditions would result in a Control Rod Out Inhibit?

A. Reactor power = 52% and Group 6 Rod 5 becomes misaligned by > 9 inches

B. Reactor power = 52% and Control Rod Group 1 loses its Group Out Limit

C. Count rate = 675 cps increasing and ALL Safety rods NOT withdrawn to their Group Out Limit and Safety Rods Out Bypass NOT enabled

D. Count rate = 675 cps increasing and Wide Range NI-1 startup rate = 2.1 dpm

2009B ONS SRO NRC Examination - Q 57

Given the following Unit 1 conditions:

- Reactor power = 100%
- ALL Absolute Position Indications (API) for Group 6 Rod 3 fail to "0"

1) A CRD Asymmetric Fault __(1)__ be generated.

2) The action that would be required in accordance with Tech Specs if <u>all</u> API <u>and</u> RPI indications for Group 6 Rod 3 become unavailable is to __(2)__.

Which ONE of the following completes the statements above?

A. 1. will
   2. immediately declare Group 6 Rod 3 NOT operable

B. 1. will
   2. begin RCS boration within 15 minutes

C. 1. will NOT
   2. immediately declare Group 6 Rod 3 NOT operable

D. 1. will NOT
   2. begin RCS boration within 15 minutes

2009B ONS SRO NRC Examination - Q 58

1) The power supply to 1NI-6 Power Range detector is __(1)__.

2) The RPS channel will __(2)__ if that power supply is deenergized.

Which ONE of the following completes the statements above?

A. 1. 1KVIB
   2. trip

B. 1. 1KVIB
   2. NOT trip

C. 1. 1KVIC
   2. trip

D. 1. 1KVIC
   2. NOT trip

2009B ONS SRO NRC Examination - Q 59

Given the following Unit 1 conditions:

Initial conditions:

- Reactor power = 100%

Current conditions:

- SBLOCA occurs
- IMAs & Symptom check complete

Which ONE of the following states the Core Exit Thermocouples (CETCs) that are being used by the Core Subcooling Monitors in ICCM?

A. ALL 47 CETCs

B. ONLY 24 qualified CETCs

C. ONLY 12 highest qualified CETCs for that train

D. ONLY 5 highest qualified CETCs for that train

2009B ONS SRO NRC Examination - Q 60

Which ONE of the following will result in automatic closure of RB Purge isolation valves 3PR-1,2,3,4,5,AND 6?

A. 3RIA-45 HIGH alarm

B. 3RIA-46 ALERT alarm

C. Manual actuation of ES channels 5 and 6

D. Low RCS pressure actuation of ES channels 1 and 2

2009B ONS SRO NRC Examination - Q 61

Given the following Unit 1 conditions:

Initial Conditions:

- Reactor in MODE 6
- Fuel Transfer Canal is full
- LPI in Normal Decay Heat Removal
- 1B Spent Fuel Cooling Pump operating and aligned to Refueling Cooling Mode in accordance with OP/1/A/1102/015 (Filling and Draining FTC)

Current conditions:

- 1SF-1 and 1SF-2 are inadvertently closed

Which ONE of the following describes the:

1) response of the Fuel Transfer Canal level?

2) position of 1SF-1 and 1SF-2 once the unit returns to MODE 1?

A. 1. Remain constant
   2. Open

B. 1. Remain constant
   2. Remain closed

C. 1. Decrease
   2. Open

D. 1. Decrease
   2. Remain closed

2009B ONS SRO NRC Examination - Q 62

Given the following Unit 2 conditions:

Initial conditions:

- Reactor in MODE 6
- Fuel Transfer Canal slightly above 21.34' mark on canal wall
- RB Hatch closed
- RB Purge is operating
- 2SF-1 AND 2SF-2 are open

Current conditions:

- RB Purge fan trips

Which ONE of the following predicts the response of actual Fuel Transfer Canal level over the next ten (10) minutes?

A. Initially increases then returns to previous level

B. Increases then remains constant

C. Initially decreases then returns to previous level

D. Decreases then remains constant

2009B ONS SRO NRC Examination - Q 63

Given the following Unit 1 conditions:

Initial conditions:

- Reactor power = 100%

Current conditions:

- Reactor trip
- "Trip Confirm" signal NOT generated by the Diamond

1) The status of the Turbine Load Status Flag is __(1)__.

2) The TBVs will control at __(2)__.

Which ONE of the following completes the statements above?

A. 1. TRUE
   2. setpoint

B. 1. TRUE
   2. setpoint + 125 psig

C. 1. FALSE
   2. setpoint

D. 1. FALSE
   2. setpoint + 125 psig

2009B ONS SRO NRC Examination - Q 64

Given the following Unit 1 conditions:

- Reactor power = 100%

1) The Hotwell Pump suction valve breakers are __(1)__.

2) The suction valve control switches are located in the __(2)__.

Which ONE of the following completes the statements above?

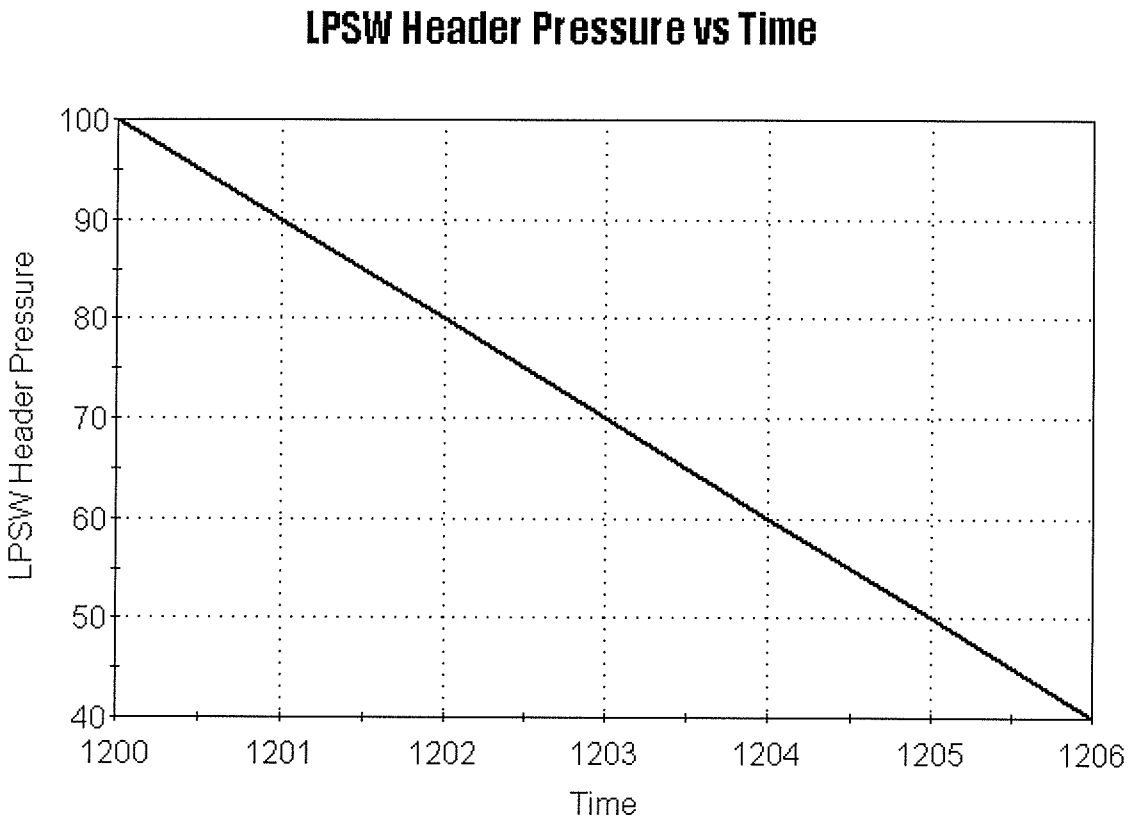
A. 1. open
   2. Turbine Building Basement adjacent to Hotwell Pump sump

B. 1. open
   2. Control Room on 1VB3

C. 1. closed
   2. Turbine Building Basement adjacent to Hotwell Pump sump

D. 1. closed
   2. Control Room on 1VB3

2009B ONS SRO NRC Examination - Q 65



- 3A LPSW pump operating
- 3B LPSW pump in AUTO
- Unit 3 LPSW system transient occurs as described in graph above

Which ONE of the following states the <u>earliest</u> time that LPSW header pressure will START the timer for the Standby LPSW pump auto start circuit?

A. 1201

B. 1203

C. 1204

D. 1205

2009B ONS SRO NRC Examination - Q 66

Which ONE of the following lists of items will always require a plant page (except for unanticipated automatic starts and emergency situations) in accordance with OMP 1-02 (Rules of Practice)?

A. ALL 4911 calls
Closing PCB-18

B. ALL AP entries
Starting and stopping 1B1 RCP

C. Starting a 4160V Motor
Closing PCB-18

D. Starting and stopping 1B1 RCP
Starting a 4160V Motor

2009B ONS SRO NRC Examination - Q 67

Which ONE of the following describes what should be used in the case of a large Hydrogen leak in accordance with OP/1/A/1106/017 (Hydrogen System) to maintain Hydrogen concentration below the lower flammability limit?

A. CO_2

B. Water

C. Halon

D. Foam fire retardant

2009B ONS SRO NRC Examination - Q 68

Given the following conditions:

Initial conditions:

- Time = 1200
- Unit 1 reactor power = 100%
- Unit 3 reactor power = 100%
- Site Area Emergency declared on Unit 2

Current conditions:

- Time = 1400
- TSC/OSC remain activated
- Unit 3 requires an NEO to place Unit 3 BWST in recirc

Which ONE of the following describes who will communicate with the NEO in accordance with OMP 1-07 (Operations Emergency Response Organization) to have them place the Unit 3 BWST in recirc?

A. OSM liaison

B. OSC Ops liaison

C. Unit 3 CRSRO

D. Unit 3 RO

2009B ONS SRO NRC Examination - Q 69

Given the following Unit 1 conditions:

- Reactor power = 100%
- Group 8 movement is required for imbalance control

Which ONE of the following:

1) describes the MINIMUM actions required prior to using the joystick to move Group 8 rods?

2) states if Group 7 rods will respond to neutron error while group 8 is selected?

A. 1. Place Diamond panel Group Select switch to Group 8 ONLY
   2. Yes

B. 1. Place Diamond panel Group Select switch to Group 8 ONLY
   2. No

C. 1. Select SEQUENCE OVERRIDE <u>and</u> Place Diamond panel Group Select switch to Group 8
   2. Yes

D. 1. Select SEQUENCE OVERRIDE <u>and</u> Place Diamond panel Group Select switch to Group 8
   2. No

2009B ONS SRO NRC Examination - Q 70

Given the following Unit 1 conditions:

Initial conditions:

- Reactor power = 100%

Current conditions:

- Keowee Emergency Start channels 1 and 2 inoperable

1) The MAXIMUM Completion Time allowed by Tech Specs to declare both KHU's NOT OPERABLE is __(1)__.

2) The resulting Tech Spec Required Action(s) is/are to __(2)__.

Which ONE of the following completes the statements above?

A. 1. Immediately
   2. energize at least ONE Standby Bus from a Lee Combustion Turbine via an isolated path within 1 hour

B. 1. One hour
   2. energize at least ONE Standby Bus from a Lee Combustion Turbine via an isolated path within 1 hour

C. 1. Immediately
   2. energize BOTH Standby Buses from a Lee Combustion Turbine via an isolated path within 1 hour

D. 1. One hour
   2. energize BOTH Standby Buses from a Lee Combustion Turbine via an isolated path within 1 hour

2009B ONS SRO NRC Examination - Q 71

Given the following Unit 1 conditions:

- Reactor power = 100%

Which ONE of the following describes a condition that would <u>require</u> entry into a Tech Spec ACTIONS table?

A. UST level = 7.6 feet

B. BWST level = 45.3 feet

C. 1C RPS NR Th fails high

D. 230KV Dacus Black and White lines isolated

2009B ONS SRO NRC Examination - Q 72

Which ONE of the following describes the actions taken and why, in accordance with the SGTR tab, to limit the activity released to the atmosphere?

A. The Condensate Steam Air Ejectors are lined up to the Main Steam system to prevent cross contamination.

B. The TD EFDW pump is placed in "Pull To Lock" to prevent feeding the affected SG with contaminated water.

C. The Auxiliary Steam systems for all three units are split to prevent cross contamination.

D. Core SCM is minimized to reduce the primary to secondary leak rate.

2009B ONS SRO NRC Examination - Q 73

Which ONE of the following describes:

1) a condition where Emergency Dose Limits (EDLs) would be in effect?

2) the MAXIMUM whole body EDL (Rem) if performing EOP actions that are NOT to save a life or protect valuable property?

A. 1. 50 gpm Primary to Secondary leak
   2. 5

B. 1. 50 gpm Primary to Secondary leak
   2. 10

C. 1. 50 gpm RCS hot leg weld leak
   2. 5

D. 1. 50 gpm RCS hot leg weld leak
   2. 10

2009B ONS SRO NRC Examination - Q 74

Which ONE of the following describes:

1) an acceptable reason for the Procedure Director to interrupt an operator who is performing Rule 2?

2) the bases behind limiting the interruptions of operators who are performing Rules?

A. 1. Perform a Crew Brief
   2. Minimize Operator errors

B. 1. Inform the operator that ACC Conditions exist
   2. Minimize Operator errors

C. 1. Perform a Crew Brief
   2. Ensures completion of Time Critical Actions

D. 1. Inform the operator that ACC Conditions exist
   2. Ensures completion of Time Critical Actions

2009B ONS SRO NRC Examination - Q 75

The following are Log entries from the Emergency Coordinator (EC/OSM) Log following an event at Oconee Unit 2.

1112 – EC/OSM declared a General Emergency based on 2RIA-57 readings.

1117 - EC/OSM provided the following offsite protective recommendations to the Offsite Communicator:

- Evacuation of Pickens A0, A1, B1, C1, and Oconee A0, D1, E1, F1
- Shelter of Pickens A2, B2, C2, and Oconee D2, E2, F2.

1120 - EC/OSM signed message form to provide offsite recommendations.

1122 - Station Manager in the control room began turnover to TSC.

1125- Present time of day.

Which ONE of the following describes the MAXIMUM number of minutes the Offsite Communicator has to initiate the notifications to State and Counties <u>from the present time</u>?

A. 2

B. 7

C. 10

D. 15

2009B ONS SRO NRC Examination - Q 76

Given the following Unit 3 conditions:

Time = 1200
- Unit shutdown in progress due to SGTL = 12.5 gpm in 3A SG
- Reactor power = 95% decreasing
- 3RC-68 fails Open

Time = 1201
- Reactor power = 90% decreasing
- MANUAL Reactor Trip initiated

Time = 1203
- ALL SCM's = 0°F

Time = 1204
- ES 1&2 actuate

Time = 1212
- ALL SCM's = 23°F increasing
- Pressurizer level = 400" stable
- 3RIA-57 = 22 R/hr slowly increasing

1) The current Emergency Plan Classification level is __(1)__.

2) An escalation of the classification would be required if __(2)__.

Which ONE of the following completes the statements above?

REFERENCE PROVIDED

A. 1. Alert
   2. 3A SG has one MSRV stuck open after the trip

B. 1. Alert
   2. 3RIA-57 = 84 R/hr at 1220 hrs

C. 1. Site Area Emergency
   2. 3A SG has one MSRV stuck open after the trip

D. 1. Site Area Emergency
   2. 3RIA-57 = 84 R/hr at 1220 hrs

2009B ONS SRO NRC Examination - Q 77

Given the following Unit 1 conditions:

- Power Range NI's = 90% stable
- Core Thermal Power = 90% stable
- Group 7 = 95% withdrawn
- Power maneuvering plan directs adding 200 gal. 1B BHUT to insert Group 7 to 90% withdrawn

1) After Control Rods respond to the water addition, NI's will become __(1)__.

2) The RPS trips associated with Power Range NI's help prevent exceeding the __(2)__ limit.

Which ONE of the following completes the statements above?

A. 1. Conservative
   2. minimum DNBR

B. 1. Conservative
   2. maximum ejected rod worth

C. 1. NON Conservative
   2. minimum DNBR

D. 1. NON Conservative
   2. maximum ejected rod worth

2009B ONS SRO NRC Examination - Q 78

Given the following Unit 1 conditions:

Initial conditions:

- Reactor in MODE 6
- Fuel Transfer Canal full
- SF-1 and SF-2 are open

Current conditions:

- Fuel Transfer Canal level slowly decreasing
- RBNS level increasing

Which one of the following describes the action(s) that would be performed FIRST, in accordance with AP/26 (Loss of Decay Heat Removal), and why?

A. Secure ALL LPI Pumps to determine if leak is on discharge of LPI Pumps

B. Secure ALL LPI Pumps in preparation for closing 1SF-1 and 1SF-2

C. Secure SF Cooling pump used for Refueling Cooling Mode to determine if leak is on discharge of SF Cooling Pump

D. Secure SF Cooling pump used for Refueling Cooling Mode in preparation for closing 1SF-1 and 1SF-2

2009B ONS SRO NRC Examination - Q 79

Given the following Unit 1 conditions:

- Reactor power = 100%
- RCS DEI = 0.35 uCi/gm stable
- AP/21 (High Activity in RCS) in progress

Which ONE of the following describes:

1) actions required by AP/21?

2) the procedure used by AP/21 to perform the power reduction when a shutdown is required?

A.
1. Maximize letdown ONLY
2. OP/1/A/1102/004 (Operation at Power)

B.
1. Maximize letdown ONLY
2. AP/29 (Rapid Unit Shutdown)

C.
1. Initiate a power reduction and maximize letdown
2. OP/1/A/1102/004 (Operation at Power)

D.
1. Initiate a power reduction and maximize letdown
2. AP/29 (Rapid Unit Shutdown)

2009B ONS SRO NRC Examination - Q 80

Given the following Unit 1 conditions:

Initial Conditions:

- Reactor power = 100%
- 1RIA-60 = 20 gpm stable

Current conditions:

- 1TA AND 1TB lockout occurs

1) The __(1)__ tab will be used to stabilize the plant following the reactor trip.

2) The __(2)__ tab will direct the unit cooldown to LPI.

Which ONE of the following completes the statements above?

A. 1. Subsequent Actions
   2. SGTR

B. 1. SGTR
   2. SGTR

C. 1. Subsequent Actions
   2. Forced Cooldown

D. 1. SGTR
   2. Forced Cooldown

2009B ONS SRO NRC Examination - Q 81

Given the following Unit 1 conditions:

- Reactor in MODE 3
- Tcold = 429°F decreasing
- 1A SG isolated due to a Main Steam Line Break in the Turbine Building
- Steam Generator Tube Rupture in the 1B SG
- 1B SG being steamed for cooldown

Which ONE of the following describes action(s) required if the 1B SG level <u>reaches</u> the Main Steam line in accordance with the EOP?

A. Unisolate and feed the 1A SG

B. Initiate Encl. 5.22 (SG Blowdown) for 1B SG

C. Stop steaming the 1B SG and initiate HPI Forced Cooling

D. Maximize 1B SG steaming rate even if TS cooldown rates are exceeded

2009B ONS SRO NRC Examination - Q 82

Given the following Unit 1 conditions:

Initial conditions:

- Reactor power = 100%
- 1SA6/B2 INVERTER 1DID SYSTEM TROUBLE actuated

Current conditions:

- NEO reports:
  - 1SA13/A8 INVERTER 1DID INPUT VOLTAGE LOW actuated
  - Inverter 1DID output voltage low

1) The MINIMUM action(s) required to restore the 1DID inverter to OPERABLE in accordance with Tech Spec 3.8.6 (Vital Inverters-Operating) is/are to restore DC input voltage __(1)__.

2) The status of 1KVID prior to any operator actions being performed is __(2)__.

Which ONE of the following completes the statements above?

A. 1. ONLY
   2. NOT energized

B. 1. AND re-connect to 1KVID
   2. NOT energized

C. 1. ONLY
   2. Energized

D. 1. AND re-connect to 1KVID
   2. Energized

2009B ONS SRO NRC Examination - Q 83

Given the following Unit 1 conditions:

Initial conditions:

- Time = 1200
- Reactor power = 100%
- TDEFWP inoperable
- Tornado results in Station Blackout

Current conditions:

- Time = 1215
- 4160V power restored

1) The Abnormal Procedure being used to align feed to the SG's between 1200 and 1215 is __(1)__.

2) The EOP tab the Procedure Director will direct once power has been restored will be __(2)__.

Which ONE of the following completes the statements above?

A. 1. AP/6 (Natural Disaster)
   2. Subsequent Actions

B. 1. AP/6 (Natural Disaster)
   2. Loss Of Heat Transfer

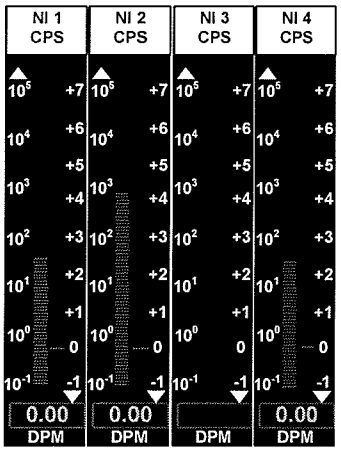
C. 1. AP/25 (Standby Shutdown Facility Emergency Operating Procedure)
   2. Subsequent Actions

D. 1. AP/25 (Standby Shutdown Facility Emergency Operating Procedure)
   2. Loss Of Heat Transfer

2009B ONS SRO NRC Examination - Q 84

Given the following Unit 1 conditions:

SOURCE RANGE



- Time = 1200
- Reactor in MODE 5
- NI-3 is NOT operable
- NI readings as indicated above:

1) Source Range __(1)__ will be used to log the hourly Source Range NI reading for PT/1/A/0600/001 (Periodic Instrument Surveillance) Enclosure 13.4 (Mode 5).

2) The source range instrumentation should be used as the primary power indication in accordance with the basis of Tech Spec 3.3.9 (Source Range Neutron Flux) __(2)__.

Which ONE of the following completes the statements above?

REFERENCE PROVIDED

A. 1. NI-1
   2. ONLY when power levels are <4E-4% RTP on Wide Range

B. 1. NI-1
   2. when in MODES 2, 3, 4, and 5

C. 1. NI-2
   2. ONLY when power levels are <4E-4% RTP on Wide Range

D. 1. NI-2
   2. when in MODES 2, 3, 4, and 5

2009B ONS SRO NRC Examination - Q 85

Given the following Unit 1 conditions:

- Reactor in MODE 3
- Startup in progress
- Safety Rod Groups 1 & 2 fully withdrawn
- Safety Rod Group 3 withdrawal in progress
- Wide Range NI-1 and NI-3 are disabled

1) The MINIMUM number of Wide Range NI's that are required by Tech Spec 3.3.10 (Wide Range Neutron Flux) __(1)__ OPERABLE.

2) The reason Wide Range Neutron Monitors are used to trigger operator actions in accordance with the basis of Tech Spec 3.3.10 is to __(2)__.

Which ONE of the following completes the statements above?

A. 1. are NOT
   2. prevent exceeding DNBR Safety Limits due to reactivity transients in MODES 3, 4, and 5

B. 1. are
   2. prevent exceeding DNBR Safety Limits due to reactivity transients in MODES 3, 4, and 5

C. 1. are NOT
   2. minimize the impact of a reactivity transient that could otherwise result in RPS actuation

D. 1. are
   2. minimize the impact of a reactivity transient that could otherwise result in RPS actuation

2009B ONS SRO NRC Examination - Q 86

Given the following Unit 1 conditions:

Initial conditions:

- Reactor in MODE 3
- 1A2 and 1B2 RCPs operating

Current conditions:

- All SCM's = 25°F stable
- Pressurizer level = 200 inches stable
- 1A2 RCP Motor
  - Upper Guide Bearing temperature = 195°F slowly increasing
  - Stator Temperature = 220°F slowly increasing
- 1B2 RCP Motor
  - Upper Guide Bearing temperature = 185°F slowly increasing
  - Stator Temperature = 255°F slowly increasing
- RCS leakage = 180 gpm stable

1) AP/16 (Abnormal Reactor Coolant Pump Operation) requires tripping the __(1)__ RCP.

2) The __(2)__ tab will provide the guidance for a plant cooldown if ALL RCP's are secured.

Which ONE of the following completes the statements above?

A. 1. 1A2
   2. Forced Cooldown

B. 1. 1A2
   2. LOCA Cooldown

C. 1. 1B2
   2. Forced Cooldown

D. 1. 1B2
   2. LOCA Cooldown

2009B ONS SRO NRC Examination - Q 87

Given the following Unit 2 conditions:

- Reactor in MODE 3
- Startup in progress
- RCS temperature = 532°F stable
- RCS pressure = 2155 psig stable
- Safety Rods Groups 1-4 fully withdrawn

Which ONE of the following would allow a MAXIMUM of 4 hours to make an Emergency Notification System notification to the NRC in accordance with NSD 202 (Reportability)?

REFERENCE PROVIDED

A. Operating Main Feedwater Pump trips

B. BOTH EFDW trains declared NOT OPERABLE

C. 1HP-120 failure resulting in RCS pressure increasing to 2375 psig

D. Turbine Bypass Valve failure resulting in RCS pressure decreasing to 1545 psig

2009B ONS SRO NRC Examination - Q 88

Given the following Unit 1 conditions:

- OP/1/A/1103/002, (Filling and Venting RCS) Enclosure 4.14 (Establishing Pzr Steam Bubble And RCS Final Vent) in progress
- Quench Tank level = 84 inches
- Quench Tank pressure = 0.5 psig
- The Pressurizer is vented to the Quench Tank for 30 minutes

1) The HIGHER of the Quench Tank pressures below that would indicate that Pzr Steam Bubble Formation is complete is __(1)__ psig.

2) Entry into MODE 4 if 1GWD-12 (QUENCH TANK VENT INSIDE RB) failed to close during system re-alignment following Pzr steam bubble formation is allowed __(2)__ in accordance with Tech Specs.

Which ONE of the following completes the statements above?

REFERENCE PROVIDED

A. 1. 0.6
   2. ONLY once 1GWD-12 is restored to OPERABLE

B. 1. 0.6
   2. if 1GWD-13 (QUENCH TANK VENT OUTSIDE RB) is closed and de-activated

C. 1. 2.5
   2. ONLY once 1GWD-12 is restored to OPERABLE

D. 1. 2.5
   2. if 1GWD-13 (QUENCH TANK VENT OUTSIDE RB) is closed and de-activated

2009B ONS SRO NRC Examination - Q 89

Given the following Unit 2 conditions:

Initial Conditions:

- Reactor power = 100%
- ACB-4 closed

Current conditions:

- KHU #1 Emergency Lockout occurs

1) PT/0/A/0620/009 (Keowee Hydro Operation) will require an __(1)__ start of KHU #2 as part of its operability verification.

2) Verifying __(2)__ SK breaker(s) can be closed meets the MINIMUM SK breaker requirements when verifying operability of the Underground powerpath.

Which ONE of the following completes the statements above?

A. 1. Automatic
   2. EITHER

B. 1. Automatic
   2. BOTH

C. 1. Emergency
   2. EITHER

D. 1. Emergency
   2. BOTH

2009B ONS SRO NRC Examination - Q 90

Given the following Unit 1 conditions:

- Reactor in MODE 5
- RB Purge in progress
- ALL 1RIA-45 (Unit Vent Particulate Monitor) functions will be unavailable for approximately 45 minutes due to scheduled PM's

Which ONE of the following describes the impact on the ability to continue the RB Purge during the PM's in accordance with SLC 16.11.3 (Radioactive Effluent Monitoring Instrumentation)?

The RB Purge release _____

A. can continue with NO additional actions if RIA outage time does NOT exceed one hour.

B. can continue however it <u>requires</u> 2 independent samples be taken for the time the RIA is not available.

C. can NOT continue because the outage time will exceed the time allowed in accordance with SLC 16.11.3.

D. can NOT continue because 1RIA-45 will NOT be able to terminate RB Purge during the PM's.

2009B ONS SRO NRC Examination - Q 91

Which ONE of the following states the MINIMUM level of approval required to place a fuel assembly into an alternate location other than the original one assigned by the Core Reload Sequence in accordance with MP/0/A/1500/009 (Defueling/Refueling Procedure)?

A. Refueling SRO Assistant

B. Reactor Building SRO

C. Refueling SRO

D. OSM

2009B ONS SRO NRC Examination - Q 92

Given the following Unit 1 conditions:

- Reactor power = 100%

1) __(1)__ would result in actuation of 1SA-18/C5 RM TBS Interlock.

2) __(2)__ gpd is the LOWEST Steam Generator tube leak rate that would require initiating a power decrease in accordance with plant procedures.

Which ONE of the following would complete the statements above?

A. 1. Loss of power to 1RIA-54
   2. 125

B. 1. Loss of power to 1RIA-54
   2. 65

C. 1. 1RIA-54 reaching the ALERT setpoint
   2. 125

D. 1. 1RIA-54 reaching the ALERT setpoint
   2. 65

2009B ONS SRO NRC Examination - Q 93

Given the following Unit 1 conditions:

Initial conditions:

- Steam line break in 1A Steam Generator has occurred from 100%
- Rule 5 (Main Steam Line Break) is complete
- EHT tab in progress

Current conditions:

- Steam line break occurs in 1B Steam Generator

1) The INITIAL EOP procedure path would be to re-perform Rule 5 and __(1)__.

2) The cooldown tab that will be directed by the Procedure Director assuming neither SG can be "Trickle Fed" would be the __(2)__ tab.

Which ONE of the following completes the statements above?

A. 1. Return to beginning of EHT tab
   2. FCD

B. 1. Return to beginning of EHT tab
   2. HPI CD

C. 1. Transfer to LOHT tab
   2. FCD

D. 1. Transfer to LOHT tab
   2. HPI CD

2009B ONS SRO NRC Examination - Q 94

Given the following Unit 1 plant conditions:

- Reactor power = 87% stable
- Chemistry results report DEI = 68 uCi/gm

1) The INITIAL Tech Spec Required Action is to __(1)__.

2) The bases behind the limit on DEI, in accordance with Tech Spec 3.4.11 (RCS Specific Activity), is to ensure __(2)__.

Which ONE of the following completes the statements above?

REFERENCE PROVIDED

A. 1. restore DEI to acceptable limits within 48 hours
   2. doses at the site boundary do not exceed 10CFR100 limits

B. 1. restore DEI to acceptable limits within 48 hours
   2. accessibility of components required to be operated in a post LOCA environment as part of Time Critical Actions

C. 1. be in MODE 3 with Tave < 500°F within 12 hours
   2. doses at the site boundary do not exceed 10CFR100 limits

D. 1. be in MODE 3 with Tave < 500°F within 12 hours
   2. accessibility of components required to be operated in a post LOCA environment as part of Time Critical Actions

2009B ONS SRO NRC Examination - Q 95

Given the following conditions:

- Unit 1 = 100% Power
- Unit 2 = Mode 5
- Unit 3 = 100% Power

1) The MINIMUM RO staffing requirements for the station in accordance with SLC 16.13.1 (Minimum Station Staffing Requirements) is __(1)__.

2) The SLC requirements __(2)__ include maintaining the ability to staff the SSF when required.

Which ONE of the following completes the statements above?

REFERENCE PROVIDED

A. 1. 5
   2. do

B. 1. 5
   2. do NOT

C. 1. 6
   2. do

D. 1. 6
   2. do NOT

2009B ONS SRO NRC Examination - Q 96

Given the following Unit 2 conditions:

- Reactor in MODE 3
- Startup in progress
- 2B LPI Train flow instrument NOT operable

1) Declare __(1)__ NOT OPERABLE.

2) Entry into MODE 2 __(2)__ allowed in accordance with Tech Specs.

Which ONE of the following completes the statements above?

A. 1. 2B LPI Train ONLY
   2. is

B. 1. 2B LPI Train ONLY
   2. is NOT

C. 1. 2B LPI Train AND 2B RBS Train
   2. is

D. 1. 2B LPI Train AND 2B RBS Train
   2. is NOT

2009B ONS SRO NRC Examination - Q 97

Given the following Unit 3 conditions:

- Reactor in MODE 6
- Fuel movement is in progress
- Fuel Handling accident occurs
- 3RIA-49 (RB Gas) HIGH alarm actuates

1) AP/9 (Spent Fuel Damage) will direct starting Outside Air Booster Fans __(1)__.

2) The bases behind the MINIMUM Fuel Transfer Canal level, in accordance with Tech Spec 3.9.6 (Fuel Transfer Canal Water Level), is to __(2)__.

Which ONE of the following completes the statements above?

A. 1. 3A and 3B ONLY
   2. reduce Iodine levels following a fuel handling accident to ensure offsite dose is maintained within allowable limits

B. 1. 3A and 3B ONLY
   2. ensure the minimum requirements of the SSF RC Makeup Pump are met for 72 hours

C. 1. A, B, 3A, and 3B
   2. reduce Iodine levels following a fuel handling accident to ensure offsite dose is maintained within allowable limits

D. 1. A, B, 3A, and 3B
   2. ensure the minimum requirements of the SSF RC Makeup Pump are met for 72 hours

2009B ONS SRO NRC Examination - Q 98

Given the following conditions:

Initial conditions:

- Date = 6/1/10
- 1A GWD tank isolated

Current conditions:

- Date = 9/29/10
- NO GWR releases in progress
- 1A GWD tank needs to be released at 1/3 Station Limit

1) The MINIMUM level of approval required for the release is __(1)__.

2) The expected response of 1RIA-46 during the 1A GWD tank release will be to read __(2)__.

Which ONE of the following completes the statements above?

A. 1. ANY SRO
   2. "0"

B. 1. ONLY the OSM
   2. "0"

C. 1. ANY SRO
   2. "on scale" but below the 1RIA-46 HIGH setpoint

D. 1. ONLY the OSM
   2. "on scale" but below the 1RIA-46 HIGH setpoint

2009B ONS SRO NRC Examination - Q 99

Which ONE of the following EAL's would result in the LOWEST level of Emergency Classification that will require on-shift NEO's to report to the OSC?

REFERENCE PROVIDED

A. Dose projection at Site Boundary = 120 mRem TEDE

B. RCS pressure spike exceeded RCS pressure safety limit, DEI= 312 uCi/ml, and H2 concentration in containment is 10.4%

C. Fire in the plant cafeteria that can NOT be extinguished for > 15 minutes

D. Valid Rx trip signal received without a SCRAM AND DSS has inserted Control Rods.

2009B ONS SRO NRC Examination - Q 100

Which ONE Of the following:

1) is the LOWEST Emergency Classification that will <u>require</u> protective action recommendations during the INITIAL notification?

2) states who is allowed to staff the NRC Communicator position in accordance with RP/1000/002 (Control Room Emergency Coordinator Procedure)?

A. 1. General Emergency due to 5 highest CETC's = 1346°F for > 15 minutes
   2. ANY licensed operator

B. 1. Site Area Emergency due to Keowee Dam Failure
   2. ANY licensed operator

C. 1. General Emergency due to 5 highest CETC's = 1346°F for > 15 minutes
   2. SRO ONLY

D. 1. Site Area Emergency due to Keowee Dam failure
   2. SRO ONLY

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CORRECT ANSWER — MARK X — TOTAL ONLY
MARK ONLY ONE

FEED IN THIS DIRECTION

	T	F	A	B	C	D	E
1		●					
2				●			
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4				●			
5			●				
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FEED IN THIS DIRECTION

NUMBER CORRECT	
PERCENT CORRECT	
ROSTER NUMBER	
SCORE	
RESCORE	

SCANTRON

COMBINED POINTS EARNED	
COMBINED PERCENT CORRECT	
LETTER GRADE	
SCORE	
RESCORE	

100 ITEM

NAME KEY

SUBJECT

PERIOD DATE

ANSWER KEY INFO.

# OF KEYS	ITEM COUNT
0 0 0	2
1 1 1	3
2 2	4
3 3	
4 4	
5 5	
6 6	
7 7	
8 8	
9 9	

PERFORMANCE ASSESSMENT

% OF TOTAL SCORE (00 = 100%)			POINTS EARNED		
0 0	0 0	0	0	0	0
1 1	1 1	1	1	1	1
2 2	2 2	2	2	2	2
3 3	3 3			3	3
4 4	4 4			4	4
5 5	5 5			5	5
6 6	6 6			6	6
7 7	7 7			7	7
8 8	8 8			8	8
9 9	9 9			9	9

EQUAL PT VALUE

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BAR CODE

MARKING INSTRUCTIONS

Use a No. 2 Pencil

Ⓐ ● Ⓒ Ⓓ Ⓔ
Fill oval completely

Ⓐ Ⓑ Ⓒ Ⓓ Ⓔ
Erase cleanly

STUDENT ID NUMBER

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4	4	4	4	4	4	4	4	4	4
5	5	5	5	5	5	5	5	5	5
6	6	6	6	6	6	6	6	6	6
7	7	7	7	7	7	7	7	7	7
8	8	8	8	8	8	8	8	8	8
9	9	9	9	9	9	9	9	9	9

Q 76.5'99

Duke Energy Oconee Nuclear Station **Emergency Classification** **Reference Use**	Procedure No. **RP/0/B/1000/001** Revision No. **027** Electronic Reference No. **OX002WOS**

PERFORMANCE

* * * * * * * * * * UNCONTROLLED FOR PRINT * * * * * * * * * *

(ISSUED) - PDF Format

Emergency Classification

NOTE:	This procedure is an implementing procedure to the Oconee Nuclear Site Emergency plan and must be forwarded to Emergency Planning within seven (7) working days of approval.

1. Symptoms

1.1 This procedure describes the immediate actions to be taken to recognize and classify an emergency condition.

1.2 This procedure identifies the four emergency classifications and their corresponding Emergency Action Levels (EALs).

1.3 This procedure provides reporting requirements for non-emergency abnormal events.

1.4 The following guidance is to be used by the Emergency Coordinator/EOF Director in assessing emergency conditions:

1.4.1 Definitions and Acronyms are italicized throughout procedure for easy recognition. The definitions are in Enclosure 4.10 (Definitions/Acronyms).

1.4.2 The Emergency Coordinator/EOF Director shall review all applicable initiating events to ensure proper classification.

1.4.3 The BASIS Document (Volume A, Section D of the Emergency Plan) is available for review if any questions arise over proper classification.

1.4.4 **IF** An event occurs on more than one unit concurrently,

THEN The event with the higher classification will be classified on the Emergency Notification Form.

A. Information relating to the problem(s) on the other unit(s) will be captured on the Emergency Notification Form as shown in RP/0/B/1000/015A, (Offsite Communications From The Control Room), RP/0/B/1000/015B, (Offsite Communications From The Technical Support Center) or SR/0/B/2000/004, (Notification to States and Counties from the Emergency Operations Facility).

1.4.5 **IF** An event occurs,

AND A lower or higher plant operating mode is reached before the classification can be made,

THEN The classification shall be based on the mode that existed at the time the event occurred.

1.4.6 The Fission Product Barrier Matrix is applicable only to those events that occur at Mode 4 (Hot Shutdown) or higher.

A. An event that is recognized at Mode 5 (Cold Shutdown) or lower shall not be classified using the Fission Product Barrier Matrix.

1. Reference should be made to the additional enclosures that provide Emergency Action Levels for specific events (e.g., Severe Weather, *Fire*, Security).

1.5 **IF** A transient event should occur,

THEN Review the following guidance:

1.5.1 **IF** An Emergency Action Level (EAL) identifies a specific duration

AND The Emergency Coordinator/EOF Director assessment concludes that the specified duration is exceeded or will be exceeded, (i.e.; condition cannot be reasonably corrected before the duration elapses),

THEN Classify the event.

1.5.2 **IF** A plant condition exceeding EAL criteria is corrected before the specified duration time is exceeded,

THEN The event is **NOT** classified by that EAL.

A. Review lower severity EALs for possible applicability in these cases.

NOTE:	Reporting under 10CFR50.72 may be required for the following step. Such a condition could occur, for example, if a follow up evaluation of an abnormal condition uncovers evidence that the condition was more severe than earlier believed.

1.5.3 **IF** A plant condition exceeding EAL criteria is not recognized at the time of occurrence, but is identified well after the condition has occurred (e.g.; as a result of routine log or record review)

AND The condition no longer exists,

THEN An emergency shall **NOT** be declared.

- Refer to NSD 202 for reportability

1.5.4 **IF** An emergency classification was warranted, but the plant condition has been corrected prior to declaration and notification

THEN The Emergency Coordinator must consider the potential that the initiating condition (e.g.; Failure of Reactor Protection System) may have caused plant damage that warrants augmenting the on shift personnel through activation of the Emergency Response Organization.

A. **IF** An ***Unusual Event*** condition exists,

THEN Make the classification as required.

1. The event may be terminated in the same notification or as a separate termination notification.

B. **IF** An ***Alert, Site Area Emergency,*** or ***General Emergency*** condition exists,

THEN Make the classification as required,

AND Activate the Emergency Response Organization.

1.6 Emergency conditions shall be classified as soon as the Emergency Coordinator/EOF Director assessment determines that the Emergency Action Levels for the Initiating Condition have been exceeded.

2. Immediate Actions

2.1 Determine the operating mode that existed at the time the event occurred prior to any protection system or operator action initiated in response to the event.

2.2 **IF** The unit is at Mode 4 (Hot Shutdown) or higher

AND The condition/event affects fission product barriers,

THEN GO TO Enclosure 4.1, (Fission Product Barrier Matrix).

2.2.1 Review the criteria listed in Enclosure 4.1, (Fission Product Barrier Matrix) and make the determination if the event should be classified).

2.3 Review the listing of enclosures to determine if the event is applicable to one of the categories shown.

2.3.1 **IF** One or more categories are applicable to the event,

2.3.2 **THEN** Refer to the associated enclosures.

2.3.3 Review the EALs and determine if the event should be classified.

A. **IF** An EAL is applicable to the event,

THEN Classify the event as required.

2.4 **IF** The condition requires an emergency classification,

THEN Initiate the following:

- for Control Room - RP/0/B/1000/002, (Control Room Emergency Coordinator Procedure)
- for TSC - RP/0/B/1000/019, (Technical Support Center Emergency Coordinator Procedure)
- for EOF - SR/0/B/2000/003, (Activation of the Emergency Operations Facility)

2.5 Continue to review the emergency conditions to assure the current classification continues to be applicable.

3. Subsequent Actions

3.1 Continue to review the emergency conditions to assure the current classification continues to be applicable.

4.0 Enclosures

	Enclosures	Page Number
4.1	Fission Product Barrier Matrix	7
4.2	System Malfunctions	8
4.3	Abnormal Rad Levels/Radiological Effluents	10
4.4	Loss Of Shutdown Functions	12
4.5	Loss of Power	14
4.6	Fires/Explosions And Security Actions	15
4.7	Natural Disasters, Hazards, And Other Conditions Affecting Plant Safety	18
4.8	Radiation Monitor Readings For Emergency Classification	21
4.9	Unexpected/Unplanned Increase In Area Monitor Readings	22
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4.11	Operating Modes Defined In Improved Technical Specifications	27
4.12	Instructions For Using Enclosure 4.1	28
4.13	References	30

Enclosure 4.1
Fission Product Barrier Matrix

DETERMINE THE APPROPRIATE CLASSIFICATION USING THE TABLE BELOW: ADD POINTS TO CLASSIFY. SEE NOTE BELOW

RCS BARRIERS (BD 5-7)		FUEL CLAD BARRIERS (BD 8-9)		CONTAINMENT BARRIERS (BD 10-12)	
Potential Loss (4 Points)	Loss (5 Points)	Potential Loss (4 Points)	Loss (5 Points)	Potential Loss (1 Point)	Loss (3 Points)
RCS Leakrate ≥ 160 gpm	RCS Leak rate that results in a loss of subcooling.	Average of the 5 highest CETC ≥ 700° F	Average of the 5 highest CETC ≥ 1200° F	CETC ≥ 1200° F ≥ 15 minutes **OR** CETC ≥ 700° F ≥ 15 minutes with a *valid* RVLS reading 0″	Rapid unexplained containment pressure decrease after increase **OR** containment pressure or sump level not consistent with LOCA
SGTR ≥ 160 gpm		*Valid* RVLS reading of 0″ NOTE: RVLS is **NOT** valid if one or more RCPs are running **OR** if LPI pump(s) are running **AND** taking suction from the LPI drop line.	Coolant activity ≥ 300 µCi/ml DEI	RB pressure ≥ 59 psig **OR** RB pressure ≥ 10 psig and no RBCU or RBS	Failure of secondary side of SG results in a direct opening to the environment with SG Tube Leak ≥ 10 gpm in the **SAME** SG
Entry into the PTS (Pressurized Thermal Shock) Operation NOTE: PTS is entered under either of the following: • A cooldown below 400°F @ > 100°F/hr. has occurred. • HPI has operated in the injection mode while **NO** RCPs were operating.	1RIA 57 or 58 reading ≥ 1.0 R/hr 2 RIA 57 reading ≥ 1.6 R/hr 2 RIA 58 reading ≥ 1.0 R/hr 3RIA 57 or 58 reading ≥ 1.0 R/hr		Hours Since SD / RIA 57 R/hr / OR RIA 58 R/hr 0 - <0.5 / ≥ 300 / ≥ 150 0.5 - < 2.0 / ≥ 80 / ≥ 40 2.0 - 8.0 / ≥ 32 / ≥ 16	Hours Since SD / RIA 57 R/hr / OR RIA 58 R/hr 0 - < 0.5 / ≥ 1800 / ≥ 860 0.5 - < 2.0 / ≥ 400 / ≥ 195 2.0 - 8.0 / ≥ 280 / ≥ 130	SG Tube Leak ≥ 10 gpm exists in one SG. **AND** the other SG has secondary side failure that results in a direct opening to the environment **AND** is being fed from the affected unit.
HPI Forced Cooling	RCS pressure spike ≥ 2750 psig			Hydrogen concentration ≥ 9%	Containment isolation is incomplete and a release path to the environment exists
Emergency Coordinator/EOF Director judgment	Emergency Coordinator/EOF Director judgment	Emergency Coordinator/EOF Director judgment	Emergency Coordinator/EOF Director judgment	Emergency Coordinator/EOF Director judgment	Emergency Coordinator/EOF Director judgment

UNUSUAL EVENT (1-3 Total Points)	ALERT (4-6 Total Points)	SITE AREA EMERGENCY (7-10 Total Points)	GENERAL EMERGENCY (11-13 Total Points)
OPERATING MODE: 1, 2, 3, 4	**OPERATING MODE:** 1, 2, 3, 4	**OPERATING MODE:** 1, 2, 3, 4	**OPERATING MODE:** 1, 2, 3, 4
4.1.U.1 Any potential loss of Containment	4.1.A.1 Any potential loss or loss of the RCS	4.1.S.1 Loss of any two barriers	4.1.G.1 Loss of any two barriers and potential loss of the third barrier
4.1.U.2 Any loss of containment	4.1.A.2 Any potential loss or loss of the Fuel Clad	4.1.S.2 Loss of one barrier and potential loss of either RCS or Fuel Clad Barriers	4.1.G.2 Loss of all three barriers
		4.1.S.3 Potential loss of both the RCS and Fuel Clad Barriers	

NOTE: An event with multiple events could occur which would result in the conclusion that exceeding the loss or potential loss threshold is **IMMINENT** (i.e., within 1-3 hours). In this IMMINENT LOSS situation, use judgment and classify as if the thresholds are exceeded.

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
1. *RCS LEAKAGE* (BD 14) OPERATING MODE: 1, 2, 3, 4 A. Unidentified leakage ≥ 10 gpm B. Pressure boundary leakage ≥ 10 gpm C. Identified leakage ≥ 25 gpm • Includes SG tube leakage 2. *UNPLANNED* LOSS OF MOST OR ALL SAFETY SYSTEM ANNUNCIATION/ INDICATION IN CONTROL ROOM FOR > 15 MINUTES (BD 15) OPERATING MODE: 1, 2, 3, 4 A.1 *Unplanned* loss of > 50% of the following annunciators on one unit for > 15 minutes: Units 1 & 3 1 SA1, 2, 3, 4, 5, 6, 7, 8, 9, 14, 15, 16, & 18 3 SA1, 2, 3, 4, 5, 6, 7, 8, 9, 14, 15, 16, & 18 Unit 2 2 SA1, 2, 3, 4, 5, 6, 7, 8, 9, 14, 15, & 16 AND A.2 Loss of annunciators or indicators requires additional personnel (beyond normal shift complement) to safely operate the unit (CONTINUED)	1. *UNPLANNED* LOSS OF MOST OR ALL SAFETY SYSTEM ANNUNCIATION/ INDICATION IN CONTROL ROOM (BD 19) OPERATING MODE: 1, 2, 3, 4 A.1 *Unplanned* loss of > 50% of the following annunciators on one unit for > 15 minutes: Units 1 & 3 1 SA1, 2, 3, 4, 5, 6, 7, 8, 9, 14, 15, 16, & 18 3 SA1, 2, 3, 4, 5, 6, 7, 8, 9, 14, 15, 16, & 18 Unit 2 2 SA1, 2, 3, 4, 5, 6, 7, 8, 9, 14, 15, & 16 AND A.2 Loss of annunciators or indicators requires additional personnel (beyond normal shift complement) to safely operate the unit AND A.3 *Significant plant transient in progress* OR A.4 Loss of the OAC and ALL PAM indications (END)	1. INABILITY TO MONITOR A SIGNIFICANT TRANSIENT IN PROGRESS (BD 21) OPERATING MODE: 1, 2, 3, 4 A.1 *Unplanned* loss of > 50% of the following annunciators on one unit for > 15 minutes: Units 1 & 3 1 SA1, 2, 3, 4, 5, 6, 7, 8, 9, 14, 15, 16, & 18 3 SA1, 2, 3, 4, 5, 6, 7, 8, 9, 14, 15, 16, & 18 Unit 2 2 SA1, 2, 3, 4, 5, 6, 7, 8, 9, 14, 15, & 16 AND A.2 A *significant transient* is in progress AND A.3 Loss of the OAC and ALL PAM indications AND A.4 *Inability to directly monitor* any one of the following functions: 1. Subcriticality 2. Core Cooling 3. Heat Sink 4. RCS Integrity 5. Containment Integrity 6. RCS Inventory (END)	

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
3. INABILITY TO REACH REQUIRED SHUTDOWN WITHIN LIMITS (BD 16) OPERATING MODE: 1, 2, 3, 4 A. Required operating mode not reached within TS LCO action statement time 4. *UNPLANNED* LOSS OF ALL ONSITE OR OFFSITE COMMUNICATIONS (BD 17) OPERATING MODE: All A. Loss of all onsite communications capability (Plant phone system, PA system, Pager system, Onsite Radio system) affecting ability to perform Routine operations B. Loss of all onsite communications capability (Selective Signaling, NRC ETS lines, Offsite Radio System, AT&T line) affecting ability to communicate with offsite authorities. 5. FUEL CLAD DEGRADATION (BD 18) OPERATING MODE: All: A. DEI - >5µCi/ml (END)			

Enclosure 4.3
Abnormal Rad Levels/Radiological Effluent

UNUSUAL EVENT

1 ANY *UNPLANNED* RELEASE OF GASEOUS OR LIQUID RADIOACTIVITY TO THE ENVIRONMENT THAT EXCEEDS TWO TIMES THE SLC LIMITS FOR 60 MINUTES OR LONGER (BD 23)

OPERATING MODE: All

A. *Valid* indication on radiation monitor RIA 33 of ≥ 4.06E+06 cpm for > 60 minutes (See Note 1)

B. *Valid* indication on radiation monitor RIA 45 of ≥ 9.35E+05 cpm for > 60 minutes (See Note 1)

C. Liquid effluent being released exceeds two times SLC 16.11.1 for > 60 minutes as determined by Chemistry Procedure

D. Gaseous effluent being released exceeds two times SLC 16.11.2 for > 60 minutes as determined by RP Procedure

NOTE 1: If monitor reading is sustained for the time period indicated in the EAL AND the required assessments (procedure calculations) cannot be completed within this period, declaration must be made on the *valid* Radiation Monitor reading.

(CONTINUED)

ALERT

1. ANY *UNPLANNED* RELEASE OF GASEOUS OR LIQUID RADIOACTIVITY TO THE ENVIRONMENT THAT EXCEEDS 200 TIMES RADIOLOGICAL TECHNICAL SPECIFICATIONS FOR 15 MINUTES OR LONGER (BD 28)

OPERATING MODE: All

A. *Valid* indication on RIA 46 of ≥ 2.09E+04 cpm for >15 minutes (See Note 1)

B RIA 33 HIGH Alarm

AND

Liquid effluent being released exceeds 200 times the level of SLC 16.11.1 for > 15 minutes as determined by Chemistry Procedure

C. Gaseous effluent being released exceeds 200 times the level of SLC 16.11.2 for >15 minutes as determined by RP Procedure

(CONTINUED)

SITE AREA EMERGENCY

1. BOUNDARY DOSE RESULTING FROM ACTUAL/IMMINENT RELEASE OF GASEOUS ACTIVITY (BD 32)

OPERATING MODE: All

A. *Valid* reading on RIA 46 of ≥ 2.09E+05 cpm for >15 minutes (See Note 2)

B. *Valid* reading on RIA 57 or 58 as shown on Enclosure 4.8 (See Note 2)

C. Dose calculations result in a dose projection at the *site boundary* of:

≥ 100 mRem TEDE or 500 mRem CDE adult thyroid

D. Field survey results indicate *site boundary* dose rates exceeding ≥100 mRad/hr expected to continue for more than one hour

OR

Analyses of field survey samples indicate adult thyroid dose commitment of ≥ 500 mRem CDE (3.84 E^{-7} µCi/ml) for one hour of inhalation

NOTE 2: If actual Dose Assessment cannot be completed within 15 minutes, then the *valid* radiation monitor reading should be used for emergency classification.

(CONTINUED)

GENERAL EMERGENCY

1. BOUNDARY DOSE RESULTING FROM ACTUAL/ IMMINENT RELEASE OF GASEOUS ACTIVITY (BD 36)

OPERATING MODE: All

A. *Valid* reading on RIA 46 of ≥ 2.09E+06 cpm for ≥15 minutes (See Note 3)

B. *Valid* reading on RIA 57 or 58 as shown on Enclosure 4.8 (See Note 3)

C. Dose calculations result in a dose projection at the *site boundary* of:

≥ 1000 mRem TEDE

OR

≥ 5000 mRem CDE adult thyroid

D. Field survey results indicate *site boundary* dose rates exceeding ≥1000 mRad/hr expected to continue for more than one hour

OR

Analyses of field survey samples indicate adult thyroid dose commitment of ≥ 5000 mRem CDE for one hour of inhalation

NOTE 3: If actual Dose Assessment cannot be completed within 15 minutes, then the *valid* radiation monitor reading should be used for emergency classification.

(END)

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
2 UNEXPECTED INCREASE IN PLANT RADIATION OR AIRBORNE CONCENTRATION (BD 25) OPERATING MODE: All A. LT 5 reading 14" and decreasing with makeup not keeping up with leakage WITH fuel in the core B. *Valid* indication of *uncontrolled* water decrease in the SFP or fuel transfer canal with all fuel assemblies remaining covered by water AND Unplanned *Valid* RIA 3, 6 or Portable Area Monitor readings increase. C. 1 R/hr radiation reading at one foot away from a damaged storage cask located at the ISFSI D. *Valid* area monitor readings exceeds limits stated in Enclosure 4.9. NOTE: This Initiating Condition is also located in Enclosure 4.4., (Loss of Shutdown Functions). High radiation levels will also be seen with this condition. (END)	2. RELEASE OF RADIOACTIVE MATERIAL OR INCREASES IN RADIATION LEVELS THAT IMPEDES OPERATION OF SYSTEMS REQUIRED TO MAINTAIN SAFE OPERATION OR TO ESTABLISH OR MAINTAIN COLD SHUTDOWN (BD 30) OPERATING MODE: All A. *Valid* radiation reading ≥ 15 mRad/hr in CR, CAS, or Radwaste CR B. *Unplanned*/unexpected *valid* area monitor readings exceed limits stated in Enclosure 4.9 3. MAJOR DAMAGE TO IRRADIATED FUEL OR LOSS OF WATER LEVEL THAT HAS OR WILL RESULT IN THE UNCOVERING OF IRRADIATED FUEL OUTSIDE THE REACTOR VESSEL (BD 31) OPERATING MODE: All A. *Valid* RIA 3*, 6, 41, OR 49* HIGH Alarm * - Applies to Mode 6 and No Mode Only B. HIGH Alarm for portable area monitors on the main bridge or SFP bridge C Report of visual observation of irradiated fuel uncovered D. Operators determine water level drop in either the SFP or fuel transfer canal will exceed makeup capacity such that irradiated fuel will be uncovered NOTE: This Initiating Condition is also located in Enclosure 4.4., (Loss of Shutdown Functions). High radiation levels will also be seen with this condition. (END)	2. LOSS OF WATER LEVEL IN THE REACTOR VESSEL THAT HAS OR WILL UNCOVER FUEL IN THE REACTOR VESSEL (BD 35) OPERATING MODE: 5, 6 A.1 Failure of heat sink causes loss of Mode 5 (Cold Shutdown) condition AND LT 5 indicates 0 inches after initiation of RCS makeup B. Failure of heat sink causes loss of Mode 5 (Cold Shutdown) condition AND Either train ultrasonic level indication less than 0 inches and decreasing after initiation of RCS makeup NOTE: This Initiating Condition is also located in Enclosure 4.4., (Loss of Shutdown Functions). High radiation levels will also be seen with this condition. (END))	

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
(CONTINUE TO NEXT PAGE)	1. FAILURE OF RPS TO COMPLETE OR INITIATE A Rx SCRAM (BD 41) OPERATING MODE 1, 2, 3 A *Valid* reactor trip signal received or required WITHOUT automatic scram AND A.1.1 DSS has inserted Control Rods OR A.1.2 Manual trip from the Control Room is successful and reactor power is less than 5% and decreasing 2. INABILITY TO MAINTAIN PLANT IN COLD SHUTDOWN (BD 43) OPERATING MODE: 5, 6 A.1 Loss of LPI and/or LPSW AND A.2 Inability to maintain RCS temperature below 200° F as indicated by either of the following: A.2.1 RCS temperature at the LPI Pump Suction OR A.2.2 Average of the 5 highest CETCs as indicated by ICCM display OR A.2.3 Visual observation (CONTINUED)	1. FAILURE OF RPS TO COMPLETE OR INITIATE A Rx SCRAM (BD 45) OPERATING MODE: 1, 2 A.1 *Valid* reactor trip signal received or required WITHOUT automatic scram AND A.2 DSS has NOT inserted Control Rods AND A.3 Manual trip from the Control Room was NOT successful in reducing reactor power to less than 5% and decreasing 2. COMPLETE LOSS OF FUNCTION NEEDED TO ACHIEVE OR MAINTAIN HOT SHUTDOWN (BD 46) OPERATING MODE: 1, 2, 3, 4 A. Average of the 5 highest CETCs ≥1200° F shown on ICCM B. Unable to maintain reactor subcritical C. EOP directs feeding SG from SSF ASWP or station ASWP (CONTINUED)	1. FAILURE OF RPS TO COMPLETE AUTOMATIC SCRAM AND MANUAL SCRAM NOT SUCCESSFUL WITH INDICATION OF CORE DAMAGE (BD 48) OPERATING MODE: 1, 2 A.1 *Valid* Rx trip signal received or required WITHOUT automatic scram AND A.2 Manual trip from the Control Room was NOT successful in reducing reactor power to < 5% and decreasing AND A.3 Average of the 5 highest CETCs ≥1200° F on ICCM (END)

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
1. UNEXPECTED INCREASE IN PLANT RADIATION OR AIRBORNE CONCENTRATION (BD 39) OPERATING MODE: All A. LT 5 reading 14" and decreasing with makeup not keeping up with leakage WITH fuel in the core B. *Valid* indication of *uncontrolled* water decrease in the SFP or fuel transfer canal with all fuel assemblies remaining covered by water AND Unplanned *Valid* RIA 3, 6 or Portable Area Monitor readings increase. C. 1 R/hr radiation reading at one foot away from a damaged storage cask located at the ISFSI D. *Valid* area monitor readings exceeds limits stated in Enclosure 4.9. NOTE: This Initiating Condition is also located in Enclosure 4.3., (Abnormal Rad Levels/Radiological Effluent). High radiation levels will also be seen with this condition. (END)	3. MAJOR DAMAGE TO IRRADIATED FUEL OR LOSS OF WATER LEVEL THAT HAS OR WILL RESULT IN THE UNCOVERING OF IRRADIATED FUEL OUTSIDE THE REACTOR VESSEL (BD 44) OPERATING MODE: All A. *Valid* RIA 3*, 6, 41, OR 49* HIGH Alarm *Applies to Mode 6 and No Mode Only B. HIGH Alarm for portable area monitors on the main bridge or SFP bridge C Report of visual observation of irradiated fuel uncovered D. Operators determine water level drop in either the SFP or fuel transfer canal will exceed makeup capacity such that irradiated fuel will be uncovered NOTE: This Initiating Condition is also located in Enclosure 4.3, (Abnormal Rad Levels/Radiological Effluent). High radiation levels will also be seen with this condition. (END)	3. LOSS OF WATER LEVEL IN THE REACTOR VESSEL THAT HAS OR WILL UNCOVER FUEL IN THE REACTOR VESSEL (BD 47) OPERATING MODE: 5, 6 A.1 Failure of heat sink causes loss of Mode 5 (Cold Shutdown) conditions AND A.2 LT-5 indicates 0 inches after initiation of RCS Makeup B.1 Failure of heat sink causes loss of Mode 5 (Cold Shutdown) conditions AND B.2 Either train ultrasonic level indication less than 0 inches and decreasing after initiation of RCS makeup NOTE: This Initiating Condition is also located in Enclosure 4.3, (Abnormal Rad Levels/Radiological Effluent). High radiation levels will also be seen with this condition. (END)	

Enclosure 4.5
Loss of Power {4}

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
1. LOSS OF ALL OFFSITE POWER TO ESSENTIAL BUSSES FOR GREATER THAN 15 MINUTES (BD 50) OPERATING MODE: All A.1 Unit auxiliaries are being supplied from Keowee or CT5 AND A.2 Inability to energize either MFB from an offsite source (either switchyard) within 15 minutes. 2. *UNPLANNED* LOSS OF REQUIRED DC POWER FOR GREATER THAN 15 MINUTES (BD 51) OPERATING MODE: 5, 6 A. *Unplanned* loss of vital DC power to required DC busses as indicated by bus voltage less than 110 VDC AND A. Failure to restore power to at least one required DC bus within 15 minutes from the time of loss (END)	1. LOSS OF ALL OFFSITE AC POWER AND LOSS OF ALL ONSITE AC POWER TO ESSENTIAL BUSSES (BD 52) OPERATING MODE: 5, 6 Defueled A.1 MFB 1 and 2 de-energized AND A.2 Failure to restore power to at least one MFB within 15 minutes from the time of loss of both offsite and onsite AC power 2. AC POWER CAPABILITY TO ESSENTIAL BUSSES REDUCED TO A SINGLE SOURCE FOR GREATER THAN 15 MINUTES (BD 53) OPERATING MODE: 1, 2, 3, 4 A. AC power capability has been degraded to a single power source for > 15 minutes due to the loss of all but one of the following: Unit Normal Transformer (backcharged) Unit SU Transformer Another Unit SU Transformer (aligned) CT4 CT5 (END)	1. LOSS OF ALL OFFSITE AC POWER AND LOSS OF ALL ONSITE AC POWER TO ESSENTIAL BUSSES (BD 54) OPERATING MODE: 1, 2, 3, 4 A.1 MFB 1 and 2 de-energized AND A.2 Failure to restore power to at least one MFB within 15 minutes from the time of loss of both offsite and onsite AC power 2. LOSS OF ALL VITAL DC POWER (BD 55) OPERATING MODE: 1, 2, 3, 4 A.1 *Unplanned* loss of *vital* DC power to required DC busses as indicated by bus voltage less than 110 VDC AND A.2 Failure to restore power to at least one required DC bus within 15 minutes from the time of loss (END)	1. PROLONGED LOSS OF ALL OFFSITE POWER AND ONSITE AC POWER (BD 57) OPERATING MODE: 1, 2, 3, 4 A.1 MFB 1 and 2 de-energized AND A.2 SSF fails to maintain Mode 3 (Hot Standby) {1} AND A.3 At least one of the following conditions exist: A.3.1 Restoration of power to at least one MFB within 4 hours is NOT likely OR A.3.2 Indications of continuing degradation of core cooling based on Fission Product Barrier monitoring (END)

Loss of Power - Emergency Action Levels (EALs) apply to the ability of electrical energy to perform its intended function, reach its intended equipment, ex. - If both MFBs, are energized but all 4160V switchgear is not available, the electrical energy can not reach the motors intended. The result to the plant is the same as if both MFBs were de-energized. {4}

Enclosure 4.6
Fire/Explosions and Security Actions {2} {3}

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
1. *FIRES/EXPLOSIONS* WITHIN THE PLANT (BD 60) OPERATING MODE: All NOTE: Within the plant means: Turbine Building Auxiliary Building Reactor Building Keowee Hydro Transformer Yard B3T B4T Service Air Diesel Compressors Keowee Hydro & associated transformers A. Fire within the plant not extinguished within 15 minutes of Control Room notification or verification of a Control Room alarm B. Unanticipated *explosion* within the plant resulting in *visible damage* to permanent structures/equipment • includes steam line break and FDW line break **(Continued)**	1. *FIRE/EXPLOSION* AFFECTING OPERABILITY OF PLANT SAFETY SYSTEMS REQUIRED TO ESTABLISH/MAINTAIN SAFE SHUTDOWN (BD 63) NOTE: Only one train of a system needs to be affected or damaged in order to satisfy this condition. A. *Fire/explosions* AND A.1.1 Affected safety-related system parameter indications show degraded performance OR A.1.2 Plant personnel report *visible damage* to permanent structures or equipment required for safe shutdown **(Continued)**	(CONTINUE TO NEXT PAGE)	(CONTINUE TO NEXT PAGE)

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
2. CONFIRMED SECURITY CONDITION OR THREAT WHICH INDICATES A POTENTIAL DEGRADATION IN THE LEVEL OF SAFETY OF THE PLANT (BD 61) OPERATING MODE: All A. Security condition that does not involve a HOSTILE ACTION as reported by the Security Shift Supervisor B. A *credible* site-specific security threat notification C. A validated notification from NRC providing information of an aircraft threat 3. OTHER CONDITIONS EXIST WHICH IN THE JUDGEMENT OF THE EMERGENCY DIRECTOR WARRANT DECLARATION OF A NOUE. OPERATING MODE: All A. Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant or indicate a security threat to facility protection has been initiated. No releases of radioactive material requiring off-site response or monitoring are expected unless further degradation of safety systems occurs. (END)	2 HOSTILE ACTION WITHIN THE OWNER CONTROLLED AREA OR AIRBORNE THREAT. (BD 64, 65) A. A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by the Security Shift Supervisor. B. A validated notification from the NRC of an AIRLINER LARGE AIRCRAFT attack threat within 30 minutes of the site. 3. OTHER CONDITIONS EXIST WHICH IN THE JUDGEMENT OF THE EMERGENCY DIRECTOR WARRANT DECLARATION OF AN ALERT OPERATING MODE: All A. Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve an actual or potential substantial degradation of 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. (END)	1. HOLTILE ACTION within the PROTECTED AREA (BD 68) OPERATING MODE: All A. A HOSTILE ACTION is occurring or has occurred within the PORTECTED AREA as reported by the Site Security force. 2. OTHER CONDITIONS EXIST WHICH IN THE JUDGEMENT OF THE EMERGENCY DIRECTOR WARRANT DECLARATION OF A SITE AREA EMERGENCY. OPERATING MODE: All A. Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress 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 of or; (2) that prevent effective access to equipment needed for the 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. (END)	1. A HOSTILE ACTION RESULTING IN LOSS OF PHYSICAL CONTROL OF THE FACILITY (BD 72) OPERATING MODE: All A. A HOSTILE ACTION has occurred such that plant personnel are unable to operate equipment required to maintain safety functions B. 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. 2. OTHER CONDITIONS EXIST WHICH IN THE JUDGMENT OF THE EMERGENCY DIRECTOR WARRANT DECLARATION OF A GENERAL EMERGENCY. OPERATING MODE: All A. Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress 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 off-site for more than the immediate site area. (END)

Enclosure 4.7
Natural Disasters, Hazards and Other Conditions Affecting Plant Safety

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
1. NATURAL AND DESTRUCTIVE PHENOMENA AFFECTING THE *PROTECTED AREA* (BD 74) OPERATING MODE: All A. Tremor felt and *valid* alarm on the strong motion accelero graph B Tornado striking within *Protected Area* Boundary C. Vehicle crash into plant structures/systems within the *Protected Area* Boundary D. Turbine failure resulting in casing penetration or damage to turbine or generator seals (CONTINUED)	1. NATURAL AND DESTRUCTIVE PHENOMENA AFFECTING THE PLANT *VITAL* AREA (BD 79) OPERATING MODE: All A. Tremor felt and seismic trigger actuates (0.05g) **NOTE:** Only one train of a safety-related system needs to be affected or damaged in order to satisfy these conditions. B. Tornado, high winds, missiles resulting from turbine failure, vehicle crashes, or other catastrophic event. AND B.1 *Visible damage* to permanent structures or equipment required for safe shutdown of the unit. OR B.2 Affected safety system parameter indications show degraded performance. (CONTINUED)	(CONTINUE TO NEXT PAGE)	(CONTINUE TO NEXT PAGE)

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
2. NATURAL AND DESTRUCTIVE PHENOMENA AFFECTING KEOWEE HYDRO CONDITION B (BD 76) OPERATING MODE: All A. Reservoir elevation ≥ 807.0 feet with all spillway gates open and the lake elevation continues to rise B. Seepage readings increase or decrease greatly or seepage water is carrying a significant amount of soil particles C New area of seepage or wetness, with large amounts of seepage water observed on dam, dam toe, or the abutments D. Slide or other movement of the dam or abutments which could develop into a failure E. Developing failure involving the powerhouse or appurtenant structures and the operator believes the safety of the structure is questionable 3. RELEASE OF TOXIC OR FLAMMABLE GASES DEEMED DETRIMENTAL TO SAFE OPERATION OF THE PLANT (BD 77) OPERATING MODE: All A. Report/detection of toxic or flammable gases that could enter within the site area boundary in amounts that can affect normal operation of the plant B. Report by local, county, state officials for potential evacuation of site personnel based on offsite event (CONTINUED)	2. RELEASE OF TOXIC/FLAMMABLE GASES JEOPARDIZING SYSTEMS REQUIRED TO MAINTAIN SAFE OPERATION OR ESTABLISH/ MAINTAIN COLD SHUTDOWN (BD 81) OPERATING MODE: All A. Report/detection of *toxic gases* in concentrations that will be life-threatening to plant personnel B. Report/detection of flammable gases in concentrations that will affect the safe operation of the plant: • Reactor Building • Auxiliary Building • Turbine Building • Control Room 3. TURBINE BUILDING FLOOD (BD 82) OPERATING MODE: All A. Turbine Building flood requiring use of AP/1,2,3/A/1700/10, (Turbine Building Flood) 4. CONTROL ROOM EVACUATION HAS BEEN INITIATED (BD 83) OPERATING MODE: All A.1 Evacuation of Control Room AND ONE OF THE FOLLOWING: AND A.1.1 Plant control IS established from the Aux shutdown Panel or the SSF OR A.1.2 Plant control IS BEING established from the Aux Shutdown Panel or SSF (CONTINUED)	1. CONTROL ROOM EVACUATION AND PLANT CONTROL CANNOT BE ESTABLISHED (BD 85) OPERATING MODE: All A.1 Control Room evacuation has been initiated AND A.2 Control of the plant cannot be established from the Aux Shutdown Panel or the SSF within 15 minutes 2. KEOWEE HYDRO DAM FAILURE (BD 86) OPERATING MODE: All A. Imminent/actual dam failure exists involving any of the following: • Keowee Hydro Dam • Little River Dam • Dikes A, B, C, or D • Intake Canal Dike • Jocassee Dam - Condition A (CONTINUED)	(CONTINUE TO NEXT PAGE)

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
4 OTHER CONDITIONS EXIST WHICH WARRANT DECLARATION OF AN *UNUSUAL EVENT* (BD 78)	5. OTHER CONDITIONS WARRANT CLASSIFICATION OF AN *ALERT* (BD 84)	3. OTHER CONDITINS WRRANT DECLARATION OF SITE AREA EMERGENCY (BD 87)	1. OTHER CONDITIONS WARRANT DECLARATION OF GENERAL EMERGENCY (BD 88)
OPERATING MODE: All	**OPERATING MODE:** All	**OPERATING MODE**: All	**OPERATING MODE:** All
A. Emergency Coordinator determines potential degradation of level of safety has occurred (END)	A.1 Emergency Coordinator judgment indicates that: A.1.1 Plant safety may be degraded **AND** A.1.2 Increased monitoring of plant functions is warranted (END)	A. Emergency Coordinator/EOF Director judgment (END)	A.1 Emergency Coordinator/EOF Director judgment indicates: A.1.1 Actual/imminent substantial core degradation with potential for loss of containment **OR** A.1.2 Potential for *uncontrolled* radionuclide releases that would result in a dose projection at the site boundary greater than 1000 mRem TEDE or 5000 mRem CDE Adult Thyroid (END)

Enclosure 4.8
Radiation Monitor Readings for Emergency Classification

All RIA values are considered GREATER THAN or EQUAL TO

HOURS SINCE REACTOR TRIPPED	RIA 57 R/hr		RIA 58 R/hr*	
	Site Area Emergency	General Emergency	Site Area Emergency	General Emergency
0.0 - < 0.5	5.9E+003	5.9E+004	2.6E+003	2.6E+004
0.5 - < 1.0	2.6E+003	2.6E+004	1.1E+003	1.1E+004
1.0 - < 1.5	1.9E+003	1.9E+004	8.6E+002	8.6E+003
1.5 - < 2.0	1.9E+003	1.9E+004	8.5E+002	8.5E+003
2.0 - < 2.5	1.4E+003	1.4E+004	6.3E+002	6.3E+003
2.5 - < 3.0	1.2E+003	1.2E+004	5.7E+002	5.7E+003
3.0 - < 3.5	1.1E+003	1.1E+004	5.2E+002	5.2E+003
3.5 - < 4.0	1.0E+003	1.0E+004	4.8E+002	4.8E+003
4.0 - < 8.0	1.0E+003	1.0E+004	4.4E+002	4.4E+003

* RIA 58 is partially shielded

Enclosure 4.9
Unexpected/Unplanned Increase In Area Monitor Readings

NOTE: This Initiating Condition is not intended to apply to anticipated temporary increases due to planned events (e.g.; incore detector movement, radwaste container movement, depleted resin transfers, etc.).

MONITOR NUMBER	UNITS 1, 2, 3	
	UNUSUAL EVENT 1000x NORMAL LEVELS mRAD/HR	*ALERT* mRAD/HR
RIA 7, Hot Machine Shop Elevation 796	150	≥ 5000
RIA 8, Hot Chemistry Lab Elevation 796	4200	≥ 5000
RIA 10, Primary Sample Hood Elevation 796	830	≥ 5000
RIA 11, Change Room Elevation 796	210	≥ 5000
RIA 12, Chem Mix Tank Elevation 783	800	≥ 5000
RIA 13, Waste Disposal Sink Elevation 771	650	≥ 5000
RIA 15, HPI Room Elevation 758	NOTE*	≥ 5000

NOTE: RIA 15 normal readings are approximately 9 mRad/hr on a daily basis. Applying 1000x normal readings would put this monitor greater than 5000 mRad/hr just for an *Unusual Event*. For this reason, an *Unusual Event* will **NOT** be declared for a reading less than 5000 mRad/hr.

1. List of Definitions and Acronyms

NOTE:	Definitions are italicized throughout procedure for easy recognition.

1.1 **ALERT** - Events are in process or have occurred which involve an actual or potential substantial degradation of 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.

1.2 **BOMB** – Refers to an explosive device suspected of having sufficient force to damage plant systems or structures.

1.3 **CONDITION A** - Failure is Imminent or Has Occurred - A failure at the dam has occurred or is about to occur and minutes to days may be allowed to respond dependent upon the proximity to the dam.

1.4 **CONDITION B** - Potentially Hazardous Situation is Developing - A situation where failure may develop, but preplanned actions taken during certain events (such as major floods, earthquakes, evidence of piping) may prevent or mitigate failure.

1.5 **CIVIL DISTURBANCE** - A group of persons violently protesting station operations or activities at the site.

1.6 **EXPLOSION** - A rapid, violent, unconfined combustion, or catastrophic failure of pressurized/energized equipment that imparts energy of sufficient force to potentially damage permanent structures, systems, or components.

1.7 **EXTORTION** - An attempt to cause an action at the station by threat of force.

1.8 **FIRE** - Combustion characterized by heat and light. Sources of smoke, such as slipping drive belts or overheated electrical equipment, do NOT constitute *fires*. Observation of flames is preferred but is NOT required if large quantities of smoke and heat are observed.

1.9 **FRESHLY OFF-LOADED CORE** - The complete removal and relocation of all fuel assemblies from the reactor core and placed in the spent fuel pool. (Typical of a "No Mode" operation during a refuel outage that allows safety system maintenance to occur and results in maximum decay heat load in the spent fuel pool system).

1.10 **GENERAL EMERGENCY** - 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 Guidelines exposure levels outside the Exclusion Area Boundary.

1.11 **HOSTAGE** - A person(s) held as leverage against the station to ensure demands will be met by the station.

1.12 **HOSTILE ACTION** - An act toward an NPP or its personnel that includes the use of violent force to destroy equipment, takes HOSTAGES, and/or intimidates the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, PROJECTILES, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. HOSTILE ACTION should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the NPP. Non-terrorism-based EALs should be used to address such activities, (e.g., violent acts between individuals in the owner controlled area.)

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

1.14 **IMMINENT** - Mitigation actions have been ineffective, additional actions are not expected to be successful, and trended information indicates that the event or condition will occur. Where IMMINENT timeframes are specified, they shall apply.

1.15 **INTRUSION** – A person(s) present in a specified area without authorization. Discovery of a BOMB in a specified area is indication of INTRUSION into that area by a HOSTILE FORCE.

1.16 **INABILITY TO DIRECTLY MONITOR** - Operational Aid Computer data points are unavailable or gauges/panel indications are NOT readily available to the operator.

1.17 **LOSS OF POWER** – Emergency Action Levels (EALs) apply to the ability of electrical energy to perform its intended function, reach its intended equipment. Ex. – If both MFBs, are energized but all 4160v switchgear is not available, the electrical energy can not reach the motors intended. The result to the plant is the same as if both MFBs were de-energized.

1.18 **PROJECTILE** – An object directed toward a NPP that could cause concern for its continued operability, reliability, or personnel safety.

1.19 **PROTECTED AREA** – Typically the site specific area which normally encompasses all controlled areas within the security PROTECTED AREA fence.

1.20 **REACTOR COOLANT SYSTEM (RCS) LEAKAGE** – RCS Operational Leakage as defined in the Technical Specification Basis B 3.4.13:

RCS leakage includes leakage from connected systems up to and including the second normally closed valve for systems which do not penetrate containment and the outermost isolation valve for systems which penetrate containment.

A. Identified LEAKAGE

LEAKAGE to the containment from specifically known and located sources, but does not include pressure boundary LEAKAGE or controlled reactor coolant pump (RCP) seal leakoff (a normal function not considered LEAKAGE).

LEAKAGE, such as that from pump seals, gaskets, or valve packing (except RCP seal water injection or leakoff), that is captured and conducted to collection systems or a sump or collecting tank;

LEAKAGE through a steam generator (SG) to the Secondary System (primary to secondary LEAKAGE): Primary to secondary LEAKAGE must be included in the total calculated for identified LEAKAGE.

B. Unidentified LEAKAGE

All LEAKAGE (except RCP seal water injection or leakoff) that is not identified LEAKAGE.

C. Pressure Boundary LEAKAGE

LEAKAGE (except primary to secondary LEAKAGE) through a nonisolable fault in an RCS component body, pipe wall or vessel wall.

1.21 **RUPTURED** (As relates to Steam Generator) - Existence of Primary to Secondary leakage of a magnitude sufficient to require or cause a reactor trip and safety injection.

1.22 **SABOTAGE** - Deliberate damage, mis-alignment, or mis-operation of plant equipment with the intent to render the equipment inoperable. Equipment found tampered with or damaged due to malicious mischief may not meet the definition of SABOTAGE until this determination is made by security supervision.

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

1.24 **SAFETY-RELATED SYSTEMS AREA** - Any area within the *Protected area* which contains equipment, systems, components, or material, the failure, destruction, or release of which could directly or indirectly endanger the public health and safety by exposure to radiation.

1.25 **SELECTED LICENSEE COMMITMENT (SLC)** -Chapter 16 of the FSAR

1.26 **SIGNIFICANT PLANT TRANSIENT** - An *unplanned* event involving one or more of the following:

(1) Automatic turbine runback>25% thermal reactor power
(2) Electrical load rejection >25% full electrical load
(3) Reactor Trip
(4) Safety Injection System Activation

1.27 **SITE AREA EMERGENCY** - Events are in process or have occurred which involve actual or likely major failures of plant functions needed for the protection of the public. or HOSTILE ACTION that results in intentional damage or malicious act; (1) toward site personnel or equipment that could lead to the likely failure of or; (2) that prevents effective access to equipment needed for the protection of the public. Any releases are NOT expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels outside the Exclusion Area Boundary.

1.28 **SITE BOUNDARY** - That area, including the *Protected Area*, in which DPC has the authority to control all activities including exclusion or removal of personnel and property (1 mile radius from the center of Unit 2).\

1.29 **TOXIC GAS** - A gas that is dangerous to life or health by reason of inhalation or skin contact (e.g.; Chlorine).

1.30 **UNCONTROLLED** - Event is not the result of planned actions by the plant staff.

1.31 **UNPLANNED** - An event or action is UNPLANNED if it is not the expected result of normal operations, testing, or maintenance. Events that result in corrective or mitigative actions being taken in accordance with abnormal or emergency procedures are UNPLANNED.

1.32 **UNUSUAL EVENT** - Events are in process or have occurred which indicate a potential degradation of 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.

1.33 **VALID** - An indication or report or condition is considered to be VALID when it is conclusively verified by: (1) an instrument channel check; or, (2) indications on related or redundant instrumentation; or, (3) by direct observation by plant personnel such that doubt related to the instrument's operability, the condition's existence, or the report's accuracy is removed. Implicit with this definition is the need for timely assessment.

1.34 **VIOLENT** - Force has been used in an attempt to injure site personnel or damage plant property.

1.35 **VISIBLE DAMAGE** - Damage to equipment or structure that is readily observable without measurements, testing, or analyses. Damage is sufficient to cause concern regarding the continued operability or reliability of affected safety structure, system, or component. Example damage: deformation due to heat or impact, denting, penetration, rupture.

1.36 **VITAL AREA** - An area within the protected area where an individual is required to badge in to gain access to the area and that houses equipment important for nuclear safety. The failure or destruction of this equipment could directly or indirectly endanger the public health and safety by exposure to radiation.

Enclosure 4.11

Operating Modes Defined In Improved Technical Specifications

MODES

MODE	TITLE	REACTIVITY CONDITION (K_{eff})	% RATED THERMAL POWER (a)	AVERAGE REACTOR COOLANT TEMPERATURE (°F)
1	Power Operation	≥0.99	> 5	NA
2	Startup	≥0.99	≤5	NA
3	Hot Standby	<0.99	NA	≥250
4	Hot Shutdown (b)	< 0.99	NA	250 > T > 200
5	Cold Shutdown (b)	< 0.99	NA	≤ 200
6	Refueling (c)	NA	NA	NA

(a) Excluding decay heat.

(b) All reactor vessel head closure bolts fully tensioned.

(c) One or more reactor vessel head closure bolts less than fully tensioned

1. Instructions For Using Enclosure 4.1 – Fission Product Barrier Matrix

1.1 If the unit was at Hot S/D or above, (Modes 1, 2, 3, or 4) and one or more fission product barriers have been affected, refer to Enclosure 4.1, (Fission Product Barrier Matrix) and review the criteria listed to determine if the event should be classified.

1.1.1 For each Fission Product Barrier, review the associated EALs to determine if there is a Loss or Potential Loss of that barrier.

NOTE: An event with multiple events could occur which would result in the conclusion that exceeding the loss or potential loss thresholds is imminent (i.e. within 1-3 hours). In this situation, use judgement and classify as if the thresholds are exceeded.

1.2 Three possible outcomes exist for each barrier. No challenge, potential loss, or loss. Use the worst case for each barrier and the classification table at the bottom of the page to determine appropriate classification.

1.3 The numbers in parentheses out beside the label for each column can be used to assist in determining the classification. If no EAL is met for a given barrier, that barrier will have 0 points. The points for the columns are as follows:

Barrier	Failure	Points
RCS	Potential Loss	4
	Loss	5
Fuel Clad	Potential Loss	4
	Loss	5
Containment	Potential Loss	1
	Loss	3

1.3.1 To determine the classification, add the highest point value for each barrier to determine a total for all barriers. Compare this total point value with the numbers in parentheses beside each classification to see which one applies.

1.3.2 Finally as a verification of your decision, look below the Emergency Classification you selected. The loss and/or potential loss EALs selected for each barrier should be described by one of the bullet statements.

EXAMPLE: Failure to properly isolate a 'B' MS Line Rupture outside containment, results in extremely severe overcooling.

PTS entry conditions were satisfied.

Stresses on the 'B' S/G resulted in failure of multiple S/G tubes.

RCS leakage through the S/G exceeds available makeup capacity as indicated by loss of subcooling margin.

Barrier	EAL	Failure	Points
RCS	SGTR > Makeup capacity of one HPI pump in normal makeup mode with letdown isolated	Potential Loss	4
	Entry into PTS operating range	Potential Loss	4
	RCS leak rate > available makeup capacity as indicated by a loss of subcooling	Loss	5
Fuel Clad	No EALs met and no justification for classification on judgment	No Challenge	0
Containment	Failure of secondary side of SG results in a direct opening to the environment	Loss	3

RCS _5_ + Fuel _0_ + Containment _3_ = Total _8_

A. Even though two Potential Loss EALs and one Loss EAL are met for the RCS barrier, credit is only taken for the worst case (highest point value) EAL, so the points from this barrier equal 5.

B. No EAL is satisfied for the Fuel Clad Barrier so the points for this barrier equal 0.

C. One Loss EAL is met for the Containment Barrier so the points for this barrier equal 3.

D. When the total points are calculated the result is 8, therefore the classification would be a *Site Area Emergency*.

E. Look in the box below "*Site Area Emergency*". You have identified a loss of two barriers. This agrees with one of the bullet statements. The classification is correct.

1 References:

1. PIP O-05-02980
2. PIP O-05-4697
3. PIP O-06-0404
4. PIP O-06-03347
5. PIP O-09-00234
6. PIP O-10-1055
7. PIP O-10-01750

1 References:

1. PIP O-05-02980
2. PIP O-05-4697
3. PIP O-06-0404
4. PIP O-06-03347
5. PIP O-09-00234
6. PIP O-10-1055
7. PIP O-10-01750

Time	RCS Temp (1) LPI or Tc	CETC	LPI Hdr Flows 'A' Hdr	LPI Hdr Flows 'B' Hdr	LPI CLR Outlet Temp (5) 'A' Clr	LPI CLR Outlet Temp (5) 'B' Clr	LPSW Flow In-Service Cooler (s) 'A' CLR	LPSW Flow In-Service Cooler (s) 'B' CLR	Total LPSW Flow (4) (✓)	NI-1, NI-2, NI-3, or NI-4 (CPS) (R.M.)	RxV Level Or PZR Level (2) (3)	Review/ Update Shift Turnover Sheet (✓)	Step 2.0	Initial
1900													N/A	
2000*													N/A	
2100													N/A	
2200													N/A	
2300													N/A	
0000														
0100													N/A	
0200*													N/A	
0300													N/A	
0400													N/A	
0500													N/A	
0600													N/A	

* Twice per shift surveillances are due at these hours.

Refer to "Notes" section on page 3 of 13.

Time	RCS Temp (1) LPI|CETC or Tc		LPI Hdr Flows 'A' Hdr	LPI Hdr Flows 'B' Hdr	LPI CLR Outlet Temp (5) 'A' Clr	LPI CLR Outlet Temp (5) 'B' Clr	LPSW Flow In-Service Cooler (s) 'A' CLR	LPSW Flow In-Service Cooler (s) 'B' CLR	Total LPSW Flow (4) (✓)	NI-1, NI-2, NI-3, or NI-4 (CPS) (R.M.)	RxV Level Or PZR Level (2) (3)	Review/ Update Shift Turnover Sheet (✓)	Initial
0700													
0800*													
0900													
1000													
1100													
1200													
1300													
1400*													
1500													
1600													
1700													
1800													

* Twice per shift surveillances are due at these hours.

Refer to "Notes" section on page 3 of 13.

1. Record required data **AND** review Shift Turnover Sheet every hour.

   1.1 Initial when all data steps completed

   1.2 Review Enclosure "Shift Turnover Sheet" to verify all turnover items updated and all required testing/surveillance items resulting from a degraded Mode per TS are performed.

   1.3 The following NOTEs apply:

   NOTE (1): **IF** LPI in High Pressure Mode, Series Mode, or Switchover Mode, record CETCs (Avg. of five highest).
   IF LPI in Normal DHR Mode, record LP Pump Suct. Temp. Line A (Gauge, C.P. O1A1322) **AND** CETC's (**IF** available, Avg. of five highest).
   IF LPI is OFF **AND** RCP operating log Tc.

   NOTE (2): **IF** 1LT-5 ≤ 84", record RxV level. **IF** 1LT-5 > 84", record Pzr level. Uncompensated PZR Level is most accurate indication when RCS is open to atmosphere.

   NOTE (3): **IF** 1LT-5 ≤ 50", perform PT/1/A/0600/001 D (Periodic Checks For RCS <50" With Fuel in Core) in parallel with this enclosure.

   NOTE (4): **WHEN** one or both Units are shutdown, U1&2 LPSW Pumps may be subject to low flow conditions. Refer to L&P of OP/1/A/1104/010 (Low Pressure Service Water) for proper pump/flow combination. Also refer to SLC 16.9.12.

   NOTE (5): Log in service LPI Cooler Outlet Temperature. **WHEN** no RCPs operating, verify LPI Cooler Outlet temperature injecting into core ≥ 71°F for NDT concerns.

2. Obtain and call in Daily Dispatcher readings at 0000 hours.

3. Procedure for Periodic Checks:

   - Review all in-progress Surveillance Evaluation enclosures in PT/1/A/0600/001 Book:
     - (N) Verify all corrective/compensatory actions still valid. (e.g., WRs, WOs, PIPs open; Procedure change(s) **NOT** yet implemented)
   - No surveillance or completion time exceeded.
   - Update in-progress Surveillance Evaluations by one-lining, initialing, and dating as required (change WR numbers to WO numbers, update resolution times).
     - RO **AND** SRO sign updated line per step 9.1 of Enclosure "Surveillance Evaluation".
   - Close out Surveillance Evaluations no longer applicable (e.g., corrective actions completed, TS/SLC no longer applicable).

- Attach completed (closed out) Surveillance Evaluations to this procedure.
- Perform periodic checks as specified.
  - **IF** check can be performed as specified **AND** is satisfied, initial appropriate block.
  - **IF** check **CANNOT** be performed as written **OR** is **NOT** satisfied, perform the following:
    - **IF** Surveillance Evaluation is **NOT** outstanding for check, perform Enclosure "Surveillance Evaluation".
    - Record Surveillance Evaluation in effect in appropriate block for periodic checks with Surveillance Evaluations issued.
    - Attach a copy of Surveillance Evaluation issued this shift to this procedure.
    - List Surveillance Evaluations in effect in Remarks section of Procedure Process Record.
  - Place Surveillance Evaluations initiated this shift in PT/1/A/0600/001 Book.

Enclosure 13.4
Mode 5

- SRO ensure all PT/1/A/0600/001 (1930-2230) or (0730-1030) SRs for current shift completed: {5}

_____	_____
SRO (1900 - 0700)	SRO (0700 - 1900)

- SRO ensure all PT/1/A/0600/001 Twice per Shift SRs for current shift completed: {5}

_____	_____
SRO (1900 - 0700)	SRO (0700 - 1900)

- SRO ensure all PT/1/A/0600/001 SRs for current shift completed:

_____	_____
SRO (1900 - 0700)	SRO (0700 - 1900)

- CRO enter in Unit Log all PT/1/A/0600/001 SRs for current shift completed:

_____	_____
CRO (1900 - 0700)	CRO (0700 - 1900)

	COMPONENT	1930-2230	0730-1030	COMPUTER	REQUIRED CONDITIONS
SR 3.3.9.1 12 Hours	Source Range Neutron Flux	_____ 1930-2100 _____ 0130-0300	_____ 0730-0900 _____ 1330-1500		Verify two Source Range Neutron Flux indications agree within 1 decade. (done on 6 hr frequency)
SR 3.3.10.1 12 Hours TS 3.3.8	Wide Range Neutron Flux	(N)	(D)		**IF** any CRD trip breaker closed and CRD system capable of rod withdrawal, verify two Wide Range Neutron Flux indications agree within 3 LED Segments.
SR 3.3.1.1 12 Hours	RPS Instrumentation NI Power Range NI-5, 6, 7, 8	(N)	(D)	O1A1544 O1A1545 O1A1546 O1A1547	**IF** any CRD trip breaker closed and CRD system capable of rod withdrawal, verify computer readouts agree within 2%. (If OAC unavailable, 4% in RPS Cab). {4} **NOTE:** **IF** the channels are off scale, the channel check will only verify that they are off scale in the same direction. (TS Bases SR 3.3.1.1)
SR 3.3.1.1 12 Hours	RPS Instrumentation RC Pressure Narrow Range	(N)	(D)	O1A1688 O1A1689 O1A1690 O1A1691	**IF** any CRD trip breaker closed and CRD system capable of rod withdrawal, verify computer readouts agree within 26 psi (If OAC unavailable, 48 psi in RPS Cab). {4} **NOTE:** **IF** the channels are off scale, the channel check will only verify that they are off scale in the same direction. (TS Bases SR 3.3.1.1) **NOTE:** Use reading from RPS Cabinets if RC Pressure < 1700 psig.

	COMPONENT	1930-2230	0730-1030	COMPUTER	REQUIRED CONDITIONS
SR 3.1.1.1 24 Hours	SDM	(N)	(D)		Verify ≥ 1% ΔK/K SDM per PT/1/A/1103/015 (Reactivity Balance Calculation). (done on 12 hr frequency)
SR 16.9.7.1 12 Hours	ECCW	(N) _____ ft	(D) _____ ft	O1E0149	Determine Absolute Keowee Lake Level by using computer point O1E0149 (CCW INTAKE LEVEL): Computer Point - 0.50 ft instrument error = Absolute Keowee Lake Level **OR** Perform Enc. 13.13 "Keowee Lake Level Determination". Record absolute Keowee lake level Verify absolute Keowee lake level within limits per SLC 16.9.7 as applied to Unit 1.
SR 3.4.7.1 12 Hours SR 3.4.8.1 12 Hours	RCS Loops	(N)	(D)		Verify one LPI Pump in operation with indicated LPI flow and capable of maintaining desired RCS Temperature. {6} **NOTE:** All DHR Loops may be removed from operation during planned heatup to MODE 4 when at least one RCS Loop is in operation.
SR 3.4.7.2 12 Hours	RCS Loops Filled	(N)	(D)		**IF** two DHR loops **NOT** operable, verify SG level ≥ 50% OR (or equivalent).
SR 3.4.12.1 12 Hours	LTOP (HPI)	(N)	(D)		**IF** LTOP Vent Path **NOT** established, verify HPI deactivated per OP/1/A/1104/049 (LTOP).
SR 3.4.12.2 12 Hours	LTOP (CFT)	(N)	(D)		**IF** CFT pressure ≥ 373 psig when SG Primary Manways installed **AND** PZR Manway installed, verify each CFT isolated per OP/1/A/1104/049 (LTOP).

	COMPONENT	1930-2230	0730-1030	COMPUTER	REQUIRED CONDITIONS
SR 3.4.12.3 12 Hours	LTOP (Pzr Level)	(N)	(D)		**IF** LTOP Vent Path **NOT** established, verify Pzr level within LTOP limits per heatup/cooldown curves in OP/0/A/1108/001 (Curves And General Information).
SR 3.4.12.6 12 Hours	LTOP (Pressurizer Heater Bank 3 or 4)	(N)	(D)		**IF** LTOP Vent Path **NOT** established, verify Pressurizer Heater Bank 3 **OR** 4 white tagged open per OP/1/A/1104/049 (Low Pressure Overpressure Protection (LTOP)).
SR 3.4.12.4 12 Hours	LTOP (1RC-4)	(N)	(D)		**IF** PORV required to be operable, verify 1RC-4 open.
SR 16.5.2.4 12 Hours	LTOP (1RC-4)	(N)	(D)		**IF** the following conditions are met: • PORV removed • 1RC-4 installed • 1RC-4 part of LTOP vent path, verify 1RC-4 open.
SR 3.4.12.6 12 Hours	LTOP (RC Pressure)	(N)	(D)		**IF** LTOP Vent Path **NOT** established, verify RCS < 375 psig.
SR 3.4.12.6 12 Hours	LTOP (RC Makeup Flow)	(N)	(D)		**IF** LTOP Vent Path **NOT** established, verify 1HP-120 travel stop operable per OP/1/A/1104/049 (LTOP) or HPI Pumps **NOT** capable of starting and injection into RCS via 1HP-120.
SR 3.4.12.6 12 Hours	LTOP (OAC Alarms) **NOTE:** The 225" Pressurizer Level Computer alarm is included in this check.	(N)	(D)	O1L3153	**IF** LTOP Vent Path **NOT** established, verify O1L3153 **NOT** in alarm **AND** OAC operable. **IF** O1L3153 in alarm, evaluate LTOP input points valid by using LTOP OAC Screen. Refer to OP/1/A/1104/049 (Low Temperature Overpressure Protection (LTOP)).

	COMPONENT	1930-2230	0730-1030	COMPUTER	REQUIRED CONDITIONS
SR 3.4.12.6 12 Hours	LTOP (OAC Alarms) (260" Pressurizer Level Computer alarm.)	(N)	(D)		**IF** LTOP Vent Path **NOT** established, verify the following: • 1SA-2/C-3 "RC Pressurizer Level High/Low" Statalarm card **NOT** Pulled. **IF** Either 1SA-18/A-3 **OR** 1SA-18/A-4 is in alarm **OR** card pulled, verify the following Computer Point status: • O1X2256 Quality = GOOD Value = FALSE • O1X2274 Quality = GOOD Value = FALSE • O1X2285 Quality = GOOD Value = FALSE
SR 3.4.12.6 12 Hours	LTOP (OAC Alarms) (315" Pressurizer Level Computer alarm.)	(N)	(D)		**IF** LTOP Vent Path **NOT** established, verify 1SA-2/C-4 "RC Pressurizer Level Emerg High/Low" Statalarm card **NOT** Pulled,.
SR 3.4.12.6 12 Hours	LTOP (HP Nitrogen System)	(N)	(D)		**IF** LTOP Vent Path **NOT** established, verify 1N-121 white tagged closed per OP/1/A/1104/049 (LTOP).
SR 16.11.3.2 24 Hours	RB Purge Recorder	(N)	(D)		**IF** RB Purge in operation, verify recorder (1MSCCR0001) indicates RB Purge Flow. Perform anytime during shift hours.

	COMPONENT	1930-2230	0730-1030	COMPUTER	REQUIRED CONDITIONS
SR 16.11.3.1 24 Hours SR 16.11.3.2 24 Hours	1RIA-35	(N)			Verify 1RIA-35 indicates > zero **AND** no low flow alarm present. **NOTE:** The analog OAC points should **NOT** be used as a substitute for the View Node because of the +/_ 25% error these points contain.
SR 16.11.3.2 24 Hours	1RIA-40	(N)			**IF** CSAEs in operation, verify 1RIA-40 indicates > zero **AND** no low flow alarm present. **NOTE:** The analog OAC points should **NOT** be used as a substitute for the View Node because of the +/_ 25% error these points contain.
SR 16.11.3.2 24 Hours	1RIA-43, 44, 45	(N)			Verify the following: 1) 1RIA-43, 44, 45 indicate > zero. 2) Unit Vent Monitor has no low flow alarm. 3) Unit Vent Flow Recorder indicates on scale. **NOTE:** The analog OAC points should **NOT** be used as a substitute for the View Node because of the +/_ 25% error these points contain.
SR 16.11.3.1 24 Hours SR 16.11.3.2 24 Hours	1RIA-54	(N)			Verify the following: 1) 1RIA-54 indicates > zero. 2) No low flow alarm present. 3) "NORMAL/BYPASS" switch in "Normal". **NOTE:** The analog OAC points should **NOT** be used as a substitute for the View Node because of the +/_ 25% error these points contain.

	COMPONENT	1930-2230	0730-1030	COMPUTER	REQUIRED CONDITIONS
SR 16.9.8a.1	EWST Level	(N)	(D)		Verify EWST level ≥ 70,000 gallons
SR 16.8.1.1 SR 16.8.1.2 SR 16.8.1.3 24 Hours	Room Temperatures Unit 1 Cable Rm. Unit 1 Equip. Rm. Unit 1&2 Control Rm.	(N) _____ _____ _____	(D) _____ _____ _____	 O1A1850 O1A1851	Record and verify room temperatures within respective temperature limits: (Computer Pt.) **OR** (Local Gage) Unit 1 Cable Rm: ≤ 78.5°F ≤ 78°F Unit 1 Equip. Rm: ≤ 83.5°F ≤ 83°F Unit 1&2 Control Room: ≤ 78°F **IF** limit is exceeded, refer to AP/1-2/A/1700/036 (Degraded Control Room Area Cooling), notify Unit Coordinator **AND** refer to SLC 16.8.1 and TS 3.7.16.
SR 16.9.11a.1	AB Flood Protection Measures	(N)	(D)		Verify with Primary NLO that the LPI hatch area curbs are in place and **NOT** obstructed. (Includes 1St & 2nd floor curbs)
SR 16.8.6.1 24 Hours	Lee/Central Alternate Power System	(N)	(D)		Verify Lee Power Source Availability per OP/0/A/1107/003 (100KV Power Supply). **IF** one LCTs or 100 KV Power Supply from Lee Steam to Oconee becomes inoperable for unplanned reasons, refer to Maintenance Rule **AND** contact Switchyard Coordinator.

	COMPONENT	1930-2230	0730-1030	COMPUTER	REQUIRED CONDITIONS
SR 16.6.11.1	Containment Debris Sources	(N)	(D)		**IF** required, perform PT/0/A/0115/013 (Surveillance of Containment Entries). • Required one time when entering MODE 5 from Mode 4 if a SD 1.3.9 Containment Entry Authorization form is **NOT** required for any materiel taken inside containment. • Required if Reactor Building is open for entry and a SD 1.3.9 Containment Entry Authorization form is required for any materiel taken inside containment. May be performed anytime during shift hours
SR 3.4.7.3 7 Days SR 3.4.8.2 7 Days	RCS Loops		**IF** Saturday (D)		**IF** RCS loops filled **AND** SG level < 50% OR (equivalent), perform Enclosure "LPI Pump Power Supply Verification". **OR** **IF** RCS loops **NOT** filled, perform Enclosure "LPI Pump Power Supply Verification". May be performed anytime during shift hours (0700-1900)

NOTE: Remaining item may be performed anytime during shift hours.

	COMPONENT	1900-0700	0700-1900	COMPUTER	REQUIRED CONDITIONS
ISFSI TS 4.1.1 ISFSI C of C 1.3.1 1.3.2	ISFSI Storage Facility	(N)			**NOTE:** **IF** AutoTour is unavailable, PT/0/A/0600/001 C (ISFSI Storage Facility Surveillance) contains needed information. When notified verify: 1) All Horizontal Storage Modules (HSM) ventilation screens (inlet and outlet) free of debris and no material accumulated between modules to block air flow. 2) All Roof Slab temperatures monitored which contain Spent Fuel < 250°F and < 80°F increase in 24 hrs. **IF** temperature limits are exceeded, issue a Work Request **AND** contact Rx Engineering. **IF** debris or material has accumulated on any Horizontal Storage Modules (HSM) ventilation screens (inlet and outlet), contact Environmental Health and Safety no later than the next day shift. {7}

Q 87

Duke Energy®

NUCLEAR POLICY MANUAL

Nuclear System Directive: 202. Reportability

Process/Program Owner: Regulatory Compliance Managers BEST

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10	12/20/98
11	04/30/99
12	02/23/00
13	12/28/00
14	01/23/01
15	11/13/01
16	01/27/03
17	05/03/04
18	02/08/05
19	07/21/05
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CATAWBA	MCGUIRE	OCONEE
Approved By/Date	Approved By/Date	Approved By/Date
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Issued By: R.L. Gill, Jr.

Manager, Nuclear Regulatory Issues & Industry Affairs

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DOCUMENT REVISION DESCRIPTION

REVISION NO.	PAGES or SECTIONS REVISED AND DESCRIPTION
10	Section 202.3 is being revised to reflect the approved version of NUREG 1022, Rev. 1. Section 202.6.1 is being revised to reflect changes made by the NRC to Part 21. Section 202.7.1, reportable example 'a' and 'b' are being revised to reflect implementation of the ITS at MNS. Section 202.7.2 is being revised to include information on the reportability policy regarding missed ASME Section XI required visual inspections after maintenance, to reflect a new philosophy concerning plants that have a delay of up to 24 hours in declaring an LCO or Tech Spec not being met (this philosophy resulted from the approval by the NRC of NUREG 1022, Rev. 1), to reflect new ITS section numbers for MNS, deletion of non-reportable example 'j' and a new example concerning multiple test failures was added under the "Reportable Examples". Section 202.8.2 is being revised to add a "Non-Reportable" example regarding Oconee's Emergency Feedwater system. Section 202.10 is being revised to add Fire Protection Program reporting information and more guidance on the reporting of Tech Spec Safety Limits violation. Appendix A, item #12 is being revised to reflect the new ITS section number for MNS.
11	Revised Section 202.4, entitled it, "Roles and Responsibilities" and renumbered remaining sections of NSD. Revised Section 202.6.4 to reflect that Tech Spec 3.03 is generic to all three sites. This is due to the implementation of the ITS. Added guidance on Past Operability in Section 202.6.4.1. Section 202.7.2 was also revised to reflect that Tech Spec 3.0.3 applies to all three sites. This is due to the implementation of the ITS at all three sites. Example 'e' was also revised to reflect to pressurizer heatup and cooldown rates no longer being in Tech Specs and Tech Specs no longer requiring D/G Special Reports. Items 'k' and 'd' were reused. Section 202.7.3 was revised to clarify what is considered a deviation and Reportable example 'b' was deleted. Section 202.7.4, item, #2 was revised to reflect renumbering of NSD sections. Section 202.8.1 was revised to reflect the new CNS ITS numbers governing tube plugging and examples that referenced these sections were also revised. Section 202.10, Tech Spec Safety Limit, was revised to reflect the change from 14 to 30 days for the submission of a written report. This is due to the implementation of the ITS. Throughout the directive, "TS" was changed to "Tech Spec".
12	Revised Section 202.6.4.1, 6th paragraph, inserted the phrase "have to" in the discussion "How far back do I look;" inserted the phrase "is generally sufficient" in place of "ago" in the 1st sentence; added the phrase "however, is there is reason to believe that the SSC was inoperable …;" deleted the sentence addressing an exception at the end of the 1st sentence, replacing it with the phrase "or until an inoperability in excess of the Tech Spec CT is discovered;" replaced the phrase "are not appropriate" with "are not necessary" in the last sentence of the paragraph.
13	NSD 202 is being revised in its entirety due to changes to 10 CFR 50.72 and 50.73. 10 CFR 50.73 reports are now required to be submitted within 60 days. The following sections are affected by this revision: TOC: The TOC was revised to reflect the new section numbering based on criteria being deleted/added. Section 202.2 (Purpose) is being revised to reflect the new eight hour reporting requirement. Section 202.3 (References) was revised to reflect approval of NUREG 1022, NSD 201 was added and the BWR Owner's Group LER/JCO Committee Consolidated Event Reporting Guidance document was deleted due to the approval of NUREG 1022, which is the primary

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	guidance document.
	Table 202-1 was added to enhance the user-friendliness of the directive.
	Section 202.5 (General Considerations) was revised to add NSD 201 as an additional reporting references.
	Section 202.6.1 (Part 21 Reporting) was revised for clarification.
	Section 202.6.3 (Emergency Notification System Reporting) was revised to reflect the new 8 hour reporting requirement.
	Section 202.6.3.2 (Reporting Multiple Events in a Single Report) was added for clarification of 60 day LER reporting.
	Section 202.6.3.4 (ENS Notification Retractions) was revised due to PIP O-99-03952.
	Section 202.6.4 (Specific Reporting Guidance to 50.72 and 50.73) was revised to delete the requirement to make an ENS notification upon entry into Tech Spec 3.0.3.
	Section 202.6.4.1 (Past Operability Determinations) was renamed "Evaluations in Support of Reportability" and clarification was added to this section to assist the user in making these determinations.
	Section 202.8.1 (Plant Shutdown Required by Technical Specifications) was deleted from the 1-hour reporting requirement and added to the 4-hour reporting requirement.
	Section 202.9 (8-hour ENS Notifications and LERs) was revised to reflect the new 8-hour reporting requirement and the reports that are reported under this requirement.
	Section 202.9.1 (Technical Specification Prohibited Operation or Condition) was revised to reflect that entry into Tech Spec 3.0.3 is not necessarily reportable and missed surveillances are reportable when the equipment is tested and found to be inoperable. Missed surveillances are also reportable when there is a programmatic breakdown (a reportable example was added to assist with this interpretation). Clarifying information was also added to the reporting requirement. Examples for this section was revised to reflect the changes in reporting.
	Section 202.9.2 (Degraded or Unanalyzed Condition) was revised to reflect deletion of the "Outside Design Bases" and "Conditions not covered by the plant's operating and emergency procedures". The "operating" and "shutdown" qualifiers for this criterion was also deleted. Additionally, examples were revised to reflect these deletions.
	Section 202.9.3 (Natural Phenomenon or Condition Threatening Safety (External Threat)) was revised by deleting the ENS reporting requirement. Examples were revise to reflect this deletion.
	Section 202.9.4 (Loss of Emergency assessment, Response, or Communications) was revised to add clarification regarding what constitutes a major loss of communication capability.
	Section 202.9.5 (Internal Threat to Plant Safety) was revised by deleting the ENS reporting requirement. Examples were revised to reflect the deletion.
	Section 202.9.6 (System Actuations) was revised by deleting the references to ESFs and instead, the NRC provided a "list of systems" that are to be reported . RPS actuations when the Reactor is critical are still 4-hour notifications; however, all other RPS actuations are to be made within 8-hours. Invalid system actuations are reportable per an LER only. Examples and Appendix A were revised to reflect these changes.
	Section 202.9.7 (Event or Condition that Could have Prevented the Fulfillment of a Safety Function) was revised to remove the term "alone" and the ENS notification requirement was revised to duplicate the LER reporting requirement. Clarifying information was also added to

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	this section to assist the user in making reportability determinations. Examples were revised accordingly. Section 202.9.9 (Airborne or Liquid Effluent Release exceeding Appendix B) was revised to delete the ENS reporting requirement. Section 202.9.11 (Single Cause that could have Prevented Fulfillment of the Safety Functions of Trains or Channels in Different Systems) is a new reporting requirement. Section 202.10 (Followup Notification) was clarified. Section 202.11 (Other Events Requiring "Immediate Notification") was revised by deleting the reports of "Loss of one working week or more of the operation of any unit" and "Damage to property in excess of $200,000). These two reporting requirements were required by Duke's Nuclear Insurer (NEIL) and they are no longer required per NEIL. The "Basis/Background for Engineered Safety Features and Associated Systems" was deleted since this information is obsolete.
14	202.9.4: Changed 50.72(b)(1)(v) to 50.72(b)(3)(xiii) 202.9.7: Changed 50.72(b)(2)(iii) to 50.72(b)(3)(v) 202.9.10: Changed 50.72(b)(2)(v) to 50.72(b)(3)(xii)
15	Section 202.2 (Purpose) is being revised to indicate limited reporting of 10 CFR 20 events. Section 202.4.3 (Engineering) is being revised to define SSC and typos were corrected. Additionally, any references to "Past Operability Evaluations" were changed to "Engineering Evaluations". This change is also applicable to Sections 202.6.3.1 and 202.6.4.1. Section 202.7.1 is being revised for clarity. Section 202.9.1 is being revised to remove the statement regarding missed surveillances being reportable due to programatic breakdowns. Example 'g' in this section is being removed to reflect this change. Section 202.9.6 (System Actuations), example 'd' is being revised to change spurious to invalid in order to be more in-line with industry terminology, example 'f' is being revised to add clarification for ONS in relation to actuations of the EFW with regards to the Steam Generator Dry-out Protection circuit. Additionally, clarification is added to this section in relation to VALID signals versus instrument drift or mis-calibration. Section 202.9.9 is being revised to remove the reference to Part 20.2202. The reporting requirements are in Part 20.2203. Section 202.11 is being revised to add ONS to the S/G Tube Plugging requirement. Appendix A is being revised to add the ONS actuation signals to the "Containment Isolation Systems", to delete the "Hydrogen Igniters" from item #3, Combustible Gas Control in Containment. Additionally, in item #5, Auxiliary/Emergency Feedwater System, the tornado pump was re-named the "Station- ASW pump.
16	Section 202.3: Added NUREG-302, Rev. 1 and a "Regulatory Position on the Reportability Requirements for a Single Train System" Section 202.6.1: This section is being updated based on guidance provided in NUREG 302. Revised to correctly reflect regulatory requirements for the completion of evaluations of deviations and failures to comply, and the regulatory guidance provided by NUREG-302, Rev. 1. (PIP #G-02-00291). Section 202.9.4: Revised this section and associated examples to reflect that Duke no longer uses the FTS-2000 telecommunication system and added clarification to the "Loss of Offsite Response Capability" section that emphasized the lost capability to alert a large segment of the population for more than an hour would warrant an immediate notification. Section 202.9.6: In the section relative to "Valid" signals the word "not" had been omitted. Section 202.9.7: Added guidance to this section to clarify when reporting of single-train

REVISION NO.	PAGES or SECTIONS REVISED AND DESCRIPTION
	systems is required. This guidance is based on information provided in the Regulatory Position on the Reportability Requirements for a Single Train System.
	Appendix A: Added "Essential or Non-Essential Isolation" to item #1 for clarification.
17	Section 202.3: Part 21 was deleted from this section and added the Federal Register Notice that amended 10 CFR 72 Event Notification Requirements;
	Section 202.5: Deleted Part 21 from this section;
	Section 202.6.1: Deleted the Part 21 guidance from this section and referenced the new Part 21 directive (NSD 229);
	Section 202.12: Revised the reporting requirements for 10 CFR 72.75 based on Event Notification Requirements Rule revision
	Table 202-1: Deleted the 50.73 reporting requirement for the transport of a contaminated person and added the 50.72 reporting requirement. There is not a 50.73 reporting requirement associated with this criteria.
18	Section 202.9.2: Added a reportable example regarding serious degradation of Steam generator tubes based on clarification published in the errata to NUREG 1022, Revision 2.
	Section 202.9.7: Deleted the reportable example regarding both trains of the VC system being inoperable based on new guidance provided by the NRC. Reference PIP #C-04-05549.
19	Change made to table 202-1 to correct the entry on "transport of a contaminated person" to be 8 hour (mistakenly placed in the 4 hour column in the previous revision).
20	Section 202.8.4.2: Corrected the spelling of "raised" under item h (non-reportable examples, last line).
	Section 202.9.6: Added clarification regarding which signals are included in "general containment isolation signals". Reference PIP M-03-05104
	Section 202.9.7: Moved this section heading to the next page
	Section 202.9.11: Corrected the spelling of "reasonable" under item a (Reportable Examples, second paragraph, sixth line down).
21	Section 202.8.4: Added clarifying information regarding formal and informal communications with local, state and other federal agencies with respect to notification to the NRC. Additionally, reformatted this section for clarity.
	Section 202.9.1: Added clarifying information to this section in response to a condition that occurred at MNS (reference PIP M-06-3281). Additionally, a Non-Reportable example "k" was added for clarification.
	Section 202.9.2: Information regarding the Fire Protection Program was removed from Section 202.11 and added to this section based on approved Amendments 230/226.
	Section 202.9.6 Clarified "Reportable Examples" section by adding to the NOTE that precedes the examples that you are to assume that actuations are the result of Valid signals. Under Reportable Example d, removed "valid or invalid". Editorial changes were made to Reportable Examples f, g, j and k.
	Section 202.12: Revised this section to expand ISFSI reporting to all three sites.

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202. REPORTABILITY

202.1 INTRODUCTION

The NRC's regulations set forth in Title 10, Chapter I, Code of Federal Regulations requires the reporting of various events or conditions in connection with licensee activities. This directive is presented to ensure nuclear station compliance and consistency with the reporting regulations of the 10CFR.

202.2 PURPOSE

The purpose of the directive is to provide guidance for use in determining reportability of station events or conditions under the provisions of the Immediate Notification Requirements of Significant Events (10CFR 50.72), the Licensee Event Reporting System (10CFR 50.73), 10CFR Part 21, Reporting of Defects and Noncompliance, 10CFR 50.9, Completeness and Accuracy of Information, to ensure proper and consistent reporting. Security event notifications are addressed in the Duke Nuclear Security Manual, a limited number of 10CFR 20 radiological notification requirements and 10CFR 72, ISFSI, notification requirements are included in the NSD. For Part 50.72 reporting, this directive only addresses [50.72(b)] one/four/eight hour notifications for non-emergency events, since adequate guidance currently exists in each stations' Emergency Plan's implementing response procedures for emergency events [50.72(a)] and their classifications.

202.3 REFERENCES

1. 10 Code of Federal Regulations 50.72, 50.73, Part 20, 72 and 50.9
2. Federal Register; Vol. 48, No. 144 and 168; July 26, 1983/August 29, 1983; "Licensee Report System" and "Immediate Notification Requirements of Significant Events"; final rule
3. NUREG 1397, Feb. '91, "An Assessment of Design Control Practices and Design Reconstitution Practices in the Nuclear Industry"
4. Generic Letter 91-18, "Operable/Operability: Ensuring the Functional Capability of a System or Component"
5. Standard Review Plan (NUREG 800)
6. Federal Register (56 FR 36081); July 31, 1991; "10 Code of Federal Regulations Parts 21 and 50.55(e)"
7. NUREG 1022, Revision 2.
8. NSD 201 (Reporting Requirements)
9. NUREG-0302, Rev. 1
10. "Regulatory Position on the Reportability Requirements for a Single Train System", dated June 28, 2002
11. Federal Register; Vol 68, No. 108; June 5, 2003; 10 CFR Parts 72 and 73, "Events Notification Requirements"; final rule

Event or Condition	ENS notification within 1 hour	ENS notification within 4 hours	ENS notification within 8 hours	60-day LER	NSD 202 Section
Deviation from Tech Specs Authorized under § 50.54 (x)	Deviation from TS authorized by 50.54 (x) [50.72 (b)(1)]			50.73 (a)(2)(i)(c)	202.7.1
Plant shutdown (S/D) required by Tech Specs		Initiation of S/D required by Tech Specs [50.72 (b)(2)(i)]		Completion of a S/D required by Tech Specs [50.73 (a)(2)(i)(A)	202.8.1
Operation or Condition Prohibited by Tech Specs				50.73 (a)(2)(i)(b)	202.9.1
Degraded or Unanalyzed Condition			50.72 (b)(3)(ii)	50.73 (a)(2)(ii)	202.9.2
External Threat or Hampering				50.73 (a)(2)(iii)	202.9.3
System Actuations		Valid ECCS discharges into Reactor Coolant system [50.72 (b)(2)(iv)(A)] RPS Actuations when Reactor is critical [50.72 (b)(2)(iv)(B)]	Valid actuation of NRC listed systems	50.73 (a)(2)(iv)(A)	202.8.2 202.8.3 202.9.6
Event or Condition that could have prevented fulfillment of a safety function			50.72 (b)(3)(v) 50.72 (b)(3)(vi)	50.73 (a)(2)(v) 50.73 (a)(2)(vi)	202.9.7
Common Cause Inoperability of independent trains or channels				50.73 (a)(2)(vii)	202.9.8
Radioactive Releases				Airborne radioactive release [50.73 (a)(2)(viii)(A)] Liquid Effluent release [50.73 (a)(2)(viii)(B)]	202.9.9
Internal Threat or Hampering				50.73 (a)(2)(x)	202.9.5
Transport of a Contaminated Person Offsite			50.72 (b)(3)(xii)		202.9.10
News Release or Notification of other Government Agency		50.72 (b)(2)(xi)			202.8.4
Loss of Emergency Preparedness Capabilities			50.72 (b)(3)(xiii)		202.9.4
Single Cause that could have prevented fulfillment of the Safety Functions of trains or channels in different systems				50.73 (a)(2)(ix)(A) 50.73 (a)(2)(ix)(B)	202.9.11

Table 202-1 Reportable Events

202.4 ROLES AND RESPONSIBILITIES

202.4.1 OPERATIONS

1. Responsible for reportability timeliness
2. Makes reportability determination (i.e., knowledgeable of issue, performs reasonableness review, ensures license requirements are met)
3. Makes notification to NRC and other required parties (Note: Regulatory Compliance may perform this function for some issues: e.g., design basis issues)
4. Ensures license and NRC requirements are met

202.4.2 REGULATORY COMPLIANCE

1. Reviews PIPs on front end (screening meeting) for potential reportability issues; follows up on screened PIPS that are unclear with respect to reportability
2. Independently assures timeliness is in accordance with NRC requirements
3. Provides lead role in making reportability recommendations to Operations Shift Manager (OSM) (For engineering related reportability evaluations, works closely with engineering to ensure the right questions are being asked and performs QV & V to ensure licensing basis assumptions are valid and proper)
4. Notifies OSM immediately when sufficient evidence exists that indicates an item is reportable
5. Provides support to the OSM as needed for NRC notification (may make NRC notification for some issues: e.g., design basis issues)
6. Ensures NSD conformance with NRC requirements
7. Ensures process is implemented in accordance with NSD

202.4.3 ENGINEERING

1. Notifies OSM and Regulatory Compliance immediately when sufficient evidence exists that indicates an item is reportable. (Note: Engineering scope for reportability also includes SSC (Structure, System, Subsystem, Component or Device) engineering evaluations (50.72 and 50.73), and Part 21 evaluations)
2. Keeps OSM and Regulatory Compliance informed of status of reportability evaluation and timeline to complete
3. Ensures engineering analysis and calculations assure SSC's can perform required functions
4. Determines if component failures meet 10 CFR 21 reporting criteria

202.5 GENERAL CONSIDERATIONS

Guidance is presented in this directive in order to ensure nuclear station compliance and consistency with the reporting regulations of 10CFR 50.72, 50.73, and 50.9. Other routine reports are outlined in NSD 201 (Reporting Requirements). Additionally, refer to NSD 201 when evaluating for reportability.

The purpose of the emergency notification requirements in 10CFR 50.72 is to inform the NRC of deficient conditions or events that have immediate safety significance or that may require NRC awareness or action in response to potential public interest. The purpose of the LER Rule in 10CFR 50.73 is to identify the types of deficient conditions or events that are significant to the NRC so that the NRC may perform engineering studies of

operational anomalies, trends and pattern analyses of operational occurrences. In general, many of the conditions and events that require immediate notification will also require an LER, as is reflected by the many parallel requirements specified in 50.72 and 50.73. Therefore, the event reporting guidance is arranged in the sequence of the Emergency Notification Rule, along with the corresponding sections of the LER Rule.

In some cases, such as discovery of an existing but previously unrecognized condition, it may be necessary to undertake an evaluation to determine if an event or condition is reportable. An evaluation should generally proceed on a schedule commensurate with the safety significance of the question. Plant operation may continue provided there is reasonable expectation that the equipment in question is operable. Whenever this reasonable expectation no longer exists, or significant doubts begin to arise, the equipment should be considered inoperable and appropriate actions, including reporting, should be taken promptly (Refer to NSD 203, "Operability" for more guidance on operable/inoperable equipment).

In evaluating a potentially reportable item, this document should be reviewed to identify all possible sections of the event reporting rules that might be applicable. It should be noted that an item can be reportable under several criteria and, in accordance with 50.72 and 50.73, a reportable item must be reported under all applicable criteria. In this directive, an example in a specific section is only evaluated for reportability under that specific criterion. In actual application, that same example might be reportable under other criteria. For ENS calls, the report should be made in accordance with the most stringent criterion that applies in order to fulfill all 50.72 requirements (e.g. an event that falls under a 1 hour and 4 hour notification should be reported within 1 hour, which also satisfies the 4 hour requirement). For LERs that are reportable under more than 1 criterion, all applicable blocks should be marked on the LER form.

202.6 SPECIFIC GUIDANCE

202.6.1 PART 21 REPORTING

NOTE: Part 21 Reporting guidance has been relocated to NSD 229 (Evaluation and Reporting of Potential Defects and Non-compliance).

202.6.2 10CFR 50.9 REPORTING

The stated intent for 10CFR 50.9(a) is that information provided to the NRC be complete and accurate in all material respects. Sections 50.72 and 50.73 contain provisions for updating and revising reports that should be used to correct material incompleteness or inaccuracies that are discovered. For example, submitting a revised LER would be appropriate to correct any previously submitted inaccuracies of a material nature.

10CFR 50.9(b) states that any licensee information with significant public health and safety, or common defense and security implications be reported to the NRC, except where a specific reporting requirement exists. The Statements of Consideration for 50.9 refer to such information as "residual information" that could affect licensed activities. The provisions of 50.9 should not be used to report information that is required to be reported under other reporting rules such as 50.72, 50.73, and Part 21.

If a condition is determined to be reportable under Part 50.9, the station shall notify the Region within 2 working days of the discovery of the information. A special report shall be written and submitted to the Region within 30 days of discovery of the event. The report should contain all relevant information pertaining to the circumstances involved, as well as, any planned corrective actions to be taken to prevent recurrence.

202.6.3 EMERGENCY NOTIFICATION SYSTEM REPORTING

202.6.3.1 Reporting Timeliness

The timing for ENS reporting is described in 10CFR 50.72 as **"immediate"** and **"as soon as practical and in all cases within one/ four/eight hour(s)"** of the occurrence of an event (depending on its significance). The intent is to require reportability decisions to be made in a timely manner so that ENS notifications are made to the NRC as soon as practical, keeping in mind the safety of the plant comes first. The event reportability time clock generally starts at the time of the event or the discovery of the condition. For example, the reportability time clock would start at the time of a Reactor trip or initiation of plant shutdown in accordance with Tech Specs. In come cases, such as discovery of an existing but previously unrecognized condition, it may be necessary to undertake an evaluation in order to determine if an event or condition is reportable. This evaluation should generally proceed on a schedule commensurate with the safety significance of the question. When evaluating more complex issues such as design basis questions, the clock should start once appropriate station management makes a decision with respect to the operability of the system or component. For example, the reportability time clock begins for an engineering evaluation once the evaluation concludes that the associated system was inoperable. When evaluating an event for reportability, consideration should be given to the requirements contained in section 202.10, Follow-up Notifications.

It is recognized that in the short time frame between the event and the ENS notification, there may not be enough time for an evaluation of the cause, effect, or compensatory measures taken. It is more important that the NRC be quickly made aware of the situation than it is for the station to answer every NRC question at the time of the initial notification. In other words, when evaluating a potentially reportable item, and there is doubt regarding whether to report or not, the NRC's policy is that licensees should make the report. Update ENS notifications should be made to provide additional information or analysis as it becomes available as appropriate.

Revisions to LERs should be submitted in a manner commensurate with its safety significance.

202.6.3.2 Reporting Multiple Events in a Single Report

More than one failure or event may be reported in a single ENS notification or LER if (1) the failures or events are related (i.e., they have the same general cause or consequences) and (2) they occurred during a single activity (e.g., a test program) over a reasonably short time (e.g., within 4 hours or 8 hours for ENS notifications, or within 60 days LER reporting).

Unrelated failures or events should be reported as separate ENS notifications to be given unique ENS numbers by the NRC. However, multiple ENS notifications may be addressed in a single telephone call.

202.6.3.3 Voluntary/Courtesy Notifications

The station may make voluntary or courtesy ENS notifications about events or conditions the NRC may be interested in. The NRC will evaluate and respond to any voluntary notification of an event or condition, as its safety significance warrants, regardless of the reporting classification of the reporting requirement. If it is determined later that the event is reportable, then another ENS notification should be made under the appropriate 50.72 criterion.

202.6.3.4 ENS Notification Retractions

If the station makes a 50.72 notification and later determines that the event or condition was not reportable, the appropriate station personnel should contact the NRC Operations Center to retract the previous notification and explain the rationale for the decision. Sound, logical bases for the withdrawal should be communicated with the retraction.

The retraction should receive a level of review commensurate with that of the initial notification. A retracted ENS report is retained in the NRC ENS database, along with the retraction.

202.6.4 SPECIFIC REPORTING GUIDANCE TO 50.72 AND 50.73

The sections that follow will address guidelines for reporting one, four and eight hour non-emergency events and LERs. Specific guidance for reporting Emergency classifications will not be provided in this directive since adequate guidance currently exists under the Emergency Plan implementing response procedures. If guidance is needed with respect to the reportability of an environmental event, Environmental Management should be contacted for assistance.

In addition to the specific guidelines given under each section, a descriptive list of examples, some of which have occurred, of nuclear station events and conditions that have been determined either reportable or non-reportable pursuant to 10CFR 50.72 and 50.73 are provided. A single event may fall under several reporting criteria. Although this list will not reflect all reporting sections applicable to a specific example, the example will be included under its most immediate reporting requirement, with the exception of those events that may also be reported under an Emergency Class declaration.

202.6.4.1 Engineering Evaluations in Support of Reportability

A Reportability Determination should first ask "Is the event or condition under consideration reportable if it is assumed that the event occurred or that the condition made an SSC inoperable for an extended period?" At a minimum, the reporting criteria that are most applicable for Reportability determinations of historical conditions should be consulted. These are: operations prohibited by Tech Specs {50.73 (a)(2)(i)(B)}, common-mode failures of independent trains or channels {50.73 (a)(2)(vii)}, events or conditions that could have prevented the fulfillment of a safety function {50.72 (b)(3)(v)}, single cause that could have prevented fulfillment of the safety functions of trains or channels in different systems {50.73 (a)(2)(ix)} and the plant in a degraded or unanalyzed condition {50.72 (b)(3)(ii)}.

However, in order to answer the above question, an Engineering Evaluation may be needed to determine the effect of a component's inoperability with respect to Tech Spec LCOs, the impact on the operability of any associated SSCs, and the ability to perform safety functions. Because the Tech Specs do not directly specify an LCO for many items that perform supporting functions, a knowledge of the plant design basis is essential to determine which support systems can affect operability.

Also, if the answer to the above question is "Yes", then an Engineering Evaluation may be needed to determine if the event/condition actually occurred, if the SSC was actually inoperable in the past due to the condition, and, if so, for how long.

Although there is no attendant duty of protecting the public, such Engineering evaluations should be completed in a timely manner. For example, there are one hour, four hours, eight hours and 60 day reporting requirements associated with past conditions, issues, or events. An event or condition may meet ENS and LER reporting criteria even though the event or condition under evaluation may have existed for years, or only minutes, or may have been corrected prior to discovery.

In most cases, it is expected that these evaluations can be made promptly (e.g., there is firm evidence that Tech Spec Completion Time has been exceeded, etc.). In other cases, additional information regarding the event or condition may be needed to complete the reportability determination. For these cases, it is expected that the required information can be obtained and the reportability determination completed within thirty days. Some few exceptional cases may take longer.

Also, in most cases, engineering judgement by a technically qualified individual is all that is needed to support the evaluation. A documented engineering analysis is not a requirement as a basis for an engineering judgement for all events or conditions – it's only necessary for particularly complex situations requiring in-depth analysis. When exercising engineering judgement, however, the NRC recommends that licensees record in writing that a judgement

was exercised by identifying the individual making the judgement and the date made, and briefly documenting the basis for the judgement.

For these evaluations, actual environmental conditions, instrument tolerance, etc., can be used rather then bounding conditions assumed in the design basis.

In general, for the purpose of evaluating the reportability of situations found during surveillance tests, it should be assumed that the situation occurred at the time of discovery, unless there is firm evidence to believe otherwise. For example, if a standby component with a seven day Tech Spec Action Statement is found to be inoperable because it was assembled improperly during maintenance conducted thirty days previously, then there is firm evidence that it had been inoperable for the entire thirty days, and the event would be reportable.

A common question when performing Reportability determinations is "How far back do I have to look?" If the SSC could have been inoperable in excess of its Tech Spec Action Statement in the past, a look back of three years is sufficient. The intent is to perform a reasonable search for a condition that could be reportable. In general, exhaustive searches or in-depth analyses are not necessary.

[**NOTE**: These Engineering Evaluations in Support of Reportability are generally in the "Past Operability" section of the associated PIP.]

202.7 1-HOUR ENS NOTIFICATIONS AND LERS

This section addresses 50.72(b)(1) 1-hour notifications for non-emergency events and the associated LER. If not reported as a declaration of an emergency class under 50.72(a), the station is required to notify the NRC as soon as practical and in all cases within 1 hour of the discovery of the event specified.

In addition to similar reporting criteria under both 10CFR 50.72 and 50.73, several requirements for only 50.72 notifications or only LERs are included in this section because of the sequential numbering scheme used.

202.7.1 TECH SPEC DEVIATION PER 10CFR 50.54(X)

§50.72(b)(1)	§50.73(a)(2)(i)(C)
Licensees shall report: "Any deviation from the plant's Technical Specifications authorized pursuant to §50.54(x) of this part."	Licensees shall report: "Any deviation from the plant's Technical Specifications authorized pursuant to §50.54(x) of this part."

1. General

   10CFR 50.54(x) generally allows the station to take reasonable action in an emergency even though the action is in violation of the License Condition or Tech Specs provided: (1) the action is immediately needed to protect the health and safety of the public (including station personnel), and (2) no action consistent with the License Conditions and Tech Specs is obvious that can immediately provide adequate protection. In accordance with 50.54(y), such action requires, as a minimum, prior approval by a licensed Senior Reactor Operator.

   Deviation from an Emergency Procedure that alters the intent of the procedure without prior approval may also be a violation of Tech Specs and should be evaluated to determine if it would require reporting under this section.

EXAMPLES

Reportable

a. With the unit at 100% power, the upper containment airlock inner door was opened to allow a technician to exit from the containment while the upper door was inoperable, resulting in a loss of containment integrity.

The Technician was inside containment when the lower airlock failed, requiring exit through the upper door.

The decision to open the upper containment airlock inner door exercised an allowable option under 10CFR 50.54(x). Immediate action was considered necessary for the technician to exit the containment for his personal safety. An ENS call was made within 1 hour of the breech of containment.

202.8 4-HOUR ENS NOTIFICATIONS AND LERS

This section addresses 50.72(b)(2) 4-hour notifications for non-emergency events and the associated LER. If not reported as a declaration of an emergency class under 50.72(a) or as a non-emergency 1 hour report under 50.72(b)(1), the station is required to notify the NRC as soon as practical and in all cases within 4 hours of the discovery of any of the events specified.

In addition to similar reporting criteria under both 10CFR 50.72 and 50.73, several requirements for only 50.72 notifications or only LERs are included in this section because of the sequential numbering scheme used.

202.8.1 PLANT SHUTDOWN REQUIRED BY TECHNICAL SPECIFICATIONS

§50.72(b)(2)(i)	50.73(a)(2)(i)(A)
Licensees shall report: "The initiation of any nuclear plant shutdown required by the plant's Technical Specifications."	Licensees shall submit a Licensee Event Report on: "The completion of any nuclear plant shutdown required by the plant's Technical Specifications."

1. 50.72

   The 50.72 reporting requirement is intended to capture those events for which Tech Specs require the initiation of reactor shutdown to provide the NRC with early warning of safety significant conditions. "Initiation" is the performance of any action to start reducing reactor power to achieve an operational condition or mode that requires the reactor to be subcritical, as a result of a Tech Spec requirement. This includes any means of power reductions such as control rod insertion or boron concentration changes.

2. 50.73

   For 50.73 reporting purposes, the phrase "completion of any nuclear plant shutdown" is defined as the point in time during a Tech Spec required shutdown when the plant enters Mode 3. Therefore, if a failure can be corrected before the unit is required to be in Mode 3 , an LER is not required. This includes a situation where the plant is shutdown, the problem is fixed and the unit is returned to power before the completion of shutdown was required by Tech Specs. The shutdown is reportable, however, if the failure cannot be corrected before the unit was required to be shutdown.

EXAMPLES

Reportable

a. Two out of three channels for a certain function failed. Tech Specs require the unit to be placed in Mode 3 within 6 hours with less than the minimum required channels operable. After 1 hour, the station began a load reduction from full power at 20% per hour. Within 15 minutes of the initial load reduction, an ENS notification was made. The station made an update ENS call 3 hours later after the equipment was repaired, the channels were declared operable, and the power reduction was stopped before completion of the shutdown.

   An ENS notification per 50.72 was required because the power reduction was an initiation of plant shutdown. {Note, however, an LER was not required because the shutdown was never completed (i.e., Mode 3 was not entered).}

b. When leakage around the primary containment ventilation exhaust dampers exceeded the maximum allowable combined secondary bypass leakage rate, the plant Tech Specs required the plant to be in Hot Shutdown within 12 hours. The station commenced a reactor shutdown at 10% per hour and made an ENS call within 10 minutes of entering the LCO Action. Hot Shutdown was reached 10 hours later.

c. While the unit was at 100%, the unit's nuclear service water pump discharge valve failed its monthly periodic test. Because the station knew repairs could not be made during the remaining time allowed by Tech Specs (72 hour Action), the unit was placed in Cold Shutdown within 1 day. The ENS call was made 30 minutes after the initial load decrease (even though there were 50 hours left on the Tech Spec clock).

Non-Reportable

d. Two out of three channels for a certain function failed. Tech Specs require the unit to be placed in Mode 3 within 6 hours with less than the minimum required channels operable. Since IAE personnel felt the repairs could be made within 3 hours, the Shift Supervisor decided to hold power for 3 hours. The equipment was repaired and the station declared the failed channels operable 4 hours later. No ENS call was made since there was no shutdown initiated.

e. While the unit was at 100%, the unit's nuclear service water pump discharge valve failed its monthly periodic test. Because the station thought repairs could be made during the remaining time allowed by Tech Specs (72 hour Action), the unit held at full power. The ENS call was not required since the valve work took only 40 of the remaining 50 hours left on the Tech Spec clock, and no power reduction had begun.

202.8.2 ECCS DISCHARGE INTO THE REACTOR COOLANT SYSTEM

§50.72(b)(2)(iv)(A)	10 CFR 50.73d
Licensees shall report: "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."	[ECCS discharge is a subset of §50.73(a)(2)(iv) (See Section 202.9.6). Therefore, an LER is required.]

1. General

Those events that result in either automatic or manual actuation of the ECCS, or should have resulted in ECCS discharge into the reactor coolant system if some component had not failed or an operator action had not been taken, are reportable. Reporting exceptions include preplanned actuations and the ECCS is properly removed from service and not required to be operable.

2. Valid Signal

Valid signal refers to those signals that are automatically initiated by the measurement of an actual physical system parameter that was within the established set point band of the sensor that provides the signal to the protection system's logic, or manually initiated in response to plant conditions. Valid signals should also include those passive system actuations that occur as a function of system conditions like differential pressure (i.e., cold leg accumulators) whereby no SSPS or other electrical signal is involved. The validity of an ECCS signal may not be determined within 4 hours, ECCS signals that result or should have resulted in injections should be considered valid until firm evidence proves otherwise. Invalid ECCS injections should be evaluated under Section 202.9.6 (§50.73 (a)(2)(iv)).

EXAMPLES

Reportable

a. While in Mode 3, valve 2NC-29 stuck open resulting in a rapid decrease in reactor coolant (NC) system pressure. This caused the Cold Leg Accumulators to actuate and inject approximately 1100 gallons of

borated water into the reactor coolant system. An ENS call is required to be made within 4 hours and an LER is required because of the actuation.

b. During a reactor vessel pressure test while in Cold Shutdown, a low pressure coolant injection pump (LPCI) automatically started when a reactor recirculation pump start caused a perturbation in reactor vessel level instrumentation readings. Because the reactor vessel pressure was above the LPCI pump shutoff head, no water was injected into the vessel. An ENS call is required because this was a valid ECCS signal that should have resulted in an ECCS discharge into the reactor vessel.

Non-Reportable

c. While surveillance testing containment isolation valves, a test pushbutton was inadvertently released, which initiated a 'B' train containment isolation and safety injection. High pressure ECCS pumps injected 300 gallons of borated water from the RWST into the reactor before pumps were secured, while the reactor remained at 94% power. The event is not reportable as a 4 hour ENS call under this section, even though it was an ECCS injection. The signal that caused the injection was an inadvertent, manual signal (i.e., plant conditions did not require a manual safety injection), thus, not a "valid" signal. The event is reportable (ref. Section 50.73 (a)(2)(iv).

202.8.3 REACTOR PROTECTION SYSTEM ACTUATION

§50.72(b)(2)(iv)(B)	10 CFR 50.73
Licensees shall report: "Any event that results in the 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 ."	[RPS actuation is a subset of §50.73(a)(2)(iv) (See Section 202.9.6). Therefore, an LER is required.]

202.8.4 NEWS RELEASE OR OTHER GOVERNMENT NOTIFICATIONS

§50.72(b)(2)(xi)	10 CFR 50.73
Licensees shall report: "Any event or situation, related to the health and safety of the public or on-site personnel, or protection of the environment, for which a news release is planned or notification to other government agencies has been or will be made. Such an event may include an on-site fatality or inadvertent release of radioactively contaminated materials."	[No corresponding Part 50.73 requirement.]

1. General

The purpose of this section is to ensure the NRC is made aware of issues that will cause heightened public or government concern related to radiological or environment events. The NRC Operations Center does not need to be made aware of every press release or offsite notification made. Only those issues that are perceived by the public to be related to the radiological health and safety of the public, onsite personnel, or protection of the environment, need be reported.

For reporting purposes, "other government agencies" refer to local, State or Federal agencies, including the FBI and local law enforcement. Notification to law enforcement agencies are governed by the Security Manual/Directive. If reportable under 50.72, they would also be reportable as one hour calls per 73.71.

If a licensee is notifying a local, state, or other federal agencies in accordance with an existing law, regulation or ordinance, then, the licensee should make its notification to the NRC under this criterion. However, if a licensee is informally communicating with a local, state, or other federal agencies (i.e., not under a specific law, regulation or ordinance), then, the licensee has discretion as to whether to informally communicate with the NRC (e.g., through the site resident inspector) or formally through the 50.72 notification process. If due to the site-specific circumstances or heightened sensitivity to the issue at the site, the issue is likely to produce strong media interest, then the licensee should consider notifying NRC under the 50.72 requirement because this is actually the underlying intent of this criterion.

The 4-hour ENS notification clock starts at the time the decision is made to report to other agencies or to make a press release. In some cases, a decision to issue a press release may not be made until long after the specific event. In all cases, however, the notification to the NRC should be made before the press release is issued to the media

Notifications to other Federal agencies does not relieve the station of the requirement to report to the NRC Operations Center via ENS. Likewise, if current procedures require reporting of events to other areas within the NRC, such as Region II, this too does not fulfill the reporting requirement of 50.72.

When in doubt, the ENS notification should be made by the station.

2. Generally, the following types of events require a report under 50.72:

Radiological

- inadvertent release of radioactively contaminated materials to public areas
- inadvertent public notification system operation for which a news release is planned
- inadvertent releases of radioactivity

Environmental

- unanticipated non-radioactive releases\spills that would generate interest from local government agencies or the EPA
- onsite plant or animal disease outbreaks including fish kills, excessive bird impaction events, or mortality or unusual occurrence of any species protected by the Endangered Species Act of 1973.
- Release of a Reportable Quantity (RQ) of a Superfund Amendment Reauthorization Act (SARA) extremely hazardous substance
- increase in nuisance organisms or conditions causally related to station operation

3. Optional or discretionary reports

For the following events, a report under 50.72 is left to the discretion of station management on a case by case basis. However, the NRC Resident inspector should be informed:

- groundwater contamination

Informal communications made to government agencies as a result of the Industry Groundwater Protection Initiative (GPI). The NRC Regional RP Inspector should receive informal notification of the event.

- underground storage tank (UST) or UST piping leak
- brown scum on the waters of the state
- release of a dye to waters of the state
- any drinking water maximum contaminate level violation for which posting is required

4. Generally, the NRC does NOT need to be informed under this section of:

- administrative matters such as company management changes, civil penalties, or media inquiries

- peaceful strikes or civil demonstrations

Environmental

- minor deviations from permitted effluent limits
- routine reports of effluent releases to other agencies
- minor onsite chemical spills that would not generate interest from local agencies of the EPA

OTHER EXAMPLES

Reportable

a. A man fell into the discharge canal while fishing and failed to resurface. The station notified the sheriff, state police, and state emergency agencies. The local media was granted onsite access to cover the event. An ENS call is required because of the fatality onsite, the other notifications made, and the media involvement.

b. The station informed the county government and other organizations of a spurious actuation of several alert sirens in a county. The station also planned a press release. An ENS notification is required within 4 hours of the initial contact with any county agency regarding the inadvertent actuation of part of the public notification system.

c. The station transported 2 secondary side filters to the county dump as non-radioactive waste, but later determined that they were contaminated. The station notified appropriate state agency and NRC resident inspector. An ENS call is required.

d. The station notified its state environmental protection agency and the NRC resident of a fish kill involving several species in the circulating water discharge canal, possibly resulting from thermal water conditions. An ENS call is required because of the state notification of a significant fish kill, which the media or public could perceive as an environmental or public hazard.

e. Oil spills to waters of the state require an ENS call.

f. A spill of ≥ 1 pound (≥ 50 ppm) of PCB's to the environment or any impervious surface requires an ENS call.

Non-Reportable

g. The station notified the state, EPA, and Dept. of Transportation that 5 gallons of diesel fuel oil had spilled onto gravel covered ground inside the protected area. The spill was cleaned up by removing the gravel and dirt. Such notifications to other agencies such as this do not require an ENS notification. These kind of events do not pertain to the radiological health and safety of the public, or protection of the environment.

h. As a result of a local newspaper article regarding the findings of an NRC regional inspection, a station representative was interviewed on local television and radio stations. The station also notified State officials and the NRC resident. An ENS call is not required in this case because the station was responding to findings raised by the NRC.

i. Notification of when a hazardous waste manifest is returned from an out-of-state facility does not require an ENS call.

j. A hazardous waste manifest not returned within the 45 day limit does not require an ENS call.

k. A licensee notified the U.S. EPA that the circulation water temperature rise exceeded the release permit allowable. This event was caused by the unexpected loss of a circulation water pump while operating at 92 percent power. The licensee reduced power to 73 percent so that the circulating water temperature would decrease to within the allowable limits until the pump could be repaired. An ENS notification is not needed because this event is routine and has little safety significance.

202.9 8-HOUR ENS NOTIFICATIONS AND LERS

This section addresses 50.72(b)(3) 8-hour notifications for non-emergency events and the associated LER. If not reported as a declaration of an emergency class under 50.72(a) or as a non-emergency 1 hour or 4 hour report under 50.72(b)(1) or 50.72(b)(2),respectively, the station is required to notify the NRC as soon as practical and in all cases within 8 hours of the discovery of any of the events specified.

In addition to similar reporting criteria under both 10CFR 50.72 and 50.73, several requirements for only 50.72 notifications or only LERs are included in this section because of the sequential numbering scheme used.

202.9.1 TECHNICAL SPECIFICATION PROHIBITED OPERATION OR CONDITION

10 CFR 50.72	§50.73(a)(2)(i)(B)
[There is no corresponding Part 50.72 requirement.	Licensees shall report: "Any operation or condition which was prohibited by the plant's Technical Specifications, except when: (1) The Tech Spec is administrative in nature; (2) The event consists solely of a case of a late surveillance test where the oversight is corrected, the test is performed, and the equipment is found to be capable of performing its specified safety functions; or (3) The Tech Spec 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."

1. General

   An LER is required under this criterion if an LCO and associated Action statement are not met. The time constraints included in the associated Action statements are based on the safety significance of the component or system being removed from service. The NRC is interested in the frequency of occurrence and the Tech Spec involved in events which a shutdown did not occur within the given time constraint. The condition is reportable even if the condition was not discovered until later and was corrected upon discovery. Therefore, if an inoperable component or system is discovered, an investigation is required in order to determine how long the component has been in the degraded condition. Reportability per this section can be determined based upon the results of the investigation.

   The LER rule does not address violations of License Conditions in documents other than Tech Specs. Such notifications are reportable as specified in a plant's license or other applicable document.

2. Inoperable upon Discovery

   If, through the course of the investigation of the inoperable component or system, it cannot be determined how long it was in the as found condition, there are 2 different assumptions to be made in order to reach a reportability decision. If the inoperable condition was discovered during the course of a surveillance, maintenance, or inspection, the condition is assumed to have occurred at the time of discovery (same philosophy as operability), provided no firm evidence exists that would indicate when the failure occurred. If, however, the condition was discovered by chance (e.g., an operator discovers a mispositioned valve during a walkdown), and it is obvious that the degraded condition was caused by some personnel action, it is assumed that the condition has existed since the component (or system) was known to be operable. Once these determinations are made, the most stringent Action statement should be compared to the time the item was inoperable. If the most limiting time constraints of the LCO Action were exceeded, the condition is reportable

per this criterion. Since these determinations may be subjective at times, the evaluator should consider what is reasonable based upon the circumstances surrounding the as found condition.

3. Tech Spec LCO 3.0.3

Tech Spec LCO 3.0.3 establishes requirements for actions when (1) an LCO is not met and the associated Actions are not met; (2) an associated ACTION is not provided, or (3) as directed by the associated ACTIONS themselves.

Entry into LCO 3.0.3 is not necessarily reportable under this criterion. However, it should be considered reportable under this criterion if the condition is not corrected within an hour, such that it is necessary to initiate actions to shutdown, cool down, etc. If a licensee responsibly concludes that plant shutdown should be delayed or corrective action can be accomplished so that an unnecessary plant transient can be avoided, such a decision is permitted as long as the shutdown times specified by the Tech Specs are observed.

4. Missed Surveillance

A missed or late surveillance test is reportable when it indicates that equipment (e.g., one train of a multiple train system) was not capable of performing its specified safety functions (and, thus, was inoperable) for a period of time longer than allowed by Tech Specs. Reporting **is not** required if an event consists solely of a case of a late surveillance test where the oversight is corrected, the test is performed, and the equipment is found to be capable of performing its specified safety functions.

For the purpose of evaluating the reportability of a discrepancy found during surveillance testing that is required by the Tech Specs:

(1) For testing that is conducted within the required time (i.e., the surveillance interval plus any allowed extension), it should be assumed that the discrepancy occurred at the time of its discovery unless there is firm evidence, based on a review of relevant information such as the equipment history and the cause of the failure, to indicate that the discrepancy existed previously.

(2) For testing that is conducted later than the required time, it should be assumed that the discrepancy occurred at the time the testing was required unless there is firm evidence to indicate that it occurred at a different time.

The purpose of this approach is two-fold. It rules out reporting of routine occurrences (i.e., occurrences where a timely surveillance test is performed, the results fall outside of acceptable limits, and the condition is corrected) unless there is firm evidence that equipment was incapable of performing its specified safety function longer than allowed. On the other hand, if the surveillance test is performed substantially late, and the equipment is not capable of performing its specified safety function, the occurrence is not routine. In this case the event is reportable unless there is firm evidence that the duration of the discrepancy was within allowed limits.

Tech Specs allow a delay of up to 24 hours in declaring an LCO or a Tech Spec requirement not met if it is found that a surveillance was not performed within its specified frequency or interval. However, this does not change the fact that the condition existed longer than allowed by Tech Specs. Failure to perform a surveillance within its frequency or interval is still reportable if it indicates that the equipment (i.e., one train of a multiple train system) was not capable of performing its specified safety functions and, thus, was inoperable for a period of time longer than allowed by Tech Specs. The delay merely specifies appropriate remedial action.

As specified previously, the event **is not** reportable if it consists solely of a case of a late surveillance test where the oversight is corrected, the test is performed, and the equipment is found to be capable of performing its specified safety functions. This type of event has not proven to be significant because the equipment remained functional.

5. Design and Analysis Defects and Deviations

A design or analysis defect or deviation is reportable under this criterion if, as a result, equipment (e.g., one train of a multiple train system) was not capable of performing its specified safety functions (and thus was inoperable) for a period of time longer than allowed by Tech Specs. Since Design and analysis conditions are long-lasting, the essential question in this case is whether the equipment was capable of performing its specified safety functions.

6. IST Requirements per Tech Spec 5.5

Tech Specs 5.5.8 covers IST requirements for ASME Class 1, 2, and 3 components. Missed or deficient IST/ASME surveillances are reportable when, as a result of the missed or deficient surveillance, a Tech Spec controlled system must be declared inoperable and the LCO action statement has been exceeded. The reportability evaluation should proceed per the guidance in Section 202.9.1, 1 through 4 (above), as applicable.

Failure to perform a visual inspection required by ASME Section XI does not in itself affect the operability of the component. As such, failure to perform visual inspections will not be reported as a condition prohibited by Technical Specifications. However, these missed inspections shall be reported to the NRC Resident Inspector for inclusion in any inspections as determined appropriate.

7. Administrative Requirements

Tech Specs include administrative requirements that are required to be followed. Violation of a Tech Spec that is administrative in nature is not reportable. For example, a change in the plant's organizational structure that has not yet been approved as a Tech Spec change would not be reportable.

An administrative procedure violation, or failure to implement a procedure, such as failure to lock a high radiation door, is generally not reportable under this criterion. Radiological conditions and events that are prohibited by Tech Specs should be evaluated for reportability under the requirements of 10CFR 20.2202 and 20.2203. Redundant reporting is not required.

EXAMPLES

Reportable

a. Doghouse water level instrumentation functional test was not performed on 1 train of channels. Tech Specs require this surveillance to be performed once every refueling outage. The missed test was discovered 1 month later and the Action statement requires continuous level monitoring with 1 or more trains inoperable.

b. The IWV Program lists valve NV-150 as a valve that requires a VST quarterly. With NV-150 inoperable, Train A of the Chemical and Volume Control (NV) system is inoperable. This valve has been tested only during Cold Shutdown.

c. Unit 1 operated at greater than 100% licensed thermal power for a period greater than the Tech Specs allow.

d. While preparing to perform a surveillance on an air operated valve, a technician discovered the instrument air line disconnected from the port. The inoperable valve renders its respective train inoperable. Upon investigation, it was determined that the line was most probably not connected properly after maintenance performed 2 weeks earlier. The Tech Spec Action for this train is 72 hours. The valve was not immediately retested following maintenance.

e. While performing surveillances on the main steam safety valves, of the 20 valves tested, 17 were out of tolerance (13 with set points above Tech Specs by as much as 4 percent). The existence of similar discrepancies in multiple valves is an indication that the discrepancies arose over a period of time and therefore reportable.

f. Operation with less than the required number of people on shift in excess of Tech Spec allowances would constitute operation prohibited by Tech Specs and therefore reportable.

Non-Reportable

g. The IWV Program lists valve NV-151 as a valve that requires a VST quarterly. With NV-151 closed and incapable of opening, Train A of the NV system is inoperable. This valve has not been tested in 9 months. Upon this discovery Operations confirmed that the valve had been in the open position for the entire period, thus, in its safety position and train 'A' NV was capable of performing its intended safety function.

Even though the IST program was violated, an LER is not required because the failure to test the valve's movement did not render its associated system or train inoperable.

h. Upon entering Mode 4, operators observed that the NC system (MNS and CNS) heatup rate had been exceeded during the removal of the 'A' reactor coolant pump. The Tech Spec LCO Action requires the rate to be restored within 30 minutes (which it was) and an engineering evaluation performed on the integrity of the pressurizer. The evaluation was performed immediately and confirmed the structural integrity acceptable, thus complying with the Action statement.

i. A certain containment isolation valve failed to meet its stroke timing test of 5.0 seconds during an outage. The subsequent investigation failed to reveal any evidence as to why the valve was slower for this surveillance. After maintenance, the valve tested in under 5.0 seconds and was declared operable.

j. Failure to perform a visual inspection after maintenance as required by ASME Section XI does not in itself affect the operability of the component. As such, failure to perform visual inspections will not be reported as a condition prohibited by Tech Specs.

k. At MNS, Unit 1 entered the Containment Isolation LCO due to an inoperable containment isolation valve. The affected penetration flow path was not isolated within the required time. This LCO also had an associated shutdown condition. Eleven minutes into the shutdown condition, the LCO was declared met. There was no need to add negative reactivity to initiate a plant shutdown at the time the LCO was declared met and the plant remained within the range of limitations as defined by the LCO. Thus, it was concluded that this event did not constitute an operation or condition prohibited by Tech Specs.

202.9.2 DEGRADED OR UNANALYZED CONDITION

§50.72(b)(3)(ii)	§50.73(a)(2)(ii)
Licensees shall report: "Any event or condition that results in:	Licensees shall report: "Any event or condition that resulted in:
(A) The condition of the nuclear power plant, including its principal safety barriers, being seriously degraded; or	(A) The condition of the nuclear power plant, including its principal safety barriers, being seriously degraded;
(B) The nuclear power plant being in an unanalyzed condition that significantly degrades plant safety."	(B) The nuclear power plant being in an unanalyzed condition that significantly degraded plant safety."

1. Definitions

a. Principal Safety Barriers: The principal safety barriers involve the functionally controlling or bounding accident and transient analysis barriers:

- Fuel Cladding
- Reactor Coolant System (RCS) Pressure Boundary
- Primary and Secondary (MNS/CNS Annulus) Containment

The specific safety function of these principal safety barriers is the protection of public health and safety through limiting the release of radioactive material. The controlling parameters for each of the principal safety barriers is contained in the UFSAR. Typical parameters include:

- Offsite Dose
- Fuel Clad Temperature
- Hydrogen Generation
- Core Geometry
- Primary Containment Integrity

- Reactor Coolant Pressure Boundary Integrity

The specific value or ranges of values chosen for each controlling parameter along with final verification of principal safety barrier performance is contained in the station's UFSAR.

b. Abnormal Degradation: Abnormal degradation is plant degradation beyond that analyzed for in the UFSAR. Abnormal degradation of a barrier may be indicated by the necessity of taking corrective action to restore the barrier's capability.

c. Unanalyzed Condition: The current UFSAR transient and accident analyses define the limiting conditions for operation and confirms the ability of the station's systems, structures, and components to prevent or mitigate the consequences of postulated transients and accidents. An event or condition that places the plant outside the bounds of any of these analyses represents an unanalyzed condition. An example of an event reportable as an unanalyzed condition that significantly degraded plant safety would be discovery that a system required to meet the single failure criterion does not do so.

d. Significantly degrades Plant Safety: An unanalyzed condition significantly degrades plant safety if it results in abnormal degradation, or has the potential to result in abnormal degradation of one of the principal safety barriers. The level of significance of these cases generally corresponds to the inability to perform a required safety function.

2. General

If the event or condition affects more than a single safety system or structure, or one of the principal safety barriers, reportability under this section should be reviewed. The stations are designed and licensed to adequately handle its Design Basis Accident along with its most limiting single failure. If an event occurs or condition exists that results in more equipment or systems being inoperable than covered by the plant's safety analysis, it may be in an unanalyzed condition. The definitions provided in 202.9.2 for these concepts need to be applied to determine reportability.

It is not intended that this section apply to minor variations in individual parameters, or to problems concerning single pieces of equipment. Any failure, or minor error in performing surveillance tests could produce a situation in which two or more often unrelated, safety-related components are out of service. Technically, this is an unanalyzed condition. However, these events should be reported only if they involve functionally related components or if they significantly compromise plant safety. For instance, if an event occurred where there could have been a failure of a safety system to properly complete a safety function, Section 202.9.7 "Event or Condition That Could Have Prevented the Fulfillment of Safety Function of Systems or Structures" should be reviewed for reportability. If an event occurred where a single cause actually made a component or group of components inoperable in redundant or independent trains or channels, of one or more systems having a safety function, Section 202.9.8 "Common-Mode Failures of Independent Trains or Channels" should be reviewed for reportability.

Section 202.9.11 "Single Cause that Could Have Prevented Fulfillment of the Safety Functions of Trains or Channels in Different Systems" should also be reviewed.

Deficiencies in the Fire Protection and App. R program should be evaluated under this criterion. In general, only programmatic breakdowns of the Fire Protection Program need to be reported. Programmatic breakdowns do not include failures to execute the program. Individual problems with Fire Protection Program Remedial Actions (e.g., fire watches) or fire protection/detection equipment should be evaluated with respect to actual cause. If the cause is related to a failure to put in place an element or elements of the Fire Protection Program, then this would be reportable. If however, the problem is related to isolated failures to execute due to human performance problems or isolated equipment failures, these conditions would not be reportable as programmatic breakdowns.

EXAMPLES

Reportable

a. Three studs were discovered missing on the horizontal missile shields located over the reactor vessel. Engineering analysis determined that the remaining studs were not sufficient to hold the shields during a postulated accident, thus, putting a principal safety barrier in a potentially degraded position.

b. Two weeks after painting the Diesel Generators, an operability test was performed and neither D/G was capable of starting since paint had been applied to key D/G components, preventing any movement. Upon discovery, the D/G components had to be scraped clean before the D/G's were able to start and load. The Unit had been at full power during the entire period.

c. Engineering determined that instrument loop inaccuracies could result in safety injection initiation on low pressurizer pressure at a lower RCS pressure than assumed in the accident analysis.

d. Fire barriers were found to be missing such that the required degree of separation for redundant safe shutdown trains is lacking. This event is reportable as an unanalyzed condition that significantly degraded plant safety.

e. With the unit in Mode 6, ultrasonic testing revealed a number of failed fuel rods (233 were identified in 88 of 109 fuel assemblies scheduled for reinsertion) that far exceed the anticipated number of failures. An ENS call is required because a principal safety barrier (fuel cladding) was found seriously degraded.

f. Steam Generator tube degradation is considered serious and reportable if the tubing fails to meet either of the following two performance criteria:

   1) Steam generator tubing shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, cooldown and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a margin of 3.0 against burst under normal steady state full power operation and a margin of 1.4 against burst under the limiting design basis accident concurrent with a safe shutdown earthquake.

   2) The primary to secondary accident induced leakage rate for the limiting design basis accident, other than a steam generator tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all steam generators and leakage rate for an individual steam generator. The licensing basis accident analyses typically assume a 1 gpm. primary to secondary leak rate per steam generator, except for specific types of degradation at specific locations where the tubes are confined, as approved by the NCR and enumerated in conjunction with the list of approved repair criteria in the licensee's design basis documents.

Non-Reportable

g. A main steam isolation valve closed while the plant was at 100% power as a result of a solenoid failure. Operations personnel reduced reactor power because of asymmetric power tilt and feedwater oscillations. No procedure existed for operating the unit in these conditions while the solenoid was being replaced. The event is not reportable because this condition would not have significantly degraded plant safety.

h. Upon review of historical test data on the 2A and 2B Component Cooling (KC) heat exchangers, both HX's were unknowingly inoperable during the same time period due to excessive fouling. {Note, although not reportable under these criteria, the condition is reportable as a loss of safety function for the KC system under Section 202.7.3 "Event or Condition That Could Have Prevented the Fulfillment of Safety Function of Systems or Structures."}

i. While in Cold Shutdown and mid-loop operation, the 2A Containment Spray (NS) pump suction valve was opened for VST with ND-1 and 2 open. This subjected the 2A NS train to Reactor Coolant pressure conditions. Overpressurization of the NS heat exchanger and a 10,000 gallon spill resulted.

j. A fire wrap, to which the licensee had committed, was missing from a safe shutdown train but another safe shutdown train was available in a different fire area, protected such that the required separation for safe shutdown is still provided, the event would not be reportable.

202.9.3 NATURAL PHENOMENON OR CONDITION THREATENING PLANT SAFETY (EXTERNAL THREAT)

§50.72	§50.73(a)(2)(iii)
There is no 50.72 requirement. Refer to the plant's Emergency Plan regarding declaration of an Emergency Class.	Licensees shall report: "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."

1. General

   This section applies only to acts of nature (e.g., tornadoes, earthquakes, fires, hurricanes, and floods) and external hazards (e.g., industrial and transportation accidents). This section requires events to be reported if the threat or actual damage challenges the ability of plant personnel to continue to operate in a safe manner, including the orderly shutdown and maintenance of safe shutdown conditions.

2. Actual Threat

   Judgment should be used to determine if a condition actually threatens the plant. For example, a small brush fire in a remote area of the site that was quickly controlled and did not present a threat to the plant need not be reported. However, a major forest fire or hurricane moving in the direction of the plant and thus threatened plant equipment may be reportable. There are no prescribed limits, but in general, situations involving only monitoring by the plant's staff are not reportable. But when preventative actions are taken or if there are serious concerns, then the situation should be carefully reviewed for reportability.

3. Significantly Hampers Personnel

   To be reportable, an event need not prevent station personnel from performing their duties. It is only necessary that they be significantly hampered, hindered, or interfered with in the performance of safety-related activities. If the condition makes performing routine safety-related functions significantly more difficult, it is reportable. For example, in a snowstorm, judgment may be based on the amount of snow, the extent to which additional assistance could have been available in an emergency, and the length of time the condition existed. If station management decides to allow all non-essential personnel to go home early as a conservative, precautionary step, considering the safety of the employees during their travel, the condition is not reportable.

EXAMPLES

Reportable

a. The National Weather Service issued a Tornado Warning for York County. A tornado was spotted approaching the Catawba Nuclear Station. The tornado touched down within the plant protected area boundary. Operations declared an Unusual Event. The tornado continued on and struck the SSF and inflicted significant damage. Operations upgraded to an Alert. The tornado left the site and after assessing damage, the Alert was terminated. An LER is also required since a safety-related structure (SSF) was significantly damaged.

Non-Reportable

b. Hurricane Hugo was within 150 miles of the plant and appeared to be heading toward the direction of the plant. Since the force of the hurricane had diminished significantly by the time it neared the station and an insignificant amount of damage was done, an LER was not required.

c. One day in February it began snowing considerably. When the accumulation reached 4 inches, station management made a decision to allow non-essential plant personnel to leave early that afternoon since the

snow was expected to continue into the night and decreasing temperatures would make traveling home especially hazardous in the evening.

d. A Tornado warning

202.9.4 LOSS OF EMERGENCY ASSESSMENT, RESPONSE, OR COMMUNICATIONS

§50.72(b)(3)(xiii)	10 CFR 50.73
Licensees shall report: "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)."	[No corresponding Part 50.73 requirement.]

1. Loss of Emergency Assessment Capability

   Emergency assessment capability is defined in the station Emergency Plan and implementing procedures. A major loss of emergency assessment capability would include those events or conditions that significantly impair the operators ability to determine the status of the key station parameters and take the proper course of actions in the event of an emergency. Engineering judgment may be needed to determine the significance of the loss in terms of the equipment and the length of time involved. For example, the unavailability of 1 redundant component or train such as a radiation monitor or OAC, for a period of time as permitted by Tech Specs or administrative procedures, generally is not reportable.

2. Loss of Offsite Response Capability

   A major loss of offsite response capability includes those events that would significantly impair the fulfillment of the station's Emergency Plan. Loss of offsite response capability may typically include the loss of plant access, emergency offsite response facilities, or public prompt notification system (a loss of more than 25% of the station's total sirens, other alerting systems (e.g., tone alert radios), or more importantly, the lost capability to alert a large segment of the population (for more than 1 hour) would be considered a major loss and, therefore, reportable per this section).

3. Loss of Communications Capability

   A major loss of communications capability would include the loss of ENS and commercial telephone lines.

   The loss of the following is considered reportable: a significant amount of control room annunciators or monitors (such as an annunciator panel, a number of annunciators on various panels, or all plant vent stack radiation monitors), Control Room or shutdown panel habitability (from complete loss to using self-contained breath apparatus), or loss of multiple independent safety assessment equipment or systems concurrently.

   A loss of the Safety Parameter Display System (SPDS) required by Three Mile Island Action Plan [Ref. NUREG-0737], is not necessarily reportable under this criterion if other (safety-related) indication(s) is (are) available for use by the operators (or Emergency Response Organization) to adequately monitor plant safety parameters. Extended losses of SPDS and related plant computer capabilities; e.g., $\geq$ 4 hours, should be considered for reporting if, coincident with other adverse conditions, such a loss would significantly hamper the plant's ability to deal with an accident or emergency.

EXAMPLES

Reportable

a. More than 25% of the stations' total alert sirens were disabled because of loss of power as a result of severe weather.

b. ENS and commercial phone lines were discovered to have been cut while crews were digging.

c. The local sheriff notified the station that all roads to and from the plant were closed because of a heavy snow storm. The station had 2 full shift crews on site to support plant operations and no emergency declaration was made. An ENS call is required because the road closing may prevent the plant staff from adequately staffing the TSC, or from fully responding to some emergencies.

Non-Reportable

d. It was observed during siren testing that 5 of 52 alert sirens around the EPZ failed to function. This was not considered to be a major loss of the offsite response capability.

e. York County was performing a scheduled quarterly full cycle siren test and as they were performing the procedure there was a step requiring the turning of a key in order to make the sirens sound. The sirens did not sound; however, within minutes the individual realized an improper arming configuration, rearmed the siren, turned the key, and the sirens function properly. This event would not be reportable because the sirens were only disabled during the quarterly test.

202.9.5 INTERNAL THREAT TO PLANT SAFETY

§50.72	§50.73(a)(2)(x)
There is no 50.72 requirement. Refer to the plant's Emergency Plan regarding declaration of an Emergency Class.	Licensees shall report: "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."

1. General

This section pertains to threats internal to the station. Fires, toxic gas releases, and radioactive releases are not the only threats that may require reporting under these provisions. The criterion to be applied in each case is whether the event poses an actual threat to the safety of the plant or significantly hampers personnel in the performance of duties necessary for the safe operation of the plant. The significant hampering criterion is pertinent to "the performance of duties necessary for safe operation of the nuclear power plant." One way to evaluate this is to ask if one could seal the room in question (or disable the function in question) for a substantial period of time and still operate the plant safely. Actions such as room evacuations that are purely precautionary would not constitute significant hampering if the performance of duties necessary for the safe operation of the plant can still be performed in a timely manner. Refer to Section 202.9.3, "Natural Phenomenon or Condition Threatening Plant Safety (External Threat)" of this directive for additional discussion on "actual threats" and "significantly hampering personnel".

EXAMPLES

Reportable

a. A turbine building evacuation was ordered when a large area of the floor was contaminated. Condensate demineralizer resin was being transferred through a cleaner to a mix-and-hold tank. As the tank was being pressurized, a mispositioned inlet valve allowed 50 to 100 gallons of water/resin to blow out into the turbine building. The ventilation system spread loose surface contamination through various turbine building locations. Eight operators and construction workers were contaminated.

An LER is required because plant operators were significantly hampered in the performance of their duties because they were evacuated from areas containing safety-related equipment and would have been delayed in their duties during an emergency.

Non-Reportable

b. A small hydrazine leak occurred in the yard as a result of transporting a drum that was inadvertently punctured. An ENS notification is not required under this reporting section since the toxic gas leak posed no threat to the safety of the plant, nor did it significantly hamper personnel in the performance of their duties necessary for plant safety. However, depending on the circumstances, this event may be reportable under other 50.72 criteria such as Section 202.8.4, "News Release or Other Government Notifications" (notification to outside agencies) if not an emergency class declaration.

202.9.6 SYSTEM ACTUATIONS

§50.72(b)(3)(iv)	§50.73(a)(2)(iv)
Licensees shall report: "(A) Any event or condition that results in valid actuation of any of the systems listed in paragraph (b)(3)(iv)(B) of this section except when the actuation results from and is part of a pre-planned sequence during testing or reactor operation. (B) The systems to which the requirements of paragraph (b)(3)(iv)(A) of this section apply are: 1. Reactor protection system (RPS) including: reactor scram and reactor trip.[5] 2. General containment isolation signals affecting containment isolation valves in more than one system or multiple main steam isolation valves (MSIVs). 3. Emergency core cooling systems(ECCS) for pressurized water reactors (PWRs) including: high-head, intermediate-head, and low-head injection systems and the low pressure injection function of residual (decay) heat removal systems 4. PWR auxiliary or emergency feedwater system. 5. Containment heat removal and depressurization systems, including containment spray and fan cooler systems. 6. Emergency ac electrical power systems, including: emergency diesel generators (EDGs); hydroelectric facilities used in lieu of EDGs at the Oconee Station." [5] Actuation of the RPS when the reactor is critical is reportable under paragraph (b)(2)(iv) of this section.	Licensees shall report: "(A) Any event or condition that resulted in manual or automatic actuation of any of the systems listed in paragraph (a) (2) (iv) (B) of this section, except when: (1) The actuation resulted from and was part of a pre-planned sequence during testing or reactor operation; or (2) The actuation was invalid and; (i) Occurred while the system was properly removed from service; or (ii) Occurred after the safety function had been completed. (B) The systems to which the requirements of paragraph (a)(2)(iv)(A) of this section apply are: 1. Reactor protection system (RPS) including: reactor scram and reactor trip.[5] 2. General containment isolation signals affecting containment isolation valves in more than one system or multiple main steam isolation valves (MSIVs). 3. Emergency core cooling systems (ECCS) for pressurized water reactors (PWRs) including: high-head, intermediate- head, and low-head injection systems and the low pressure injection function of residual (decay) heat removal systems. 4. PWR auxiliary or emergency feedwater system. 5. Containment heat removal and depressurization systems, including containment spray and fan cooler systems. 6. Emergency ac electrical power systems, including: emergency diesel generators (EDGs); hydroelectric facilities used in lieu of EDGs at the Oconee Station. 7. Emergency service water systems that do not normally run and that serve as ultimate heat sinks.

1. Definitions

   a. Valid actuations are those actuations that result from "valid signals" or from intentional manual initiation, unless it is part of a preplanned test. Valid signals are those signals that are initiated in response to actual plant conditions or parameters satisfying the requirements for initiation of the safety function of the system. They do not include those that are the result of other signals

   b. Invalid actuations are, by definition, those that do not meet the criteria for being valid. Thus, invalid actuations include actuations that are not the result of valid signals and are not intentional manual actuations.

      Some invalid actuations are still reportable (see examples).

   These systems specific to each station are listed in Appendix A.

   c. RPS Actuation: (1) Receipt of a Solid State Protection System (SSPS) signal(s) necessary to activate the RPS system, or (2) manual or automatic actions that activate the RPS system without the presence of an SSPS signal(s).

   d. Actuation of multichannel systems is defined as actuation of enough channels to complete the minimum actuation logic. Therefore, single channel actuations, whether caused by failures or otherwise, are not reportable if they do not complete the minimum actuation logic. Note, however, that if only a single logic channel actuates when, in fact, the system should have actuated in response to plant parameters, this would be reportable under these paragraphs as well as under 10 CFR 50.72(b)(3)(v) and 10 CFR 50.73(a)(2)(v) (event or condition that could have prevented the fulfillment of the safety function of ....).

   e. Preplanned Actuation: A preplanned system actuation is the initiation of a particular system as called for by an approved operating or testing procedure.

   f. Properly Removed From Service: The component or system is intentionally mechanically or electrically disabled such that it is not capable of performing its intended safety function, and station procedures for removing equipment from service have been implemented (e.g., required clearance documentation, equipment and control board tagging, etc.).

   g. General Containment Isolation Signals: General containment isolation signals affecting containment isolation valves in more than one system or multiple main steam isolation valves do not include the containment ventilation isolation (S_H). The referenced signals include those signals that are provided to systems that mitigate the consequences of a significant event and are credited in the Chapter 15 of the UFSAR. The S_H is not credited in Chapter 15 of the UFSAR nor is it credited as an engineered safety feature.

2. Reportability

   These paragraphs require events to be reported whenever one of the specified systems actuates either manually or automatically.

   These systems are provided to mitigate the consequences of a significant event and, therefore:

   a. they should work properly when called upon, and

   b. they should not be challenged frequently or unnecessarily.

   The NRC is interested both in events where a system was needed to mitigate the consequences (whether or not the equipment performed properly) and events where a system operated unnecessarily. Generally, the NRC would not consider this to include single component actuations because single components of complex systems, by themselves, usually do not mitigate the consequences of significant events. However, in some cases a component would be sufficient to mitigate the event (i.e., perform the safety function) and its actuation would then be reportable.

   Since single trains do mitigate the consequences of significant events, train level actuations are reportable. In this regard, actuation of a diesel-generator is considered to be an actuation of a train and not an actuation of a single component because a diesel generator is needed to mitigate the event (performs the safety function) .

The ECCS contains systems that have no other operating function as well as systems that are shared with other systems. Actuations of ECCS systems which are shared with other systems is reportable only when they are performing their safety function.

An actuation of any of the systems named in § 50.73(a)(2)(iv)(B) is reportable under §50.73(a)(2)(iv)(A) [a 60-day *report]* unless the actuation resulted from and was part of a preplanned sequence during testing or reactor operation or the actuation was invalid and occurred while the system was properly removed from service or occurred after the safety function had been already completed. As indicated in § 50.73(a)(1), in the case of an invalid actuation reported under § 50.73(a)(2)(iv)(A) other than actuation of the reactor protection system (RPS) when the reactor is critical the licensee may, at its option, provide a telephone notification to the NRC Operations Center within 60 days after discovery of the event instead of submitting a written LER. In these cases the telephone 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.

VALID Signals versus instrument drift or mis-calibration: If a transient is in progress such that a plant parameter approaches an actuation setpoint, and an actuation of a listed system occurs sooner than expected, (i.e., outside of the allowed tolerances), the signal should initially be considered VALID and reported per 50.72 (This position assumes that the transient would have reached the actuation setpoint if not terminated or mitigated by the actuation. The 50.72 notification can be retracted if subsequent analysis concludes that the transient could not have reached the setpoint if the actuation were assumed not to occur. It is not expected that such an analysis could be completed within the 8 hour notification requirement.). Conversely, if a listed system is operating at essentially steady-state (i.e.,within normal operating range) and is apparently actuated due to calibration drift of the instrumentation outside of the allowed tolerances, then, the signal should be treated as INVALID.

3. Reporting Exceptions

Except for critical scrams, invalid actuations are not reportable by telephone under § 50.72. In addition, invalid actuations are not reportable under § 50.73 in any of the following circumstances:

(A) The invalid actuation occurred when the system is already properly removed from service. This means all requirements of plant procedures for removing equipment from service have been met. It includes required clearance documentation, equipment and control board tagging, and properly positioned valves and power supply breakers.

(B) The invalid actuation occurred after the safety function has already been completed. An example would be RPS actuation after the control rods have already been inserted into the core.

However, if one of the specified systems actuate during the planned operation or test in a way that is not part of the planned procedure, such as at the wrong step, that event is reportable.

EXAMPLES

Reportable

Note: {For the reportable examples provided, assume the actuation is the result of a valid signal, is not part of a pre-planned sequence in a procedure and the system has not been removed from service.} This note applies to examples a-k.

a. Any manual or automatic actuation of the reactor trip switchgear is reportable.

b. Initiation of a containment isolation signal constitutes an actuation whether or not the containment isolation valve actually repositions.

c. The opening of a Hydrogen Skimmer fan header isolation valve and the subsequent starting of a Hydrogen Skimmer fan is an actuation.

d. The starting or speed change of a Reactor Building Cooling Unit fan, as a result of an ES Channel 5 or 6 signal, is reportable. (ONS)

e. The starting of any of the ECCS pumps to mitigate the consequences of a significant event is an activation.

f. Any manual or automatic actuation of the Auxiliary (CA)/Emergency Feedwater(EFW) system is reportable. At ONS, The Steam Generator Dry-out Protection Circuit does not perform a credited safety function. By definition, it is an INVALID signal so EFW actuations due to it are NOT reportable under 50.72 but are reportable under 50.73.

g. Unplanned Diesel Generator starts, and Keowee starts resulting from ES Channel 1 or 2 signals, are reportable.

h. Emergency power switching logic actuations of 4160V breakers which result from ES 1 or 2 signals are reportable. (ONS)

i. During a significant operational transient, an "ice condenser door open" alarm was received in the Control Room. This is a reportable event because if the Ice Condenser doors are off their seals, the equipment is considered actuated.

j. Swaps of Nuclear Service Water pump's suction from the lake to the Standby Nuclear Service Water pond are reportable under 50.73. However, they are NOT reportable to the NRC Operations Center under 10 CFR 50.72(b)(3)(iv).

k. Actuation of the SSF- ASW pump is reportable under 50.73. However, it is NOT reportable to the NRC Operations Center under 10 CFR 50.72(b)(3)(iv). (ONS)

Non-Reportable

a. Equipment actuation because of a signal generated by EMF's (radiation monitors) is not reportable.

b. RPS actuates after all control rods and banks have already been inserted in the core.

c. During surveillance testing of the main steam isolation valves (MSIVs), an operator incorrectly closed MSIV "D" when the procedure specified closing MSIV "C". This event is not reportable because the event is an inadvertent actuation of a component of a system.

d. Movement of a single valve swapped the suction of the Nuclear Service Water System to the Auxiliary Feedwater pump suction. Since only a single component was actuated and the valve could not mitigate the consequences of an event by itself, the valve movement is not reportable as an actuation.

202.9.7 EVENT OR CONDITION THAT COULD HAVE PREVENTED THE FULFILLMENT OF SAFETY FUNCTION OF SYSTEMS OR STRUCTURES

§50.72(b)(3)(v)	§50.73(a)(2)(v)
Licensees shall report: "Any event or condition that at the time of discovery could have prevented the fulfillment of the safety function of structures or systems that are needed to: a. Shut down the reactor and maintain it in a safe shutdown condition; b. Remove residual heat; c. Control the release of radioactive material; or d. Mitigate the consequences of an accident." §50.72(b)(3)(vi) "Events covered in paragraph (b)(3)(v) of this section may include one or more procedural errors, equipment failures, and/or discovery of design, analysis, fabrication, construction, and/or procedural inadequacies. However, individual component failures need not be reported pursuant to paragraph (b)(3)(v) of this section if redundant equipment in the same system was operable and available to perform the required safety function".	Licensees shall report: "Any event or condition that could have prevented the fulfillment of the safety function of structures or systems that are needed to: a. Shut down the reactor and maintain it in a safe shutdown condition; b. Remove residual heat; c. Control the release of radioactive material; or d. Mitigate the consequences of an accident." §50.73(a)(2)(vi) "Events covered in paragraph (a)(2)(v) of this section may include one or more procedural errors, equipment failures, and/or discovery of design, analysis, fabrication, construction, and/or procedural inadequacies. However, individual component failures need not be reported pursuant to paragraph (a)(2)(v) of this section if redundant equipment in the same system was operable and available to perform the required safety function".

Note: This reporting criterion shall be reviewed for applicability to any plant condition that caused both trains to be inoperable.

1. General

The intent of this section is to capture those events where there could have been a failure of a safety system to properly complete a safety function, regardless of when the failures were discovered or whether the system was needed at the time. The event must be reported regardless of the situation or condition that caused the system to be unavailable, and regardless of whether or not an alternate safety system could have been used to perform the safety function.

The level of judgement for reporting an event or condition under this criteria is a reasonable expectation of preventing fulfillment of safety function.

If the event or condition could have prevented fulfillment of the safety function at the time of discovery an ENS notification is required. If it could have prevented fulfillment of the safety function at any time within 3 years of the date of discovery an LER is required.

The applicability of this section includes those safety systems designed to mitigate the consequences of an accident (e.g., containment isolation). Hence, minor operational events involving a specific component such as valve packing leaks, which could be considered a lack of control of radioactive material, should not be reported under this section.

In determining the reportability of an event or condition that affects a system, it is not necessary to assume an additional random single failure in that system; however, it is necessary to consider other existing plant conditions.

It should be noted that a reportable condition does not exist if you knowingly disable a safety function by removing all trains of a safety system from service provided this is done within a Tech Spec action statement time limit and within an approved procedure (Which is presumed to have properly evaluated the risk and need

for compensatory actions or controls.). However, it appears that it is the intent of the NRC to report any failure, or discovery of a procedure deficiency, that unexpectedly prevented the fulfillment of the safety function (In such a case the approved procedure was NOT approved with the thought in mind that it defeated a safety function.).

This logic is extended to the situation where one train of a system is removed from service for maintenance or testing (in accordance with Tech Specs) and the redundant train fails or is discovered to be inoperable. The resulting condition is an unplanned loss of safety function and the event is reportable. Again, planned removal would not be reportable.

In regards to single-train systems, there are a limited number of single-train systems that perform safety functions. At Duke, specifically at ONS, the single-train system is the Standby Shutdown Facility (SSF). The SSF is included in ONS' Tech Specs but is not credited in Chapter 15 of the plant's safety analysis. Existing guidance in NUREG 1022 specifies that a loss of a single-train system is reportable even though the plant's Tech Specs may allow such a condition to exist for a limited length of time. The guidance was modified and clarified by specifying that the inclusion of the system in the plant's Tech Specs indicated that the system was "needed" to perform one of the designated safety functions and was therefore subject to the reporting requirement. The reference to the Tech Specs was based on an assumption that if a system was included in the Tech Specs, then credit for the system was taken in the UFSAR. However, the NRC has reconsidered this position and now concludes that in order for the failure of a single-train system to be reportable as a loss of safety function, the system must be credited in mitigating design basis accidents described in Chapter 15 of the plant's safety analysis.

Therefore, failure of the SSF at ONS is not reportable per the criterion of this section simply because it is in the plant's Tech Specs. In order for the failure to be reportable as a loss of safety function, the system must be credited in the plant's safety analysis.

2. Single Train/Common-Mode Failure

These reporting criteria are not meant to require reporting of a single, independent component failure that makes only one functionally redundant train inoperable. The following conditions, however, are reportable:

- an actual single event or condition that disabled multiple trains of a safety-related system
- an actual event or condition that disabled one train of a safety-related system and could have affected a redundant train
- a condition or potential single event that could have disabled multiple trains of a safety-related system

Engineering judgement should be used when these criteria are applied to those few systems with more than 2 redundant trains (e.g., MNS/CNS CA system).

3. Non-Reportable Events or Conditions

- failures that affect inputs or services to systems that have no safety function
- defective component(s) that has not been installed
- unrelated component failures in several different safety systems
- a single stuck control rod that alone would not have prevented the fulfillment of a reactor shutdown
- removal of a system or part of a system from service as part of a planned evolution for maintenance or surveillance testing when done in accordance with an approved procedure and the plant's Tech Specs (unless a condition is discovered that could have prevented the system from performing its function)

A design or analysis defect or deviation is reportable under this criterion if it could have prevented fulfillment of the safety function of structures or systems defined in the rules. Reportability of a design or analysis defect or deviation under this criterion should be judged on the same basis that is used for other conditions, such as operator errors and equipment failures. That is, the condition is reportable if there is a reasonable expectation of preventing fulfillment of the safety function. Alternately stated, the condition is reportable if there was reasonable doubt that the safety function would have been fulfilled if the structure or system had been called upon to perform it.

OTHER EXAMPLES

Reportable

a. During a refueling outage, the equipment hatch was discovered open 1/4" after containment integrity had been established.

b. While train 'A' VC/YC was inoperable due to maintenance, the 'B' train YC chiller tripped and could not be restarted. Tech Spec 3.0.3 was entered for 90 minutes because both trains of the VC system were inoperable. There was no load reduction since operators felt that 1 of the trains would be back in service within 2 hours. This event is reportable as a loss of safety function.

Non-Reportable

c. While performing a main steam line Pressure Instrument Functional Test and Calibration, a switch was found to actuate at 853 psig. The Tech Spec limit is 825 + 15 psig head correction. The redundant switches were operable. The cause of the occurrence was setpoint drift. The switch was recalibrated, tested successfully per procedure and returned to service. The event is not reportable due to the drift of a single pressure switch unless it could have caused a system to fail to fulfill its safety function.

202.9.8 COMMON-MODE FAILURES OF INDEPENDENT TRAINS OR CHANNELS

10 CFR 50.72	§50.73(a)(2)(vii)
[No corresponding Part 50.72 requirement.]	Licensees shall report: "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."

1. General

This section requires those events to be reported where a single cause made a component or group of components to become inoperable in redundant or independent trains or channels, of one or more systems having a safety function (common-mode failures). Failures reported under this part of the rule should be actual failures, not potential ones.

Such failures can be simultaneous which occur from a single initiating cause, or sequential (i.e., cascade failures), such as the case where a single component failure results in the failure of one or more additional components.

To be reportable, however, the event or failure must result in or involve the failure of independent portions of more than one train or channel in the same or different systems. For example, if a single cause or condition resulted in inoperable components in Train "A" of the KC System and Train "B" of the Nuclear Service Water (RN) system (i.e., train that is assumed in the safety analysis to be independent) the event is reportable. Additionally, one function of the "B" train of the RN system is to provide cooling for the "B" train of the KC System, and since "B" train of the RN system cannot perform its cooling function, then "B" train of the KC

system is also inoperable. Thus, both trains of the KC system are inoperable and unable to perform their safety function.

EXAMPLES

Reportable

a. Events reportable under Section 202.9. 7, "Event or Condition That Could Have Prevented the Fulfillment of Safety Function of Systems or Structures" of this directive are also reportable per this section provided: (1) the system involved has 2 or more trains or channels, and (2) the inoperable condition is as a result of "actual" failures.

b. The station found 11 inoperable snubbers during periodic testing. All the snubbers failed to lock up when required. These failures rendered trains in 3 systems inoperable. This condition is reportable because the condition indicated a generic common-mode problem that caused numerous multiple independent trains in one or more safety systems to become inoperable.

Non-Reportable

c. Design investigation indicated that electrical power feed to the VE filter train heaters can be postulated to drop to a sustained voltage that would place power dissipation outside the required range. Both trains of VE were considered inoperable. This condition is not reportable under this section because the condition was not an actual failure of both trains, but a postulated event that "could have" prevented the fulfillment of the safety function of the VE system, and is reportable under Section 202. 9. 7, "Event or Condition That Could Have Prevented the Fulfillment of Safety Function of Systems or Structures," ref. Example 2.

202.9.9 AIRBORNE OR LIQUID EFFLUENT RELEASE EXCEEDING 20 TIMES APPENDIX B

§50.72	50.73(a)(2)(viii)
There is no 50.72 requirement. Refer to the plant's Emergency Plan regarding declaration of an Emergency Class. Refer to § 50.72(b)(2)(xi) below regarding a news release or notification of another agency. Refer to § 20.2202 regarding events reportable under that section.	Licensees shall report: 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." "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."

1. General

This section is similar to Part 20.2203, but places a lower threshold for reporting events at commercial power reactors. The lower threshold is based on the significance of the breakdown of the station's program necessary to have a release of this size, rather than on the significance of the impact of the actual release. For a release that takes less than 1 hour, normalize the release to 1 hour (e.g., release of 15 minutes, multiply by 4). For releases that last more than 1 hour, use the highest release for any continuous 60 minute period. It often takes a period of time to assess the magnitude of a radioactive release

EXAMPLE

Reportable

a. During routine maintenance on a pressure actuated valve in the waste gas system, an unplanned radioactive release to the environment was detected by a radiation alarm. The release occurred when an isolation valve, required to be closed, was inadvertently left open. This allowed radioactive gas from the waste gas decay tank to escape through a pressure gage connection that had been opened to vent the system. The concentration at the site boundary, averaged over 1 hour, was estimated by the station to exceed the limits specified in §50.73 (a)(2)(viii).

202.9.10 CONTAMINATED PERSON REQUIRING TRANSPORT TO OFFSITE MEDICAL FACILITY

§50.72(b)(3)(xii)	10 CFR 50.73
Licensees shall report: "Any event requiring the transport of a radioactively contaminated person to an offsite medical facility for treatment."	[No corresponding Part 50.73 requirement.]

1. General

Contaminated, in this case, refers to either contaminated clothing, the person, or both. If the initial onsite survey is incomplete and there is a potential for contamination, the station should assume the individual is contaminated and make the ENS notification. Often the full extent of radioactive contamination on an injured individual may not be known until after arrival at the hospital. If no potential for contamination is present, reporting of the transport to offsite medical facilities is not required.

EXAMPLES

Reportable

a. A contract worker experienced a back injury lifting a tool while working in the reactor building and was considered to be potentially contaminated because his back could not be surveyed. An ENS call was made immediately. The individual was later found not to be contaminated and an update ENS notification was made.

Non-Reportable

b. The station transported a high school student from its PAP to a medical office because the student had stomach pains. This event is not reportable because no potential for contamination was present.

c. An employee cut his head in the containment pipe chase. RP reported that the individual was not contaminated but was being transported to the hospital. The event is not reportable because no potential for contamination was present.

202.9.11 SINGLE CAUSE THAT COULD HAVE PREVENTED FULFILLMENT OF THE SAFETY FUNCTIONS OF TRAINS OR CHANNELS IN DIFFERENT SYSTEMS

§50.72	§ 50.73 (a)(2)(ix)
There is no corresponding requirement in requirement in § 50.72.	(A) Any event or condition that as a result of a single cause could have prevented the fulfillment of a safety function for two or more trains or channels in different systems that are needed to: 1) Shut down the reactor and maintain it in a safe shutdown condition; 2) Remove residual heat; 3) Control the release of radioactive material; or

	4) Mitigate the consequences of an accident. (B) Events covered in paragraph (ix)(A) of this section may include cases of procedural error, equipment failure, and/or discovery of a design, analysis, fabrication, construction, and/or procedural inadequacy. However, licensees are not required to report an event pursuant to paragraph (ix)(A) of this section if the event results from: 1) A shared dependency among trains or channels that is a natural or expected consequence of the approved plant design; or 2) Normal and expected wear or degradation."

1. General

Subject to the two exclusions stated in the rule, this criterion captures those events where a single cause could have prevented the fulfillment of the safety function of multiple trains or channels, but the event:

(1) Would not be captured by §§ 50.73(a)(2)(v) and 50.72(b)(3)(v) [event or condition that could have prevented fulfillment of the safety function of structures and systems needed to ...] because the affected trains or channels are in different systems; and

(2) Would not be captured by § 50.73(a)(2)(vii) [common cause inoperability of independent trains or channels] because the affected trains or channels are either:

(a) Not assumed to be independent in the plant's safety analysis; or

(b) Not both considered to be inoperable.

This criterion is closely related to §§ 50.73(a)(2)(v) and 50.72(b)(3)(v) [event or condition that could have prevented fulfillment of the safety function of structures and systems needed to: shut down the reactor and maintain it in a safe shutdown condition; remove residual heat; control the release of radioactive material; or mitigate the consequences of an accident]. Specifically:

- The meaning of the term "could have prevented the fulfillment of the safety function" is essentially the same for this criterion as it is for §§ 50.73(a)(2)(v) and 50.72(b)(3)(v) [i.e., there was a reasonable expectation of preventing the fulfillment of the safety function(s) involved]. However, in contrast to §§ 50.73(a)(2)(v) and 50.72(b)(3)(v), reporting under this criterion applies to trains or channels in different systems. Thus, for this criterion, the safety function that is affected may be different in different trains or channels.

- In contrast to §§ 50.73(a)(2)(v) and 50.72(b)(3)(v), reporting under this criterion applies only to a single cause. Also, in contrast to §§ 50.73(a)(2)(v) and 50.72(b)(3)(v), this criterion does not apply to an event that results from a shared dependency among trains or channels that is a natural or expected consequence of the approved plant design. For example, this criterion does not capture failure of a common electrical power supply that disables Train A of AFW and Train A of HPSI, because their shared dependency on the single power supply is a natural or expected consequence of the approved plant design.

- Similar to §§ 50.73(a)(2)(v) and 50.72(b)(3)(v), this criterion does not capture events or conditions that result from normal and expected wear or degradation. For example, consider pump bearing wear that is within the normal and expected range. In the case of two pumps in different systems, this criterion categorically excludes normal and expected wear. In the case of two pumps in the same system, normal and expected wear should be adequately addressed by normal plant operating and maintenance practices and thus should not indicate a reasonable expectation of preventing fulfillment of the safety function of the system.

The level of judgment for reporting an event or condition under this criterion is a reasonable expectation of preventing fulfillment of a safety function. In the discussions that follow, several different expressions such as "would have," "could have," "alone could have," and "reasonable doubt" are used to characterize this standard. In the staff's view, all of these should be judged on the basis of a reasonable expectation of preventing fulfillment of

the safety function.

The intent of this criterion is to capture those events where, as a result of a single cause, there would have been a failure of two or more trains or channels to properly complete their safety function, regardless of whether there was an actual demand. For example if, as a result of a single cause, a train of the high pressure safety injection system and a train of the auxiliary feedwater system failed, the event would be reportable even if there was no demand for the systems' safety functions.

Examples of a single cause responsible for a reportable event may include cases of procedural error, equipment failure, and/or discovery of a design, analysis, fabrication, construction, and/or procedural inadequacy. They may also include such factors as high ambient temperatures, heat up from energization, inadequate preventive maintenance, oil contamination of air systems, incorrect lubrication, or use of non-qualified components.

The event is reportable if, as a result of a single cause, there would have been a failure of two or more trains or channels to properly complete their safety function, regardless of whether the problem was discovered in both trains at the same time.

Trains or channels for reportability purposes are defined as those trains or channels designed to provide protection against single failures. Many systems containing active components are designed as at least a two-train system. Each train in a two-train system can normally satisfy all the system functions.

This criterion does not include those cases where trains or channels are removed from service as part of a planned evolution, in accordance with the plant's Tech Specs. For example, if a licensee removes two trains from service to perform maintenance, and the Tech Specs permit the resulting configuration, and the trains are returned to service within the time limits specified in the Tech Specs, the action need not be reported under this paragraph. However, if, while the trains or channels are out of service, the licensee identifies a single cause that could have prevented the trains from performing their safety functions (e.g., the licensee finds a set of relays that is wired incorrectly), that condition must be reported.

The definition of the systems included in the scope of this criterion is provided in the rule itself. It includes systems required by the Tech Specs to be operable to perform one of the four functions specified in the rule. It is not determined by the phrases " safety- related," "important to safety," or "ESF."

Trains or channels must operate long enough to complete their intended safety functions as defined in the safety analysis report.

Generic Letter 91-18 provides guidance on determining whether a system is operable.

The application of this reporting criterion and other reporting criteria involves the use of engineering judgment. In the case of this criterion, a technical judgment must be made as to whether a failure or operator action that did actually disable one train or channel, could have, but did not, disable another train or channel. If so, this would constitute an event that "could have prevented" the fulfillment of the safety function of multiple trains or channels, and, accordingly, must be reported.

Reporting is required if one train or channel fails and, as a result of a single cause, there is reasonable doubt that another train or channel would remain operational until it completed its safety function or is repaired. For example, if a pump fails because of improper lubrication, and engineering judgment indicates that there is a reasonable expectation that another pump in a different system, which was also improperly lubricated, would have also failed before it completed its safety function, then the event is reportable under this criterion.

Reportable conditions under this criterion include the following:

- an event or condition that disabled multiple trains because of a single cause
- an event or condition where one train is disabled; in addition, (1) the underlying cause that disabled one train of a system could have failed another train and (2) there is reasonable expectation that the second train would not complete its safety function if it were called upon to do so
- an observed or identified event or condition that could have prevented fulfillment of the safety function of multiple trains or channels as a result of a single cause

The following types of events or conditions generally are not reportable under this criterion:

- failures that affect inputs or services to systems that have no safety function (unless it could have prevented the performance of a safety function of an adjacent or interfacing system)
- a defective component that was delivered, but not installed
- removal of trains or channels from service as part of a planned evolution for maintenance or surveillance testing when done in accordance with the plant's Technical Specifications (unless a condition is discovered that could have prevented multiple trains or channels from performing their safety functions)
- independent failure of a single component (unless it is indicative of a generic problem, which could have caused failure of multiple trains or channels)
- a procedure error that could have resulted in defeating the safety function of multiple trains or channels but was discovered before procedure approval
- a failure of a system used only to warn the operator where no credit is taken for it in any safety analysis and it does not directly control any of the four safety functions in the rule
- a single stuck control rod that would not have prevented the fulfillment of a reactor shutdown
- unrelated component failures in different trains or channels

Minor operational events involving a specific component such as valve packing leaks, which could be considered a lack of control of radioactive material, should not be reported under this criterion.

A design or analysis defect or deviation is reportable under this criterion if it could have prevented fulfillment of the safety function of multiple trains or channels. Reportability of a design or analysis defect or deviation under this criterion should be judged on the same basis that is used for other conditions, such as operator errors and equipment failures. That is, the condition is reportable if there is a reasonable expectation of preventing fulfillment of the safety function(s) of multiple trains or channels. Alternatively stated, the condition is reportable if there was reasonable doubt that the safety functions of multiple trains or channels would have been fulfilled if there were demands for them.

EXAMPLES

Reportable

a) Solenoid Operated Valve Deficiency

During testing, two containment isolation valves failed to function as a result of improper air gaps in the solenoid operated valves that controlled the supply of instrument air to the containment isolation valves.

The valves were powered from the same electrical division. Thus, § 50.73(a)(2)(vii) [common cause inoperability of independent trains or channels] would not apply. The two valves isolated fluid process lines in two different systems. Thus § 50.73(a)(2)(v) [condition that could have prevented fulfillment of the safety function of a structure or system] would apply only if engineering judgment indicates there was a reasonable expectation of preventing fulfillment of the safety function for redundant valves within the same system. Or, alternatively, there was reasonable doubt that the safety function would have been fulfilled if the affected trains had been called upon to perform them. However, this criterion would certainly apply if a single cause (such as a design inadequacy) induced the improper air gaps, thus preventing fulfillment of the safety function of two trains or channels in different systems.

b) Degraded Valve Stems

A motor operated valve in one train of a system was found with a crack 75 percent through the stem. Although the valve stem did not fail, engineering evaluation indicated that further cracking would occur which could have prevented fulfillment of its safety function. As a result, the train was not considered capable of performing its specified safety function. The valve stem was replaced with a new one.

The root cause was determined to be environmentally assisted stress corrosion cracking which resulted from installation of an inadequate material some years earlier. The same inadequate material had been installed in a similar valve in a different system at the same time. The similar valve was exposed to similar environmental conditions as the first valve.

c) Overpressure due to Thermal Expansion

It was determined that a number of liquid-filled and isolated containment penetration lines in multiple safety systems were not adequately designed to accommodate the internal pressure buildup that could occur because of thermal expansion caused by heatup after a design basis accident. The problem existed because the original design failed to consider this effect following a postulated accident.

The condition is reportable under this criterion because there was a reasonable expectation of preventing fulfillment of the safety function of multiple trains or channels as a result of a single cause.

d) Cable Degradation

One of three component cooling water pumps tripped due to a ground fault on a power cable leading to the pump. The likely cause was determined to be moisture permeation into the cable insulation over time in a section of cable that was exposed to water.

e) Overstressed Valve Yokes

The event is reportable under this criterion if engineering judgment indicates that there was a reasonable expectation of preventing fulfillment of the safety function of an additional train in a different system as a result of the same cause. For example, if cable testing indicates that another cable to safety related equipment was likely to fail as a result of the same cause the event is reportable.

It was determined that numerous motor operated valve yokes experienced over thrusting that exceeded design basis stress levels.

The cause was lack of knowledge that resulted in inadequate design engineering at the time the designs were performed.

Some of the motor operated valve yokes, in different systems, were being over stressed enough during routine operations that, although they were currently capable of performing their specified safety functions, the over stressing would, with the passage of time, render them incapable of performing those functions. The condition is reportable under this criterion if engineering judgment indicates there was a reasonable expectation of preventing fulfillment of the safety function of trains or channels in two or more different systems.

Non-Reportable

f) Heat Exchanger Fouling

Periodic monitoring of heat exchanger performance indicated that two heat exchangers in two different systems required cleaning in order to ensure they would remain operable. The degree of fouling was within the range of normal expectations upon which the monitoring and maintenance procedures were based.

The event is not reportable under this criterion because there was not a reasonable expectation of preventing the fulfillment of the safety function of the heat exchangers.

g) Pump Vibration

Based on increasing vibration trends, identified by routine vibration monitoring, it was determined that a pump's bearings required replacement. Other pumps in different systems with similar designs and service histories experience similar bearing degradation. However, it is expected that the degradation will be detected and corrected before failure occurs. Such bearing degradation is not reportable under this criterion because it is normal and expected.

202.10 FOLLOWUP NOTIFICATION

§ 50.72 (c)	§ 50.73
"Followup Notification. With respect to the telephone notifications made under paragraphs (a) and (b) of this section, in addition to making the required initial notification, each licensee shall, 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 any of the Emergency Classes, if such a declaration has not been previously made, or (ii) any change from one Emergency Class to another, or (iii) a termination of the Emergency Class. (2) *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. (3) Maintain an open, continuous communication channel with the NRC Operations Center upon request by the NRC."	*There is no corresponding requirement in § 50.73.*

General

10CFR 50.72(c), "Followup Notification", is in addition to making the required initial ENS notification under 50.72(a) or (b). Reporting under this section is intended to provide the NRC with timely notification when an event becomes more serious or additional information or new analysis clarify the event.

It is important that the station record the NRC 50.72 Report number in the appropriate procedure for the initial ENS phone call, so when notifications are made per this section, the station can provide the NRC the proper report number. Any new information to be given will be recorded as such on the NRC's original 50.72 report as an update.

The followup notification is required for data or analysis results that clarify the plant conditions. Anytime a determination is made that a followup notification is required under 10CFR 50.72c, a formal notification shall be made using the ENS phone. Notification to the NRC Resident, other NRC representatives on site, or informally communicating on the open ENS line during an event is not a substitute for a 50.72 notification.

Since this criterion primarily deals with changes in plant status or analyses associated with emergency events, no discussion on the specific parts of the rule will be included in this directive, since current Emergency Plan implementing procedures provide adequate guidance (as stated in the Purpose of this directive).

202.11 OTHER EVENTS REQUIRING "IMMEDIATE NOTIFICATION"

This section addresses immediate notification requirements for sections other than 50.72. The station is required to notify the NRC as soon as practical and in all cases within 1 hour of the occurrence of any of the events specified. There are no examples available for these reporting sections.

10CFR 20.2202a

Each Licensee shall immediately report any incident involving byproduct, source or special nuclear material that may have caused or threatens to cause the following:

1. Individual Exposure
   Greater than or equal to 25 Rem total effective dose equivalent (TEDE)

   or

   Greater than or equal to 75 Rem eye dose equivalent (EDE)

   or

   Greater than or equal to 250 Rads shallow dose equivalent to the skin or extremities (SDE)
2. Release of radioactive material, inside or outside of a restricted area, so that, had an individual been present for 24 hours, the individual could have received an intake five times the annual limit on intake (the provisions of this paragraph do not apply to locations where personnel are not normally stationed during routine operations, such as hot cells or process enclosures).

10CFR20.1906(d)(1) and (d)(2)

Notification to the NRC Regional Office, Region II, Atlanta, GA. following receipt of a package of radioactive materials where:

Removable radioactive surface contamination exceeds the limits of 10 CFR 71.87 (i)

or

External radiation levels exceed the limits of 10 CFR 71.47

Steam Generator Tube Plugging

Catawba Tech Spec 5.5.9 and ONS Tech Spec Table 5.5.10-1 delineates the requirements for the SG tube surveillance program. If the results of the SG tube inspection requires a SG to be classified as Category C-3, prompt notification to the NRC pursuant to 10 CFR 50.72 is required.

Tech Spec Safety Limit Violation

The station is required to notify the NRC as soon as practical and in all cases within 1 hour of the occurrence of a Safety Limit violation. A follow-up written report is to be submitted within 30 days of the event.

202.12 INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI) REPORTING REQUIREMENTS

Reporting Requirements

Independent Spent Fuel Storage Installation (ISFSI)

10CFR72.75 (a) Emergency Notifications

Adequate guidance currently exist in each site's Emergency Plan's implementing response procedures for emergency events and their classifications. For an ISFSI that is located on the site of a nuclear power reactor licensed for operation by the Commission, the emergency plan required by 10CFR50.47 shall be deemed to satisfy the requirements of this section.

The sections that follow address guidelines for reporting four and twenty-four hour notifications for non-emergency events and the associated event report. There are no examples available for these sections.

10CFR72.75 (b) Four hour reports

The station is required to notify the NRC as soon as practical and in all cases within 4 hours of the occurrence of any of the following events or conditions involving spent fuel or high-level radioactive waste:

1. An action taken in an emergency that departs from a condition or a technical specification contained in a license or certificate of compliance issued under this part when the action is immediately needed to protect the public health and safety and no action consistent with license or certificate of compliance conditions or technical specifications that can provide adequate or equivalent protection is immediately apparent.
2. An event or situation, related to the health and safety of the public or on-site personnel, or protection of the environment, for which a news release is planned or notification to other government agencies has been or will be made.

10CFR72.75 (c) Eight hour reports

1. A defect in any spent fuel, high-level radioactive waste, or reactor-related Greater then Class C (GTCC) waste storage structure, system, or component important to safety.
2. A significant reduction in the effectiveness of any spent fuel, high-level radioactive waste, or reactor-related GTCC storage confinement system during use.
3. An event that requires the transport of a radioactively contaminated person to an offsite medical facility for treatment.

10CFR72.75 (d) Twenty-four hour reports

The station is required to notify the NRC as soon as practical and in all cases within 24 hours of the occurrence of any of the following events or conditions involving spent fuel or high-level radioactive waste:

1. An event in which safety equipment is disabled or fails to function as designed when:
   a. The equipment is required by regulation, license condition, or certificate of compliance to be available and operable to prevent releases that could exceed regulatory limits, to prevent exposures to radiation or radioactive materials that could exceed regulatory limits, or to mitigate the consequences of an accident; and
   b. No redundant equipment was available and operable to perform the required safety function.

10CFR72.75(g) Written report

The station is required to submit a written followup report within 60 days of an initial report required by paragraph (b) or (c) of this section. A written followup report is not required for (b)(2) and (c)(3). The reports and copies that licensees are required to submit to the Commission under the provisions of this section must be of sufficient quality to permit legible reproduction and micrographic processing. Written reports prepared pursuant to other regulations may be submitted to fulfill this requirement if the reports contain all of the necessary information and the appropriate distribution is made.

APPENDIX A.202. SYSTEM ACTUATIONS

System Actuations	CNS	MNS	ONS
1. Containment Isolation Systems			
a. Phase A/ES Chl 1 & 2 (Non-Essential Isolation)	X	X	X
b. Phase B/ES Chl 5 & 6 (Essential Isolation)	X	X	X
c. NW	X		
2. Containment Heat Removal			
a. Ice Condenser	X	X	
b. Air Return Fans	X	X	
c. Containment/Reactor Building Spray	X	X	X
d. Reactor Building Cooling Units			X
3. Combustible Gas Control in Containment			
a. Hydrogen Recombiners	X	X	
b. Air Return and Skimmer Fans	X	X	
c. Hydrogen Purge	X	X	
4. Emergency Core Cooling System			
a. NV/HPI	X	X	X
b. NI	X	X	
c. ND/LPI	X	X	X
d. CLA/CFT	X	X	X
e. FWST/BWST	X	X	X
1. Containment Sump Swapover	X	X	
5. Auxiliary/Emergency Feedwater System	X	X	X
a. Station- ASW Pump (10 CFR 50.73 only)			X
b. SSF-ASW Pump (10 CFR 50.73 only)			X
6. Diesel Generator starts	X	X	
7. Keowee starts (see Section 202.9.6 , example " g ")			X
8. Reactor Protection System	X	X	X
9. Nuclear Service Water System Suction Transfer - (10 CFR 50.73 only)	X	X	
10. Steam Line Isolation	X	X	
11. 4KV Undervoltage	X	X	X

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3.6 CONTAINMENT SYSTEMS

3.6.3 Containment Isolation Valves

LCO 3.6.3 Each containment isolation valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

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

1. Penetration flow paths except for 48 inch purge valve penetration flow paths may be unisolated intermittently under administrative controls.
2. Separate Condition entry is allowed for each penetration flow path.
3. Enter applicable Conditions and Required Actions for system(s) made inoperable by containment isolation valves.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ----------NOTE---------- Only applicable to penetration flow paths with two containment isolation valves. ---------- One or more penetration flow paths with one containment isolation valve inoperable.	A.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, one closed and de-activated non-automatic power operated valve, closed manual valve, blind flange, or check valve with flow through the valve secured. AND	4 hours (continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2 ----------NOTE---------- Isolation devices in high radiation areas may be verified by use of administrative means. ---------- Verify the affected penetration flow path is isolated.	Once per 31 days for isolation devices outside containment AND Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days for isolation devices inside containment
B. ----------NOTE---------- Only applicable to penetration flow paths with two containment isolation valves. ---------- One or more penetration flow paths with two containment isolation valves inoperable.	B.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, one closed and de-activated non-automatic power operated valve, closed manual valve, or blind flange.	1 hour

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. ----------NOTE---------- Only applicable to penetration flow paths with only one containment isolation valve and a closed system. ---------- One or more penetration flow paths with one containment isolation valve inoperable.	C.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, one closed and de-activated non-automatic power operated valve, closed manual valve, or blind flange.	4 hours
	AND	
	C.2 ----------NOTE---------- Isolation devices in high radiation areas may be verified by use of administrative means. ---------- Verify the affected penetration flow path is isolated.	Once per 31 days
D. Required Action and associated Completion Time not met.	D.1 Be in MODE 3.	12 hours
	AND	
	D.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.3.1	Verify each 48 inch purge valve is sealed closed.	31 days
SR 3.6.3.2	----------NOTE---------- Valves and blind flanges in high radiation areas may be verified by use of administrative means. ---------- Verify each containment isolation manual and non-automatic power operated valve and blind flange that is located outside containment and not locked, sealed, or otherwise secured and required to be closed during accident conditions is closed, except for containment isolation valves that are open under administrative controls.	31 days
SR 3.6.3.3	----------NOTE---------- Valves and blind flanges in high radiation areas may be verified by use of administrative means. ---------- Verify each containment isolation manual and non-automatic power operated valve and blind flange that is located inside containment and not locked, sealed, or otherwise secured and required to be closed during accident conditions is closed, except for containment isolation valves that are open under administrative controls.	Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.3.4	Verify the isolation time of each automatic power operated containment isolation valve is within limits.	In accordance with the Inservice Testing Program
SR 3.6.3.5	Verify each automatic containment isolation valve that is not locked, sealed, or otherwise secured in position, actuates to the isolation position on an actual or simulated actuation signal.	18 months

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3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.11 RCS Specific Activity

LCO 3.4.11 The specific activity of the reactor coolant shall be within limits.

APPLICABILITY: MODES 1 and 2,
MODE 3 with RCS average temperature ≥ 500°F.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. DOSE EQUIVALENT I-131 > 1.0 μCi/gm.	----------NOTE---------- LCO 3.0.4 is not applicable. ----------	
	A.1 Verify DOSE EQUIVALENT I-131 within the acceptable region of Figure 3.4.11-1.	Once per 4 hours
	AND	
	A.2 Restore DOSE EQUIVALENT I-131 to within limit.	48 hours
B. Required Action and associated Completion Time of Condition A not met. OR DOSE EQUIVALENT I-131 in unacceptable region of Figure 3.4.11-1.	B.1 Be in MODE 3 with RCS average temperature < 500°F.	12 hours

(continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Gross specific activity of the coolant not within limit.	C.1 Be in MODE 3 with RCS Average Temperature < 500°F.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.11.1	Verify reactor coolant gross specific activity $\leq 100/\bar{E}$ μCi/gm.	7 days
SR 3.4.11.2	----------NOTE---------- Only required to be performed in MODE 1. ---------- Verify reactor coolant DOSE EQUIVALENT I-131 specific activity ≤ 1.0 μCi/gm.	14 days AND Between 2 and 6 hours after THERMAL POWER change of ≥ 15% RTP within a 1 hour period
SR 3.4.11.3	----------NOTE---------- Not required to be performed until 31 days after a minimum of 2 EFPD and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for ≥ 48 hours. ---------- Determine $\bar{E}$.	184 days

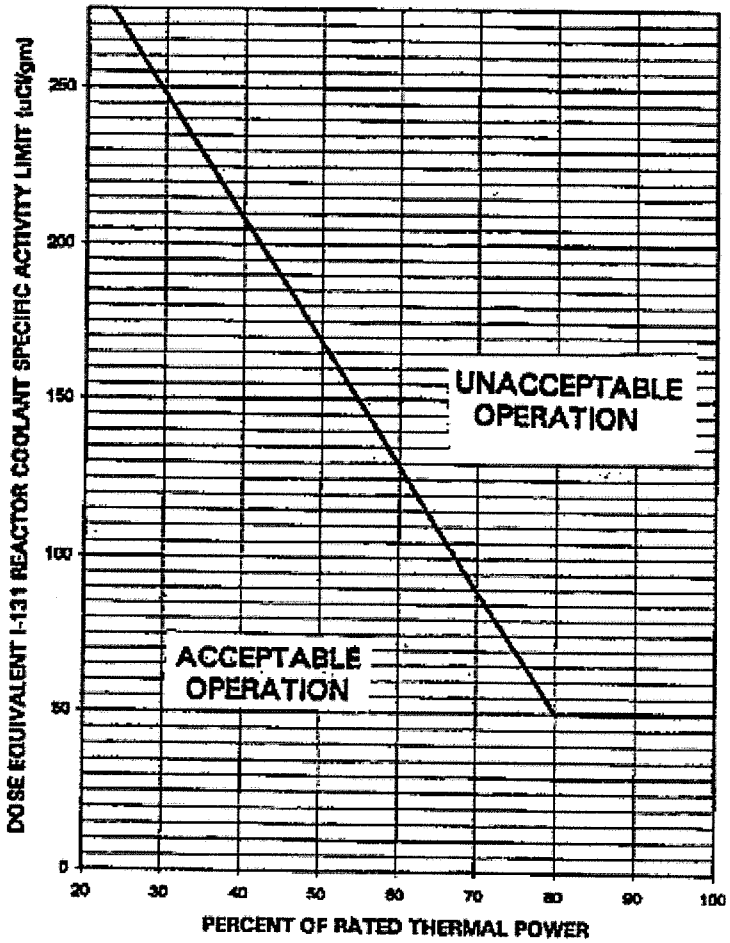

Figure 3.4.11-1 (page 1 of 1)
Reactor Coolant DOSE EQUIVALENT I-131 Specific Activity Limit
Versus Percent of RATED THERMAL POWER With Reactor Coolant
Specific Activity > 1.0 μCi/gm DOSE EQUIVALENT I-131

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16.13 CONDUCT OF OPERATIONS

16.13.1 Minimum Station Staffing Requirements

COMMITMENT a. Minimum station staffing shall be as indicated in Table 16.13.1-1 and shall meet the following additional requirements:

1. At least one RO per unit shall be present in the control room when fuel is in the reactor. In addition, while the unit is in MODES 1, 2, 3, or 4, at least one licensed SRO shall be present in the control room.

2. At least one licensed operator shall be in the reactor building when fuel handling operations in the reactor building are in progress. In addition, during CORE ALTERATIONS including fuel loading and transfer, an SRO or an SRO limited to fuel handling shall be present to directly supervise the activity and, during this time, shall not be assigned to other licensed activities

3. If the computer for a reactor is inoperable for more than eight hours, an operator, in addition to those specified in ITS 5.2.2.b and 10 CFR 50.54(m) shall supplement the control room staff.

b. The Shift Technical Advisor shall be an experienced SRO.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements for minimum station staffing not met.	A.1 Restore minimum station staffing levels.	2 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 16.13.1.1 N/A	N/A

Table 16.13.1-1

MINIMUM STATION STAFFING REQUIREMENTS

	THREE UNITS IN MODES 1-4	TWO UNITS IN MODES 1-4 CONTROLLED FROM TWO CONTROL ROOMS	TWO UNITS IN MODES 1-4 CONTROLLED FROM ONE CONTROL ROOM	ONE UNIT IN MODES 1 - 4	THREE UNITS IN MODES 5 OR 6 OR NO MODE
OSM	1	1	1	1	1
STA	1	1	1	1	1
SRO[1]	5	5	4	4	3
RO[3]	6	5	5	4	3
NLO[1,2]	8	8	8	8	7
SPOC	7	7	7	7	6
Chemistry Technician	1	1	1	1	1
RP Technician	3	3	3	3	3

1 SRO number can be reduced by one when a qualified NLO is designated the fire brigade leader. The NLO number must be increased by one, or one fire brigade member must be supplied from another organization.

2 NLO number must be increased by two when in ITS 3.5.2 Condition B.

3 RO number must be increased by one when in ITS 3.5.2 Condition B.

BASES

Some of the requirement(s) of this SLC section were relocated from TS 6.1.1.9 and TS Table 6.1-1 during the conversion to ITS. These requirements were initially relocated to SLC 16.13.5, "Additional Operating Shift Requirements," dated 3/27/97.

The requirements of this SLC consolidate ONS station staffing requirements into one document. This SLC includes the shift manning requirements of ITS 5.2.2, 10 CFR 50 Appendix R Section III.H, 10 CFR 50.54.m, Operations Management Procedures (OMPs), NSD 112, and the Emergency Plan. This SLC also includes the old requirements of SLC 16.13.1, "Fire Brigade," dated 3/27/99 and SLC 16.13.5; "Additional Operating Shift Requirements dated 3/27/99. The numbers for each position per shift are additive. For example, Table 16.13.1-1 requires a total of 5 SROs per shift (3 SROs required by 10 CFR 50.54(m)(2)(i) plus 1 additional SRO for the Fire Brigade and 1 additional for the ERO). The bases for the numbers in the first column of SLC Table 16.13.1-1 are as follows:

1 OSM (active SRO)	Required by 50.54(m)(2)(ii) (implemented by OMP).
1 STA (active or inactive SRO)	Required by ITS 5.2.2.g which indicates the individual fulfilling the STA position is the Shift Work Manager (implemented by OMP). Revision 50 to OMP 2-1 renamed the person fulfilling this position an STA. Note that pre-conversion TS Table 6.1-1, which implemented NUREG-0737 requirements, did not require an STA on shift when no units were in MODES 1-4. The SLC Table is more restrictive in that it requires an STA on shift at all times.
3 SRO's (active SRO)	Required by 10 CFR 50.54(m)(2)(i). Per ITS 5.2.2.b and 10 CFR 50.54(m)(2)(iii) at least 2 SRO's must be in the control room.
1 SRO (active or inactive) or NLO - Fire Brigade	Required by Appendix R Section III.H. Implemented by OMP and NSD. Individual fulfilling position shall be a SRO or an NLO who is qualified to be a fire brigade leader. Per OMP this individual functions as fire brigade leader and is not available for control room activities when directing the fire brigade. Appendix R does not specify that the brigade leader be an SRO, it only specifies that the fire brigade leader have sufficient training in or knowledge of plant safety-related systems to understand the effects of fire and fire suppression systems on safe shutdown capability. When an

	NLO is serving as the fire brigade leader, the SRO number for each column in Table 16.13.1-1 may be reduced by one.
1 SRO (~~active or inactive~~licensed or previously licensed) - ERO	_____Required by Volume A, Section B, Figure B-2 of the Emergency Plan. Implemented by OMP. SRO serves as the offsite communicator and the NRC communicator in the CR/TSC. This is permissible since the offsite communicator role is completed prior to the NRC communicator role starting.
5 RO's	Required by 10 CFR 50.54(m)(2)(i).
1 RO - SSF	Required by ITS 5.2.2.h, implemented by OMP. Per ITS 5.2.2.h, the manpower necessary to operate the SSF will be exclusive of the fire brigade and the minimum operating shift that is required to be present in the Control Room. ITS 5.2.2.b and10 CFR 50.54(m)(2)(iii) require 3 of the 5 RO's required by 10 CFR 50.54(m)(2)(i) to be present in the control room when fuel is in the reactor. When all three units are in MODES 1-3, one RO per unit must be available to be dispatched to the SSF. Since 3 RO's must be present in the Control Room only two are available to dispatch to the SSF. Therefore, one additional RO, beyond what is required by 10 CFR 50.54(m)(2)(i), is required.
1 RO - ADV	Amendment 314, 314, 314 requires that staffing level be increased by an additional RO beyond what is required in Table 16.13.1-1 when in Condition B of ITS 3.5.2. The additional RO is designated to respond to an event requiring activation of the SSF for the unit operating under ITS 3.5.2 Condition B. The additional RO role may be fulfilled by an SRO as long as the SRO is not being counted towards the number of required SROs listed in Table 16.13.1-1 and is qualified to tasks involving operation of the SSF systems. When all three units are in MODES 1-4, a total of seven ROs are required; five per 10 CFR 50.54(m)(2)(I), one per ITS 5.2.2.h, and one per ITS 3.5.2 Condition B.
8 NLO's	Required by 10 CFR 50 Appendix R Part III.H, ITS

5.2.2.a, Volume A, Appendix 8, Spill Prevention and Control and Counter Measures Plan, Revision 98-04, 10/98 of the Emergency Plan, and Volume A, Section B, Figure B-8 of the Emergency Plan. Implemented by OMP. (Four for fire brigade, one NLO per Unit to complete critical AP and EOP actions and 1 for SSF equipment verification for the design basis Appendix R fire.) When an NLO is serving as the fire brigade leader, the NLO number for each column in Table 16.13.1-1 must be increased by one or one fire brigade member must be supplied from another organization.

The number of NLOs that are fire brigade qualified may be reduced provided that a like number of fire brigade qualified personnel are provided from other organizations. This does not change the total number of NLOs required; only the number required to be fire brigade qualified.

2 NLO's – ADV

License Amendment 314, 314, 314 requires that staffing levels be increased by an additional two NLO's beyond those required in Table 16.13.1-1 when in Condition B of ITS 3.5.2. The additional NLO's are designated for the purpose of operating the Atmospheric Dump Valves (ADVs) for the unit under ITS 3.5.2 Condition B. In addition, the NLO's with the responsibility for operating the ADVs will be designated to respond to the control room within five minutes and will not be given duties that will prevent this from happening.

7 SPOC

Required by Volume A, Section B, Figure B-8 of the Emergency Plan and the Fire Plan (Volume A, Appendix 8, Spill Prevention and Control and Counter Measures Plan, Revision 98-04, 10/98). Implemented by OMP 1-7 and NSD 112. Consists of two I&E technicians ERO qualified and knowledgeable of IP/0/A0050/003 (Power SSF Submersible Pump), two MM technicians ERO qualified and knowledgeable of MP/0/A/1300/059 (Install SSF Submersible Pump), one supervisor or temporary supervisor qualified to establish the OSC and perform the OSC Maintenance Supervisor functions, and one additional person to help with pump installation as directed by SPOC

supplied by one of the following groups in the order listed: SPOC, other maintenance personnel onsite, C&F, Chemistry, RP and Maintenance Overtime resources. Security will automatically supply one Security Guard to open doors and gates who will also assist with any maintenance activities to be performed. The Security Guard is counted as one of a total of 6 people needed to install the submersible pump. One other person is needed to establish the OSC for a total of 7. In the event of a fire, SPOC will respond to the fire until directed to install the submersible pump. A total of 5 SPOC personnel are assigned to the fire brigade. Per PIP 4-O99-2987 problem evaluation, it is acceptable to consider these additional 5 Fire Brigade members to be available for other duties, such as installation of the SSF pump. This is based on Oconee Fire Brigade Guide #2, which contains guidance that allows fire brigade members to be released from the brigade for operational needs at the discretion of the OSC/TSC.

The number of SPOC personnel qualified as fire brigade members may be reduced, provided that the qualified fire brigade members from other organizations are increased by a like number. This does not change the total number of SPOC personnel required, only the number required to be fire brigade qualified.

1 Chemistry Technician - ERO

Required by Volume A, Section B, Figure B-8 of the Emergency Plan. Implemented by OMP and Station Chemistry Manual 2.6. A Chemistry Technician who is fire brigade qualified may be credited toward fulfilling the ERO requirement and the fire brigade requirement. In the event of a fire, the Chemistry technician will respond to the fire until directed otherwise.

3 RP Technicians

Three are required by Volume A, Section B, Figure B-8 of the Emergency Plan. One is required by ITS 5.2.2.d and may be counted towards fulfilling the ERO requirement. Implemented by HP/0/B/1000/054. RP technicians who are fire brigade qualified may be credited toward fulfilling the ERO and TS requirements and the fire brigade

requirement. In the event of a fire, the RP technician will respond to the fire until directed otherwise.

Minimum Station Staffing numbers for the SRO and RO positions in Table 16.13.1-1 change as a function of the number of units in MODES 1-4 and whether the operating Units are controlled from one or two Control Rooms. The number for the remaining positions in Table 16.13.1-1 is not affected by operational condition of the units.

- 10 CFR 50.54(m)(2)(i) requires 3 SROs when two units are in MODES 1-4 and controlled from two Control Rooms, 2 SROs when two units are in MODES 1-4 and controlled from a common control room, 2 SROs when one unit is MODES 1-4 and 1 SRO when no units are MODES 1-4. Thus considering fire brigade and ERO requirements, this results in the requirement for 5 SROs when two units are in MODES 1-4 and controlled from two Control Rooms, 4 SROs when two units are in MODES 1-4 and controlled from a common control room, 4 SROs when one unit is MODES 1-4 and 3 SROs when no units are MODES 1-4.

- 10 CFR 50.54(m)(2)(i) requires 5 ROs when two units are in MODES 1-4 and controlled from two Control Rooms, 4 ROs when two units are in MODES 1-4 and controlled from a common control room, 4 ROs when one unit is MODES 1-4 and 3 ROs when no units are MODES 1-4. OMPs require 2 ROs to man the SSF when two units are in MODES 1-3 and 1 RO when one unit is MODES 1-3. None are required when no units are in MODES 1-3. Therefore, no additional RO's are required beyond what is required by 10 CFR 50.54(m)(2)(i) when less than three units are in MODES 1-3 with one exception. When two units are in MODES 1-3 and controlled from one Control Room one additional RO is required since 10 CFR 50.54(m)(2)(i) only requires 4 RO's when the two operating units (Units 1 and 2) are controlled from one control room. Since one RO (or SRO) must be present in the Control Room when fuel is in the reactor vessel, the two RO's required to man the SSF for the operating units are exclusive of the one RO required for each unit. Therefore, a total of 5 RO's are required for this configuration.

The minimum staffing number for the SPOC and NLO positions is reduced by one when all three units are in MODE 4 or below. This reduction is allowed since the SSF is not required to be OPERABLE in these MODES. Therefore, there is no need for SPOC to provide a qualified individual to establish the OSC and no need for an NLO to perform SSF equipment verification.

SLC 16.13.1.a.1 requires at least one RO per unit to be present in the control room when fuel is in the reactor and one SRO to be present in the control room while in MODES 1-4. This requirement is based on 10 CFR 50.54(m)(2)(iii) and ITS 5.2.2.b. The first part of SLC 16.13.1.a.2, which requires at least one licensed operator to be in the reactor building when fuel handling operations in the reactor building were in progress, was relocated during the ITS conversion from TS Table 6.1-1, Additional Requirement 3. This requirement has existed since the initial issuance of Oconee Technical Specifications. The second part of SLC 16.13.a.2, which requires that a SRO or an SRO limited to fuel handling activities be present to directly supervise CORE ALTERATIONS including fuel loading or transfer and be assigned no other

duties, is based on 10 CFR 50.54(m)(2)(iv). SLC 16.13.1.a.3 which requires an operator, in addition to those specified in ITS 5.2.2.b to supplement the control room staff if the computer for a reactor is inoperable for more than eight hours, was relocated during the ITS conversion from TS Table 6.1-1, Additional Requirement 6. This requirement has also existed since the initial issuance of Oconee Technical Specifications. SLC 16.13.1.b, which specifies the STA shall be an experienced SRO was relocated during the ITS conversion from TS 6.1.1.9.

The primary purpose of the Fire Protection Program is to minimize both the probability and consequences of postulated fires. Despite designed active and passive Fire Protection Systems installed throughout the plant, a properly trained and equipped Fire Brigade organization of at least ten (Reference 8) members is needed to provide immediate response to fires that may occur at the site. This number is the result of a corrective action from Reference 10. This Fire Brigade requirement is normally met by using one SRO (or NLO qualified to be a fire brigade leader), 4 NLOs, and 5 SPOC personnel. However, this requirement can also be met by using personnel from other organizations (e.g., Chemistry, Radiation Protection, and Security).

Fire Brigade equipment and training conform to Oconee's commitments to Appendix A to Branch Technical Position 9.5-I and supplemental NRC Staff guidelines including Nuclear Plant Functional Responsibilities, Administrative Controls and Quality Assurance.

This SLC is part of the Oconee Fire Protection Program and therefore subject to the provisions of Oconee Facility Operating License Conditions.

The following requirement was relocated from the TS 6.1.1.8 during the conversion to ITS. A training program for the fire brigade shall meet or exceed the requirements of Section 27 of the NFPA Code-1975, except that training sessions may be held quarterly.

ACTIONS A.1

With the requirements for minimum station staffing not met, the minimum station staffing levels shall be restored within 2 hours. The 2 hour Completion Time is consistent with ITS 5.2.2.c and d which allows 2 hours to accommodate unexpected absence of on-duty shift crew members provided that immediate action is taken to restore the shift crew composition to within the minimum requirements.

REFERENCES:

1. Oconee UFSAR, Chapter 9.5.1.
2. Oconee Fire Protection SER dated August 11, 1978.
3. Oconee Fire Protection Review, (currently contained in the Fire Protection DBD) as revised.
4. Duke letter of January 16, 1978 to NRC in response to "Nuclear Plant Functional Responsibilities, Administrative Controls, and Quality Assurance".

5. ITS 5.2.2, Amendment 300/300/300.
6. 10 CFR 50.54(m).
7. Emergency Plan, Volume A, Section B, Figure B-8, Revision 97-01, 7/97.
8. Emergency Plan, Volume A, Appendix 8, Spill Prevention and Control and Counter Measures Plan, Revision 98-04, 10/98.
9. Station Chemistry Manual 2.6.
10. Problem Investigation Report Serial No. 1-089-0001.
11. Problem Investigation Process (PIP) Serial No. 4-O99-2987.
12. ITS 3.5.2, Amendment 314/314/314.
13. NRC Regulatory Issue Summary (RIS) 2002-16, "Current Incident Response Issues."
14. PIP O-03-0233

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