Question # 1

Given the following conditions:

- Unit 3 and Unit 4 are at 100% power.
- A fire is reported in the control room electrical chase.
- Units are tripped in accordance with 0-ONOP-105, Control Room Evacuation.

Which one of the following completes the statements below?

In accordance with 0-ONOP-105, the Third Licensed Reactor Operator will mechanically trip the ___(1)___ AFW pump.

When the ASP transfer switch for MOV-6459B, B AFW PUMP T&T VALVE, is placed in LOCAL, a ___(1)___ AFW pump may occur.

A. (1) A
(2) trip of a running

B. (1) A
(2) start of a non-running

C. (1) C
(2) trip of a running

D. (1) C
(2) start of a non-running

Question # 2

Given the following initial conditions:

- Unit 3 trips from full power.
- RCS pressure is 1200 psig and lowering.
- Indicated PRZ level is 80% and rising.

Subsequently:

- The crew transitions to 3-EOP-E-1, Loss of Reactor or Secondary Coolant.
- Containment temperature is 160°F.
- RCS pressure stabilizes at 590 psig.

Which one of the following completes the statements below?

The crew is responding to a break on the PRZ ___(1)___ line.

RHR stop criteria ___(2)___ met.

A. (1) surge
(2) is NOT

B. (1) surge
(2) is

C. (1) safety
(2) is NOT

D. (1) safety
(2) is

Question # 3

Given the following conditions:

- 3-EOP-ES-1.2, Post LOCA Cooldown and Depressurization, is in progress.
- PRZ narrow range pressure is 1500 psig.
- RCS wide range pressure is 1400 psig.
- CET temperatures are 500°F.

Which one of the following completes the statements below?

Subcooling based on CETs is ___(1)___ .

Subcooling is monitored on the foldout page to determine if ___(2)___ .

A. (1) 88°F
(2) SI re-initiation is required

B. (1) 88°F
(2) voiding will occur during depressurization

C. (1) 97°F
(2) SI re-initiation is required

D. (1) 97°F
(2) voiding will occur during depressurization

Question # 4

Given the following initial conditions:

- Unit 3 experiences a reactor trip due to a LOOP from 100% power.
- 3B EDG fails to start.
- The crew transitions to 3-EOP-E-0, Reactor Trip or Safety Injection.

Subsequently:

- A Large Break LOCA occurs.
- Containment pressure is 28 psig.
- The crew transitions to 3-EOP-E-1, Loss of Reactor or Secondary Coolant.

Which one of the following completes the statements below?

The AFW System will start and ___(1)___ required to support core decay heat removal during this accident.

During the injection phase of the accident, ICW pumps will be loaded on the 3A EDG for ___(2)___ removal.

A. (1) is NOT
(2) containment heat

B. (1) is NOT
(2) core decay heat

C. (1) is
(2) containment heat

D. (1) is
(2) core decay heat

Question # 5

Given the following conditions:

- Unit 3 is at 100% power.
- Component Cooling Water flow is lost and can NOT be established in either header.
- 3B RCP lower guide bearing reaches 235°F.
- The crew enters 3-ONOP-030, Component Cooling Water Malfunction and 3-ONOP-041.1, Reactor Coolant Pump Off-Normal.

Which one of the following completes the statements below?

The crew will next trip the reactor and then secure ___(1)___ .

The crew will operate the charging pumps at ___(2)___ speed.

A. (1) only the 3B RCP
(2) maximum

B. (1) only the 3B RCP
(2) minimum

C. (1) all RCPs
(2) maximum

D. (1) all RCPs
(2) minimum

Question # 6

Given the following initial conditions:

- Unit 3 trips from full power.
- The crew enters 3-EOP-ES-0.1, Reactor Trip Response.
- RCS temperature stabilizes at 547°F.
- PRZ level stabilizes on program.

Subsequently:

- A loss of all charging occurs.
- PRZ level is lowering by 1% every four minutes.

Which one of the following identifies the MINIMUM time to restore charging flow BEFORE letdown is automatically isolated?

(Assuming no operator action)

A. 2 - 10 minutes

B. 30 - 40 minutes

C. 80 - 90 minutes

D. 170 - 180 minutes

Question # 7

Given the following initial conditions:

- Unit 3 is in Mode 5.
- PRZ cold cal level is 25% and stable.
- 3A RHR Pump is in service.
- RHR flow is 3100 gpm.

Subsequently:

- The 3A RHR Pump's motor amps and flow lower and become erratic.
- The crew stops the 3A RHR Pump in accordance with 3-ONOP-050, Loss of RHR.

Which one of the following completes the statements below?

The crew will first attempt to start the ___(1)___ RHR Pump.

When RCS temperature is subsequently stabilized, FI-3-605, RHR flow indicator, will measure flow through ___(2)___ .

A. (1) 3A
(2) both the RHR HXs and the bypass flow line

B. (1) 3A
(2) the RHR HXs only

C. (1) 3B
(2) the RHR HXs and the bypass flow line

D. (1) 3B
(2) the RHR HXs only

Question # 8

Given the following conditions:

- Unit 3 is in Mode 2.
- PRZ pressure control is in automatic.
- Pressurizer pressure is 2150 psig and lowering.
- 3-ONOP-041.5, Pressurizer Pressure Control Malfunction, is entered.

Which one of the following completes the statements below?

The operator will manually ___(1)___ the output on PC-3-444J, PZR PRESS CONTROL, to stabilize pressure.

When adjusting the output on PC-3-444J, PCV-3-455C, PZR PORV, ___(2)___ expected to immediately automatically open if the controller demand setpoint is driven to 2335 psig.

A. (1) raise
(2) is

B. (1) lower
(2) is

C. (1) raise
(2) is NOT

D. (1) lower
(2) is NOT

Question # 9

Which one of the following describes how the AMSAC (ATWS Mitigating System Actuation Circuit) trips the reactor?

A. Energizes both Control Rod MG set input breaker trip coils.

B. Energizes both Control Rod MG set output breaker trip coils.

C. De-energizes the Shunt Trip Coils on both Reactor Trip Breakers and Bypass Breakers.

D. De-energizes the Undervoltage Trip Coils on both Reactor Trip Breakers and Bypass Breakers.

Question # 10

Given the following conditions:

- The crew is performing 4-ONOP-071.2, Steam Generator Tube Leakage.
- Reactor power is 50%.
- Tavg and Tref are matched.
- Pressurizer level starts lowering.
- All charging pumps are running with individual controllers in AUTOMATIC.

Which one of the following completes the statement below?

The crew is required to manually initiate safety injection when the master charging pump controller's demand is ___(1)___ in MANUAL with letdown isolated OR when PRZ level is ___(2)___ and lowering.

A. (1) 100%
(2) 40.5%

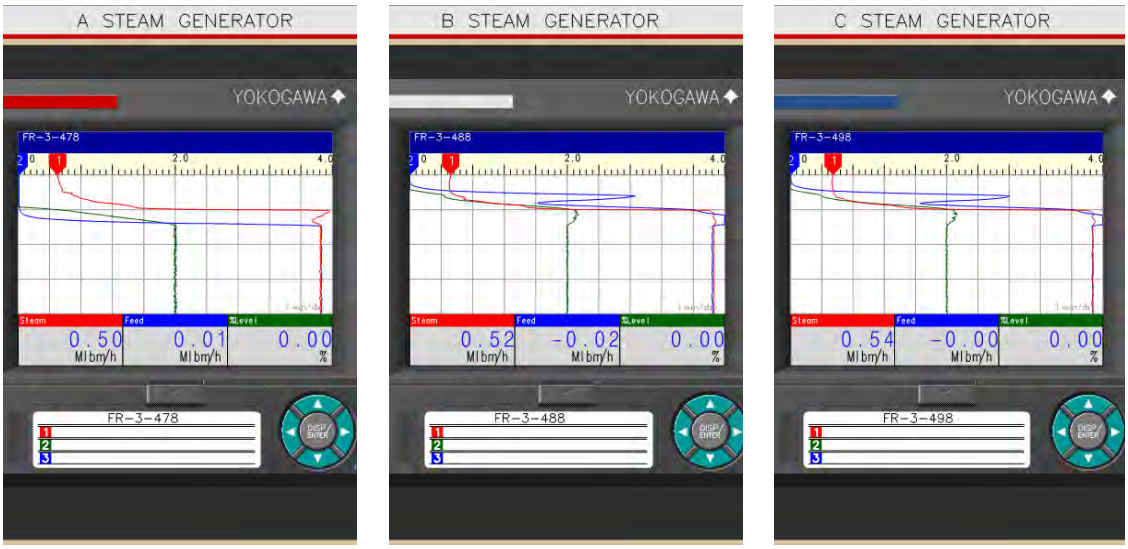
B. (1) 100%
(2) 29.5%

C. (1) 0%
(2) 40.5%

D. (1) 0%
(2) 29.5%

Question # 11

Post trip steam generator trends on Unit 3 are as follows:



TRENDS ALSO PROVIDED AS REFERENCE IN LARGER FORMAT

Which one of the following identifies the initiating event?

A. One main feed regulation valve failed closed.

B. A feedwater isolation occurred.

C. One steam dump to condenser failed open.

D. A steamline isolation occurred.

Question # 12

Given the following initial conditions:

- Unit 3 experienced a safety injection at full power.

Subsequently:

- SG NR levels are 4% and rising.
- AFW flow is 375 gpm.
- CET temperatures are 670°F and rising.
- CET subcooling is 0°F.
- Containment temperature is 160°F and rising.
- A station blackout occurs on Unit 3.
- 3-EOP-ECA-0.0, Loss of All AC Power, is entered.

Which one of the following completes the statements below?

QSPDS ___(1)___ available for monitoring.

If power is immediately restored, the crew will transition to ___(2)___ .

A. (1) is
(2) 3-EOP-FR-H.1

B. (1) is
(2) 3-EOP-FR-C.2

C. (1) is NOT
(2) 3-EOP-FR-H.1

D. (1) is NOT
(2) 3-EOP-FR-C.2

Question # 13

Given the following initial conditions:

- Unit 3 is operating at 100% power.
- A loss of offsite power occurs.
- The crew enters 3-EOP-ES-0.1, Reactor Trip Response.

While performing Attachment 1, Natural Circulation Indications, the following plant conditions are observed:

- CET temperatures = 595°F and stable
- RCS hot leg temperatures = 585°F and stable
- RCS pressure = 1900 psig and stable
- All SG pressures = 985 psig and stable
- RCS cold leg temperatures = 540°F and stable

Which one of the following (1) identifies the status of natural circulation and (2) the action required to enhance or establish natural circulation in accordance with 3-EOP-ES-0.1?

A. (1) Established
(2) Raise steam flow through steam dumps to condenser

B. (1) Established
(2) Raise steam flow through steam dumps to atmosphere

C. (1) NOT established
(2) Raise steam flow through steam dumps to condenser

D. (1) NOT established
(2) Raise steam flow through steam dumps to atmosphere

Question # 14

Given the following conditions:

- Unit 3 experiences a LOOP-LOCA.
- 3P07, Vital Instrument AC bus, de-energizes.

Which one of the following identifies why safeguards equipment must be manually started in accordance with Attachment 3, Prompt Action Verifications of 3-EOP-E-0?

A. 3A train Emergency Sequencer loses power.

B. 3A train Safety Injection bistables fail to actuate.

C. 3B train Emergency Sequencer loses power.

D. 3B train Safety Injection bistables fail to actuate.

Question # 15

Which one of the following (1) identifies the signal that automatically closes ICW/TPCW Isolation Valves POV-3-4882 and POV-3-4883, and (2) the reason for the automatic closure?

A. (1) High ICW flow
(2) Prevents ICW pump runout

B. (1) High ICW flow
(2) Ensures ICW is dedicated to CCW

C. (1) Safety Injection
(2) Prevents ICW pump runout

D. (1) Safety Injection
(2) Ensures ICW is dedicated to CCW

Question # 16

Given the following conditions:

- Unit 3 is at 100% power.
- PRZ level is 60%.
- A loss of instrument air is in progress.
- The crew has entered 3-ONOP-013, Loss of Instrument Air.
- Instrument Air header pressure on PI-3-1444 (VPA) is 65 psig and lowering slowly.
- Pressure drop across the operating air dryer is 8 psid.
- All available Instrument Air Compressors are running.

Which one of the following completes the statements below?

In accordance with 3-ONOP-013, the crew is required to __(1)__ .

Once the plant is shutdown, PRZ level band is maintained by cycling __(1)__ as necessary.

A. (1) perform a fast load reduction
(2) letdown valves

B. (1) perform a fast load reduction
(2) charging pumps

C. (1) trip the plant
(2) letdown valves

D. (1) trip the plant
(2) charging pumps

Question # 17

Given the following conditions:

- The crew enters 3-EOP-FR-H.1, Loss of Secondary Heat Sink.
- Bleed and feed criteria is met.

Which one of the following describes (1) the MINIMUM number of pressurizer PORV(s) opened in order to establish bleed and feed and (2) the reason for taking this action?

A. (1) One
(2) Prevents RCS pressure from rising to the pressurizer safety valve setpoint, leading to further loss of coolant inventory

B. (1) One
(2) Allows SI flow to inject when the RCS is depressurized, leading to core decay heat removal

C. (1) Two
(2) Prevents RCS pressure from rising to the pressurizer safety valve setpoint, leading to further loss of coolant inventory

D. (1) Two
(2) Allows SI flow to inject when the RCS is depressurized, leading to core decay heat removal

Question # 18

Given the following conditions:

- The crew enters 3-EOP-ECA-2.1, Uncontrolled Depressurization of All Steam Generators.
- RCS cooldown rate is 65°F/hr.
- AFW is unavailable.
- A standby steam generator feed water pump is in service.
- All SG NR levels are 20%.
- Containment temperature is 185°F.

Which one of the following completes the statements below?

A minimum feed flow of 50 gpm ___(1)___ required to be maintained to each SG.

Low range flow indication ___(2)___ available when using standby feed.

A. (1) is
(2) is NOT

B. (1) is
(2) is

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

D. (1) is NOT
(2) is

Question # 19

Given the following conditions:

- A downpower is in progress in accordance with 3-GOP-103, Power Operation to Hot Standby.
- TCS is in MW control.
- Rods are in automatic.
- T_{avg} is 571.2°F.
- T_{ref} is 570.0°F.
- TCS is placed in HOLD.
- Control Bank D rod M8 drops fully into the core.

Which one of the following completes the statements below?

(Assuming no operator action)

Control rods ___(1)___ insert to close the T_{avg}-T_{ref} mismatch.

TCS ___(2)___ automatically adjust to maintain MW output.

A. (1) will NOT
(2) will

B. (1) will
(2) will

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

D. (1) will
(2) will NOT

Question # 20

Given the following conditions:

- Unit 3 is at 75% power during a plant startup.
- A rod withdrawal of 3 steps is initiated in manual.
- ANN B 9/2, AXIAL FLUX TILT alarms.
- Rod withdrawal is stopped.
- ANN B 9/3, SHUTDOWN ROD OFF TOP/ DEVIATION alarms.
- Tavg rises 0.2°F and stabilizes.

Which one of the following describes (1) the event and (2) the cause of the AXIAL FLUX TILT alarm?

A. (1) One RCCA fully drops during withdrawal
(2) Exceeding a maximum delta between any two PR channels

B. (1) One RCCA fully drops during withdrawal
(2) Exceeding a maximum delta between upper and lower detectors on any PR channel

C. (1) One RCCA sticks during withdrawal
(2) Exceeding a maximum delta between any two PR channels

D. (1) One RCCA sticks during withdrawal
(2) Exceeding a maximum delta between upper and lower detectors on any PR channel

Question # 21

Given the following initial conditions:

- Unit 3 is reducing power from 100% power for maintenance.
- B 8/2, ROD BANK A/B/C/D EXTRA LO LIMIT alarms concurrently with a rod control card issue which prevents further rod motion.
- The unit is stabilized at 75%.

Subsequently:

- Rod Control repairs are complete after one hour.
- The crew initiates an emergency boration in accordance with 3-ONOP-046.1, Emergency Boration, to restore from excessive rod motion.
- Turbine control is in manual.

Which one of the following completes the statements below?

During the boration, the RCO ensures charging flow is a minimum of ___(1)___ on FI-3-122A in accordance with 3-ONOP-046.1.

With the same initial boration flow and after enough time elapses for the effects of the boration, a rise of reactor power will next require the RCO to ___(2)___ .

A. (1) 45 gpm
(2) start an additional Boric Acid Transfer Pump

B. (1) 40 gpm
(2) start an additional Boric Acid Transfer Pump

C. (1) 45 gpm
(2) verify FCV-3-114A, Primary Water to Blender, is closed

D. (1) 40 gpm
(2) verify FCV-3-114A, Primary Water to Blender, is closed

Question # 22

Given the following conditions:

- Unit 3 is in Mode 6.
- A fuel element drops into the core during fuel movement.
- Gas bubbles are rising from the dropped fuel element.

Which one of the following completes the statements below?

For the event in progress, ANN B 4/1, SOURCE RANGE HIGH FLUX AT SHUTDOWN, ___(1)___ the ONLY clad damage indication.

ANN B 4/1 ___(2)___ automatically cause a containment evacuation alarm to occur.

A. (1) is
(2) will

B. (1) is
(2) will NOT

C. (1) is NOT
(2) will

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

Question # 23

Given the following conditions:

- Unit 3 trips due to a LOOP.
- Secondary radiation levels rise.
- 3C SG is identified as being ruptured and is isolated.
- The cooldown to required Core Exit Temperature is complete.

Which one of the following describes a concern during the initial RCS depressurization of 3-EOP-E-3, Steam Generator Tube Rupture and how the depressurization will be performed?

A. Voiding in the reactor vessel upper head when using Auxiliary Spray.

B. Loss of RCS subcooling when using Auxiliary Spray.

C. Voiding in the reactor vessel upper head when using a PRZ PORV.

D. Loss of RCS subcooling when using RCS Vent Valves.

Question # 24

Which one of the following is the correct responses after acknowledgement of a Fire Alarm Operator Workstation C41 alarm?

Dispatch ___(1)___ to inspect the alarming zone(s) for fire or smoke.

The crew will reset the alarming fire detector ___(2)___ .

A. (1) the fire brigade
(2) locally at the detector

B. (1) an operator
(2) locally at the detector

C. (1) the fire brigade
(2) on Fire Alarm Operator Workstation C41

D. (1) an operator
(2) on Fire Alarm Operator Workstation C41

Question # 25

Given the following conditions:

- 3-EOP-ES-1.2, Post LOCA Cooldown and Depressurization, is entered.
- Charging pumps become gas bound.
- Unit 3 HHSI pumps are running at shutoff head.
- Containment temperature is 195°F.
- PRZ level is 10%.

Which one of the following completes the statements below?

The pressurizer level requirement ___(1)___ met for terminating HHSI.

HHSI pumps will be rotated to limit continuous runtime of any pump to a maximum of ___(2)___ minutes.

A. (1) is
(2) 44

B. (1) is
(2) 30

C. (1) is NOT
(2) 44

D. (1) is NOT
(2) 30

Question # 26

Given the following conditions:

- Unit 3 tripped from 100% power.
- A LOCA is in progress.
- Attachment 3 of 3-EOP-E-0, Reactor Trip or Safety Injection, is complete.
- The running HHSI pumps trip.
- ANN A 4/2, QSPDS INADEQUATE CORE COOLING, alarms.
- CET temperatures are 1250°F and rising.
- RCPs are secured.

Which one of the following completes the statements below?

The highest priority Core Cooling CSF is a/an ___(1)___ path.

In accordance with the required Core Cooling FRP, the 1st priority is to ___(2)___ .

A. (1) RED
(2) start RCPs for forced flow

B. (1) RED
(2) establish HHSI flow

C. (1) ORANGE
(2) start RCPs for forced flow

D. (1) ORANGE
(2) establish HHSI flow

Question # 27

Given the following condition:

- The crew enters 4-EOP-FR-Z.2, Response to Containment Flooding.

Which one of the following identifies an unexpected source of water to containment?

A. Water from the Accumulators may exceed the containment level design basis criterion when injected in an uncontrolled manner.

B. Water from the RCS lower vessel may cause a thermal stress on the core when re-injected during the recirculation phase of accident.

C. Water from the accident unit RWST may block the sump filters with contaminants if pumped below the low level setpoint.

D. Water from the opposite unit RWST may reach critical plant components necessary for plant recovery and may be damaged.

Question # 28

Given the following conditions:

- Unit 3 is at 14% power.
- The 3B 4kV Bus de-energizes due to an undervoltage condition.

Which one of the following completes the statement below?

___(1)___ RCP(s) will trip and the reactor ___(2)___ automatically trip.

A. (1) Only one
(2) will

B. (1) Only one
(2) will NOT

C. (1) Two
(2) will

D. (1) Two
(2) will NOT

Question # 29

Given the following conditions:

- Unit 3 is at 100% power.
- NRHX outlet temperature on TI-3-144 is 151°F and rising.
- Letdown pressure on PI-3-145 is 295 psig.
- NRHX (Non Regenerative Heat Exchanger) CCW flow is 105 gpm.
- TCV-3-143, Letdown Demin Divert Valve, remains aligned to the demineralizers.

Which one of the following identifies the effect if no action is taken?

A. Demineralizer vessel over pressure condition

B. Demineralizer bed high temperature degradation

C. Flashing of the letdown line upstream of the NRHX

D. Thermal stress on NRHX tubes due to insufficient cooling

Question # 30

Given the following conditions:

- Unit 3 is at 100% power.
- An inadvertent letdown isolation occurred.
- The cause has been identified and corrected.
- PRZ level is 68% and rising.

Which one of the following describes (1) a requirement for re-establishing letdown, and (2) the maximum allowable letdown flow when letdown is in service?

A. (1) Orifice Isolation Valves CV-3-200A/B/C must be open prior to opening Letdown Isolation valve LCV-3-460.
(2) 120 gpm

B. (1) Orifice Isolation Valves CV-3-200A/B/C must be open prior to opening Letdown Isolation valve LCV-3-460.
(2) 165 gpm

C. (1) Letdown Isolation Valve LCV-3-460 must be open prior to opening Orifice Isolation Valves CV-3-200A/B/C
(2) 120 gpm

D. (1) Letdown Isolation Valve LCV-3-460 must be open prior to opening Orifice Isolation Valves CV-3-200A/B/C
(2) 165 gpm

Question # 31

Given the following conditions:

- Unit 4 is in Mode 3.
- Plant startup is in progress in accordance with 4-GOP-301, Hot Standby to Power Operation.

Which one of the following identifies (1) the power supply to RHR Isolation Valve MOV-4-750 and (2) the current status of power to MOV-4-750.

A. (1) 4D MCC
(2) energized

B. (1) 4D MCC
(2) de-energized

C. (1) 4B MCC
(2) energized

D. (1) 4B MCC
(2) de-energized

Question # 32

Given the following conditions:

- A plant heatup is in progress in accordance with 3-GOP-503, Cold Shutdown to Hot Standby.
- RCS temperature is 520°F and rising.
- Pressurizer pressure is 2075 psig and rising.

Which one of the following completes the statements below?

The BLOCK LOW PRZ PRESS SI status light on VPA is ___(1)___ .

The LOW PRZ PRESS SI BLOCKED status light on VPA is ___(2)___ .

A. (1) lit
(2) lit

B. (1) lit
(2) NOT lit

C. (1) NOT lit
(2) lit

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

Question # 33

Given the following conditions:

- Unit 3 is at 100% power.
- RCS pressure is 2235 psig and stable.
- RCS leakage has risen by 0.5 gpm.

Which one of the following indications can SOLELY be used to distinguish a leaking PRZ PORV from a leaking PRZ Safety?

A. PRZ relief tank level, LI-3-470

B. PRZ relief tank pressure, PI-3-472

C. PRZ relief line temperature, TI-3-463

D. PRZ PORV/safety acoustic monitor, S-3-6303

Question # 34

Given the following conditions:

- Unit 3 is operating at 100% power.
- The 3B CCW Pump is in operation.
- The following alarms are received:
  - H 8/2, CCW PP A/B/C/ MOTOR OVERLOAD
  - H 8/1, CCW PP A/B/C TRIP
  - H 8/3, CCW HEADER LO PRESS

Which one of the following identifies the subsequent automatic action?

A. 3A CCW pump will start immediately after initiating signal

B. 3A CCW pump will start after at least 10 seconds

C. 3C CCW pump will start immediately after initiating signal

D. 3C CCW pump will start after at least 10 seconds

Question # 35

Given the following conditions:

- All systems are in automatic when a vital instrument AC bus is lost.
- PRZ spray control indicates as follows :

Which one of the following completes the statements below?

120V Vital Instrument Bus power ___(1)___ is lost to the PCV-3-455B controller.

Due to the loss of power, PCV-3-455B LOOP B spray valve will ___(2)___ .

(Assuming no operator action)

A. (1) 3P06
(2) fail CLOSED

B. (1) 3P06
(2) remain AS IS

C. (1) 3P09
(2) fail CLOSED

D. (1) 3P09
(2) remain AS IS

Question # 36

Given the following conditions:

- Unit 3 is stable at 75% power.
- Control bank D is at 210 steps.

Which one of the following completes the statements below?

The Tech Spec maximum allowed rod position misalignment from group step counter demand is __(1)__ steps.

If two C bank control rods drop into the core (one dropped rod RPI is at 2 steps and the other dropped rod RPI indicates 4 steps), the crew will __(2)__ .

A. (1) 18
(2) trip the reactor

B. (1) 12
(2) trip the reactor

C. (1) 18
(2) stabilize the plant at power

D. (1) 12
(2) stabilize the plant at power

Question # 37

Given the following conditions:

- Unit 3 is at 100% power.
- 3-SMI-063.01A, Train A Safeguards Matrix Logic Test, is in progress.
- Blue channel STM HEADER PRESSURE, PT-3-466, fails HIGH.
- The test is stopped and all test switches are returned to normal.
- I&C wants to open the associated channel rack.

Which one of the following completes the statements below?

The ___(1)___ PROTECTION RACK OPEN annunciator is expected to alarm when I&C performs the investigation.

If one of the remaining steam header pressure protection transmitters subsequently fails LOW, the 3A train sequencer ___(2)___ start safeguards equipment.

A. (1) CHANNEL II
(2) will

B. (1) CHANNEL III
(2) will

C. (1) CHANNEL II
(2) will NOT

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

Question # 38

Given the following conditions:

- Unit 3 is at 100% power.
- 3-OSP-055.1, Emergency Containment Cooler Operability Test, is in progress.

Which one of the following completes the statement below?

In accordance with 3-OSP-055.1, a minimum of ___(1)___ CCW Heat Exchangers must be in service to prevent exceeding the ___(2)___ during an ECC test.

A. (1) two
(2) 550 gpm individual ECC coil flow rate limit

B. (1) two
(2) 6840 gpm individual CCW heat exchanger flow rate limit

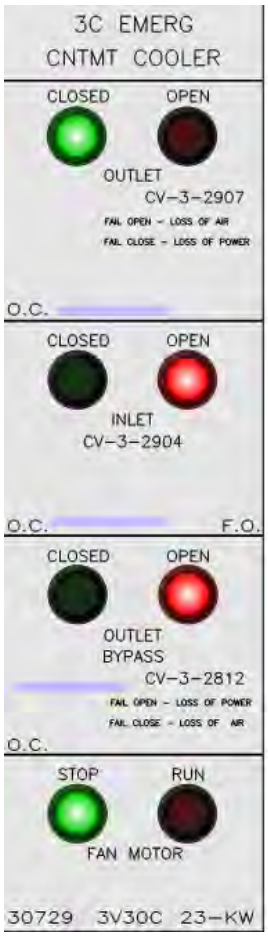
C. (1) three
(2) 550 gpm individual ECC coil flow rate limit

D. (1) three
(2) 6840 gpm individual CCW heat exchanger flow rate limit

Question # 39

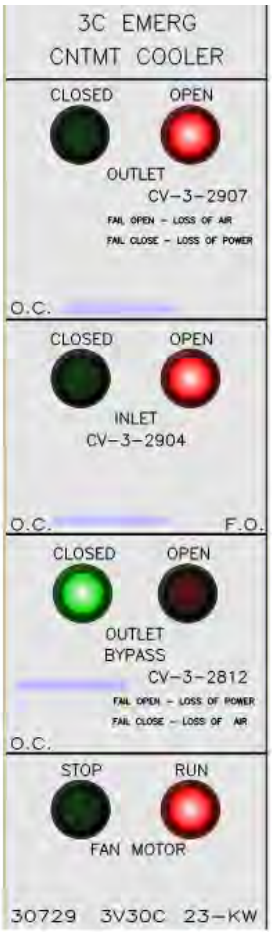
Which one of the following matches the letters identifying the alignments of the 3C ECC CCW valves in the (1) normal standby and (2) following a safety injection signal?

A.

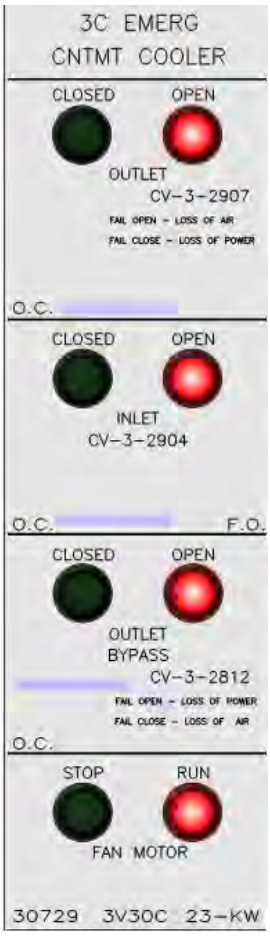


B.

C.



D.



A. (1) normal standby alignment- A
(2) safety injection alignment- D

B. (1) normal standby alignment- B
(2) safety injection alignment- D

C. (1) normal standby alignment- A
(2) safety injection alignment- C

D (1) normal standby alignment- B
(2) safety injection alignment- C

Question # 40

Given the following conditions:

- Unit 3 trips from full power due to a steam line break inside containment.
- Containment pressure on DCS is 22 psig.
- The following is observed on VPB:

☆ identifies a lit bistable

Which one of the following completes the statements below?

(Assuming no operator action)

Containment spray pumps ___(1)___ automatically started.

An ORANGE path on the Containment critical safety function, ___(2)___ present.

A. (1) have
(2) is

B. (1) have NOT
(2) is

C. (1) have
(2) is NOT

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

Question # 41

Given the following conditions:

- Unit 3 experiences a steam header break from 50% power.
- RCS pressure is 1800 psig.
- RCS Tavg is 525°F.
- Containment pressure is 0.1 psig.
- All SG pressures are 720 psig.
- All SG steam flows are 40% of full-scale.

Which one of the following completes the statements below?

(Assuming no operator action)

The Turbine Stop valves and MSIVs ___(1)___ receive an automatic closure signal.

MSR stop valves and MSIV bypasses ___(2)___ receive an automatic closure signal.

A. (1) will NOT
(2) will NOT

B. (1) will NOT
(2) will

C. (1) will
(2) will NOT

D. (1) will
(2) will

Question # 42

Given the following conditions:

- Unit 3 is performing a power ascension.
- The 3B Steam Generator Feed Pump is OOS.

Which one of the following identifies the MAXIMUM power level allowed in accordance with 3-GOP-301, Hot Standby to Power Operation?

A. 88%

B. 85%

C. 58%

D. 50%

Question # 43

Given Unit 3 is at 15% power.

Which one of the following completes the statements below?

A Main Steam Line Isolation signal ___(1)___ directly cause an automatic Auxiliary Feedwater actuation.

A feedwater isolation signal ___(2)___ close Main Feedwater Control Valves FCV-3-478, FCV-3-488 and FCV-3-498.

A. (1) will
(2) will

B. (1) will NOT
(2) will

C. (1) will
(2) will NOT

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

Question # 44

Given the following conditions:

- Unit 4 trips due to a LOOP.
- The crew is securing AFW to Unit 4.
- SG pressures are 200 psig.

Which one of the following identifies (1) which valve(s) is / are required to be throttled open PRIOR to starting the B Standby Feed Pump (SSGFP) and (2) the amount of pump run time that the B SSGFP fuel tank capacity will provide at full flow?

A. (1) FCV-4-479, FCV-4-489 and FCV-4-499, FW Bypass Valves
(2) 11 hours

B. (1) DWDS-4-012, STBY SGFP Discharge Header to Unit 4 Isolation Valve
(2) 11 hours

C. (1) FCV-4-479, FCV-4-489 and FCV-4-499, FW Bypass Valves
(2) 72 hours

D. (1) DWDS-4-012, STBY SGFP Discharge Header to Unit 4 Isolation Valve
(2) 72 hours

Question # 45

Given the following conditions:

- Unit 4 trips due to a loss of main feedwater.
- 4B 4kV Bus is locked out.
- Only the A AFW Pump is in service.
- A AFW Pump speed begins to lower due to a malfunctioning governor.

Which one of the following describes (1) how the change in AFW flow will initially affect pressurizer level and (2) the reason why?

A. (1) pressurizer level will rise initially then lower
(2) changing saturation conditions in the vessel head

B. (1) pressurizer level will rise initially then lower
(2) changing primary to secondary heat transfer rate

C. (1) pressurizer level will lower and stabilize
(2) changing saturation conditions in the vessel head

D. (1) pressurizer level will lower and stabilize
(2) changing primary to secondary heat transfer rate

Question # 46

Given the following conditions:

- A loss of off-site power and SI has occurred on Unit 4.
- 4A EDG is powering the 4A 4KV Bus.
- 4A EDG is loaded to 2500 KW with essential loads.
- The RO is directed to load additional equipment in the following order:

A containment spray pump	212 KW
A battery charger	56 KW
A CRDM fan	48 KW
A computer room chiller	43 KW
A battery room ac	30 KW
An electrical equipment room A/C	25 KW

Which one of the following indicates the maximum number of components (if any) that may be started, in the order listed, before steady state loading limits on the 4A EDG are exceeded?

A. 0

B. 1

C. 4

D. 6

Question # 47

Given the following conditions:

- A loss of all AC power has occurred on Unit 3.
- 3-EOP-ECA-0.0, Loss of All AC Power, is entered.
- 4kV bus power can NOT be restored.

Which one of the following completes the statements below?

Operators are directed to shed non-essential loads in order to __(1)__ .

Vital DC battery amps will be monitored by reading Vital DC bus voltages __(2)__ .

A. (1) lower battery discharge rates to lengthen availability of vital equipment
(2) on DCS

B. (1) lower battery hydrogen generation rates while no ventilation is available
(2) on DCS

C. (1) lower battery discharge rates to lengthen availability of vital equipment
(2) at VPA

D. (1) lower battery hydrogen generation rates while no ventilation is available
(2) at VPA

Question # 48

Given the following conditions:

Unit 3 is at 10% power.
A loss of Vital DC bus 3D23 occurs.

Which one of the following describes the direct effect on the RTBs?

A. 3A RTB opens due to loss of power to the undervoltage trip coil

B. 3B RTB opens due to loss of power to the undervoltage trip coil

C. 3A RTB opens due to loss of power to the shunt trip coil

D. 3B RTB opens due to loss of power to the shunt trip coil

Question # 49

Given the following conditions:

- Unit 3 at 100% power.
- ANN F 8/2, EDG A TROUBLE, alarms.
- ANN 1/10, LOW AIR PRESSURE, is locked in at the local panel.
- The left-side Air Receivers for the 3A EDG are at 195 psig.
- The right-side Air Receivers for the 3A EDG are at 205 psig.

Which one of the following completes the statement below?

In accordance with 3-OP-023, Emergency Diesel Generator, the 3A EDG ___(1)___ available to start and the right-side air receivers ___(2)___ be aligned to supply both the left-side and right-side set of Air Start Motors.

A. (1) is
(2) will

B. (1) is NOT
(2) will

C. (1) is
(2) will NOT

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

Question # 50

Given the following initial conditions:

- Unit 3 is at 100% power.

Subsequently:

- Annunciator H-1/6, PRMS CHANNEL FAILURE actuates.
- R-19, SG Blowdown Radiation Monitor, is failed.

Which one of the following completes the statements below?

Blowdown Isolation Valves, CV-3-6275A, B and C ___(1)___ auto close.

S/G Liquid Sample Valves, MOV-3-1425, 1426 and 1427 ___(2)___ auto close.

A. (1) will
(2) will

B. (1) will
(2) will NOT

C. (1) will NOT
(2) will

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

Question # 51

Given the following initial conditions:

- Unit 3 is at 8% power, MOL.
- 3-GOP-301, Hot Standby to Power Operation, startup is in progress.
- The main generator is rolling unloaded at 1800 rpm.
- Steam Dump to Atmosphere (SDTA) valves are controlled as follows:
  - 3A is throttled in manual.
  - 3B is closed in manual.
  - 3C is throttled in AUTO.

Subsequently:

- 3C SDTA setpoint is lowered by 20 psig.

Which one of the following completes the statements below?

With no additional operator action, reactor power will ___(1)___ .

Control rods ___(2)___ automatically compensate for the SDTA adjustment.

A. (1) lower
(2) will

B. (1) rise
(2) will

C. (1) lower
(2) will NOT

D. (1) rise
(2) will NOT

Question # 52

Given the following conditions:

- Unit 3 is at 75% power.
- Annunciator F 1/1, RCP MOTOR/SHAFT HI VIB, is received.
- R-3-369 RCP Vibration Recorder indicates the following for 3B RCP:
  - Shaft vibration is 20 mils and rising slowly.
  - Motor frame vibration is 4 mils and stable.

Which one of the following completes the statement below?

The crew will ___(1)___ in accordance with ___(2)___ .

A. (1) trip the reactor then trip the 3B RCP
(2) 3-ARP-097.CR.F, Control Room Annunciator Response Panel F

B. (1) trip the reactor then trip the 3B RCP
(2) 3-ONOP-041.1, Reactor Coolant Pump Off-Normal

C. (1) perform a fast load reduction
(2) 3-GOP-100, Fast Load Reduction

D. (1) perform a fast load reduction
(2) 3-GOP-103, Power Operation to Hot Standby

Question # 53

Given the following conditions:

- Unit 3 is at 100% power.
- The 3A and 3B ICW pumps are running.
- The ICW header piping ruptures.

Which one of the following completes the statements below?

An ICW header __(1)__ alarm will come in.

3-ONOP-019, Intake Cooling Water Malfunction, will direct __(2)__ start of the 3C ICW pump.

A. (1) low pressure
(2) a manual

B. (1) low pressure
(2) verifying an automatic

C. (1) high flow
(2) a manual

D. (1) high flow
(2) verifying an automatic

Question # 54

Given the following conditions:

- Unit 3 is at 100% power.
- Instrument Air Header Pressure on PI-3-1444 is 85 psig and lowering slowly.
- Containment pressure is 0.3 psig and rising slowly.
- A Field Operator closes 3-40-337, Header Supply to Containment Isolation.

Which one of the following completes the statement below?

Normal letdown ___(1)___ expected to be lost and 3C loop pressurizer spray valve is expected to fail ___(2)___ .

A. (1) is NOT
(2) closed

B. (1) is
(2) open

C. (1) is
(2) closed

D. (1) is NOT
(2) open

Question # 55

Given the following conditions:

- Unit 3 is in Mode 6 with refueling in progress.
- A spent fuel assembly is damaged in containment.
- 3-ONOP-033.3, Accidents Involving New or Spent Fuel, is entered.
- The Containment Evacuation alarm is actuated.

Which one of the following completes the statements below:

The sound of the containment evacuation alarm can be described as a ___(1)___ .

After the plant page is made to evacuate containment, the crew must next stop the ___(2)___ .

A. (1) beeping tone
(2) Containment Purge Fans

B. (1) beeping tone
(2) Normal Containment Cooler Fans

C. (1) wailing tone
(2) Containment Purge Fans

D. (1) wailing tone
(2) Normal Containment Cooler Fans

Question # 56

Given the following initial conditions:

- Unit 3 is at 50% power.
- All RCP currents indicate 600 amps.

Subsequently:

- 3A RCP current drops and stabilizes at 150 amps.

Which one of the following completes the statement below?

The reactor will automatically trip on _____ .

A. RCS loop loss of flow

B. 4kV bus voltage

C. OPΔT

D. OTΔT

Question # 57

Given the following initial conditions:

- Unit 3 is at 40% power.
- LT-3-461, PRZ LEVEL PROT / CONT channel failed.
- All bistables for the failed channel are tripped.
- The plant is stabilized and all systems are in automatic.

Subsequently:

PRZ LEVEL PROT / CONTROL channel:

- LI-3-459 is 90% and rising.
- LI-3-460 is 94% and rising.
- LI-3-461 is 85% and stable.

Which one of the following identifies a condition the crew is currently responding to?

A. Reactor trip breakers opening

B. Charging pumps tripping

C. PRZ heaters tripping

D. Backup PRZ heaters energizing

Question # 58

Given the following conditions:

- A load reduction is in progress in accordance with 3-GOP-100, Fast Load Reduction.
- Rod Control is in AUTO.
- Control bank C RPIs indicate 188 steps.
- Control bank D RPIs indicate 60 steps.
- Reactor power is 50%.

Which one of the following completes the statements below?

The Rod Insertion Limit Monitor compares C & D bank steps with ___(1)___ to determine if limits are exceeded.

The rod insertion limits ___(2)___ exceeded.

REFERENCE PROVIDED

A. (1) median loop Tavg
(2) are

B. (1) median loop Tavg
(2) are NOT

C. (1) median loop ΔT
(2) are

D. (1) median loop ΔT
(2) are NOT

Question # 59

Given the following conditions:

- Unit 4 is at 100% power.
- A Core Exit Thermocouple fails.
- ANN A4/2 QSPDS INADEQUATE CORE COOLING, alarms.

Which one of the following completes the statements below?

QSPDS ___(1)___ automatically bypass the failed inputs.

ANN A4/2, can also be actuated by a failed ___(2)___.

A. (1) will
(2) Tavg module

B. (1) will NOT
(2) Tavg module

C. (1) will
(2) RVLMS sensor

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

Question # 60

Given the following conditions:

- Unit 3 is in Mode 5.
- A Containment purge is initiated in accordance with 3-NOP-053, Containment Purge System.

Which one of the following completes the statements below?

Containment Purge flow is monitored on ___(1)___ .

A high alarm on noble gas monitor 3-R-12 ___(2)___ cause the Unit 3 purge exhaust and supply fans to trip.

A. (1) DCS ONLY
(2) will

B. (1) DCS and VPB
(2) will

C. (1) DCS ONLY
(2) will NOT

D. (1) DCS and VPB
(2) will NOT

Question # 61

Given the following initial conditions:

- Unit 3 is in Mode 5.
- Spent Fuel Pit (SFP) level is 56’ 11”.
- SFP boron concentration is 2300 ppm.

Subsequently:

- A tube inside of the in-service Spent Fuel Pool Heat Exchanger breaks.
- ANN H 8/6 CCW HEAD TANK HI/LO LEVEL alarms.
- A crew is dispatched to isolate the heat exchanger.

Which one of the following correctly completes the statement below?

The crew must take action to prevent Spent Fuel Pool ___(1)___ from violating the Tech Spec LCO by performing a ___(2)___ to the SFP in accordance with 3-NOP-033, Spent Fuel Pit Cooling System.

A. (1) level
(2) direct boration

B. (1) level
(2) primary water fill

C. (1) boron concentration
(2) direct boration

D. (1) boron concentration
(2) primary water fill

Question # 62

Given the following conditions:

- A 3C Steam Generator tube rupture occurs.
- A loss of offsite power occurs.
- The crew enters 3-EOP-E-3, Steam Generator Tube Rupture.
- ECCS flow has been terminated.
- Offsite power has been restored to the emergency buses.
- The crew is performing step 37, Evaluate RCP Status.

Which one of the following describes the preferred operation of the RCPs?

A. Start the 3C RCP, then 3B RCP.

B. Start ONLY the 3B RCP.

C. Start ONLY the 3C RCP.

D. Start ONLY the 3A RCP.

Question # 63

Given the following conditions:

- Unit 3 is at 75% power and stable.
- Absolute condenser pressure is 2 in Hg on DCS.
- A steam leak occurs downstream of steam jet air ejector common steam supply valve, 3-30-020.

Which one of the following completes the statements below?

Absolute condenser pressure will ___(1)___ .

Turbine exhaust hood temperatures will ___(2)___ .

A. (1) RISE
(2) LOWER

B. (1) RISE
(2) RISE

C. (1) LOWER
(2) LOWER

D. (1) LOWER
(2) RISE

Question # 64

Given the following conditions:

- Discharge of a Waste Monitor Tank is in progress.
- H 1/4, PRMS HI RADIATION, is received.
- R-18 HI alarm is lit.

Which one of the following describes the action required by the crew?

A. Manually trip the in-service Waste Monitor Tank discharge pump.

B. Manually close RCV-018, Liquid Waste Effluent Isolation Valve.

C. Verify that the in-service Waster Monitor Tank discharge pump automatically trips.

D. Verify that RCV-018, Liquid Waste Effluent Isolation Valve automatically closes.

Question # 65

Given the following conditions:

- Both units are at 100% power.
- ANN I-6/1 INSTRUMENT AIR HI TEMP/LO PRESS alarms on both units.
- Both units implement 3/4-ONOP-013, Loss of Instrument Air.
- Instrument air pressure on Unit 3 is 85 psig and slowly lowering.
- Instrument air pressure on Unit 4 is 90 psig and stable.
- NO additional Instrument Air Compressors can be started.

Which one of the following describes the required actions in accordance with 3/4-ONOP-013?

A. Manually close Header Pressure Control Valve CV-4-1605 and then close CV-3-1605.

B. Manually close Header Pressure Control Valve CV-3-1605 and then close CV-4-1605.

C. Locally start available Service Air Compressors and open Service Air Supply to Unit 3 / Unit 4 tie valve 40-2059.

D. Locally start available Service Air Compressors and open Units 1 and 2 Instrument / Service air supply valve 40-1331.

Question # 66

Which one of the following completes the statement below?

The following shall be OPERABLE for core alterations:

At a minimum, ___(1)___ Source Range Neutron Flux Monitor(s) with continuous visual indication in the control room and audible indication in the containment and control room, and one ___(2)___ channel with continuous visual indication in the control room.

A. (1) one
(2) Intermediate Range

B. (1) two
(2) Intermediate Range

C. (1) one
(2) Gamma Metrics

D. (1) two
(2) Gamma Metrics

Question # 67

Given the following conditions:

- Unit 3 is at 100% power.
- A 3C loop Tcold fails HIGH.
- The following alarms are received:
  - B 3/6, OTΔT
  - B 4/5, RCS HI/LO TAVG
  - B 5/5, OTΔT/OPΔT ROD STOP
  - B 5/6, ΔT DEVIATION
  - J 7/4, EAGLE 21 TROUBLE
  - J 9/5, RTD CHANNEL FAILURE

Which one of the following completes the statements below?

Median Tavg on TR-3-408 will ___(1)___ .

3C loop ___(2)___ setpoint on VPA will indicate lower.

A. (1) indicate higher
   (2) OTΔT

B. (1) remain the same
   (2) OTΔT

C. (1) indicate higher
   (2) OPΔT

D. (1) remain the same
   (2) OPΔT

Question # 68

Given the following conditions:

- A control room evacuation is ordered in accordance with 0-ONOP-105, Control Room Evacuation.

Which one of the following completes the statements below?

The Unit 3 ___(1)___ will report to the Unit 3 Alternate Shutdown Panel to control the plant.

The ___(2)___ is required to report to the Unit 3 and 4 Main Turbine Front Standards and ensure the Turbines are tripped.

A. (1) Reactor Operator
(2) Non-Fire Brigade Nuclear Plant Operator

B. (1) Reactor Operator
(2) Third Licensed Reactor Operator

C. (1) Unit Supervisor
(2) Non-Fire Brigade Nuclear Plant Operator

D. (1) Unit Supervisor
(2) Third Licensed Reactor Operator

Question # 69

Given the following conditions:

- Both units are at 100% power.

Which one of the following completes the statements below?

In accordance with 3-OSP-075.1, Auxiliary Feedwater Train 1 Operability Verification, Train 1 of AFW ___(1)___ be declared OPERABLE with AFW feedwater control valves in manual.

When the test is complete, AFW flow controllers will be set to ___(2)___ .

A. (1) will
(2) 130 gpm

B. (1) will
(2) 135 gpm

C. (1) will NOT
(2) 130 gpm

D. (1) will NOT
(2) 135 gpm

Question # 70

Given the following conditions:

- Unit 3 is at 100% power.
- All circulating water pumps are running.
- MOV-3-1415, 3A2 CIRC WTR PUMP DISCHARGE VLV, is de-energized open on a clearance.
- The next step in the clearance is to stop the 3A2 Circulating Water Pump (CWP).

Which one of the following completes the statements below?

REFERENCES PROVIDED

The 3A2 CWP ___(1)___ be stopped at this time.

Breaker 3AA01 ___(2)___ be used to complete the 3A2 CWP tag out.

A. (1) will NOT
(2) will NOT

B. (1) will NOT
(2) will

C. (1) will
(2) will NOT

D. (1) will
(2) will

Question # 71

Given the following conditions:

- Unit 3 is raising power from 25% to 100%.
- Containment radiation monitors indicate rising trends.
- R-11/12, Containment Air Particulate / Gas Radiation Monitors are in HIGH alarm.

Which one of the following completes the statements below?

In accordance with 3-ONOP-067, Radioactive Effluent Release, to check the channel operability of R-11/12, the operator must ___(1)___.

The R-11/12 bypass switch on VPB ___(2)___ a keylock switch.

A. (1) depress the CHECK SOURCE pushbutton and ensure that the readout rises slightly
(2) is

B. (1) depress the CHECK SOURCE pushbutton and ensure that the readout rises slightly
(2) is NOT

C. (1) depress the FAIL/TEST pushbutton and ensure that the readout equals 288K or 289K
(2) is

D. (1) depress the FAIL/TEST pushbutton and ensure that the readout equals 288K or 289K
(2) is NOT

Question # 72

Given the following conditions:

- The crew is maneuvering the unit in accordance with 3-GOP-301, Hot Standby to Power Operation.
- Power is stabilized at 5% power.

Which one of the following completes the statements below?

During the next rod withdrawal, the maximum startup rate allowed is ___(1)___ .

When syncing to the grid, steam load will transfer from the steam dumps to ___(2)___ to the main turbine.

A. (1) 1 dpm
(2) atmosphere

B. (1) 1 dpm
(2) condenser

C. (1) 0.5 dpm
(2) atmosphere

D. (1) 0.5 dpm
(2) condenser

Question # 73

Given the following conditions:

- An imminent airborne threat has been confirmed by Plant Security.

Which one of the following completes the statements below?

The ___(1)___ will become the Emergency Coordinator.

If ONLY the EC is incapacitated, the Third Licensed Reactor Operator, which does NOT hold an active SRO license, ___(2)___ be next in line to relieve the EC.

A. (1) Shift Manager
(2) will NOT

B. (1) Security Shift Supervisor
(2) will NOT

C. (1) Shift Manager
(2) will

D. (1) Security Shift Supervisor
(2) will

Question # 74

Given the following initial conditions:

- Unit 3 trips from 25% power due to a Startup Transformer failure.
- The Startup Transformer must be replaced.
- Safety Injection is NOT required.

Which one of the following completes the statements below:

The crew will transition from 3-EOP-E-0, Reactor Trip or Safety, to a(n) __(1)__ to cooldown the plant.

When performing non-accident EOPs, all Technical Specification surveillances __(2)__ required to be complied with.

A. (1) Functional Restoration Procedure
(2) are NOT

B. (1) Functional Restoration Procedure
(2 are

C. (1) Optimal Recovery Procedure
(2) are NOT

D. (1) Optimal Recovery Procedure
(2) are

Question # 75

Which one of the following identifies a Control Board Instrument that may be available in a post-accident condition, and the required color of the instrument label, in accordance with 0-ADM-209, Equipment Tagging and Labeling?

A. PI-3-444, Pressurizer Pressure; blue

B. PI-3-444, Pressurizer Pressure; purple

C. TI-3-410A, Loop A T-cold Wide Range; blue

D. TI-3-410A, Loop A T-cold Wide Range; purple

Question # 76

Given the following conditions:

- Unit 3 is at 7% power.
- An automatic reactor trip signal is generated, but the reactor fails to trip.
- 3-EOP-E-0, Reactor Trip or Safety Injection, is entered.
- The crew completes their immediate operator actions.
- The unit is at 6% power and lowering.

Which one of the following completes the statements below?

The SRO ___(1)___ direct the RO to perform a boration in accordance with 0-OP-046, CVCS Boron Concentration Control for the failure.

If an automatic SI fails to occur when required, the SRO will ___(2)___ .

A. (1) will NOT
(2) direct a transition back to 3-EOP-E-0, Reactor Trip or Safety Injection

B. (1) will NOT
(2) continue in current procedure while aligning equipment in Attachment 3 of 3-EOP-E-0

C. (1) will
(2) direct a transition back to 3-EOP-E-0, Reactor Trip or Safety Injection

D. (1) will
(2) continue in current procedure while aligning equipment in Attachment 3 of 3-EOP-E-0

Question # 77

Given the following conditions:

- Unit 3 is heating up in Mode 3.
- RCS pressure is 480 psig at 355F.
- Shortly after starting 3A RCP, the following is observed:
  - ANN A 7/5, RCP TROUBLE, alarms.
  - #1 seal ΔP is 200 psid.
  - #2 seal ΔP is 100 psid.
  - #3 seal ΔP is 180 psid.
  - Controlled Bleed Off (CBO) is 0.4 gpm.

Which one of the following describes the correct response?

The SRO will address a ___(1)___ seal failure and the CBO flow condition in accordance with ___(2)___ .

A. (1) #1
(2) 3-GOP-103, Power Operation to Hot Standby

B. (1) #2
(2) 3-GOP-103, Power Operation to Hot Standby

C. (1) #1
(2) 3-ONOP-041.1, Reactor Coolant Pump Off-Normal

D. (1) #2
(2) 3-ONOP-041.1, Reactor Coolant Pump Off-Normal

Question # 78

Given the following conditions:

- Unit 3 is at 100% power and rising.
- ANN H8/6, CCW HEAD TANK HI/LO LEVEL alarms.
- ANN A4/6, VCT HI/LO LEVEL alarms.
- CCW Head Tank level is 0%.
- LI-3-115, VCT level is 81% and rising.
- Tavg is rising slowly.
- Control rods are automatically inserting.
- CCW Surge Tank level stabilizes with MOV-3-832, Component Cooling Water Surge Tank Makeup, fully opened.

Which one of the following completes the statement below?

The SRO will direct isolating the ___(1)___ in accordance with ___(2)___ .

A. (1) Seal Water Return Heat Exchanger
(2) 3-ONOP-030, Component Cooling Water Malfunction

B. (1) Non-Regen Heat Exchanger
(2) 3-ONOP-030, Component Cooling Water Malfunction

C. (1) Seal Water Return Heat Exchanger
(2) 3-ONOP-041.3, Excessive Reactor Coolant System Leakage

D. (1) Non-Regen Heat Exchanger
(2) 3-ONOP-041.3, Excessive Reactor Coolant System Leakage

Question # 79

Given the following initial conditions:

- Unit 3 is at 100% power.
- A Steam Generator tube leak is in progress.

Subsequently:

- An ICW system malfunction causes secondary temperatures to rise.

Which one of the following completes the statements below?

The SRO will direct a reactor trip when ___(1)___ .

If the crew manually performs a safety injection following the trip due to degrading PRZ pressure, ___(2)___ of 3-EOP-E-0 will be used to verify proper ICW system operation.

A. (1) TPCW Supply Header temperature is 110°F
(2) ATTACHMENT 3, Prompt Action Verifications

B. (1) Turbine Bearing temperature is 185°F
(2) ATTACHMENT 3, Prompt Action Verifications

C. (1) TPCW Supply Header temperature is 110°F
(2) ATTACHMENT 5, Align Plant Equipment

D. (1) Turbine Bearing temperature is 185°F
(2) ATTACHMENT 5, Align Plant Equipment

Question # 80

Given the following conditions:

- Unit 3 is at 100% power.
- A loss of instrument air (IA) is in progress.
- The crew is unable to restore IA pressure and initiates a manual reactor trip.
- RCS pressure is 2150 psig.
- The crew initiates a controlled cooldown to 470°F.
- During the cooldown, SI actuates on Main Steam Line High DP.
- The Shift Manager decides to declare an Unusual Event

Which one of the following describes the maximum time the SRO is allowed to notify the NRC Operations Center (NRCOC) in accordance with LI-AA-102-1001, Regulatory Reporting?

A. immediately

B. 15 minutes

C. 1 hour

D. 4 hours

Question # 81

Given the following initial conditions:

- Reactor trip and safety injection occurred on Unit 4.
- The crew enters 4-EOP-ECA-1.2, LOCA Outside Containment.

Subsequently:

- MOV-4-843A and MOV-4-843B, SI to Cold Leg Isolation valves, are closed to isolate the leak.
- 4A Charging Pump is running.
- RCS pressure is 1700 psig and rising slowly.
- PZR level is 10% and rising.
- SG levels are 16% and rising.
- CET subcooling is 34°F and stable.

Which one of the following identifies the required procedure transition and EOP strategy?

Transition to...

A. 4-EOP-E-1, Loss of Reactor or Secondary Coolant, and then to 4-EOP-ES-1.1, SI Termination.

B. 4-EOP-E-1, Loss of Reactor or Secondary Coolant, and then to 4-EOP-ES-1 .2, Post LOCA Cooldown and Depressurization.

C. 4-EOP-ECA-1.1, Loss of Emergency Coolant Recirculation, and then to 4-EOP-ES-1 .2, Post LOCA Cooldown and Depressurization.

D. 4-EOP-ECA-1.1, Loss of Emergency Coolant Recirculation, and then to 4-EOP-ECA-1.2, LOCA Outside Containment.

Question # 82

Given the following conditions:

- Both Units are at 100% power.

On Unit 3:
- The crew is preparing to release a Gas Decay Tank.
- R-14, Plant Vent Radiation Monitor alarm setpoint cannot be adjusted properly and is declared inoperable.
- RAD-6304, Plant Vent SPING Radiation Monitor, is inoperable.

On Unit 4:
- Fuel shuffle is in progress in the Spent Fuel Pool.

Which one of the following completes the statement below?

Tech Spec LCO(s) __(1)__ is / are applicable, and the most time restrictive action required is __(2)__ .

REFERENCE PROVIDED

A. (1) 3.3.3.1 (Unit 4)
(2) initiate the preplanned alternate method of monitoring the appropriate parameter(s), within 72 hours

B. (1) 3.3.3.1 (Unit 4)
(2) immediately suspend spent fuel manipulations in the Unit 4 Spent Fuel Pool

C. (1) 3.3.3.1 (Unit 4) and 3.3.3.3 (Both units)
(2) initiate the preplanned alternate method of monitoring the appropriate parameter(s), within 72 hours

D. (1) 3.3.3.1 (Unit 4) and 3.3.3.3 (Both units)
(2) immediately suspend spent fuel manipulations in the Unit 4 Spent Fuel Pool

Question # 83

Given the following conditions:

- Unit 3 is in Mode 4 during a plant heatup.
- RI-3-6311A, Containment High Range Radiation Monitor fails.
- RI-3-6311B is unaffected.

Which one of the following identifies the action required in accordance with Technical Specifications?

Declare the channel inoperable and...

REFERENCE PROVIDED

A. continue RCS heatup to Mode 3.

B. remain in Mode 4 until it is restored to operable status.

C. initiate the preplanned alternate method of monitoring the parameter within 14 days.

D. restore the inoperable channel(s) to operable status within 7 days or make a special report to the NRC.

Question # 84

Given the following conditions:

- Unit 3 is in Mode 1.
- A containment entry is in progress.
- The gasket on the inner air lock door is damaged.
- There is a gouge across the entire sealing surface approximately 1/8 inch deep and 1/2 inch wide.
- Air flow could be heard through the gouge when pressure is equalized across the inner door.
- When personnel are exiting the hatch, the outer door fails to close.
- No corrective maintenance has started due to parts unavailability.

Which one of the following identifies (1) the applicable Technical Specification LCO action(s), and (2), the MOST restrictive action required?

REFERENCE PROVIDED

A. (1) 3.6.1.3 ONLY
(2) Restore the air lock door within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

B. (1) 3.6.1.3 ONLY
(2) Restore the inoperable air lock to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

C. (1) 3.6.1.1 AND 3.6.1.3
(2) Restore containment integrity within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

D. (1) 3.6.1.1 AND 3.6.1.3
(2) Restore the inoperable air lock to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

Question # 85

Given the following conditions:

- A LOCA has occurred on Unit 3.
- The SRO transitions to 3-EOP-E-1, Loss of Reactor or Secondary Coolant.
- RCS pressure is 900 psig.
- RWST level is 180,000 gallons.
- ALL Charging Pumps are tripped.
- Phase B isolation inadvertently actuated and has NOT been reset.
- AFW is operating as required.
- The SRO is transitioning out of 3-EOP-E-1.

Which one of the following identifies (1) the subsequent procedure entry required, and (2) the operation of the RCPs once transition is performed?

A. (1) 3-EOP-ES-1.2, Post LOCA Cooldown and Depressurization
(2) ONE RCP will be started for mixing of the RCS and to prevent void formation in the head

B. (1) 3-EOP-ES-1.2, Post LOCA Cooldown and Depressurization
(2) RCPs may NOT be started until a status evaluation has been performed

C. (1) 3-EOP-ES-1.3, Transfer to Cold Leg Recirculation
(2) ONE RCP will be started for mixing of the RCS and to prevent void formation in the head

D. (1) 3-EOP-ES-1.3, Transfer to Cold Leg Recirculation
(2) RCPs may NOT be started until a status evaluation has been performed

Question # 86

Given the following conditions:

- Unit 3 is at 100% power when a loss of all charging flow occurs.
- 3-ONOP-047.1, Loss of Charging flow in Modes 1-4, is entered.
- PZR level is 55% and lowering.
- VCT level is 2% and lowering.
- LCV-3-115B, Emergency Make up to Charging Pumps, is closed and will not open.

Which one of the following completes the statements below?

Tech Spec 3.0.3 ___(1)___ applicable.

If a charging pump is subsequently restored, the Unit Supervisor will direct the crew to ___(2)___ restore charging pump suction to the RWST in accordance with Attachment 2, Establish Charging Flow.

A. (1) is
(2) manually

B. (1) is NOT
(2) manually

C. (1) is
(2) locally

D. (1) is NOT
(2) locally

Question # 87

Given the following conditions:

- Unit 4 tripped from 100% power.
- 4B DC Bus voltage is zero.
- 4P08 Vital Instrument Bus Inverter has failed.

Which one of the following describes (1) the effect on Unit 4 and (2) the most restrictive Technical Specification (TS) actions required?

A. (1) 4B ESF Load Sequencer and AFW actuation for bus stripping are lost
(2) Comply with actions of TS 3.3.2, ESF System Actuation Instrumentation

B. (1) 4B ESF Load Sequencer and AFW actuation for bus stripping are lost
(2) Comply with actions of TS 3.0.3

C. (1) Startup Transformers are inoperable due to loss of power to protection circuits
(2) Comply with actions of TS 3.8.1.1, AC Sources

D. (1) Startup Transformers are inoperable due to loss of power to protection circuits
(2) Comply with actions of TS 3.0.3

Question # 88

Given the following conditions:

- Unit 3 has experienced a LOCA.
- RCS pressure is 1250 psig.
- Containment temperature is 187°F.
- SG narrow range levels are 5%.
- Tavg is 552°F
- All AFW pumps are tripped and will NOT restart.
- The crew transitions out of 3-EOP-E-0, Reactor Trip or Safety Injection.

Which one of the following describes the condition of the Unit and the action required?

A. RCS bleed and feed requirements are met. Enter 3-EOP-FR-H.1, Loss of Secondary Heat Sink, trip RCPs and initiate Bleed and Feed.

B. RCS bleed and feed requirements are NOT met. Enter 3-EOP-FR-H.1 and attempt to establish Standby Feedwater flow to at least ONE SG.

C. RCS bleed and feed requirements are NOT met. Enter 3-EOP-FR-H.1 and attempt to establish Main Feedwater flow to at least ONE SG.

D. Secondary Heat Sink is NOT required. Enter 3-EOP-FR-H.1 and then immediately exit.

Question # 89

Given the following initial conditions:

- Core offload is in progress on Unit 3.
- 3D 4kV bus is OOS on a clearance.
- 3B RHR pump is in service.
- RCS temperature is 127°F.

Subsequently:

- The switchyard is de-energized.
- 3A EDG is manually loaded onto the 3A 4kV bus after a 10 minute delay.
- 3B EDG fails to start.
- Switchyard expected time of restoration is 2 hours.

Which one of the following completes the statement below?

The SRO will declare an __(1)__ in accordance with 0-EPIP-20101 and will reinitiate core cooling by __(2)__ .

REFERENCE PROVIDED

A. (1) Unusual Event
(2) loading an RHR pump on an EDG in accordance with 3-ONOP-004, Loss of Offsite Power

B. (1) Unusual Event
(2) establishing a secondary heat sink with a SSGFP in accordance 3-ONOP-050, Loss of RHR

C. (1) Alert
(2) loading an RHR pump on an EDG in accordance with 3-ONOP-004, Loss of Offsite Power

D. (1) Alert
(2) establishing a secondary heat sink with a SSGFP in accordance 3-ONOP-050, Loss of RHR

Question # 90

Given the following conditions:

- E-Waste Gas Decay Tank release is in progress.
- R-14, Plant Vent Gaseous Monitor, FAIL indicator light illuminates with high indication.
- Initial sample results for release of the tank were acceptable in accordance with 0-NCOP-004, Preparation of Gas Release Permits.

Which one of the following completes the statement below?

The release ___(1)___ terminated and ___(2)___.

REFERENCE PROVIDED

A. (1) will be automatically
(2) R-14 must be returned to operability prior to reinitiating the release

B. (1) must be manually
(2) R-14 must be returned to operability prior to reinitiating the release

C. (1) will be automatically
(2) compensatory actions are required to reinitiate the release with R-14 inoperable

D. (1) must be manually
(2) compensatory actions are required to reinitiate the release with R-14 inoperable

Question # 91

Given the following conditions:

- An unplanned load reduction is in progress on Unit 3.
- ANN B 9/3, SHUTDOWN ROD OFF TOP/DEVIATION, alarms and locks in.
- Control Bank D, rod H-12, is 20 steps above the remainder of control bank D.
- Control rod H-12 is mechanically bound at its current position.

Which one of the following completes the statements below?

The Rod Deviation Monitor ___(1)___ OPERABLE.

In accordance with Tech Specs, a comparison of the Demand Position Indication System and the Analog Rod Position Indication System must be performed at a minimum of once per ___(2)___ hours.

REFERENCE PROVIDED

A. (1) is NOT
(2) 5

B. (1) is NOT
(2) 4

C. (1) is
(2) 5

D. (1) is
(2) 4

Question # 92

Given the following initial conditions:

- Unit 3 calorimetric power on DCS is 99.98% and stable on the hourly.

Subsequently:

- A steam dump to condenser valve fails open.
- The 5 minute calorimetric is 101% power.
- The hourly calorimetric is 100.05% power.
- The 8-hr calorimetric is 99.99% power.

Which one of the following identifies (1) the effect on the main generator and (2) the NRC Operations Center (NRCOC) notification requirement if applicable?

A. (1) main generator load will not change
(2) NRCOC must be notified

B. (1) main generator load will lower
(2) NRCOC must be notified

C. (1) main generator load will not change
(2) NRCOC is NOT required to be notified

D. (1) main generator load will lower
(2) NRCOC is NOT required to be notified

Question # 93

Given the following conditions:

- Unit 3 experiences a reactor trip and SI due to a SBLOCA.
- 3-EOP-FR-C.2, Response to Degraded Core Cooling, is entered.
- All CETs are 735°F and rising.
- RVLMS head indicates full.

Which one of the following identifies (1) the current Emergency Classification, and (2) the MAXIMUM amount of time to Notify the State of Florida?

REFERENCE PROVIDED

A. (1) Site Area Emergency
(2) 15 Minutes

B. (1) Site Area Emergency
(2) 60 minutes

C. (1) General Emergency
(2) 15 minutes

D. (1) General Emergency
(2) 60 minutes

Question # 94

Given the following conditions:

- The date is 8/5/2016.
- Both Units are in Mode 1.
- Shift complement is at MINIMUM allowed by Technical Specifications.
- One of the ROs becomes ill and must be transported to the hospital.
- TWO potential replacements are identified for call-in.
- BOTH replacements have been assigned to OPS Support for the last year.
- The last times they were on shift are as follows:

Operator A	Operator B
12 hours on June 24 BOP	12 hours on June 21 RCO
12 hours on May 23 BOP	12 hours on June 20 RCO
12 hours on May 22 BOP	12 hours on June 19 RCO
12 hours on April 19 BOP	12 hours on June 18 RCO
12 hours on April 18 BOP	12 hours on May 5 WCC

Which one of the following completes the statements below?

In accordance with Technical Specifications, action must be taken to ensure the RO is replaced within a maximum of ___(1)___ .

Operator ___(2)___ will be selected as the replacement.

REFERENCE PROVIDED

A. (1) 2 hours
(2) A

B. (1) 2 hours
(2) B

C. (1) 1 hour
(2) A

D. (1) 1 hour
(2) B

Question # 95

Which one of the following completes the statements below?

In accordance with 0-ADM-701, Control Of Plant Work Activities, a ___(1)___ shall be designated as a liason between the Shift Manager and the troubleshooting team for troubleshooting sensitive or load threatening equipment.

When the plant is in a 3.0.3 Tech Spec action or a load threatening condition, the Shift Manager ___(2)___ allowed to authorize work without formal planning and prior to obtaining QC approval.

A. (1) maintenance planner
(2) is NOT

B. (1) senior reactor operator
(2) is NOT

C. (1) maintenance planner
(2) is

D. (1) senior reactor operator
(2) is

Question # 96

A required annunciator must be defeated in accordance with 0-OSP-200.5, Miscellaneous Tests, Checks, and Operating Evolutions, Attachment 1.

Which one of the following identifies the maximum time the (1) annunciator may be defeated prior to performing a PCR to the ARP, and (2) allowed for an out of service annunciator prior to performing a 10CFR50.59 applicability/screening review?

A. (1) 7 days
(2) 7 days

B. (1) 7 days
(2) 60 days

C. (1) 60 days
(2) 7 days

D. (1) 60 days
(2) 60 days

Question # 97

Given the following conditions:

- Refueling is in progress.
- A spent fuel element is being moved from the reactor to the upender.
- The spent fuel element is dropped to the bottom of the canal and ruptures.

Which one of the following identifies what products released from the ruptured spent fuel element will present the MOST immediate hazard?

A. Alpha radiation from Argon decay

B. Alpha radiation from Fluorine decay

C. Gamma radiation from Cobalt and Nickel solids

D. Gamma radiation from Iodine and Krypton gases

Question # 98

Given the following conditions:

- At 0000 on 7/31/2016, a Unit 4 shutdown to MODE 3 is commenced.
- At 0630 on 7/31/2016, Unit 4 enters MODE 3.
- At 2200 on 8/01/2016, the first reactor vessel head stud is de-tensioned.

Which one of the following completes the statements below?

In accordance with Technical Specifications, the EARLIEST time that fuel movement may commence is ___(1)___ .

The basis for this time requirement is to ensure that ___(2)___ .

A. (1) 0630 on 08/03/16
(2) the release of fission product radioactivity, subsequent to a fuel handling accident, results in doses that are well within the values specified in the safety analysis

B. (1) 0630 on 08/03/16
(2) the heat load assumptions specified in the safety analysis are met, so that boiling may be prevented in the Spent Fuel Pool

C. (1) 2200 on 08/04/16
(2) the release of fission product radioactivity, subsequent to a fuel handling accident, results in doses that are well within the values specified in the safety analysis

D. (1) 2200 on 08/04/16
(2) the heat load assumptions specified in the safety analysis are met, so that boiling may be prevented in the Spent Fuel Pool

Question # 99

Given the following conditions:

Unit 3 is in Mode 3.
A fire alarm has been received in Fire Zone 086, Unit 3 Main and Startup Transformer Area.
The fire alarm has been confirmed by field operators.

Which one of the following completes the statements below?

The ___(1)___ Train is credited for Safe Shutdown for a fire in this area.

The ___(2)___ is responsible for calling in off-site assistance, if required, in fighting the fire.

A. (1) 3A
(2) Shift Manager

B. (1) 3A
(2) Security Shift Supervisor

C. (1) 3B
(2) Shift Manager

D. (1) 3B
(2) Security Shift Supervisor

Question # 100

Given the following conditions:

- A Steam Generator Tube Rupture has occurred on Unit 3.
- The crew is performing 3-EOP-E-3, Steam Generator Tube Rupture, when a loss of vital instrument bus 3P07 occurs.

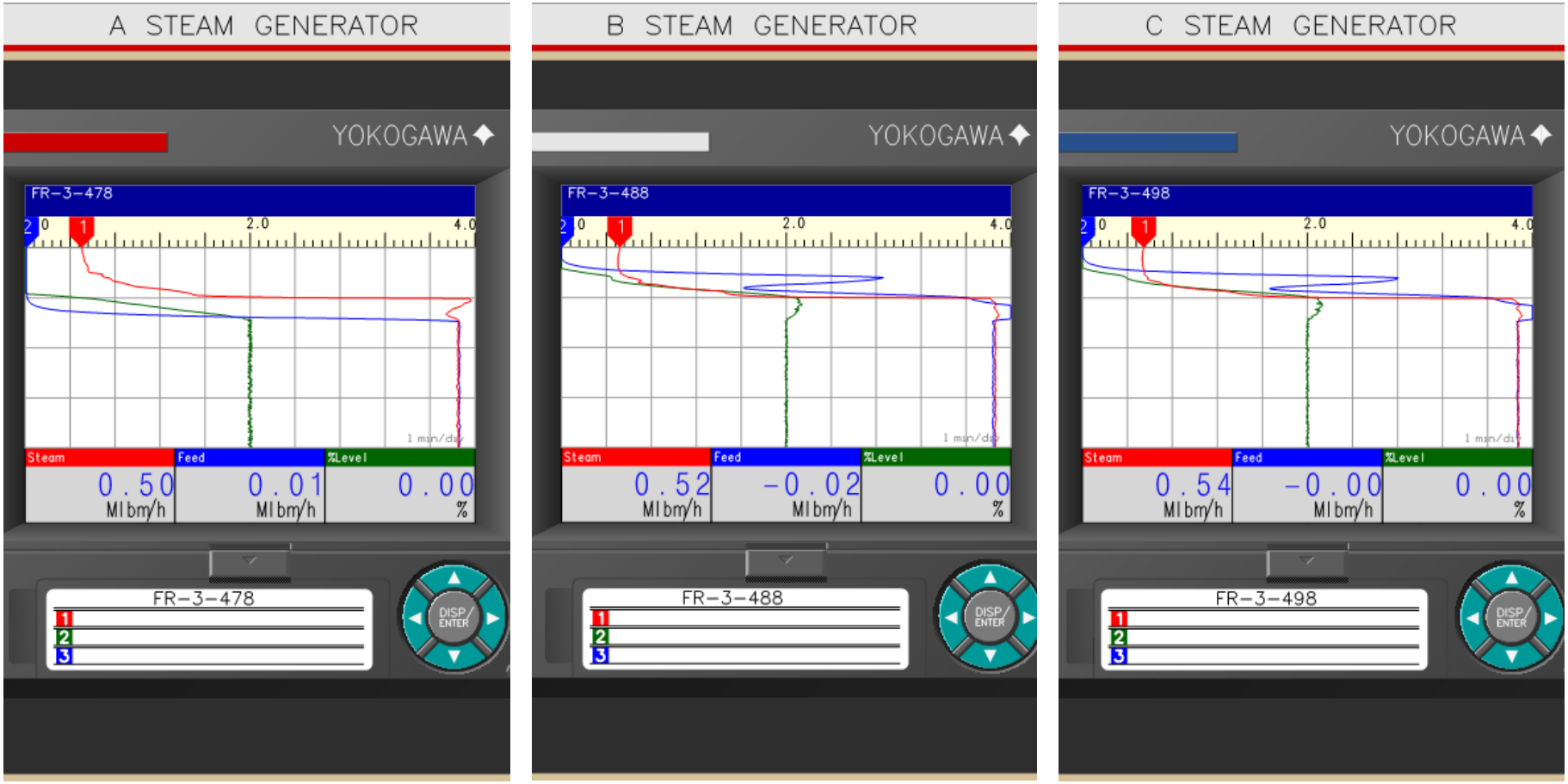
In accordance with 0-ADM-211, Emergency and Off-Normal Operating Procedure Usage, which one of the following identifies the use of 3-ONOP-003.7, Loss of 120 Volt Vital Bus 3P07, while performing 3-EOP-E-3?

A. Continue in 3-EOP-E-3. Perform actions of 3-ONOP-003.7 only after exiting the EOP network.

B. Stabilize the unit in 3-EOP-E-3. Exit 3-EOP-E-3. Perform actions to restore 3P07 in accordance with 3-ONOP-003.7 and then re-enter 3-EOP-E-3 to continue event mitigation.

C. At US discretion, perform steps of 3-ONOP-003.7 concurrently with the performance of 3-EOP-E-3 to mitigate both events.

D. The US must obtain Assistant Operations Manager authorization prior to implementation of any part of 3-ONOP-003.7 while 3-EOP-E-3 is in effect.


A STEAM GENERATOR
YOKOGAWA
FR-3-478
0
1
2.0
4.0
1 min/div
Steam
Feed
%Level
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Mlbm/h
0.01
Mlbm/h
0.00
%
FR-3-478
1
2
3
DISP/ENTER
B STEAM GENERATOR
YOKOGAWA
FR-3-488
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2.0
4.0
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Steam
Feed
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Mlbm/h
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0.00
%
FR-3-488
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2
3
DISP/ENTER
C STEAM GENERATOR
YOKOGAWA
FR-3-498
0
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2.0
4.0
1 min/div
Steam
Feed
%Level
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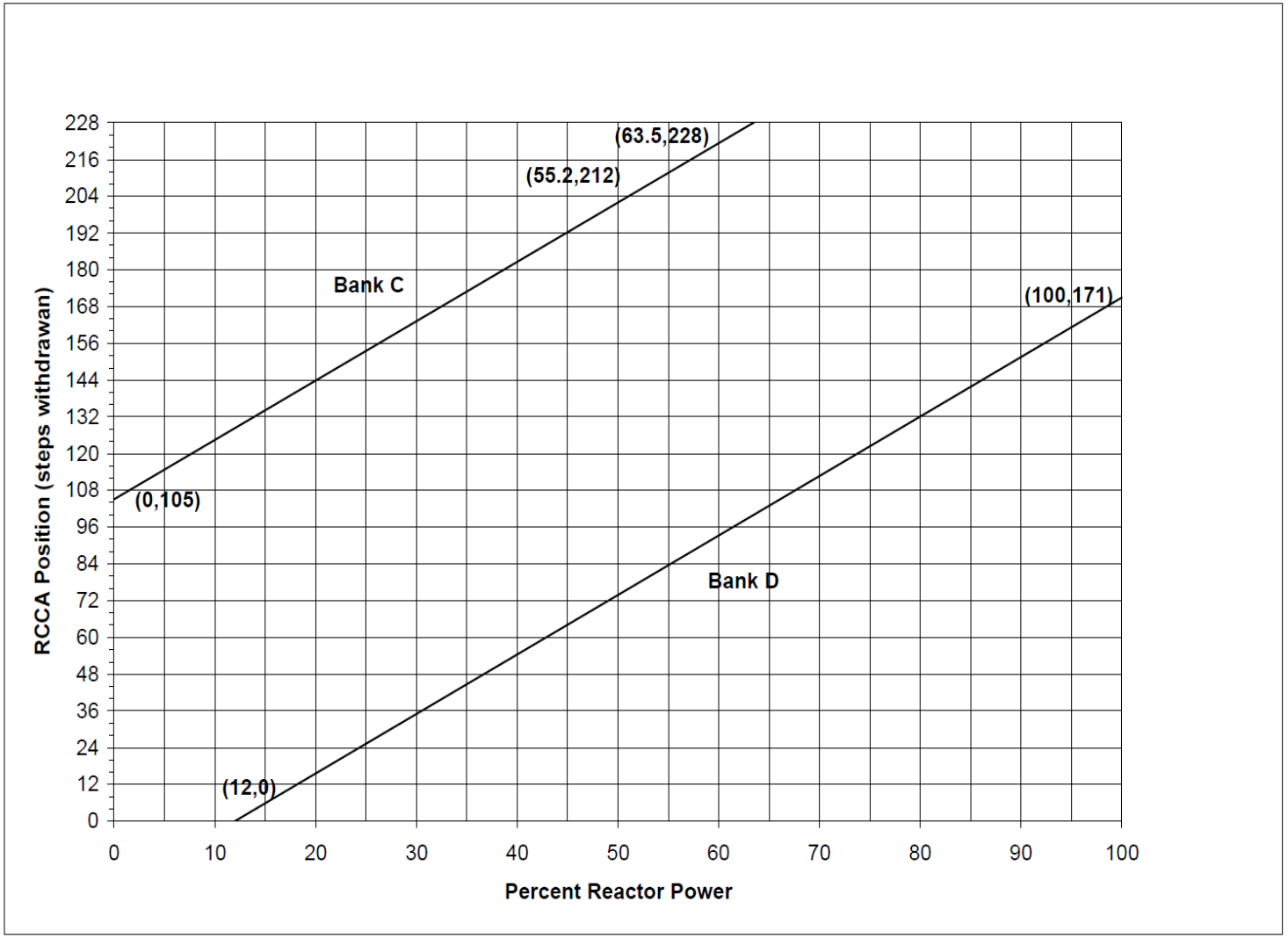
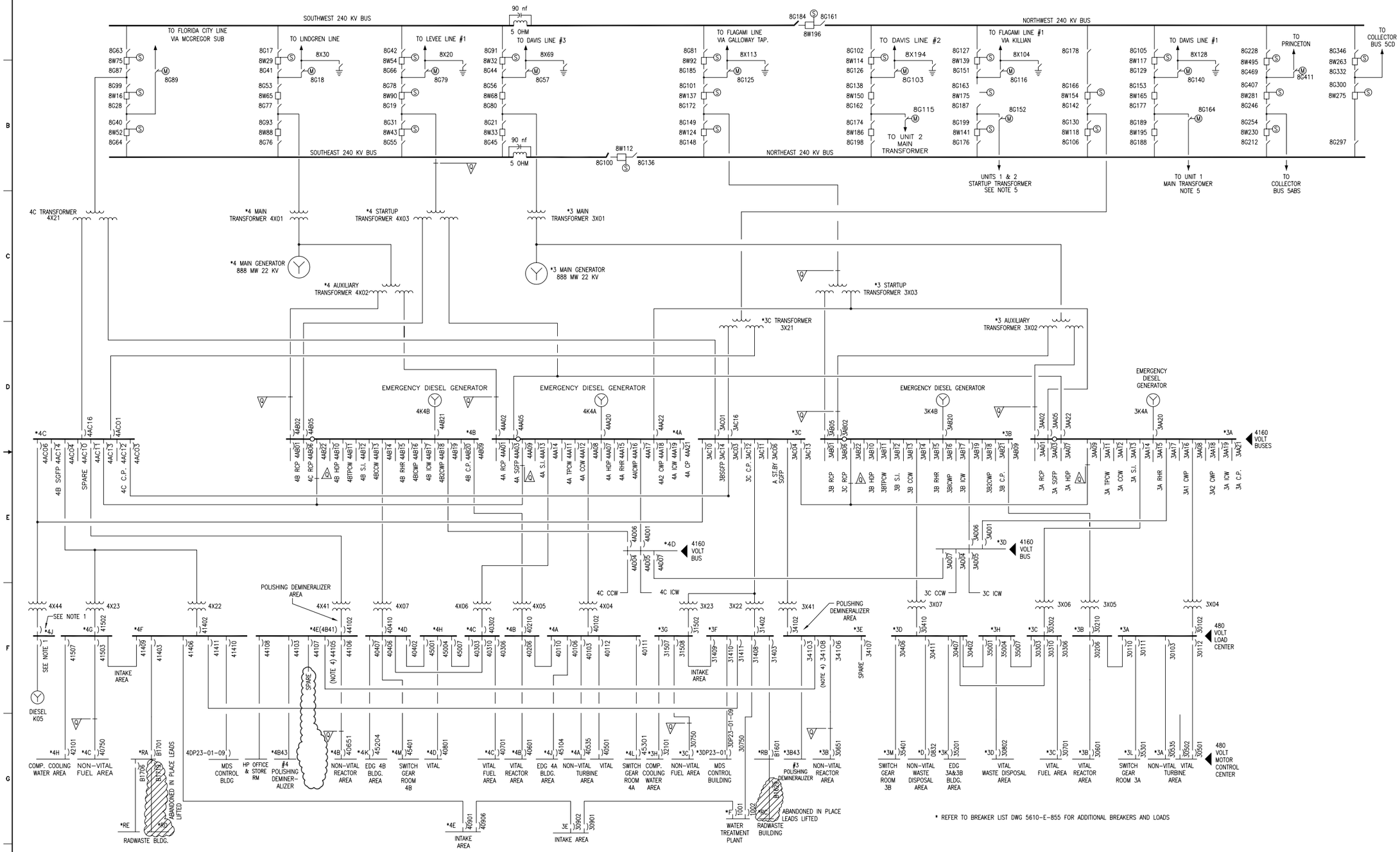
Mlbm/h
-0.00
Mlbm/h
0.00
%
FR-3-498
1
2
3
DISP/ENTER

FIGURE A3

Turkey Point Unit 3 Cycle 28 Rod Insertion Limits vs Thermal Power
ARO = 228 Steps Withdrawn, Overlap = 100 Steps



DWG NO 5610-T-E-1591
SHT 1



* REFER TO BREAKER LIST DWG 5610-E-855 FOR ADDITIONAL BREAKERS AND LOADS

NOTES:

1) SEE 5610-E-1522 FOR ADDITIONAL INFORMATION
2) DELETED
3) DELETED
4) 480V TIE-LINE BETWEEN LOAD CENTERS 3E & 4E IS A TEMPORARY MAINTENANCE TIE AND ITS USE IS CONTROLLED BY PROCEDURE 3/4-NOP-006.
5) NOTIFY NPS PRIOR TO COMMENCING WORK ON THIS SYSTEM.

REV	DATE	REVISION	BY	CH	APP	APP
71	3/19/13	ISSUED AS-BUILT PER EC 249331 (ITOP 13-03-150).	RH	BB	–	ERJ
70	02-09-12	ISSUED AS-BUILT PER EC 242437 (ITOP 13-02-034)	RH	RV	–	PMB
69	12-08-12	ISSUED AS-BUILT PER EC-DCR 277655.	RV	BB	–	MA
68	08-30-12	ISSUED AS-BUILT PER EC 246916 (PC/M 06-167) .	RV	BB	DA	JD
67	08-23-12	ISSUED AS-BUILT PER EC 275192	RV	RH	–	PMB
66	06-11-12	ISSUED AS-BUILT PER EC 275192	RV	BB	–	PDS
65	4/25/12	ISSUED AS-BUILT PER EC 249330 AND INCORP. CRN-038.	RV	BB	–	AH
64	2/23/12	ISSUED AS-BUILT PER EC 249330. (PARTIAL – BREAKER ONLY)	RV	RH	–	LG
63	1/27/12	ISSUED AS-BUILT PER EC 246989(PC/M 09-105).	RV	RH	–	LG
74	06-13-13	ISSUED AS-BUILT PER EC 246917 (ITOP 13-06-052).	RV	RH	–	PMB
73	04-02-13	ISSUED AS-BUILT PER EC-DCR 278017.	RV	BB	–	DBJ
72	3-29-13	ISSUED AS-BUILT PER EC 246904 & INCORP. CRN-017 (ITOP 13-03-168).	RH	RV	–	TK
REV	DATE	REVISION	BY	CH	APP	APP

SAFETY RELATED

POD

TURKEY POINT NUCLEAR UNITS 3 & 4

FLORIDA POWER & LIGHT

OPERATING DIAGRAM ELECTRICAL DISTRIBUTION

DRAWING NUMBER 5610-T-E-1591

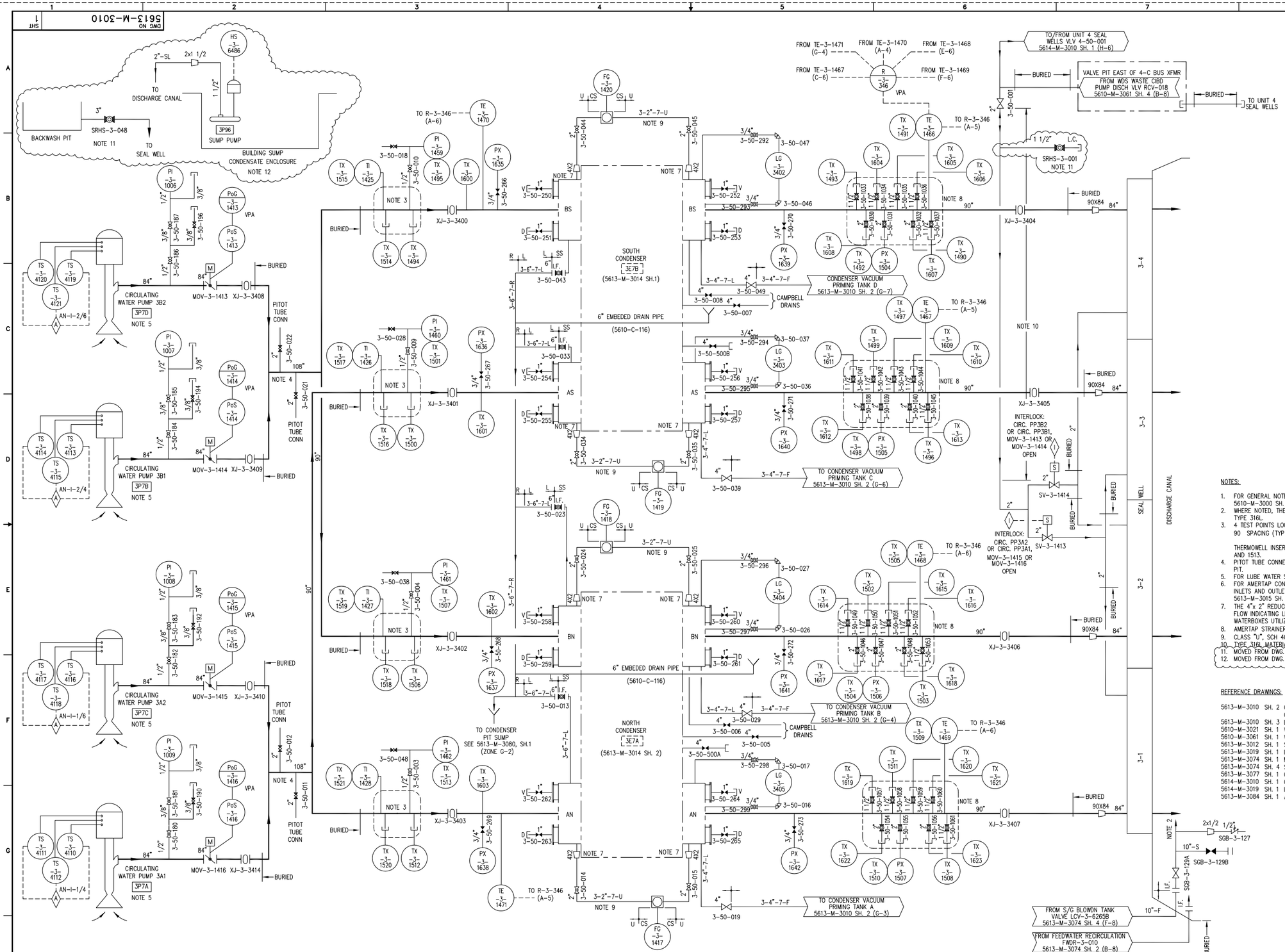
SYS –

REV 74

SHEET: 1

5610-T-E-1591_SH 1.DWG

DWG NO 5613-M-3010 SHT 1



NOTES:

1. FOR GENERAL NOTES & LEGEND SEE DWG NO. NO. 5610-M-3000 SH. 2.
2. WHERE NOTED, THE PIPING MATERIAL IS SCHEDULE 40, TYPE 316L.
3. 4 TEST POINTS LOCATED RADIALLY AROUND PIPE AT APPROX. 90 SPACING (TYP 8 PLCS).

   THERMOWELL INSERT REMOVED FROM TX-3-1495, 1501, 1507, AND 1513.
4. PITOT TUBE CONNECTIONS LOCATED IN PITOT TUBE ACCESS PIT.
5. FOR LUBE WATER SYSTEM SEE 5613-M-3010 SH. 3
6. FOR AMERTAP CONDENSER TUBE CLEANING CONNECTIONS ON INLETS AND OUTLETS TO EACH CONDENSER SECTION SEE 5613-M-3015 SH. 1.
7. THE 4"x 2" REDUCING FLANGES LOCATED IN THE PRIMING FLOW INDICATING LINES BETWEEN THE INLET AND OUTLET WATERBOXES UTILIZE SLIP-ON TYPE FLANGES.
8. AMERTAP STRAINER SECTION (TYP 4 PLCS).
9. CLASS "U", SCH 40 INSTALLED.
10. TYPE 316L MATERIAL USED IN LIEU OF 304L.
11. MOVED FROM DWG. 5613-M-3077, SH. 3.
12. MOVED FROM DWG. 5613-M-3077, SH. 2

REFERENCE DRAWINGS:

5613-M-3010 SH. 2 CIRCULATING WATER SYSTEM – CONDENSER WATER BOX PRIMING
5613-M-3010 SH. 3 LUBE WATER TO CIRC WATER PUMPS
5610-M-3021 SH. 1 WATER TREATMENT PLANT
5610-M-3061 SH. 1 WASTE DISPOSAL SYSTEM–LIQUIDS
5613-M-3012 SH. 1 SERVICE WATER SYSTEM
5613-M-3019 SH. 1 INTAKE COOLING WATER SYSTEM
5613-M-3074 SH. 1 MAIN FEEDWATER
5613-M-3074 SH. 4 STEAM GENERATOR BLOWDOWN RECOVERY
5613-M-3077 SH. 1 CONDENSATE POLISHING
5614-M-3010 SH. 1 CIRCULATION WATER SYSTEM
5614-M-3019 SH. 1 INTAKE COOLING WATER SYSTEM
5613-M-3084 SH. 1 AUXILIARY STEAM SYSTEM

POD

THIS DRAWING SUPERSEDES DRAWINGS:

5610-M-5	SH. 1	REV. 41
5610-M-342	SH. 1	REV. 9
5610-M-364		REV. 2
5610-M-365	SH. 1	REV. 6

NOTE: THIS DWG IS MADE FROM:
FPL POD 5610-T-E-4065 SH. 2 REV. 58

REV	DATE	REVISION	BY	CH	APP	APP
28	4-25-14	ISSUED AS-BUILT PER EC 242495.	RH	RV	–	RSV
27	4-17-14	ISSUED AS-BUILT PER EC 242051 (242051-FINAL).	RH	AFG	–	EB
26	2-12-13	ISSUED AS-BUILT PER EC 249149 (ITOP 13-02-065).	RH	DS	–	RSV
25	8-29-12	ISSUED AS-BUILT PER EC 246914 (PC/M 08-165) INCORP. CRN'S-028, 029 & 043	RV/RH	BB	DA	JD
24	7-19-12	ISSUED AS-BUILT PER EC 246914. (PARTIAL)	RH	RV	–	JL
23	12-17-11	ISSUED AS-BUILT PER EC-DCR 274982.	RH	RV	–	BB
22	12-1-10	ISSUED AS-BUILT PER EC 242487 (PC/M 09-041).	RH	RV	RN	TS
21	11-02-10	ISSUED AS-BUILT PER EC 242489.	RV	CBW	PJV	MG
20	8-13-09	ISSUED AS-BUILT PER CRN's M-12628 & M-12629 (PC/M 09-005).	RV	BB	–	PJV
19	4-24-09	ISSUED AS-BUILT PER PC/M 09-043 (PARTIAL) & INCORP. CRN M-12706.	RH	BB	ASD	PRB
0	1-29-91	THIS DWG CREATED PER THE P & ID RECONSTITUTION PROJECT SCOPE	TAM	MD	SVM RAS	LRB
——	——	AND ISSUED INTO THE FPL DWG SYSTEM PER DCR-TPM-90-541.	——	——	——	——

FPL

TURKEY POINT NUCLEAR UNIT 3

P & I D

CIRCULATING WATER SYSTEM

STONE & WEBSTER ENGINEERING CORP. FT. LAUDERDALE, FLORIDA

DRAWING NUMBER 5613-M-3010

SYS 010

REV 28

SHEET 1

8 5613-M-3010_SH_1.DWG

INSTRUMENTATION

3/4.3.3 MONITORING INSTRUMENTATION

RADIATION MONITORING FOR PLANT OPERATIONS

LIMITING CONDITION FOR OPERATION

3.3.3.1 The radiation monitoring instrumentation channels for plant operations shown in Table 3.3-4 shall be OPERABLE with their Alarm/Trip Setpoints within the specified limits.

APPLICABILITY: As shown in Table 3.3-4.

ACTION:

a. With a radiation monitoring channel Alarm/Trip Setpoint for plant operations exceeding the value shown in Table 3.3-4, adjust the Setpoint to within the limit within 4 hours or declare the channel inoperable.

b. With one or more radiation monitoring channels for plant operations inoperable, take the ACTION shown in Table 3.3-4.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.1 Each radiation monitoring instrumentation channel for plant operations shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and ANALOG CHANNEL OPERATIONAL TEST for the MODES and at the frequencies shown in Table 4.3-3.

TABLE 3.3-4

RADIATION MONITORING INSTRUMENTATION FOR PLANT OPERATIONS

FUNCTIONAL UNIT	CHANNELS TO TRIP/ALARM	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	ACTION
1. Containment					
a. Containment Atmosphere Radioactivity-High (Particulate or Gaseous (See Note 1.))	1	1*	All*	Particulate $\leq 6.1 \times 10^5$ CPM Gaseous See Note 2.	26 for MODES 1, 2, 3, 4 or 27 for MODES 5 and 6
b. RCS Leakage Detection Particulate Radio-activity or Gaseous Radioactivity	N.A.	1	1, 2, 3, 4	N.A.	26
2. Spent Fuel Storage Pool Areas					
a. Unit 3 Radioactivity – High Gaseous	1	1	**	$< 5.5 \times 10^{-2}$ $\frac{\mu Ci}{cc}$	28
b. Unit 4 Radioactivity – High Gaseous#	1	1	**	$< 2.8 \times 10^{-2}$ $\frac{\mu Ci}{cc}$ (SPING) or $< 1.0 \times 10^6$ CPM (PRMS)	28

TABLE 3.3-4 (Continued)
TABLE NOTATIONS

* During CORE ALTERATIONS or movement of irradiated fuel within the containment comply with Specification 3/4.9.13.

** With irradiated fuel in the spent fuel pits.

Unit 4 Spent Fuel Pool Area is monitored by Plant Vent radioactivity instrumentation.

Note 1 Either the particulate or gaseous channel in the OPERABLE status will satisfy this LCO.

Note 2 Containment Gaseous Monitor Setpoint = $\frac{(3.2 \times 10^4)}{(F)}$ CPM,

Where $F = \frac{\text{Actual Purge Flow}}{\text{Design Purge Flow (35,000 CFM)}}$

Setpoint may vary according to current plant conditions provided that the release rate does not exceed allowable limits provided in the Offsite Dose Calculation Manual.

ACTION STATEMENTS

ACTION 26 - In MODES 1 thru 4: With both the Particulate and Gaseous Radioactivity Monitoring Systems inoperable, operation may continue for up to 7 days provided:

1) A Containment sump level monitoring system is OPERABLE,

2) Appropriate grab samples are obtained and analyzed at least once per 24 hours,

3) A Reactor Coolant System water inventory balance is performed at least once per 8*** hours except when operating in shutdown cooling mode, and

4) Containment Purge, Exhaust and Instrument Air Bleed Valves are maintained closed.****

Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours (ACTION 27 applies in MODES 5 and 6).

*** Not required to be performed until 12 hours after establishment of steady state operation.

**** Instrument Air Bleed Valves may be opened intermittently under administrative controls.

TABLE 3.3-4 (Continued)

ACTION STATEMENTS (Continued)

ACTION 27 - In MODES 5 or 6 (except during CORE ALTERATION or movement of irradiated fuel within the containment): With the number of OPERABLE Channels less than the Minimum Channels OPERABLE requirement perform the following:

1) Obtain and analyze appropriate grab samples at least once per 24 hours, and

2) Monitor containment atmosphere with area radiation monitors.

Otherwise, isolate all penetrations that provide direct access from the containment atmosphere to the outside atmosphere.

During CORE ALTERATION or movement of irradiated fuel within the containment: With the number of OPERABLE Channels less than the Minimum Channels OPERABLE requirements, comply with ACTION statement requirements of Specification 3.9.9 and 3.9.13.

ACTION 28 - With the number of OPERABLE channels less than the Minimum Channels OPERABLE requirement, immediately suspend operations in the Spent Fuel Pool area involving spent fuel manipulations.

TABLE 4.3-3
RADIATION MONITORING INSTRUMENTATION FOR PLANT
OPERATIONS SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Containment				
a. Containment Atmosphere Radioactivity--High	SFCP	SFCP	SFCP	All
b. RCS Leakage Detection				
1) Particulate Radio-activity	SFCP	SFCP	SFCP	1, 2, 3, 4
2) Gaseous Radioactivity	SFCP	SFCP	SFCP	1, 2, 3, 4
2. Spent Fuel Pool Areas				
a. Unit 3 Radioactivity--High Gaseous	SFCP	SFCP	SFCP	*
b. Unit 4 (Plant Vent) Radioactivity--High Gaseous# (SPING and PRMS)	SFCP	SFCP	SFCP	*

TABLE NOTATIONS

* With irradiated fuel in the fuel storage pool areas.

Unit 4 Spent Fuel Pool Area is monitored by Plant Vent radioactivity instrumentation.

INSTRUMENTATION

MOVABLE INCORE DETECTORS

LIMITING CONDITION FOR OPERATION

3.3.3.2 The Movable Incore Detection System shall be OPERABLE with:

a. At least 16 detector thimbles when used for recalibration and check of the Excore Neutron Flux Detection System and monitoring the QUANDRANT POWER TILT RATIO*****, and at least 38 detector thimbles when used for monitoring $F^N_{\Delta H}$, $F_Q(Z)$ and $F_{xy}(Z)$.

b. A minimum of two detector thimbles per core quadrant, and

c. Sufficient movable detectors, drive, and readout equipment to map these thimbles.

APPLICABLITY: When the Movable Incore Detection System is used for:

a. Recalibration of the Excore Neutron Flux Detection System, or

b. Monitoring the QUADRANT POWER TILT RATIO*****, or

c. Measurement of $F^N_{\Delta H}$, $F_Q(Z)$ and $F_{xy}(Z)$.

ACTION:

With the Movable Incore Detection System inoperable, do not use the system for the above applicable monitoring or calibration functions. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.2 The Movable Incore Detection System shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program by normalizing each detector output when required for:

a. Recalibration of the Excore Neutron Flux Detection System, or

b. Monitoring the QUADRANT POWER TILT RATIO*****, or

c. Measurement of $F^N_{\Delta H}$, $F_Q(Z)$ and $F_{xy}(Z)$.

***** Exception to the 16 detector thimble requirement of monitoring the QUADRANT POWER TILT RATIO is acceptable when performing Specification 4.2.4.2 using two sets of four symmetric thimbles.

INSTRUMENTATION

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.3 The accident monitoring instrumentation channels shown in Table 3.3-5 shall be OPERABLE.

APPLICABILITY: As shown in Table 3.3-5.

ACTION:

a. As shown in Table 3.3-5.

b. The provisions of Specification 3.0.4 are not applicable to ACTIONS in Table 3.3-5 that require a shutdown.

c. Separate Action entry is allowed for each Instrument.

SURVEILLANCE REQUIREMENTS

4.3.3.3 Each accident monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION at the frequencies shown in Table 4.3-4.

TABLE 3.3-5

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE	APPLI-CABLE MODES	ACTIONS
1. Containment Pressure (Wide Range)	2	1	1, 2, 3	31, 32
2. Containment Pressure (Narrow Range)	2	1	1, 2, 3	36
3. Reactor Coolant Outlet Temperature T_{HOT} (Wide Range)	2-2 Detectors per Channel	1-2 Detectors per Channel	1, 2, 3	31, 32
4. Reactor Coolant Inlet Temperature T_{COLD} (Wide Range)	2-2 Detectors per Channel	1-2 Detectors per Channel	1, 2, 3	31, 32
5. Reactor Coolant Pressure – Wide Range	2	1	1, 2, 3	31, 32
6. Pressurizer Water Level	2	1	1, 2, 3	31, 32
7. Auxiliary Feedwater Flow Rate	2/steam generator	1/steam generator	1, 2, 3	31, 32
8. Reactor Coolant System Subcooling Margin Monitor	2(2)	1(2)	1, 2, 3	31, 32
9. PORV Position Indicator (Primary Detector)	1/valve	1/valve	1, 2, 3	33
10. PORV Block Valve Position Indicator	1/valve	1/valve	1, 2, 3	33
11. Safety Valve Position Indicator (Primary Detector)	1/valve	1/valve	1, 2, 3	32
12. Containment Water Level (Narrow Range)	2	1	1, 2, 3	36
13. Containment Water Level (Wide Range)	2	1	1, 2, 3	31, 32

TABLE 3.3-5 (Continued)

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE	APPLI-CABLE MODES	ACTIONS
14. In Core Thermocouples (Core Exit Thermocouples)	4/core quadrant	2/core quadrant	1, 2, 3	31, 32
15. Containment High Range Area Radiation	2	1	1, 2, 3	34
16. Reactor Vessel Level Monitoring System	2(1)	1(1)	1, 2, 3	37, 38
17. Neutron Flux, Backup NIS (Wide Range)	2	1	1, 2, 3	31, 32
18. DELETED				
19. High Range-Noble Gas Effluent Monitors				
a. Plant Vent Exhaust	1	1	ALL	34
b. Unit 3-Spent Fuel Pit Exhaust	1	1	ALL	34
c. Condenser Air Ejectors	1	1	1, 2, 3	34
20. RWST Water Level	2	1	1, 2, 3	31, 32
21. Steam Generator Water Level (Narrow Range)	2/stm. Gen.	1/stm. Gen.	1, 2, 3	31, 32
22. Containment Isolation Valve Position Indication*	1/valve	1/valve	1, 2, 3	39

TABLE NOTATIONS

1. A channel is eight sensors in a probe. A channel is OPERABLE if a minimum of four sensors are OPERABLE.

2. Inputs to this instrument are from instrument items 3, 4, 5 and 14 of this Table.

* Applicable for containment isolation valve position indication designated as post-accident monitoring instrumentation (containment isolation valves which receive containment isolation Phase A, Phase B, or containment ventilation isolation signals).

TABLE 3.3-5 (Continued)

ACTION STATEMENTS

ACTION 31 With the number of OPERABLE accident monitoring instrumentation channel(s) less than the Total Number of Channels either restore the inoperable channel(s) to OPERABLE status within 30 days, or submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 14 days outlining the action taken, the cause of the inoperability, and the plans and schedule for restoring the system to OPERABLE status.

ACTION 32 With the number of OPERABLE accident monitoring instrumentation channels less than the Minimum Channels OPERABLE, either restore the inoperable channel(s) to OPERABLE status within 7 days, or be in at least HOT STANDBY within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

ACTION 33 Close the associated block valve and open its circuit breaker.

ACTION 34 With the number of OPERABLE Channels less than required by the Minimum Channels OPERABLE requirements, initiate the preplanned alternate method of monitoring the appropriate parameter(s), within 72 hours, and:

1) Either restore the inoperable channel(s) to OPERABLE status within 7 days of the event, or

2) Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 14 days outlining the action taken, the cause of the inoperability, and the plans and schedule for restoring the system to OPERABLE status.

ACTION 35 DELETED

ACTION 36 With the number of OPERABLE accident monitoring instrumentation channels less than the Minimum Channel OPERABLE, either restore the inoperable channel to OPERABLE status within 30 days, or be in at least HOT STANDBY within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

ACTION 37 With the number of OPERABLE channels one less than the Total Number of Channels, restore the system to OPERABLE status within 30 days. If repairs are not feasible without shutting down, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 14 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

TABLE 3.3-5 (Continued)

ACTION STATEMENTS

ACTION 38 With the number of OPERABLE channels less than the Minimum Channels OPERABLE requirements, restore the inoperable channel(s) to OPERABLE status within 7 days. If repairs are not feasible without shutting down:

1. Initiate an alternate method of monitoring the reactor vessel inventory; and

2. Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 14 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status; and

3. Restore at least one channel to OPERABLE status at the next scheduled refueling.

ACTION 39 With the number of OPERABLE channels less than the Minimum Channels OPERABLE requirement, verify position by an alternate means (e.g. administrative controls, ERDADS, alternate position indication, or visual observation) within 2 hours, and restore the inoperable channel(s) within 7 days, or comply with the provisions of Specification 3.6.4 for an inoperable containment isolation valve.

TABLE 4.3-4

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT		CHANNEL CHECK	CHANNEL CALIBRATION
1.	Containment Pressure (Wide Range)	SFCP	SFCP
2.	Containment Pressure (Narrow Range)	SFCP	SFCP
3.	Reactor Coolant Outlet Temperature - T_{HOT} (Wide Range)	SFCP	SFCP
4.	Reactor Coolant Inlet Temperature - T_{COLD} (Wide Range)	SFCP	SFCP
5.	Reactor Coolant Pressure - Wide Range	SFCP	SFCP
6.	Pressurizer Water Level	SFCP	SFCP
7.	Auxiliary Feedwater Flow Rate	SFCP	SFCP
8.	Reactor Coolant System Subcooling Margin Monitor	SFCP	SFCP
9.	PORV Position Indicator (Primary Detector)	SFCP	SFCP
10.	PORV Block Valve Position Indicator	SFCP	SFCP
11.	Safety Valve Position Indicator (Primary Detector)	SFCP	SFCP
12.	Containment Water Level (Narrow Range)	SFCP	SFCP
13.	Containment Water Level (Wide Range)	SFCP	SFCP
14.	In Core Thermocouples (Core Exit Thermocouples)	SFCP	SFCP
15.	Containment - High Range Area Radiation Monitor	SFCP	SFCP*
16.	Reactor Vessel Level Monitoring System	SFCP	SFCP
17.	Neutron Flux, Backup NIS (Wide Range)	SFCP	SFCP
18.	DELETED		
19.	High Range - Noble Gas Effluent Monitors		
	a. Plant Vent Exhaust	SFCP	SFCP
	b. Unit 3 - Spent Fuel Pit Exhaust	SFCP	SFCP
	c. Condenser Air Ejectors	SFCP	SFCP
20.	RWST Water Level	SFCP	SFCP
21.	Steam Generator Water Level (Narrow Range)	SFCP	SFCP
22.	Containment Isolation Valve Position Indication	SFCP	SFCP

*Acceptable criteria for calibration are provided in Table II.F.1-3 of NUREG-0737.

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained.*

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

Without primary CONTAINMENT INTEGRITY, restore CONTAINMENT INTEGRITY within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.1 CONTAINMENT INTEGRITY shall be demonstrated:

a. In accordance with the Surveillance Frequency Control Program by verifying that all penetrations** not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in their closed positions;

b. By verifying that each containment air lock is in compliance with the requirements of Specification 3.6.1.3.

* Exception may be taken under Administrative Controls for opening of valves and airlocks necessary to perform surveillance, testing requirements and/or corrective maintenance. In addition, Specification 3.6.4 shall be complied with.

** Except valves, blind flanges, and deactivated automatic valves which are located inside the containment and are locked, sealed or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except that such verification need not be performed more often than once per 92 days.

CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Containment leakage rates shall be limited in accordance with the Containment Leakage Rate Testing Program.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With the measured overall integrated containment leakage rate exceeding 1.0 L_a within one hour, initiate action to be in HOT STANDBY within the next 6 hours, and COLD SHUTDOWN within the following 30 hours. Restore the overall integrated leakage rate to less than 0.75 L_a and the combined leakage rate for all penetrations subject to Type B and C tests to less than 0.60 L_a prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.2 The containment leakage rates shall be demonstrated at the required test schedule and shall be determined in conformance with the criteria specified in the Containment Leakage Rate Testing Program.

CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each containment air lock shall be OPERABLE with:

a. Both doors closed except when the air lock is being used for normal transit entry and exit through the containment, or during the performance of containment air lock surveillance and/or testing requirements, then at least one air lock door shall be closed, and

b. An overall air lock leakage rate in accordance with the Containment Leakage Rate Testing Program.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With one containment air lock door inoperable:

1. Maintain at least the OPERABLE air lock door closed and either restore the inoperable air lock door to OPERABLE status within 24 hours or lock the OPERABLE air lock door closed;

2. Operation may then continue until performance of the next required overall air lock leakage test provided that the OPERABLE air lock door is verified to be locked closed at least once per 31 days;

3. Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With the containment air lock inoperable, except as the result of an inoperable air lock door, maintain at least one air lock door closed; restore the inoperable air lock to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

a. Following each closing, at the frequency specified in the Containment Leakage Rate Testing Program, by verifying that the seals have not been damaged and have seated properly by vacuum testing the volume between the door seals in accordance with approved plant procedures.

b. By conducting overall air lock leakage tests in accordance with the Containment Leakage Rate Testing Program.

c. In accordance with the Surveillance Frequency Control Program by verifying that only one door in each air lock can be opened at a time.

COLD CONDITIONS TABLE
(RCS ≤ 200°F)

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COLD

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COLD

		General Emergency	Site Area Emergency	Alert	Unusual Event
R – ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENT	RADIOLOGICAL EFFLUENT	**RG1** Off-site Dose Resulting from an Actual or IMMINENT Release of Gaseous Radioactivity Greater Than 1000 mrem TEDE or 5000 mrem Thyroid CDE for the Actual or Projected Duration of the Release Using Actual Meteorology. Pg. 6	**RS1** Off-site Dose Resulting from an Actual or IMMINENT Release of Gaseous Radioactivity Greater Than 100 mrem TEDE or 500 mrem Thyroid CDE for the Actual or Projected Duration of the Release. Pg. 6	**RA1** Any Release of Gaseous or Liquid Radioactivity to the Environment Greater Than 200 Times the ODCM for 15 Minutes or Longer. Pg. 6	**RU1** Any Release of Gaseous or Liquid Radioactivity to the Environment Greater Than 2 Times the ODCM for 60 Minutes or Longer. Pg. 6
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					CU2 UNPLANNED loss of RCS inventory. Pg. 12
	AC POWER			**CA3** Loss of all Off-site and all On-site AC power to emergency busses for 15 minutes or longer. Pg. 12	**CU3** AC power capability to emergency busses reduced to a single power source for 15 minutes or longer such that any additional single failure would result in unit blackout. Pg. 12
	DECAY HEAT			**CA4** Inability to Maintain Plant in Cold Shutdown. Pg. 14	**CU4** UNPLANNED Loss of Decay Heat Removal Capability with Irradiated Fuel in the Reactor Vessel Pg. 14
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	CRITICALITY				**CU8** Inadvertent Criticality. Pg. 14
	COMM.				**CU6** Loss of All On-site or Off-site Communications Capabilities. Pg. 16

		General Emergency	Site Area Emergency	Alert	Unusual Event
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EPIP Reference		Complete actions listed in EPIP-20101 Section 5.8	Complete actions listed in EPIP-20101 Section 5.7	Complete actions listed in EPIP-20101 Section 5.6	Complete actions listed in EPIP-20101 Section 5.5

INITIATING CONDITIONS MATRIX

COLD

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COLD

RECOGNITION CATEGORY R

ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENT

Rec. Cat.	General Emergency	Site Area Emergency	Alert	Unusual Event
R – ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENT	**RG1 – Basis:** This IC addresses radioactivity releases that result in doses at or beyond the site boundary that exceed the EPA Protective Action Guides (PAGs). Public protective actions will be necessary. Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public and likely involve fuel damage. <u>Threshold Values</u> The monitor list includes effluent monitors on all potential release pathways. Since dose assessment is based on actual meteorology, whereas the monitor reading Threshold is not, the results from these assessments may indicate that the classification is not warranted, or may indicate that a higher classification is warranted. For this reason, emergency implementing procedures call for the timely performance of dose assessments using actual meteorology and release information. If the results of these dose assessments are available when the classification is made (e.g., initiated at a lower classification level), the dose assessment results override the monitor reading EAL.	**RS1 – Basis:** This IC addresses radioactivity releases that result in doses at or beyond the site boundary that exceed 10% of the EPA Protective Action Guides (PAGs). Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public. <u>Threshold Values</u> The monitor list in Threshold #1 includes effluent monitors on all potential release pathways. Since dose assessment is based on actual meteorology, whereas the monitor reading Threshold is not, the results from these assessments may indicate that the classification is not warranted, or may indicate that a higher classification is warranted. For this reason, emergency implementing procedures call for the timely performance of dose assessments using actual meteorology and release information. If the results of these dose assessments are available when the classification is made (e.g., initiated at a lower classification level), the dose assessment results override the monitor reading EAL.	**RA1 – Basis:** The Emergency Coordinator should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time. This IC addresses an actual or substantial potential decrease in the level of safety of the plant as indicated by a radiological release that exceeds regulatory commitments for an extended period of time. Nuclear power plants incorporate features intended to control the release of radioactive effluents to the environment. Further, there are administrative controls established to prevent unintentional releases, or control and monitor intentional releases. These controls are located in the Off-site Dose Calculation Manual (ODCM). The occurrence of extended, uncontrolled radioactive releases to the environment is indicative of a degradation in these features and/or controls. The ODCM multiples are specified in RU1 and RA1 only to distinguish between non-emergency conditions, and from each other. While these multiples obviously correspond to an off-site dose or dose rate, the emphasis in classifying these events is the degradation in the level of safety of the plant, not the magnitude of the associated dose or dose rate. This Threshold includes any release for which a radioactivity discharge permit was not prepared, or a release that exceeds the conditions (e.g., minimum dilution flow, maximum discharge flow, alarm setpoints, etc.) on the applicable permit. <u>Threshold #1</u> This Threshold is intended for effluent monitoring on non-routine release pathways for which a discharge permit would not normally be prepared. The threshold values of this EAL are to be 200 times the ODCM limit or 100 times the threshold values used in RU1. This is true except for the thresholds given for the liquid release pathways (PRMS-R-18 and PRMS-3[4]-R-19). On both PRMS, the calculated values exceeded the range of the monitors. Therefore, the values used in the EAL are 95% of full scale. This lesser threshold is still indicative of a loss of control of radioactive material well in excess of that constituting an Unusual Event. <u>Threshold #2</u> This Threshold addresses radioactivity releases, that for whatever reason, cause effluent radiation monitor readings to exceed the threshold identified in the IC established by the radioactivity discharge permit. This value may be associated with a planned batch release, or a continuous release path. <u>Threshold #3</u> This Threshold addresses uncontrolled releases that are detected by sample analyses, particularly on unmonitored pathways, e.g., spills of radioactive liquids into storm drains, heat exchanger leakage in river water systems, etc. The underlying basis of this Threshold involves the degradation in the level of safety of the plant implied by the uncontrolled release.	**RU1 – Basis:** The Emergency Coordinator should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time. This IC addresses a potential decrease in the level of safety of the plant as indicated by a radiological release that exceeds regulatory commitments for an extended period of time. Nuclear power plants incorporate features intended to control the release of radioactive effluents to the environment. Further, there are administrative controls established to prevent unintentional releases, or control and monitor intentional releases. These controls are located in the Off-site Dose Calculation Manual (ODCM). The occurrence of extended, uncontrolled radioactive releases to the environment is indicative of a degradation in these features and/or controls. The ODCM multiples are specified in RU1 and RA1 only to distinguish between non-emergency conditions, and from each other. While these multiples obviously correspond to an off-site dose or dose rate, the emphasis in classifying these events is the degradation in the level of safety of the plant, not the magnitude of the associated dose or dose rate. Releases should not be prorated or averaged. For example, a release exceeding 4x ODCM for 30 minutes does not meet the threshold. This Threshold includes any release for which a radioactivity discharge permit was not prepared, or a release that exceeds the conditions (e.g., minimum dilution flow, maximum discharge flow, alarm setpoints, etc.) on the applicable permit. <u>Threshold #1</u> This Threshold addresses radioactivity releases, that for whatever reason, cause effluent radiation monitor readings to exceed the threshold identified in the IC. This Threshold is intended for sites that have established effluent monitoring on non-routine release pathways for which a discharge permit would not normally be prepared. <u>Threshold #2</u> This Threshold addresses radioactivity releases, that for whatever reason, cause effluent radiation monitor readings to exceed the threshold identified in the IC established by the radioactivity discharge permit. This value may be associated with a planned batch release, or a continuous release path. <u>Threshold #3</u> This Threshold addresses uncontrolled releases that are detected by sample analyses, particularly on unmonitored pathways, e.g., spills of radioactive liquids into storm drains, heat exchanger leakage in river water systems, etc. The underlying basis of this Threshold involves the degradation in the level of safety of the plant implied by the uncontrolled release.

COLD

General Emergency

RG1 Off-site Dose Resulting from an Actual or IMMINENT Release of Gaseous Radioactivity Greater Than 1000 mrem TEDE or 5000 mrem Thyroid CDE for the Actual or Projected Duration of the Release Using Actual Meteorology.

Operating Mode Applicability: **All**

Threshold Values:

Note: The Emergency Coordinator should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time. If dose assessment results are available, declaration should be based on dose assessment instead of radiation monitor values. Do not delay declaration awaiting dose assessment results.

1. VALID reading on ANY of the following radiation monitors greater than the value shown for 15 minutes or longer:

Monitor		Pathway	Value
RAD-6304		Plant Vent SPING	2.8E+0 µCi/cc
RAD 6418	Unit 4 Via Plant Vent	Spent Fuel Pool Vent SPING	1.1E+1 µCi/cc
MSL 3[4] RAD-6426A, B, C (DAM-1)		Main Steam Line Monitor	2.4E+2 µCi/cc
RAD-3[4]-6417		SJAE SPING	7.5E+3 µCi/cc

OR

2. Dose assessment using actual meteorology indicates doses greater than 1000 mrem TEDE or 5000 mrem thyroid CDE at or beyond the site boundary.

OR

3. Field survey results indicate closed window dose rates greater than 1000 mR/hr expected to continue for 60 minutes or longer; or analyses of field survey samples indicate thyroid CDE greater than 5000 mrem for one hour of inhalation, at or beyond site boundary.

GUIDANCE BOX FOR USE OF SPING MONITORS IN RADIOLOGICAL EFFLUENT TABLES

The calculations for the SPING monitors listed are based on NOBLE GAS levels; use SPING channels 5, 7, and 9 as appropriate for the indicated values in the tables. (Ref. DBD PTN-NEI-99-01)

Channel		Range	Typical Reading
Ch-5	Low Range	1.0E-07 - 6.0E-02	5.0E-07
Ch-7	Mid Range	2.5E-02 - 4.0E+02	1.0E-04
Ch-9	High Range	1.0E+00 - 1.0E+05	1.0E-01

DEFINITION BOX

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.

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

Site Area Emergency

RS1 Off-site Dose Resulting from an Actual or IMMINENT Release of Gaseous Radioactivity Greater Than 100 mrem TEDE or 500 mrem Thyroid CDE for the Actual or Projected Duration of the Release.

Operating Mode Applicability: **All**

Threshold Values:

Note: *The Emergency Coordinator should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time. If dose assessment results are available, declaration should be based on dose assessment instead of radiation monitor values. Do not delay declaration awaiting dose assessment results.*

1. VALID reading on ANY of the following radiation monitors greater than the value shown for 15 minutes or longer:

Monitor		Pathway	Value
RAD-6304		Plant Vent SPING	2.8E-1 µCi/cc
RAD 6418	Unit 4 Via Plant Vent	Spent Fuel Pool Vent SPING	1.1E+0 µCi/cc
MSL 3[4] RAD-6426A, B, C (DAM-1)		Main Steam Line Monitor (DAM-1)	2.4E+1 µCi/cc
RAD-3[4]-6417		SJAE SPING	7.5E+2 µCi/cc

OR

2. Dose assessment using actual meteorology indicates doses greater than 100 mrem TEDE or 500 mrem thyroid CDE at or beyond the site boundary.

OR

3. Field survey results indicate closed window dose rates greater than 100 mR/hr expected to continue for 60 minutes or longer; or analyses of field survey samples indicate thyroid CDE greater than 500 mrem for one hour of inhalation, at or beyond the site boundary.

GUIDANCE BOX FOR USE OF SPING MONITORS IN RADIOLOGICAL EFFLUENT TABLES

The calculations for the SPING monitors listed are based on NOBLE GAS levels; use SPING channels 5, 7, and 9 as appropriate for the indicated values in the tables. (Ref. DBD PTN-NEI-99-01)

Channel		Range	Typical Reading
Ch-5	Low Range	1.0E-07 - 6.0E-02	5.0E-07
Ch-7	Mid Range	2.5E-02 - 4.0E+02	1.0E-04
Ch-9	High Range	1.0E+00 - 1.0E+05	1.0E-01

Alert

RA1 Any Release of Gaseous or Liquid Radioactivity to the Environment Greater Than 200 Times the Radiological Effluent Technical Specifications/ODCM for 15 Minutes or Longer.

Operating Mode Applicability: **All**

Threshold Values:

Note: The Emergency Coordinator should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.

1. VALID reading on ANY of the following radiation monitors greater than the value shown for 15 minutes or longer:

Monitor		Pathway	Value
RAD-6304		Plant Vent SPING	3.2E-2 µCi/cc
RAD 6418	Unit 4 Via Plant Vent	Spent Fuel Pool Vent SPING	1.3E-1 µCi/cc
MSL 3[4] RAD-6426A, B, C (DAM-1)		Main Steam Line Monitor (DAM-1)	2.8E+0 µCi/cc
RAD-3[4]-6417		SJAE SPING	8.6E+1 µCi/cc
PRMS R-18		Liquid Rad Waste Effluent Line	2.4E+5 cpm
PRMS-3[4]-R-19		Any one Steam Generator Blowdown	2.4E+5 cpm

OR

2. VALID reading on any effluent monitor reading that greater than 200 times the alarm setpoint established by a current radioactivity discharge permit for 15 minutes or longer.

OR

3. Confirmed sample analyses for gaseous or liquid releases indicates concentrations or release rates greater than 200 times ODCM limits for 15 minutes or longer.

GUIDANCE BOX FOR USE OF SPING MONITORS IN RADIOLOGICAL EFFLUENT TABLES

The calculations for the SPING monitors listed are based on NOBLE GAS levels; use SPING channels 5, 7, and 9 as appropriate for the indicated values in the tables. (Ref. DBD PTN-NEI-99-01)

Channel		Range	Typical Reading
Ch-5	Low Range	1.0E-07 - 6.0E-02	5.0E-07
Ch-7	Mid Range	2.5E-02 - 4.0E+02	1.0E-04
Ch-9	High Range	1.0E+00 - 1.0E+05	1.0E-01

Unusual Event

RU1 Any Release of Gaseous or Liquid Radioactivity to the Environment Greater Than 2 Times the ODCM for 60 Minutes or Longer.

Operating Mode Applicability: **All**

Threshold Values:

Note *The Emergency Coordinator should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.*

1. VALID reading on ANY of the following radiation monitors greater than the value shown for 60 minutes or longer.

Monitor		Pathway	Value
RAD-6304		Plant Vent SPING	3.2E-4 µCi/cc
RAD 6418	Unit 4 Via Plant Vent	Spent Fuel Pool Vent SPING	1.3E-3 µCi/cc
MSL 3[4] RAD-6426A, B, C (DAM-1)		N/A	N/A
RAD-3[4]-6417		SJAE SPING	8.6E-1 uCi/cc
PRMS R-18		Liquid Rad Waste Effluent Line	1.6E+4 cpm
PRMS-3[4]-R-19		Any one Steam Generator Blowdown	1.6E+4 cpm
PRMS R-14		Plant Vent (PV)	6.4E+4 cpm

OR

2. VALID reading on any effluent monitor reading greater than 2 times the alarm setpoint established by a current radioactivity discharge permit for 60 minutes or longer.

OR

3. Confirmed sample analyses for gaseous or liquid releases indicates concentrations or release rates greater than 2 times ODCM Limits for 60 minutes or longer.

GUIDANCE BOX FOR USE OF SPING MONITORS IN RADIOLOGICAL EFFLUENT TABLES

The calculations for the SPING monitors listed are based on NOBLE GAS levels; use SPING channels 5, 7, and 9 as appropriate for the indicated values in the tables. (Ref. DBD PTN-NEI-99-01)

Channel		Range	Typical Reading
Ch-5	Low Range	1.0E-07 - 6.0E-02	5.0E-07
Ch-7	Mid Range	2.5E-02 - 4.0E+02	1.0E-04
Ch-9	High Range	1.0E+00 - 1.0E+05	1.0E-01

Recognition Category

R – ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENT

RADIOLOGICAL EFFLUENT

Rec. Cat.	General Emergency	Site Area Emergency	Alert	Unusual Event
R – ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENT			**RA2 – Basis:** This IC addresses increases in radiation dose rates within plant buildings, and may be a precursor to a radioactivity release to the environment. These events represent a loss of control over radioactive material and represent an actual or substantial potential degradation in the level of safety of the plant. <u>Threshold #1</u> A specific value is not specified in that the elevation of any fuel bundle in transit must be considered for evaluating this condition. <u>Threshold #2</u> This Threshold addresses radiation monitor indications of fuel uncovery and/or fuel damage. Increased ventilation monitor readings may be indication of a radioactivity release from the fuel, confirming that damage has occurred. Increased background at the ventilation monitor due to water level decrease may mask increased ventilation exhaust airborne activity and needs to be considered. While a radiation monitor could detect an increase in dose rate due to a drop in the water level, it might not be a reliable indication of whether or not the fuel is covered. Escalation of this emergency classification level, if appropriate, would be based on RS1 or RG1. **RA3 – Basis:** This IC addresses increased radiation levels that: impact continued operation in areas requiring continuous occupancy to maintain safe operation or to perform a safe shutdown. The cause and/or magnitude of the increase in radiation levels is not a concern of this IC. The Emergency Coordinator must consider the source or cause of the increased radiation levels and determine if any other IC may be involved. Areas requiring continuous occupancy are the Control Room and CAS The value of 15mR/hr is derived from the GDC 19 value of 5 rem in 30 days with adjustment for expected occupancy times. Although Section III.D.3 of NUREG-0737, "Clarification of TMI Action Plan Requirements", provides that the 15 mR/hr value can be averaged over the 30 days, the value is used here without averaging, as a 30 day duration implies an event potentially more significant than an Alert.	**RU2 – Basis:** This IC addresses increased radiation levels as a result of water level decreases above irradiated fuel or events that have resulted, or may result, in UNPLANNED increases in radiation dose rates within plant buildings. These radiation increases represent a loss of control over radioactive material and represent a potential degradation in the level of safety of the plant. Threshold #1 The refueling pathway is the combination of the refueling cavity, transfer canal and spent fuel pool. While a radiation monitor could detect an increase in dose rate due to a drop in the water level, it might not be a reliable indication of whether or not the fuel is covered. For refueling events where the water level drops below the RPV flange classification would be via CU2. This event escalates to an Alert per RA2 if irradiated fuel outside the reactor vessel is uncovered. For events involving irradiated fuel in the reactor vessel, escalation would be via the Fission Product Barrier Table for events in operating modes 1-4. Threshold #2 This Threshold addresses increases in plant radiation levels that represent a loss of control of radioactive material resulting in a potential degradation in the level of safety of the plant. This Threshold excludes radiation level increases that result from planned activities such as use of radiographic sources and movement of radioactive waste materials. A specific list of ARMs is not required as it would restrict the applicability of the Threshold. The intent is to identify loss of control of radioactive material in any monitored area. This event escalates to an Alert per RA3 if the increase in dose rates impedes personnel access necessary for safe operation. **GUIDANCE BOX FOR USE OF TEMPORARY MONITORS IN ABNORMAL RADIATION TABLES** Portable radiation monitors staged as compensatory actions for OOS monitors should be used in place of installed equipment for the purpose of this EAL.

COLD

General Emergency	Site Area Emergency	Alert	Unusual Event	Recognition Category

RA2 Damage to Irradiated Fuel or Loss of Water Level that Has Resulted or Will Result in the Uncovering of Irradiated Fuel Outside the Reactor Vessel.

Operating Mode Applicability: **All**

Threshold Values:

1. A water level drop in the reactor refueling cavity, spent fuel pool or fuel transfer canal that will result in irradiated fuel becoming uncovered.

   OR

2. A VALID elevated reading of greater than or equal to 1.0 E+4 mR/hr on ANY of the following due to damage to irradiated fuel or loss of water level.

Unit 3	Unit 4
RI-3-1402B, UNIT 3 CONTAINMENT OPERATING FLOOR	RI-4-1405B, UNIT 4 CONTAINMENT OPERATING FLOOR
RI-3-1407B, UNIT 3 SPENT FUEL PIT CANAL AREA	RI-4-1408B, UNIT 4 SPENT FUEL PIT CANAL AREA
RI-3-1421B UNIT 3 SPENT FUEL PIT NORTH WALL	RI-4-1422B UNIT 4 SPENT FUEL PIT SOUTH WALL

RA3 Rise in Radiation Levels Within the Facility That Impedes Operation of Systems Required to Maintain Safe Plant Safety Functions.

Operating Mode Applicability: **All**

Threshold Values:

1. Dose rate greater than 15 mR/hr in **ANY** of the following areas requiring continuous occupancy to maintain plant safety functions:
   - Control Room
   - CAS

RU2 UNPLANNED Rise in Plant Radiation Levels.

Operating Mode Applicability: **All**

Threshold Values:

1.a. UNPLANNED water level drop in a reactor refueling pathway as indicated by LT-3[4]-651 56’ 10”.

AND

b. VALID Area Radiation Monitor reading rise on any of the following:

Unit 3	Unit 4
RI-3-1402B, UNIT 3 CONTAINMENT OPERATING FLOOR	RI-4-1405B, UNIT 4 CONTAINMENT OPERATING FLOOR
RI-3-1407B, UNIT 3 SPENT FUEL PIT CANAL AREA	RI-4-1408B, UNIT 4 SPENT FUEL PIT CANAL AREA
RI-3-1421B UNIT 3 SPENT FUEL PIT NORTH WALL	RI-4-1422B UNIT 4 SPENT FUEL PIT SOUTH WALL

OR

2. UNPLANNED VALID Area Radiation Monitor readings or survey results indicate a rise by a factor of 1000 over normal* levels.

*Normal can be considered as the highest reading in the past twenty-four hours excluding the current peak value.

GUIDANCE BOX FOR RU2, Threshold 1

This event addresses increased radiation levels as a result of decreasing water level above irradiated fuel. A Spent Fuel Pool level resulting from routine evaporation accompanied by normal* and expected area radiation levels are not subject to this EAL.

GUIDANCE BOX FOR USE OF MONITORS IN ABNORMAL RADIATION TABLES

Area Radiation monitors in the Control Room indicate in mr/hr up to 100 mr/hr. Monitors automatically switch to Rem/hr as noted at the monitor.

GUIDANCE BOX FOR USE OF TEMPORARY MONITORS IN ABNORMAL RADIATION TABLES

Portable radiation monitors staged as compensatory actions for OOS monitors may be used in place of installed equipment, only if data is available to the control room, for the purpose of this EAL.

DEFINITION BOX

UNPLANNED – A parameter change or an event that is not the result of an intended evolution and requires corrective or mitigative actions.

VALID – An indication, report, or condition, is considered to be VALID when it is verified by (1) an instrument channel check, (2) indications on related or redundant indicators, or (3) by direct observation by plant personnel, such that doubt related to the indicator’s operability, the condition’s existence, or the report’s accuracy is removed. Implicit in this definition is the need for timely assessment.

ABNORMAL RAD LEVELS

R – ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENT

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COLD

RECOGNITION CATEGORY C

COLD SHUTDOWN / REFUELING

SYSTEM MALFUNCTIONS

COLD

Rec. Cat.	General Emergency	Site Area Emergency	Alert	Unusual Event
C – COLD SHUTDOWN / REFUELING SYSTEM MALFUNCTIONS	**CG1 – Basis:** This IC represents the inability to restore and maintain RPV level to above the top of active fuel with containment challenged. Fuel damage is probable if RPV level cannot be restored, as available decay heat will cause boiling, further reducing the RPV level. With the CONTAINMENT breached or challenged then the potential for unmonitored fission product release to the environment is high. This represents a direct path for radioactive inventory to be released to the environment. This is consistent with the definition of a GE. The GE is declared on the occurrence of the loss or IMMINENT loss of function of all three barriers. These Threshold Values are based on concerns raised by Generic Letter 88-17, Loss of Decay Heat Removal, SECY 91-283, Evaluation of Shutdown and Low Power Risk Issues, NUREG-1449, Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States, and, NUMARC 91-06, Guidelines for Industry Actions to Assess Shutdown Management. A number of variables can have a significant impact on heat removal capability challenging the fuel clad barrier. For example, mid-loop, reduced level/flange level, head in place, cavity flooded, RCS venting strategy, decay heat removal system design, vortexing pre-disposition, and steam generator U-tube draining. Analysis indicates that core damage may occur within an hour following continued core uncovery therefore, 30 minutes was conservatively chosen. If CONTAINMENT CLOSURE is re-established prior to exceeding the 30 minute core uncovery time limit then escalation to GE would not occur. Threshold #1 10% is approximately the bottom of the Hot Leg. Threshold #2 Sump and tank level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of RCS leakage. If all level indication were to be lost during a loss of RCS inventory event, the operators would need to determine that RPV inventory loss was occurring by observing sump and tank level changes. Sump and tank level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of RCS leakage. PTN-NEI-99-01 / Recognition Category C assumes the reactor head and upper internals are removed so the only shielding is from the water level above the top of the active fuel. As water level in the RPV lowers, the dose rate increases due to core shine and should result in monitor indication with possible alarm. Of the two CHRRM channels RAD-6311B has the most reliable response to the core shine due to its location on the 58 foot operating floor of both units. The “A” Channels located near the personnel hatch on the mezzanine level will not detect the upward shine from the core effectively. Post-TMI studies indicated that the installed nuclear instrumentation will operate erratically when the core is uncovered and that this should be used as a tool for making such determinations.	**CS1 – Basis:** Under the conditions specified by this IC, continued decrease in RCS level is indicative of a loss of inventory control. Inventory loss may be due to an RCS breach, pressure boundary leakage, or continued boiling in the RPV. Thus, declaration of a Site Area Emergency is warranted. Escalation to a General Emergency is via CG1 or AG1. Threshold #1 15% is below the minimum level for operation of RHR. Threshold #2 10% is nominally the bottom of the Hot Leg. Threshold #3 The 30-minute duration allows sufficient time for actions to be performed to recover inventory control equipment. PTN-NEI-99-01 / Recognition Category C assumes the reactor head and upper internals are removed so the only shielding is from the water level above the top of the active fuel. As water level in the RPV lowers, the dose rate above the core increases due to core shine and should result in monitor indication with possible alarm. Of the two CHRRM channels RAD-6311B has the most reliable response to the core shine due to its location on the 58 foot operating floor of both units. The “A” Channels located near the personnel hatch on the mezzanine level will not detect the upward shine from the core effectively. Post-TMI studies indicated that the installed nuclear instrumentation will operate erratically when the core is uncovered and that this should be used as a tool for making such determinations. **Alert** **CA1 – Basis:** These Threshold Values serve as precursors to a loss of ability to adequately cool the fuel. The magnitude of this loss of water indicates that makeup systems have not been effective and may not be capable of preventing further RPV level decrease and potential core uncovery. This condition will result in a minimum emergency classification level of an Alert. Threshold #1 The alarm is at the top of the Hot Leg and provides an indicator for escalation to the SAE and GE. The inability to restore and maintain level after reaching this setpoint would therefore be indicative of a failure of the RCS barrier. Threshold #2 If all level indication were to be lost during a loss of RCS inventory event, the operators would need to determine that RPV inventory loss was occurring by observing sump and tank level changes. Sump and tank level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of RCS leakage. The 15-minute duration for the loss of level indication was chosen because it is half of the CS1 Site Area Emergency Threshold duration. Significant fuel damage is not expected to occur until the core has been uncovered for greater than 1 hour per the analysis referenced in the CG1 basis. Therefore this Threshold meets the definition for an Alert. If RPV level continues to lower then escalation to Site Area Emergency will be via CS1.	**CA3 – Basis:** Loss of all AC power compromises all plant safety systems requiring electric power including RHR, ECCS, Containment Heat Removal, Spent Fuel Heat Removal and the Ultimate Heat Sink. An SBO crosstie, from the unaffected unit through each unit’s D 4KV buses, that can be accomplished within the 15 minutes will provide an alternate power source. The event can be classified as an Alert when in cold shutdown, refueling, or defueled mode because of the significantly reduced decay heat and lower temperature and pressure, increasing the time to restore one of the emergency busses, relative to that specified for the Site Area Emergency EAL. Escalating to Site Area Emergency, if appropriate, is by Abnormal Rad Levels / Radiological Effluent ICs. Fifteen minutes was selected as a Threshold to exclude transient or momentary power losses. **Unusual Event** **CU1 – Basis:** This IC is considered to be a potential degradation of the level of safety of the plant. The inability to maintain or restore level for the current configuration is indicative of loss of RCS inventory. Relief valve normal operation should be excluded from this IC. However, a relief valve that operates and fails to close per design should be considered applicable to this IC if the relief valve cannot be isolated. Prolonged loss of RCS Inventory may result in escalation to the Alert emergency classification level via either CA1 or CA4. **GUIDANCE BOX FOR CU1** Planned evolutions to test, manipulate, repair, perform maintenance or modifications to systems and equipment that result in an EAL value being met or exceeded are not subject to classification and activation requirements as long as the evolution proceeds as planned and is within the operational limitations imposed by the specific operating license. [Turkey Point Technical Basis, Section 3.9] **GUIDANCE BOX FOR CU1** The difference between CU1 and CU2 deals with the RCS conditions that exist between cold shutdown and refueling modes. In the refueling mode the RCS is not intact and Reactor Pressure Vessel level and inventory are monitored by different means. In cold shutdown the RCS will normally be intact and standard RCS inventory and level monitoring means are available. [NEI 99-01, page 45 of 167]	**CU2 – Basis:** Refueling evolutions that decrease RCS water level below the RPV flange are carefully planned and procedurally controlled. An UNPLANNED event that results in water level decreasing below the RPV flange, or below the planned RCS water level for the given evolution (if the planned RCS water level is already below the RPV flange), warrants declaration of a NOUE due to the reduced RCS inventory that is available to keep the core covered. The allowance of 15 minutes was chosen because it is reasonable to assume that level can be restored within this time frame using one or more of the redundant means of refill that should be available. If level cannot be restored in this time frame then it may indicate a more serious condition exists. Continued loss of RCS Inventory will result in escalation to the Alert emergency classification level via either CA1 or CA4. Threshold #1 This Threshold involves a decrease in RCS level below the top of the RPV flange that continues for 15 minutes due to an UNPLANNED event. This Threshold is not applicable to decreases in flooded reactor cavity level, which is addressed by RU2 Threshold 1 until such time as the level decreases to the level of the vessel flange. If RPV level continues to decrease and reaches the Bottom ID of the RCS Loop then escalation to CA1 would be appropriate. Threshold #2 This Threshold addresses conditions in the refueling mode when normal means of core temperature indication and RCS level indication may not be available. Redundant means of RPV level indication will normally be installed (including the ability to monitor level visually) to assure that the ability to monitor level will not be interrupted. However, if all level indication were to be lost during a loss of RCS inventory event, the operators would need to determine that RPV inventory loss was occurring by observing sump and tank level changes. Sump and tank level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of RCS leakage. Escalation to the Alert emergency classification level would be via either CA1 or CA4. **CU3 – Basis:** The condition indicated by this IC is the degradation of the off-site and on-site AC power systems such that any additional single failure would result in a unit blackout. This condition could occur due to a loss of off-site power with a concurrent failure of all but one emergency generator to supply power to its emergency busses. The subsequent loss of this single power source would escalate the event to an Alert in accordance with CA3. Fifteen minutes was selected as a threshold to exclude transient or momentary losses of power.

General Emergency

CG1 Loss of RCS Inventory Affecting Fuel Clad Integrity with Containment Challenged.

Operating Mode Applicability: **5, 6**

Threshold Value:

Note: *The Emergency Coordinator should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.*

1. a. RCS level less than 10% on EITHER LIS-3[4]-6421 or LIS-3[4]-6423 for 30 minutes or longer.

AND

b. **ANY** containment challenge indication (see Table):

OR

2. a. RCS level cannot be monitored with core uncovery indicated by **ANY** of the following for 30 minutes or longer.

- RAD-3(4)-6311B reading greater than 1.4E+3 R/hr
- Erratic source range monitor indication.
- Unexplained level rise in
  - Containment Sump
  - #1 Waste Holdup Tank
  - CVCS Holdup Tank
  - RWST

AND

b. **ANY** containment challenge indication:

Containment Challenge Indications
• CONTAINMENT CLOSURE not established.
• 4% H_2 inside containment.
• UNPLANNED rise in containment pressure.

DEFINITION BOX

CONTAINMENT CLOSURE – The procedurally defined actions taken to secure containment and its associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions.

UNPLANNED – A parameter change or an event that is not the result of an intended evolution and requires corrective or mitigative actions.

Site Area Emergency

CS1 Loss of RCS Inventory Affecting Core Decay Heat Removal Capability.

Operating Mode Applicability: **5, 6**

Threshold Values:

Note: *The Emergency Coordinator should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.*

1. With CONTAINMENT CLOSURE not established, RCS level less than 15% on EITHER LIS-3[4]-6421 or LIS-3[4]-6423.

OR

2. With CONTAINMENT CLOSURE established, RCS level less than 10% on EITHER LIS-3[4]-6421 or LIS-3[4]-6423.

OR

3. RCS level cannot be monitored for 30 minutes or longer with a loss of RCS inventory as indicated by **ANY** of the following:

- RAD-3(4)-6311B reading greater than 8.7E+1 R/h
- Erratic Source Range Monitor Indication.
- Unexplained level rise in
  - Containment Sump
  - #1 Waste Holdup Tank
  - CVCS Holdup Tank
  - RWST

GUIDANCE BOX FOR CU1

Planned evolutions to test, manipulate, repair, perform maintenance or modifications to systems and equipment that result in an EAL value being met or exceeded are not subject to classification and activation requirements as long as the evolution proceeds as planned and is within the operational limitations imposed by the specific operating license. [Turkey Point Technical Basis, Section 3.9]

GUIDANCE BOX FOR CU1

The difference between CU1 and CU2 deals with the RCS conditions that exist between cold shutdown and refueling modes. In the refueling mode the RCS is not intact and Reactor Pressure Vessel level and inventory are monitored by different means. In cold shutdown the RCS will normally be intact and standard RCS inventory and level monitoring means are available. [NEI 99-01, page 45 of 167]

Alert

CA1 Loss of RCS Inventory.

Operating Mode Applicability: **5, 6**

Threshold Values:

Note: *The Emergency Coordinator should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.*

1. Loss of RCS inventory as indicated by RX VESSEL DRAINDOWN LO-LO LEVEL Alarm I 7/3 (24% on either LIS-3[4]-6421 or LIS-3[4]-6423)

OR

2. RCS level cannot be monitored for 15 minutes or longer with a loss of RCS inventory as indicated by an unexplained level rise in ANY of the following:

- Containment Sump
- #1 Waste Holdup Tank
- CVCS Holdup Tank
- RWST

CA3 Loss of All Off-site and All On-Site AC Power to Emergency Busses For 15 minutes Or Longer.

Operating Mode Applicability: **5, 6, Defueled**

Threshold Value:

Note: *The Emergency Coordinator should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.*

1. Loss of all Off-Site and all On-Site AC Power to 3[4]A 4KV and 3[4]B 4KV Buses for 15 minutes or longer.

Unusual Event

CU2 UNPLANNED Loss of RCS Inventory.

Operating Mode Applicability: **6**

Threshold Values:

Note: *The Emergency Coordinator should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.*

1. UNPLANNED RCS level drop as indicated by either of the following:

- RCS water level drop below the RPV flange for 15 minutes or longer when the RCS level band is established above the RPV flange.
- RCS water level drop below the RCS level band for 15 minutes or longer when the RCS level band is established below the RPV flange.

OR

2. RCS level cannot be monitored with a loss of RCS inventory as indicated by an unexplained level rise in:

- Containment Sump
- #1 Waste Holdup Tank
- CVCS Holdup Tank
- RWST

CU1 RCS Leakage.

Operating Mode Applicability: **5**

Threshold Values:

Note: *The Emergency Coordinator should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.*

1. RCS leakage results in the inability to maintain or restore level within the pressurizer or RCS level target band for 15 minutes or longer.

CU3 AC Power Capability To Emergency Busses Reduced to a Single Power Source for 15 Minutes or longer such that Any Additional Single Failure Would Result in Unit Blackout.

Operating Mode Applicability: **5, 6**

Threshold Value:

Note: *The Emergency Coordinator should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.*

1. a. AC power capability to 3[4]A 4KV and 3[4]B 4KV Buses reduced to a single power source for 15 minutes or longer.

AND

b. Any additional single power source failure will result in unit blackout.

Recognition Category

RCS LEAKAGE / INVENTORY

AC POWER

C – COLD SHUTDOWN / REFUELING SYSTEM MALFUNCTIONS

COLD

Rec. Cat.	General Emergency	Site Area Emergency	Alert	Unusual Event
C – COLD SHUTDOWN / REFUELING SYSTEM MALFUNCTIONS			**CA4 – Basis:** For Threshold 1, the RCS Reheat Duration Threshold table addresses complete loss of functions required for core cooling for greater than 60 minutes during refueling and cold shutdown modes when RCS integrity is established. RCS integrity should be considered to be in place when the RCS pressure boundary is in its normal condition for the cold shutdown mode of operation (e.g., no freeze seals or nozzle dams). The status of CONTAINMENT CLOSURE in this condition is immaterial given that the RCS is providing a high pressure barrier to fission product release to the environment. The 60 minute time frame should allow sufficient time to restore cooling without there being a substantial degradation in plant safety. The RCS Reheat Duration Threshold table also addresses the complete loss of functions required for core cooling for greater than 20 minutes during refueling and cold shutdown modes when CONTAINMENT CLOSURE is established but RCS integrity is not established or RCS inventory is reduced. The allowed 20 minute time frame was included to allow operator action to restore the heat removal function, if possible. Complete loss of functions required for core cooling during refueling and cold shutdown modes when neither CONTAINMENT CLOSURE nor RCS integrity are established. No delay time is allowed because the evaporated reactor coolant that may be released into the Containment during this heatup condition could also be directly released to the environment. The note (*) indicates that this Threshold is not applicable if actions are successful in restoring an RCS heat removal system to operation and RCS temperature is being reduced within the specified time frame. In Threshold 2, the 10 psi pressure increase addresses situations where, due to high decay heat loads, the time provided to restore temperature control, should be less than 60 minutes. Escalation to Site Area Emergency would be via CS1 should boiling result in significant RPV level loss leading to core uncovery. NRC analyses show that there are sequences that can cause core uncovery in 15 to 20 minutes and severe core damage within an hour after decay heat removal is lost. A loss of Technical Specification components alone is not intended to constitute an Alert. The same is true of a momentary UNPLANNED excursion above the Technical Specification cold shutdown temperature limit when the heat removal function is available. The Emergency Coordinator must remain alert to events or conditions that lead to the conclusion that exceeding the Threshold is IMMINENT. If, in the judgment of the Emergency Coordinator, an IMMINENT situation is at hand, the classification should be made as if the Threshold has been exceeded.	**CU4 – Basis:** This IC is be a precursor of more serious conditions and, as a result, is considered to be a potential degradation of the level of safety of the plant. In cold shutdown, the ability to remove decay heat relies primarily on forced cooling flow. Operation of the systems that provide this forced cooling may be jeopardized due to the unlikely loss of electrical power or RCS inventory. Since the RCS usually remains intact in the cold shutdown mode a large inventory of water is available to keep the core covered. During refueling, the level in the RPV will normally be maintained above the RPV flange. Refueling evolutions that decrease water level below the RPV flange are carefully planned and procedurally controlled. Loss of forced decay heat removal at reduced inventory may result in more rapid increases in RCS temperatures depending on the time since shutdown. Normal means of core temperature indication and RCS level indication may not be available in the refueling mode. Redundant means of RPV level indication are therefore procedurally installed to assure that the ability to monitor level will not be interrupted. However, if all level and temperature indication were to be lost in either the cold shutdown of refueling modes, Threshold 2 would result in declaration of a NOUE if both temperature and level indication cannot be restored within 15 minutes from the loss of both means of indication. Escalation to Alert would be via CA1 based on an inventory loss or CA4 based on exceeding its temperature criteria. **CU6 – Basis:** The purpose of this IC and its associated Threshold Values is to recognize a loss of communications capability that either defeats the plant operations staff ability to perform routine tasks necessary for plant operations or the ability to communicate issues with off-site authorities. The loss of off-site communications ability is expected to be significantly more comprehensive than the condition addressed by 10 CFR 50.72. The availability of one method of ordinary off-site communications is sufficient to inform federal, state, and local authorities of plant issues. This Threshold is intended to be used only when extraordinary means (e.g., relaying of information from radio transmissions, individuals being sent to off-site locations, etc.) are being utilized to make communications possible. The list for on-site and off-site communications loss encompasses the loss of all means of routine communications and all means of communications with off-site authorities. **CU7 – Basis:** The purpose of this IC and its associated Threshold Values is to recognize a loss of DC power compromising the ability to monitor and control the removal of decay heat during Cold Shutdown or Refueling operations. It is intended that the loss of the operating (operable) train is to be considered. If this loss results in the inability to maintain cold shutdown, the escalation to an Alert will be per CA4. Fifteen minutes was selected as a Threshold to exclude transient or momentary power losses. **CU8 – Basis:** This IC addresses criticality events that occur in Cold Shutdown or Refueling modes such as fuel mis-loading events and inadvertent dilution events. This IC indicates a potential degradation of the level of safety of the plant, warranting a NOUE classification. The term “sustained” is used in order to allow exclusion of expected short term startup rates from planned fuel bundle or control rod movements during core alteration. These short term positive startup rates are the result of the increase in neutron population due to subcritical multiplication. Escalation would be by Emergency Coordinator Judgment.

DEFINITION BOX

CONTAINMENT CLOSURE – The procedurally defined actions taken to secure containment and its associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions.

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.

COPY

General Emergency	Site Area Emergency	Alert	Unusual Event	Recognition Category

DEFINITION BOX

UNPLANNED – A parameter change or an event that is not the result of an intended evolution and requires corrective or mitigative actions.

CA4 Inability to Maintain Plant in Cold Shutdown.

Operating Mode Applicability: **5, 6**

Threshold Values:

Note: *The Emergency Coordinator should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.*

1. An UNPLANNED event results in RCS temperature greater than 200 F for greater than the specified duration on table.

Table: RCS Reheat Duration Thresholds

RCS	Containment Closure	Duration
Intact (Vent Path NOT Established) (but not Reduced Inventory)	N/A	60 minutes*
Not Intact (Vent Path Established) or Reduced Inventory	Established	20 minutes*
	Not Established	0 minutes

* If an RCS heat removal system is in operation within this time frame and RCS temperature is being reduced, the EAL is not applicable.

OR

2. An UNPLANNED event results in RCS pressure increase greater than 10 psi due to a loss of Shutdown Cooling. (This Threshold does not apply in Solid Plant conditions.)

CU4 UNPLANNED Loss of Decay Heat Removal Capability with Irradiated Fuel in the Reactor Vessel

Operating Mode Applicability: **5, 6**

Threshold Values:

Note: *The Emergency Coordinator should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.*

1. UNPLANNED event results in RCS temperature exceeding 200 F.

OR

2. Loss of all RCS temperature and RCS level indication for 15 minutes or longer.

CU7 Loss of Required DC Power for 15 Minutes Or Longer.

Operating Mode Applicability: **5, 6**

Threshold Values:

Note

The Emergency Coordinator should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

1. Less than 105 VDC on ANY 2 of the following required Vital DC buses for 15 minutes or longer:
   - 3D01
   - 3D23
   - 4D01
   - 4D23

CU8 Inadvertent Criticality.

Operating Mode Applicability: **5, 6**

Threshold Values:

1. UNPLANNED sustained positive startup rate observed on nuclear instrumentation.

DECAY HEAT

DC POWER

CRITICALITY

C – COLD SHUTDOWN / REFUELING SYSTEM MALFUNCTIONS

COLD

Rec. Cat.	General Emergency	Site Area Emergency	Alert	Unusual Event
C – COLD SHUTDOWN / REFUELING SYSTEM MALFUNCTIONS				**CU6 – Basis:** The purpose of this IC and its associated Threshold Values is to recognize a loss of communications capability that either defeats the plant operations staff ability to perform routine tasks necessary for plant operations or the ability to communicate issues with off-site authorities. The loss of off-site communications ability is expected to be significantly more comprehensive than the condition addressed by 10 CFR 50.72. The availability of one method of ordinary off-site communications is sufficient to inform federal, state, and local authorities of plant issues. This Threshold is intended to be used only when extraordinary means (e.g., relaying of information from radio transmissions, individuals being sent to off-site locations, etc.) are being utilized to make communications possible. The list for on-site and off-site communications loss encompasses the loss of all means of routine communications and all means of communications with off-site authorities.

COLD

General Emergency	Site Area Emergency	Alert	Unusual Event	Recognition Category	
			CU6 Loss of All On-site or Off-site Communications Capabilities. **Operating Mode Applicability:** **5, 6, Defueled** Threshold Values: 1. Loss of **all** of the following on-site communication methods affecting the ability to perform routine operations. • Plant Page • Portable Radios • Commercial Telephones (PBX) **OR** 2. Loss of all of the following off-site communication methods affecting the ability to perform either of the following offsite notifications. • Commercial Telephones * • Federal Telecommunications System (FTS)	COMMUNICATIONS	C – COLD SHUTDOWN / REFUELING SYSTEM MALFUNCTIONS

GUIDANCE FOR CU6

The following communications systems are available for off-site notifications from the control room:

a. State and County Notifications:
   - Commercial Telephones *
   - Hot Ring Down (HRD)
   - Emergency Satellite Communications

b. NRC Notification
   - Emergency Notification System (ENS)
   - Commercial Telephones

* Commercial phones include plant phones, direct outside lines and Installed cell phones in the Control Room, but not personal cell phones

COLD

COLD

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RECOGNITION CATEGORY H

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

COLD

Rec. Cat.	General Emergency	Site Area Emergency	Alert	Unusual Event
H – HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY	**DEFINITION BOX** **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. **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 flame is preferred but is NOT required if large quantities of smoke and heat are observed. **NORMAL PLANT OPERATIONS –** Activities at the plant site associated with routine testing, maintenance, or equipment operations, in accordance with normal operating or administrative procedures. Entry into abnormal or emergency operating procedures, or deviation from normal security or radiological controls posture, is a departure from NORMAL PLANT OPERATIONS. **PROJECTILE –** An object directed toward a Nuclear Power Plant (NPP) that could cause concern for its continued operability, reliability, or personnel safety. **PROTECTED AREA –** The area (within the Owner Controlled Area) occupied by the nuclear units and associated equipment and facilities enclosed within the security perimeter fence. The area within which accountability of personnel is maintained in an emergency. This does not include the ISFSI Protected Area. **VISIBLE DAMAGE –** Damage to equipment or structure that is readily observable without measurements, testing, or analysis. Damage is sufficient to cause concern regarding the continued operability or reliability of the affected structure, system, or component. Example damage includes: deformation due to heat or impact, denting, penetration, rupture, cracking, and paint blistering. Surface blemishes (e.g., paint chipping, scratches) should not be included. **VITAL AREAS –** Areas within the PROTECTED AREA, that 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.		**HA1 – Basis:** These Thresholds escalate from HU1 in that the occurrence of the event has resulted in VISIBLE DAMAGE to plant structures or areas containing equipment necessary for a safe shutdown, or has caused damage to the safety systems in those structures evidenced by Control Room indications of degraded system response or performance. The occurrence of VISIBLE DAMAGE and/or degraded system response is intended to discriminate against lesser events. The initial report should not be interpreted as mandating a lengthy damage assessment prior to classification. No attempt is made in this Threshold to assess the actual magnitude of the damage. The significance here is not that a particular system or structure was damaged, but rather, that the event was of sufficient magnitude to cause this degradation. Escalation of this emergency classification level, if appropriate, would be based on System Malfunction ICs. Threshold #1 Seismic events of this magnitude can result in a VITAL AREA being subjected to forces beyond design limits, and thus damage may be assumed to have occurred to plant safety systems. The Modified Mercalli Intensity V (0.04 – 0.09 g) is indicated by: • Felt by nearly everyone • Light damage, some windows broken or unstable objects overturned. The ETNA Seismograph Event Recorder Red LED Indicator ON indicates an event has been recorded. The red LED indicator flashing indicates an event is being recorded or the device is writing to memory; I&C should be contacted immediately. The USGS National Earthquake Information Center can confirm if an earthquake has occurred in the area of the plant. Threshold #2 This Threshold is based on a tornado striking (touching down) or high winds that have caused VISIBLE DAMAGE to structures containing functions or systems required for safe shutdown of the plant. Threshold #3 This Threshold addresses the effect of internal flooding caused by events such as component failures, equipment misalignment, or outage activity mishaps. It is based on the degraded performance of systems, or that has created industrial safety hazards (e.g., electrical shock) that preclude necessary access to operate or monitor safety equipment. The inability to access, operate or monitor safety equipment represents an actual or substantial potential degradation of the level of safety of the plant. Flooding as used in this Threshold describes a condition where water is entering the room faster than installed equipment is capable of removal, resulting in a rise of water level within the room. Classification of this Threshold should not be delayed while corrective actions are being taken to isolate the water source. Threshold #5 This Threshold addresses vehicle crashes within the PROTECTED AREA that results in VISIBLE DAMAGE to VITAL AREAS or indication of damage to safety structures, systems, or components containing functions and systems required for safe shutdown of the plant. Threshold #6 This Threshold addresses other natural phenomena that result in VISIBLE DAMAGE to VITAL AREAS or results in indication of damage to safety structures, systems, or components containing functions and systems required for safe shutdown of the plant in the event of a hurricane and the associated storm surge that can also be precursors of more serious events.	**HU1 – Basis:** These Thresholds are categorized on the basis of the occurrence of an event of sufficient magnitude to be of concern to plant operators. Threshold #1 Damage may be caused to some portions of the site, but should not affect ability of safety functions to operate. The Modified Mercalli Intensity IV (0.01 – 0.04 g) is indicated by: • Felt indoors by many, outdoors by few • No damage As defined in the EPRI-sponsored Guidelines for Nuclear Plant Response to an Earthquake, dated October 1989, a "felt earthquake" is: An earthquake of sufficient intensity such that: (a) the vibratory ground motion is felt at the nuclear plant site and recognized as an earthquake based on a consensus of Control Room operators on duty at the time, and (b) for plants with operable seismic instrumentation, the seismic switches of the plant are activated. The ETNA Seismograph Event Recorder Red LED Indicator ON indicates an event has been recorded. The red LED indicator flashing indicates an event is being recorded or the device is writing to memory; I&C should be contacted immediately. The USGS National Earthquake Information Center can confirm if an earthquake has occurred in the area of the plant. Threshold #2 This Threshold is based on a tornado striking (touching down) or high winds within the PROTECTED AREA. The high wind Threshold Value (100 mph) is based on approaching the FSAR design basis maximum wind speed value(145 mph) and is within the calibrated range of control room instrumentation. Escalation of this emergency classification level, if appropriate, would be based VISIBLE DAMAGE via HA1. Threshold #3 This EAL addresses the effect of internal flooding caused by events such as component failures, equipment misalignment, or outage activity mishaps. Escalation of this emergency classification level, if appropriate, would be based VISIBLE DAMAGE via HA1, or by other plant conditions. Threshold #4 This Threshold addresses main turbine rotating component failures of sufficient magnitude to cause observable damage to the turbine casing or to the seals of the turbine generator. Generator seal damage observed after generator purge does not meet the intent of this Threshold because it did not impact normal operation of the plant. Of major concern is the potential for leakage of combustible fluids (lubricating oils) and gases (hydrogen cooling) to the plant environs. Actual FIRES and flammable gas build up are appropriately classified via HU2 and HU3. This Threshold is consistent with the definition of a NOUE while maintaining the anticipatory nature desired and recognizing the risk to non-safety related equipment. Escalation of this emergency classification level, if appropriate, would be to HA1 based on damage done by PROJECTILES generated by the failure. Threshold #5 This Threshold addresses natural phenomena for a hurricane and the associated storm surge that can also be precursors of more serious events.

General Emergency	Site Area Emergency	Alert	Unusual Event	Recognition Category

DEFINITION BOX

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.

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 flame is preferred but is NOT required if large quantities of smoke and heat are observed.

NORMAL PLANT OPERATIONS – Activities at the plant site associated with routine testing, maintenance, or equipment operations, in accordance with normal operating or administrative procedures. Entry into abnormal or emergency operating procedures, or deviation from normal security or radiological controls posture, is a departure from NORMAL PLANT OPERATIONS.

PROJECTILE – An object directed toward a Nuclear Power Plant (NPP) that could cause concern for its continued operability, reliability, or personnel safety.

PROTECTED AREA – The area (within the Owner Controlled Area) occupied by the nuclear units and associated equipment and facilities enclosed within the security perimeter fence. The area within which accountability of personnel is maintained in an emergency. This does not include the ISFSI Protected Area.

VISIBLE DAMAGE – Damage to equipment or structure that is readily observable without measurements, testing, or analysis. Damage is sufficient to cause concern regarding the continued operability or reliability of the affected structure, system, or component. Example damage includes: deformation due to heat or impact, denting, penetration, rupture, cracking, and paint blistering. Surface blemishes (e.g., paint chipping, scratches) should not be included.

VITAL AREAS – Areas within the PROTECTED AREA, that 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.

HA1 Natural or Destructive Phenomena Affecting VITAL AREAS.

Operating Mode Applicability: **All**

Threshold Values:

1. a. Earthquake confirmed by **ANY** of the following:
   - Earthquake felt in plant
   - Modified Mercalli Intensity V evaluated
   - USGS National Earthquake Information Center

   AND

   b. Control Room indication of degraded performance of systems required for the safe shutdown of the plant.

   OR

   c. Seismic event greater than Operating Basis Earthquake (OBE) as indicated by the ETNA Seismograph Event Recorder indicating greater than or equal to 0.05 g.

   OR

2. Tornado striking OR high winds greater than 100 mph resulting in VISIBLE DAMAGE to **ANY** of the Table H1 plant structures containing safety systems or components **OR** Control Room indication of degraded performance of those safety systems.

   OR

3. Internal flooding in the RHR Pump Rooms resulting in an electrical shock hazard that precludes access to operate or monitor safety equipment **OR** Control Room indication of degraded performance of those safety systems.

   OR

4. Turbine failure-generated PROJECTILES resulting in VISIBLE DAMAGE to or penetration of **ANY** of the Table H1 plant structures containing safety systems or components **OR** control room indication of degraded performance of those safety systems.

   OR

5. Vehicle crash resulting in VISIBLE DAMAGE to ANY of the Table H1 plant structures containing safety systems or components OR Control Room indication of degraded performance of those safety systems.

   OR

6. Natural occurrences resulting in VISIBLE DAMAGE to **ANY** of the Table H1 plant structures containing safety systems or components **OR** Control Room indication of degraded performance of those safety systems.

TABLE H1

Turbine Building	Feedwater Platforms
Control Building/Roof	Condensate Storage Tanks
EDG Buildings	U3 DOST
Containment Buildings	Spent Fuel Areas
Auxiliary Building/Roof	Component Cooling Water Area
¾ C Bus/XFMRs and ¾ E LC	HHSI Pump Area
DWST	Fuel Handling Building
Main/Aux/Startup XFMRs	Refueling Water Storage Tanks
U4 Laydown Area	Primary Water Storage Tanks
Main Steam Platforms	Intake

HU1 Natural or Destructive Phenomena Affecting the PROTECTED AREA.

Operating Mode Applicability: **All**

Threshold Value:

1. Seismic event identified by **ANY** 2 of the following:
   - Seismic event confirmed by the ETNA Seismograph Event Recorder Red LED Indicator ON.
   - Earthquake felt in plant
   - Modified Mercalli Intensity IV evaluated
   - USGS National Earthquake Information Center

   OR

2. Tornado striking within PROTECTED AREA boundary or high winds greater than 100 mph.

   OR

3. Internal flooding in the RHR Pump Rooms that has the potential to affect safety related equipment required by Technical Specifications for the current operating mode:

   OR

4. Turbine failure resulting in casing penetration or damage to turbine or generator seals.

   OR

5. Natural occurrences affecting the PROTECTED AREA.
   - Confirmed hurricane warning is in effect.
   - Flood surge that prevents land access to the site

H – HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

NATURAL OR MAN-MADE EVENTS

COLD

Rec. Cat.	General Emergency	Site Area Emergency	Alert	Unusual Event
H – HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY	**DEFINITION BOX** **VITAL AREAS –** Areas within the PROTECTED AREA, that 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. **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. **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 flame is preferred but is NOT required if large quantities of smoke and heat are observed. **NORMAL PLANT OPERATIONS –** Activities at the plant site associated with routine testing, maintenance, or equipment operations, in accordance with normal operating or administrative procedures. Entry into abnormal or emergency operating procedures, or deviation from normal security or radiological controls posture, is a departure from NORMAL PLANT OPERATIONS. **PROJECTILE –** An object directed toward a Nuclear Power Plant (NPP) that could cause concern for its continued operability, reliability, or personnel safety. **PROTECTED AREA –** The area (within the Owner Controlled Area) occupied by the nuclear units and associated equipment and facilities enclosed within the security perimeter fence. The area within which accountability of personnel is maintained in an emergency. This does not include the ISFSI Protected Area. **VISIBLE DAMAGE –** Damage to equipment or structure that is readily observable without measurements, testing, or analysis. Damage is sufficient to cause concern regarding the continued operability or reliability of the affected structure, system, or component. Example damage includes: deformation due to heat or impact, denting, penetration, rupture, cracking, and paint blistering. Surface blemishes (e.g., paint chipping, scratches) should not be included.		**HA2 – Basis:** VISIBLE DAMAGE is used to identify the magnitude of the FIRE or EXPLOSION and to discriminate against minor FIRES and EXPLOSIONS. The reference to structures containing safety systems or components is included to discriminate against FIRES or EXPLOSIONS in areas having a low probability of affecting safe operation. The significance here is not that a safety system was degraded but the fact that the FIRE or EXPLOSION was large enough to cause damage to these systems. The use of VISIBLE DAMAGE should not be interpreted as mandating a lengthy damage assessment prior to classification. The declaration of an Alert and the activation of the Technical Support Center will provide the Emergency Coordinator with the resources needed to perform detailed damage assessments. The Emergency Coordinator also needs to consider any security aspects of the EXPLOSION. Escalation of this emergency classification level, if appropriate, will be based on System Malfunctions, Fission Product Barrier Degradation or Abnormal Rad Levels / Radiological Effluent ICs. **HA3 – Basis:** Gases in a VITAL AREA can affect the ability to safely operate or safely shutdown the reactor. The fact that SCBA may be worn does not eliminate the need to declare the event. Declaration should not be delayed for confirmation from atmospheric testing if the atmosphere poses an immediate threat to life and health or an immediate threat of severe exposure to gases. This could be based upon documented analysis, indication of personal ill effects from exposure, or operating experience with the hazards. If the equipment in the stated area was already inoperable, or out of service, before the event occurred, then this EAL should not be declared as it will have no adverse impact on the ability of the plant to safely operate or safely shutdown beyond that already allowed by Technical Specifications at the time of the event. An asphyxiant is a gas capable of reducing the level of oxygen in the body to dangerous levels. Most commonly, asphyxiants work by merely displacing air in an enclosed environment. This reduces the concentration of oxygen below the normal level of around 19%, which can lead to breathing difficulties, unconsciousness or even death. An uncontrolled release of flammable gasses within a facility structure has the potential to affect safe operation of the plant by limiting either operator or equipment operations due to the potential for ignition and resulting equipment damage/personnel injury. Flammable gasses, such as hydrogen and acetylene, are routinely used to maintain plant systems (hydrogen) or to repair equipment/components (acetylene - used in welding). This Threshold assumes concentrations of flammable gasses which can ignite/support combustion. Escalation of this emergency classification level, if appropriate, will be based on System Malfunctions, Fission Product Barrier Degradation or Abnormal Rad Levels / Radioactive Effluent ICs.	**HU2 – Basis:** This Threshold addresses the magnitude and extent of FIRES or EXPLOSIONS that may be potentially significant precursors of damage to safety systems. It addresses the FIRE / EXPLOSION, and not the degradation in performance of affected systems that may result. As used here, detection is visual observation and report by plant personnel or sensor alarm indication. Threshold #1 The 15 minute time period begins with a credible notification that a FIRE is occurring, or indication of a fire detection system alarm/actuation. Verification of a fire detection system alarm/actuation includes actions that can be taken within the Control Room or other nearby location to ensure that it is not spurious. An alarm is assumed to be an indication of a FIRE unless it is disproved within the 15 minute period by personnel dispatched to the scene. In other words, a personnel report from the scene may be used to disprove a sensor alarm if received within 15 minutes of the alarm, but shall not be required to verify the alarm. The intent of this 15 minute duration is to size the FIRE and to discriminate against small FIRES that are readily extinguished (e.g., smoldering waste paper basket). Threshold #2 This Threshold addresses only those EXPLOSIONS of sufficient force to damage permanent structures or equipment within the PROTECTED AREA. No attempt is made to assess the actual magnitude of the damage. The occurrence of the EXPLOSION is sufficient for declaration. The Emergency Coordinator also needs to consider any security aspects of the EXPLOSION, if applicable. Escalation of this emergency classification level, if appropriate, would be based on HA2. **HU3 – Basis:** This Threshold is based on the release of toxic, corrosive, asphyxiant or flammable gases of sufficient quantity to affect NORMAL PLANT OPERATIONS. The fact that SCBA may be worn does not eliminate the need to declare the event. This IC is not intended to require significant assessment or quantification. It assumes an uncontrolled process that has the potential to affect plant operations. This would preclude small or incidental releases, or releases that do not impact structures needed for plant operation. An asphyxiant is a gas capable of reducing the level of oxygen in the body to dangerous levels. Most commonly, asphyxiants work by merely displacing air in an enclosed environment. This reduces the concentration of oxygen below the normal level of around 19%, which can lead to breathing difficulties, unconsciousness or even death. Escalation of this emergency classification level, if appropriate, would be based on HA3.

COLD

General Emergency	Site Area Emergency	Alert	Unusual Event	Recognition Category	
		HA2 FIRE or EXPLOSION Affecting the Operability of Plant Safety Systems Required to Establish or Maintain Safe Shutdown. **Operating Mode Applicability: All** Threshold Value: 1. FIRE or EXPLOSION resulting in VISIBLE DAMAGE to **ANY** of the following structures containing safety systems or components **OR** Control Room indication of degraded performance of those safety systems.	**HU2** FIRE Within the PROTECTED AREA Not Extinguished Within 15 Minutes of Detection OR EXPLOSION within the PROTECTED AREA. **Operating Mode Applicability: All** Threshold Value: ***Note:*** *The Emergency Coordinator should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the duration has exceeded, or will likely exceed, the applicable time.* 1. FIRE in any of the following areas not extinguished within 15 minutes of a FIRE Alarm or Control Room notification.	FIRE / EXPLOSION	H – HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY
		HA3 Access to a VITAL AREA Is Prohibited Due To Toxic, Corrosive, Asphyxiant or Flammable Gases Which Jeopardize Operation of Operable Equipment Required to Maintain Safe Operations or Safely Shutdown the Reactor. **Operating Mode Applicability: All** Threshold Values: ***Note:*** *If the equipment in the stated area was already inoperable, or out of service, before the event occurred, then this EAL should not be declared as it will have no adverse impact on the ability of the plant to safely operate or safely shutdown beyond that already allowed by Technical Specifications at the time of the event.* 1. Access to a VITAL AREA is prohibited due to toxic, corrosive, asphyxiant or flammable gases which jeopardize operation of systems required to maintain safe operations or safely shutdown the reactor.	**HU3** Release of Toxic, Corrosive, Asphyxiant, or Flammable Gases Deemed Detrimental to NORMAL PLANT OPERATIONS. **Operating Mode Applicability**: **All** Threshold Values: 1. Toxic, corrosive, asphyxiant or flammable gases in amounts that have or could adversely affect NORMAL PLANT OPERATIONS. **OR** 2. Report by local, county or state officials for evacuation or sheltering of site personnel based on an off-site event.	TOXIC / FLAMMABLE GAS	

DEFINITION BOX

VITAL AREAS – Areas within the PROTECTED AREA, that 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.

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.

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 flame is preferred but is NOT required if large quantities of smoke and heat are observed.

NORMAL PLANT OPERATIONS – Activities at the plant site associated with routine testing, maintenance, or equipment operations, in accordance with normal operating or administrative procedures. Entry into abnormal or emergency operating procedures, or deviation from normal security or radiological controls posture, is a departure from NORMAL PLANT OPERATIONS.

PROJECTILE – An object directed toward a Nuclear Power Plant (NPP) that could cause concern for its continued operability, reliability, or personnel safety.

PROTECTED AREA – The area (within the Owner Controlled Area) occupied by the nuclear units and associated equipment and facilities enclosed within the security perimeter fence. The area within which accountability of personnel is maintained in an emergency. This does not include the ISFSI Protected Area.

VISIBLE DAMAGE – Damage to equipment or structure that is readily observable without measurements, testing, or analysis. Damage is sufficient to cause concern regarding the continued operability or reliability of the affected structure, system, or component. Example damage includes: deformation due to heat or impact, denting, penetration, rupture, cracking, and paint blistering. Surface blemishes (e.g., paint chipping, scratches) should not be included.

Turbine Building	Feedwater Platforms
Control Building/Roof	Condensate Storage Tanks
EDG Buildings	U3 DOST
Containment Buildings	Spent Fuel Areas
Auxiliary Building/Roof	Component Cooling Water Area
¾ C Bus/XFMRs and ¾ E LC	HHSI Pump Area
DWST	Fuel Handling Building
Main/Aux/Startup XFMRs	Refueling Water Storage Tanks
U4 Laydown Area	Primary Water Storage Tanks
Main Steam Platforms	Intake

Turbine Building	Feedwater Platforms
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Main/Aux/Startup XFMRs	Refueling Water Storage Tanks
U4 Laydown Area	Primary Water Storage Tanks
Main Steam Platforms	Intake

OR

2. EXPLOSION within the PROTECTED AREA.

GUIDANCE BOX FOR HU3, HA3

Planned controlled activities, such as Containment entry at power, do not meet the intent of HU3 or HA3.

Rec. Cat.	General Emergency	Site Area Emergency	Alert	Unusual Event
H – HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY	**HG1 – Basis:** Threshold #1 This Threshold encompasses conditions under which a HOSTILE ACTION has resulted in a loss of physical control of VITAL AREAS (containing vital equipment or controls of vital equipment) required to maintain safety functions and control of that equipment cannot be transferred to and operated from another location. These safety functions are reactivity control, RCS inventory, and secondary heat removal. If control of the plant equipment necessary to maintain safety functions can be transferred to another location, then the threshold is not met. Threshold #2 This Threshold addresses failure of spent fuel cooling systems as a result of HOSTILE ACTION if IMMINENT fuel damage is likely, such as when a freshly off-loaded reactor core is in the spent fuel pool. **DEFINITION BOX** **AIRLINER –** Airliner is meant to be a large aircraft with the potential for causing significant damage to the plant. **FRESHLY OFF-LOADED REACTOR CORE -** A freshly off-loaded Reactor core in the Spent Fuel Pool exists during the period of time when core off-load begins until core reload is complete. **HOSTILE ACTION –** An act toward a NPP or its personnel that includes the use of violent force to destroy equipment, take HOSTAGES, and/or intimidate the licensee to achieve an end. 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 (i.e., this may include violent acts between individuals in the owner controlled area). **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. **OWNER CONTROLLED AREA –** That portion of FPL property surrounding and including the Turkey Point Nuclear Power Plant which is subject to limited access and control as deemed appropriate by FPL. **PROTECTED AREA –** The area (within the Owner Controlled Area) occupied by the nuclear units and associated equipment and facilities enclosed within the security perimeter fence. The area within which accountability of personnel is maintained in an emergency. This does not include the ISFSI Protected Area. **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. **VISIBLE DAMAGE –** Damage to equipment or structure that is readily observable without measurements, testing, or analysis. Damage is sufficient to cause concern regarding the continued operability or reliability of the affected structure, system, or component. Example damage includes: deformation due to heat or impact, denting, penetration, rupture, cracking, and paint blistering. Surface blemishes (e.g., paint chipping, scratches) should not be included.	**HS4 – Basis:** This condition represents an escalated threat to plant safety above that contained in the Alert in that a HOSTILE FORCE has progressed from the OWNER CONTROLLED AREA to the PROTECTED AREA. This Threshold addresses the contingency for a very rapid progression of events, such as that experienced on September 11, 2001. It is not premised solely on the potential for a radiological release. Rather the issue includes the need for rapid assistance due to the possibility for significant and indeterminate damage from additional air, land or water attack elements. The fact that the site is under serious attack with minimal time available for further preparation or additional assistance to arrive requires ORO readiness and preparation for the implementation of protective measures. This Threshold addresses the potential for a very rapid progression of events due to a HOSTILE ACTION. It is not intended to address incidents that are accidental events or acts of civil disobedience, such as small aircraft impact, hunters, or physical disputes between employees within the PROTECTED AREA. Those events are adequately addressed by other Thresholds. This initiating condition and EAL do not apply to an attack solely in the ISFSI Protected Area. An attack on the ISFSI Protected Area should be considered an attack within the Owner Controlled Area and classified as an Alert per Initiating Condition HA4. Escalation of this emergency classification level, if appropriate, would be based on actual plant status after impact or progression of attack. **HS2 – Basis:** The intent of this IC is to capture those events where control of the plant cannot be reestablished in a timely manner. In this case, expeditious transfer of control of safety systems has not occurred (although fission product barrier damage may not yet be indicated). The intent of the Threshold is to establish control of important plant equipment and knowledge of important plant parameters in a timely manner. Primary emphasis should be placed on those components and instruments that supply protection for and information about safety functions. These safety functions are reactivity control, RCS inventory, and secondary heat removal. The determination of whether or not control is established at the remote shutdown panel is based on Emergency Coordinator judgment. The Emergency Coordinator is expected to make a reasonable, informed judgment within the time for transfer that the licensee has control of the plant from the remote shutdown panel. Escalation of this emergency classification level, if appropriate, would be by Fission Product Barrier Degradation or Abnormal Rad Levels/Radiological Effluent EALs.	**HA4 – Basis:** ***Note:*** *Timely and accurate communication between Security Shift Supervision and the Control Room is crucial for the implementation of effective Security EALs.* These Thresholds address the contingency for a very rapid progression of events, such as that experienced on September 11, 2001. They are not premised solely on the potential for a radiological release. Rather the issue includes the need for rapid assistance due to the possibility for significant and indeterminate damage from additional air, land or water attack elements. The fact that the site is under serious attack or is an identified attack target with minimal time available for further preparation or additional assistance to arrive requires a heightened state of readiness and implementation of protective measures that can be effective (such as on-site evacuation, dispersal or sheltering). Threshold #1 This Threshold addresses the potential for a very rapid progression of events due to a HOSTILE ACTION. It is not intended to address incidents that are accidental events or acts of civil disobedience, such as small aircraft impact, hunters, or physical disputes between employees within the OCA. Those events are adequately addressed by other EALs. Note that this EAL is applicable for any HOSTILE ACTION occurring, or that has occurred, in the OWNER CONTROLLED AREA. This includes ISFSI's that may be outside the PROTECTED AREA but still within the OWNER CONTROLLED AREA. Threshold #2 This Threshold addresses the immediacy of an expected threat arrival or impact on the site within a relatively short time. The intent of this Threshold is to ensure that notifications for the airliner attack threat are made in a timely manner and that OROs and plant personnel are at a state of heightened awareness regarding the credible threat. Airliner is meant to be a large aircraft with the potential for causing significant damage to the plant. This Threshold is met when a plant receives information regarding an airliner attack threat from NRC and the airliner is within 30 minutes of the plant. Only the plant to which the specific threat is made need declare the Alert. The NRC Headquarters Operations Officer (HOO) will communicate to the licensee if the threat involves an airliner (airliner is meant to be a large aircraft with the potential for causing significant damage to the plant).. The status and size of the plane may be provided by NORAD through the NRC. **HA5 – Basis:** With the Control Room evacuated, additional support, monitoring and direction through the Technical Support Center and/or other emergency response facilities may be necessary. Inability to establish plant control from outside the Control Room will escalate this event to a Site Area Emergency.	**HU4 – Basis:** ***Note:*** *Timely and accurate communication between Security Shift Supervision and the Control Room is crucial for the implementation of effective Security EALs.* Security events which do not represent a potential degradation in the level of safety of the plant are reported under 10 CFR 73.71 or in some cases under 10 CFR 50.72. Security events assessed as HOSTILE ACTIONS are classifiable under HA4, HS4 and HG1. A higher initial classification could be made based upon the nature and timing of the security threat and potential consequences. The licensee shall consider upgrading the emergency response status and emergency classification level in accordance with the site's Physical Security Plan and Emergency Plan. Threshold #1 Reference is made to security shift supervision because these individuals are the designated personnel on-site qualified and trained to confirm that a security event is occurring or has occurred. Training on security event classification confirmation is closely controlled due to the strict secrecy controls placed on the plant Physical Security Plan. Physical Security Plans are based on guidance provided by NEI 03-12. Threshold #2 This threshold is included to ensure that appropriate notifications for the security threat are made in a timely manner. This includes information of a credible threat. Only the plant to which the specific threat is made need declare the Notification of an Unusual Event. The determination of "credible" is made through use of information found in the Physical Security Plan. Threshold #3 The intent of this Threshold is to ensure that notifications for the aircraft threat are made in a timely manner and that OROs and plant personnel are at a state of heightened awareness regarding the credible threat. It is not the intent of this Threshold to replace existing non-hostile related Thresholds involving aircraft. This Threshold is met when a plant receives information regarding an aircraft threat from NRC. Validation is performed by calling the NRC or by other approved methods of authentication. Only the plant to which the specific threat is made need declare the Unusual Event. The NRC Headquarters Operations Officer (HOO) will communicate to the licensee if the threat involves an airliner (airliner is meant to be a large aircraft with the potential for causing significant damage to the plant). The status and size of the plane may be provided by NORAD through the NRC. Escalation to Alert emergency classification level would be via HA4 would be appropriate if the threat involves an airliner within 30 minutes of the plant.

COLD

General Emergency	Site Area Emergency	Alert	Unusual Event	Recognition Category	
HG1 HOSTILE ACTION Resulting in Loss of Physical Control of the Facility. **Operating Mode Applicability:** **All** Threshold Values: **1.** A HOSTILE ACTION has occurred such that plant personnel are unable to operate equipment required to maintain safety functions. **OR** **2.** A HOSTILE ACTION has caused failure of Spent Fuel Cooling Systems and IMMINENT fuel damage is likely for A FRESHLY OFF-LOADED REACTOR CORE in pool.	**HS4** HOSTILE ACTION within the PROTECTED AREA **Operating Mode Applicability:** **All** Threshold Value: **1.** A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the Security Shift Supervisor. **2.** This threshold does not apply to a hostile action to the ISFSI Protected Area only.	**HA4** HOSTILE ACTION within the OWNER CONTROLLED AREA or Airborne Attack Threat **Operating Mode Applicability:** **All** Threshold Values: **1.** A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by the Security Shift Supervisor. **OR** **2.** A validated notification from NRC of an airliner attack threat within 30 minutes of the site.	**HU4** Confirmed SECURITY CONDITION or Threat Which Indicates a Potential Degradation in the Level of Safety of the Plant. **Operating Mode Applicability:** **All** Threshold Values: **1.** A SECURITY CONDITION that does **NOT** involve a HOSTILE ACTION as reported by the Security Shift Supervisor. **OR** **2.** A credible Turkey Point security threat notification. **OR** **3.** A validated notification from NRC providing information of an aircraft threat.	SECURITY	H – HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY
	HS2 Control Room Evacuation Has Been Initiated and Plant Control Cannot be Established. **Operating Mode Applicability:** **All** Threshold Values: **1.** **a.** Control Room evacuation has been initiated. **AND** **b.** Control of the plant cannot be established within 15 minutes.	**HA5** Control Room Evacuation has been Initiated. **Operating Mode Applicability:** **All** Threshold Value: **1.** 0-ONOP-105, Control Room Evacuation, requires Control Room evacuation.		CR EVACUATION	

DEFINITION BOX

AIRLINER – Airliner is meant to be a large aircraft with the potential for causing significant damage to the plant.

FRESHLY OFF-LOADED REACTOR CORE IN POOL – A freshly off-loaded reactor core, in the Spent Fuel Pool, exists during the period of time when core off-load begins until core reload is complete.

HOSTILE ACTION – An act toward a Nuclear Power Plant (NPP) or its personnel that includes the use of violent force to destroy equipment, take HOSTAGES, and/or intimidate the licensee to achieve an end. 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 (i.e., this may include violent acts between individuals in the owner controlled area).

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.

OWNER CONTROLLED AREA – That portion of FPL property surrounding and including the Turkey Point Nuclear Power Plant which is subject to limited access and control as deemed appropriate by FPL. [EPlan]

PROJECTILE – An object directed toward a Nuclear Power Plant (NPP) that could cause concern for its continued operability, reliability, or personnel safety.

PROTECTED AREA – The area (within the Owner Controlled Area) occupied by the nuclear units and associated equipment and facilities enclosed within the security perimeter fence. The area within which accountability of personnel is maintained in an emergency. This does not include the ISFSI Protected Area.

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.

VISIBLE DAMAGE – Damage to equipment or structure that is readily observable without measurements, testing, or analysis. Damage is sufficient to cause concern regarding the continued operability or reliability of the affected structure, system, or component. Example damage includes: deformation due to heat or impact, denting, penetration, rupture, cracking, and paint blistering. Surface blemishes (e.g., paint chipping, scratches) should not be included.

VITAL AREAS – Areas within the PROTECTED AREA, that 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.

COLD

Rec. Cat.	General Emergency	Site Area Emergency	Alert	Unusual Event
H – HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY	**HG2 – Basis:** This EAL addresses unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Coordinator to fall under the emergency classification level for General Emergency.	**HS3 – Basis:** This EAL addresses unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Coordinator to fall under the emergency classification level description for Site Area Emergency.	**HA6 – Basis:** This EAL addresses unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Coordinator to fall under the Alert emergency classification level.	**HU5 – Basis:** This EAL addresses unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Coordinator to fall under the Notification of Unusual Event (NOUE) emergency class.

COLD

General Emergency	Site Area Emergency	Alert	Unusual Event	Recognition Category	
HG2 Other Conditions Existing Which in the Judgment of the Emergency Coordinator Warrant Declaration of General Emergency. **Operating Mode Applicability:** **All** Threshold Value: **1.** Other conditions exist which in the judgment of the Emergency Coordinator 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. **DEFINITION BOX** **HOSTILE ACTION –** An act toward a Nuclear Power Plant (NPP) or its personnel that includes the use of violent force to destroy equipment, take HOSTAGES, and/or intimidate the licensee to achieve an end. 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 (i.e., this may include violent acts between individuals in the owner controlled area). **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. **PROJECTILE –** An object directed toward a Nuclear Power Plant (NPP) that could cause concern for its continued operability, reliability, or personnel safety.	**HS3** Other Conditions Existing Which in the Judgment of the Emergency Coordinator Warrant Declaration of Site Area Emergency. **Operating Mode Applicability:** **All** Threshold Value: **1.** Other conditions exist which in the judgment of the Emergency Coordinator 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.	**HA6** Other Conditions Existing Which in the Judgment of the Emergency Coordinator Warrant Declaration of an Alert. **Operating Mode Applicability:** **All** Threshold Value: **1.** Other conditions exist which in the judgment of the Emergency Coordinator indicate that events are in progress or have occurred which involve 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.	**HU5** Other Conditions Existing Which in the Judgment of the Emergency Coordinator Warrant Declaration of a Notification of Unusual Event (NOUE). **Operating Mode Applicability:** **All** Threshold Value: **1.** Other conditions exist which in the judgment of the Emergency Coordinator 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.	DISCRETIONARY	H – HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

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RECOGNITION CATEGORY E

EVENTS RELATED TO ISFSI

COLD

Rec. Cat.	General Emergency	Site Area Emergency	Alert	Unusual Event
E – EVENTS RELATED TO ISFSI			**DEFINITION BOX** **CONFINEMENT BOUNDRY –** The barrier(s) between areas containing radioactive substances and the environment. **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. **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 flame is preferred but is NOT required if large quantities of smoke and heat are observed. **PROJECTILE –** An object directed toward a Nuclear Power Plant (NPP) that could cause concern for its continued operability, reliability, or personnel safety.	**E-HU1 – Basis:** A Notification of Unusual Event (NOUE) in this IC is categorized on the basis of the occurrence of an event of sufficient magnitude that a loaded cask CONFINEMENT BOUNDARY is damaged or violated. This includes classification based on a loaded fuel storage cask CONFINEMENT BOUNDARY loss leading to the degradation of the fuel during storage or posing an operational safety problem with respect to its removal from storage as indicated by elevated radiation readings from the loaded fuel storage cask. The results of the ISFSI Safety Analysis Report (SAR) per NUREG 1536 or SAR referenced in the cask's Certificate of Compliance and the related NRC Safety Evaluation Report identify natural phenomena events and accident conditions that could potentially effect the CONFINEMENT BOUNDARY. This EAL addresses a dropped cask, a tipped over cask, EXPLOSION, PROJECTILE damage, FIRE damage or natural phenomena affecting a cask (e.g., seismic event, tornado, etc.).

COLD

General Emergency	Site Area Emergency	Alert	Unusual Event	Recognition Category	
			E-HU1 Damage to a loaded cask CONFINEMENT BOUNDARY. **Operating Mode Applicability:** Not applicable Threshold Value: **1.** Damage to a loaded cask CONFINEMENT BOUNDARY.	**ISFSI**	**E – EVENTS RELATED TO ISFSI**

DEFINITION BOX

CONFINEMENT BOUNDARY – The barrier(s) between areas containing radioactive substances and the environment.

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3.0 RADIOACTIVE GASEOUS EFFLUENT

CONTROL 3.1: Radioactive Gaseous Effluent Monitoring Instrumentation; Operability and Alarm/Trip Setpoints

The radioactive gaseous effluent monitoring instrumentation channels shown in Table 3.1-1 shall be OPERABLE / FUNCTIONAL with their Alarm / Trip Setpoints set to ensure that the limits of Control 3.2 are not exceeded. The Alarm / Trip Setpoints of these channels meeting Control 3.2 shall be determined and adjusted in accordance with the methodology and parameters in this ODCM.

APPLICABILITY: As shown in Table 3.1-1

ACTION:

a. With a radioactive gaseous effluent monitoring instrumentation channel Alarm / Trip Setpoint less conservative than required by the above control, immediately suspend the release of radioactive gaseous effluents monitored by the affected channel, or declare the channel inoperable / non-functional or change the setpoint so it is acceptably conservative.

b. With less than the minimum number of radioactive gaseous effluent monitoring instrumentation channels OPERABLE / FUNCTIONAL, take the ACTION shown in Table 3.1-1. Restore the inoperable / non-functional instrumentation to OPERABLE / FUNCTIONAL status within 30 days and, if unsuccessful explain in the next Annual Radioactive Effluent Release Report pursuant to Administrative Control 1.3 why this in operability / non-functional was not corrected in a timely manner.

c. The provisions of Administrative Control section 1.6.3 are not applicable.

SURVEILLANCE REQUIREMENTS

Each radioactive gaseous effluent monitoring instrumentation channel shall be demonstrated OPERABLE / FUNCTIONAL by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION and ANALOG CHANNEL OPERATIONAL TEST at the frequencies shown in Table 3.1-2.

3.0 RADIOACTIVE GASEOUS EFFLUENT

CONTROL 3.1 Radioactive Gaseous Effluent Monitoring Instrumentation, Operability / Functionality and Alarm/Trip Setpoints, (Cont'd)

TABLE 3.1-1

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

	INSTRUMENT	MINIMUM CHANNELS OPERABLE / FUNCTIONAL	APPLICABILITY	ACTION
1.	GAS DECAY TANK SYSTEM			
	a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release (Plant Vent Monitor)	1	*	3.1.1
	b. Effluent System Flow Rate Measuring Device	1	*	3.1.2
2.	Condenser Air Ejector Vent System			
	a. Noble Gas Activity Monitor (SPING or PRMS)	1	#	3.1.3
	b. Iodine Sampler	1	##	3.1.6
	c. Particulate Sampler	1	##	3.1.6
	d. Effluent System Flow Rate Measuring Device	1	##	3.1.2
	e. Sampler Flow Rate Measuring Device	1	##	3.1.5

3.0 RADIOACTIVE GASEOUS EFFLUENT

CONTROL 3.1: Radioactive Gaseous Effluent Monitoring Instrumentation; Operability / Functionality and Alarm / Trip Setpoints, (Cont'd**)**

TABLE 3.1-1 (Cont'd)
RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

	INSTRUMENT	MINIMUM CHANNELS OPERABLE / FUNCTIONAL	APPLICABILITY	ACTION
3.	Plant Vent System (Include Unit 4's Spent Fuel Pool)			
a.	Noble Gas Activity Monitor (SPING or PRMS)	1	*	3.1.3
b.	Iodine Sampler	1	*	3.1.4
c.	Particulate Sampler	1	*	3.1.4
d.	Effluent System Flow Rate Measuring Device	1	*	3.1.2
e.	Sampler Flow Rate Measuring Device	1	*	3.1.5
4.	Unit 3 Spent Fuel Pit Building Vent			
a.	Noble Gas Activity Monitor	1	*	3.1.3
b.	Iodine Sampler	1	*	3.1.4
c.	Particulate Sampler	1	*	3.1.4
d.	Sampler Flow Rate Measuring Device	1	*	3.1.5

3.0 RADIOACTIVE GASEOUS EFFLUENT

CONTROL 3.1: Radioactive Gaseous Effluent Monitoring Instrumentation; Operability / Functionality and Alarm / Trip Setpoints (Cont'd)

TABLE 3.1-1 (Cont'd)
TABLE NOTATION

* At all times.
\# Applies during Mode 1, 2, 3, and 4.
\## Applies during Mode 1, 2, 3, and 4 when primary to secondary leakage is detected.

ACTION 3.1.1 - With the number of channels FUNCTIONAL less than required by the Minimum Channels FUNCTIONAL requirement, the contents of the tank(s) may be released to the environment provided that prior to initiating the release:

a. At least two independent samples of the tank's contents are analyzed, **and**

b. At least two technically qualified members of the facility staff independently verify the release rate calculations and discharge valve lineup;

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 3.1.2 - With the number of channels FUNCTIONAL less than required by the Minimum Channels FUNCTIONAL requirement, effluent releases via this pathway may continue provided the flow rate is estimated at least once per 4 hours.

ACTION 3.1.3 - With the number of channels OPERABLE / FUNCTIONAL less than required by the Minimum Channels OPERABLE / FUNCTIONAL requirement, effluent releases via this pathway may continue provided grab samples are taken at least once per 12 hours and these samples are analyzed for radioactivity within 24 hours. These monitors may have Technical Specification requirements and action statements.

ACTION 3.1.4 With the number of channels FUNCTIONAL less than required by the Minimum Channels FUNCTIONAL requirement, effluent releases via the affected pathway may continue provided continuous sample collection with auxiliary equipment as required by Table 3.2-1 is installed within 4 hours of the channel being declared non-functional, and analyzed at least weekly.

3.0 RADIOACTIVE GASEOUS EFFLUENT

CONTROL 3.1: Radioactive Gaseous Effluent Monitoring Instrumentation; Operability / Functionality and Alarm / Trip Setpoints (Cont'd)

TABLE 3.1-1 (Cont'd)
TABLE NOTATION

ACTION 3.1.5 With the number of channels FUNCTIONAL less than required by the Minimum Channels FUNCTIONAL requirement, effluent releases via the affected pathway may continue provided auxiliary equipment is installed, AND the sample flow rate is verified at least once per 4 hours.

ACTION 3.1.6 Continuous collection of iodine and particulate in the Condenser Air Ejector is not required. If Primary to Secondary leakage is detected, effluent releases via the affected pathway may continue provided that the iodine and particulate curies released are determined using the methodology of Reference 3.

3.0 RADIOACTIVE GASEOUS EFFLUENT

CONTROL 3.1: Radioactive Gaseous Effluent Monitoring Instrumentation; Operability / Functionality and Alarm / Trip Setpoints, (Cont'd**)**

TABLE 3.1-2

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. GAS DECAY TANK SYSTEM					
a. Noble Gas Activity Monitor Providing Alarm and Automatic Termination of Release (Plant Vent Monitor)	P	P	R (3)	Q (1)	*
b. Effluent System Flow Rate Measuring Device	P (4)	N.A.	R	N.A.	*
2. Condenser Air Ejector Vent System					
a. Noble Gas Activity Monitor (SPING or PRMS)	D	M	R (3)	Q (2)	#
b. Effluent System Flow Rate Measuring Device	D (4)	N.A.	R	N.A.	##
c. Sample Flow Rate Measuring Device	D (4)	N.A.	R	N.A.	##

3.0 RADIOACTIVE GASEOUS EFFLUENT

CONTROL 3.1: Radioactive Gaseous Effluent Monitoring Instrumentation; Operability / Functionality and Alarm / Trip Setpoints, (Cont'd**)**

TABLE 3.1-2 (Cont'd)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
3. Plant Vent System (Include Unit 4's Spent Fuel Pool)					
a. Noble Gas Activity Monitor (SPING or PRMS)	D	M	R (3)	Q (2)	*
b. Iodine Sampler	W (5)	N.A.	N.A.	N.A.	*
c. Particulate Sampler	W (5)	N.A.	N.A.	N.A.	*
d. Effluent System Flow Rate Measuring Device	D (4)	N.A.	R	N.A.	*
e. Sampler Flow Rate Measuring Device	D (4)	N.A.	R	N.A.	*
4. Unit 3 Spent Fuel Pit Building Vent					
a. Noble Gas Activity Monitor	D	M	R (3)	Q (2)	*
b. Iodine Sampler	W (5)	N.A.	N.A.	N.A.	*
c. Particulate Sampler	W (5)	N.A.	N.A.	N.A.	*
d. Sampler Flow Rate Measuring Device	D (4)	N.A.	R	N.A.	*

REDACTED INFORMATION- BLOCKED OUT

REACTIVITY CONTROL SYSTEMS

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

GROUP HEIGHT

LIMITING CONDITION FOR OPERATION

3.1.3.1 All full length (shutdown and control) rods shall be OPERABLE and positioned within the Allowed Rod Misalignment between the Analog Rod Position Indication and the group step counter demand position within one hour after rod motion. The Allowed Rod Misalignment shall be defined as:

APPLICABILITY: MODES 1* and 2*

ACTION:

* See Special Test Exceptions 3.10.2 and 3.10.3.

REDACTED INFORMATION- BLOCKED OUT

d. With one full length rod inoperable due to causes other than addressed by ACTION a, above, or misaligned from its group step counter demand position by more than the Allowed Rod Misalignment of Specification 3.1.3.1, POWER OPERATION may continue provided that within one hour either:

1. The rod is restored to OPERABLE status within the Allowed Rod Misalignment of Specification 3.1.3.1, or

2. The remainder of the rods in the bank with the inoperable rod are aligned to within the Allowed Rod Misalignment of Specification 3.1.3.1 of the inoperable rod while maintaining the rod sequence and insertion limits of Specification 3.1.3.6; the THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation, or

3. The rod is declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. POWER OPERATION may then continue provided that:

a) The THERMAL POWER level is reduced to less than or equal to 75% of RATED THERMAL POWER within one hour and within the next 4 hours the power range neutron flux high trip setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER. THERMAL POWER shall be maintained less than or equal to 75% of RATED THERMAL POWER until compliance with ACTIONS 3.1.3.1.d.3.c and 3.1.3.1.d.3.d below are demonstrated, and

b) The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is determined at least once per 12 hours, and

c) A power distribution map is obtained from the movable incore detectors and $F_Q(Z)$ and $F^N_{\Delta H}$ are verified to be within their limits within 72 hours, and

d) A reevaluation of each accident analysis of Table 3.1-1 is performed within 5 days; this reevaluation shall confirm that the previously analyzed results of these accidents remain valid for the duration of operation under these conditions.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position of each full length rod shall be determined to be within the Allowed Rod Misalignment of the group step counter demand position in accordance with the Surveillance Frequency Control Program (allowing for one hour thermal soak after rod motion) except during time intervals when the Rod Position Deviation Monitor is inoperable, then verify the group positions at least once per 4 hours.

4.1.3.1.2 Each full length rod not fully inserted in the core shall be determined to be OPERABLE by movement of at least 10 steps in any one direction in accordance with the Surveillance Frequency Control Program.

HOT CONDITIONS TABLE
(RCS > 200°F)

HOT

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HOT

INITIATING CONDITIONS MATRIX

		General Emergency	Site Area Emergency	Alert	Unusual Event
R – ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENT	RADIOLOGICAL EFFLUENT	**RG1** Off-Site Dose Resulting from an Actual or IMMINENT Release of Gaseous Radioactivity Greater Than 1000 mRem TEDE or 5000 mRem Thyroid CDE for the Actual or Projected Duration of the Release Using Actual Meteorology. Pg. 6	**RS1** Off-site Dose Resulting from an Actual or IMMINENT Release of Gaseous Radioactivity Greater Than 100 mRem TEDE or 500 mRem Thyroid CDE for the Actual or Projected Duration of the Release. Pg. 6	**RA1** Any Release of Gaseous or Liquid Radioactivity to the Environment Greater Than 200 Times the ODCM for 15 Minutes or Longer. Pg. 6	**RU1** Any Release of Gaseous or Liquid Radioactivity to the Environment Greater Than 2 Times the ODCM for 60 Minutes or Longer. Pg. 6
	ABNORMAL RAD LEVELS			**RA2** Damage to Irradiated Fuel or Loss of Water Level that Has Resulted or Will Result in the Uncovering of Irradiated Fuel Outside the Reactor Vessel. Pg. 8	**RU2** UNPLANNED Rise in Plant Radiation Levels. Pg. 8
				RA3 Rise in Radiation Levels Within the Facility That Impedes Operation of Systems Required to Maintain Safe Plant Safety Functions. Pg. 8	
F – FISSION PRODUCT BARRIER		**FG1** Loss of ANY Two Barriers AND Loss or Potential Loss of the third barrier. Pgs. 11-12	**FS1** Loss or Potential Loss of ANY two barriers. Pgs. 11-12	**FA1** ANY Loss or ANY Potential Loss of EITHER Fuel Clad OR RCS. Pgs. 11-12	**FU1** ANY Loss or ANY Potential Loss of Containment. Pgs. 11-12
S – SYSTEM MALFUNCTIONS	AC POWER	**SG1** Prolonged Loss of All Off-site and All On-Site AC Power to Emergency Busses. Pg. 16	**SS1** Loss of All Off-site and All On-site AC Power to Emergency Busses for 15 minutes or longer. Pg. 16	**SA5** AC Power Capability To Emergency Busses Reduced To A Single Power Source For 15 Minutes or Longer Such That Any Additional Single Failure Would Result In Unit Blackout. Pg. 16	**SU1** Loss of All Off-site AC Power to Emergency Busses for 15 Minutes or Longer. Pg. 16
	FAILURE OF RX PROTECTION / CRITICALITY	**SG2** Automatic Trip and All Manual Actions Fail to Shutdown the Reactor AND Indication of an Extreme Challenge to the Ability to Cool the Core Exists. Pg. 16	**SS2** Automatic Trip Fails to Shutdown the Reactor AND Manual Actions Taken in the Control Room are **NOT** Successful in Shutting Down the Reactor. Pg. 16	**SA2** Automatic Trip Fails to Shutdown the Reactor AND the Manual Actions Taken in the Control Room are Successful in Shutting Down the Reactor. Pg. 16	**SU8** Inadvertent Criticality. Pg. 16
	DC POWER		**SS3** Loss of All Vital DC Power for 15 Minutes or Longer. Pg. 16		
	ANNUNCIATORS		**SS6** Inability to Monitor a Significant Transient in Progress. Pg. 18	**SA4** UNPLANNED Loss of Safety System Annunciation or Indication in the Control Room With Either (1) a Significant Transient in Progress, or (2) Compensatory Indicators Unavailable. Pg. 18	**SU3** UNPLANNED Loss of Safety System Annunciation or Indication in the Control Room for 15 Minutes or Longer. Pg. 18
	RCS LEAK				**SU5** RCS Leakage. Pg. 18
	FUEL CLAD				**SU4** Fuel Clad Degradation. Pg. 18
	TECH SPECS				**SU2** Inability to Reach Required Shutdown Within Technical Specification Limits. Pg. 18
	COMM.				**SU6** Loss of All On-site or Off-site Communications Capabilities. Pg. 20

		General Emergency	Site Area Emergency	Alert	Unusual Event
H – HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY	NATURAL OR MAN-MADE EVENTS			**HA1** Natural or Destructive Phenomena Affecting VITAL AREAS. Pgs. 24	**HU1** Natural or Destructive Phenomena Affecting the PROTECTED AREA. Pgs. 24
	FIRE / EXPLOSION			**HA2** FIRE or EXPLOSION Affecting the Operability of Plant Safety Systems Required to Establish or Maintain Safe Shutdown. Pg. 26	**HU2** FIRE Within the PROTECTED AREA Not Extinguished Within 15 Minutes of Detection OR EXPLOSION within the PROTECTED AREA. Pg. 26
	TOXIC / FLAMMABLE GAS			**HA3** Access to an VITAL AREA Is Prohibited Due To Release of Toxic, Corrosive, Asphyxiant or Flammable Gases Which Jeopardize Operation of Systems Required to Maintain Safe Operations or Safely Shutdown the Reactor. Pg. 26	**HU3** Release of Toxic, Corrosive, Asphyxiant, or Flammable Gases Deemed Detrimental to NORMAL PLANT OPERATIONS. Pg. 26
	SECURITY	**HG1** HOSTILE ACTION Resulting in Loss of Physical Control of the Facility. Pg. 28	**HS4** HOSTILE ACTION within the PROTECTED AREA Pg. 28	**HA4** HOSTILE ACTION within the OWNER CONTROLLED AREA or Airborne Attack Threat Pg. 28	**HU4** Confirmed SECURITY CONDITION or Threat Which Indicates a Potential Degradation in the Level of Safety of the Plant. Pg. 28
	CR EVAC.		**HS2** Control Room Evacuation Has Been Initiated and Plant Control Cannot be Established. Pg. 28	**HA5** Control Room Evacuation has been Initiated. Pg. 28	
	DISCRETIONARY	**HG2** Other Conditions Existing Which in the Judgment of the Emergency Coordinator Warrant Declaration of General Emergency. Pg. 30	**HS3** Other Conditions Existing Which in the Judgment of the Emergency Coordinator Warrant Declaration of Site Area Emergency. Pg. 30	**HA6** Other Conditions Existing Which in the Judgment of the Emergency Coordinator Warrant Declaration of an Alert. Pg. 30	**HU5** Other Conditions Existing Which in the Judgment of the Emergency Coordinator Warrant Declaration of a Notification of Unusual Event (NOUE). Pg. 30
E – EVENTS RELATED TO ISFSI					**E-HU1** Damage to a loaded cask confinement BOUNDARY. Pg. 34
EPIP Reference		Complete actions listed in EPIP-20101 Section 5.8	Complete actions listed in EPIP-20101 Section 5.7	Complete actions listed in EPIP-20101 Section 5.6	Complete actions listed in EPIP-20101 Section 5.5

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HOT

RECOGNITION CATEGORY R

ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENT

HOT

Rec. Cat.	General Emergency	Site Area Emergency	Alert	Unusual Event
R – ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENT	**RG1 – Basis:** This IC addresses radioactivity releases that result in doses at or beyond the site boundary that exceed the EPA Protective Action Guides (PAGs). Public protective actions will be necessary. Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public and likely involve fuel damage. Threshold Values The monitor list includes effluent monitors on all potential release pathways. Since dose assessment is based on actual meteorology, whereas the monitor reading Threshold is not, the results from these assessments may indicate that the classification is not warranted, or may indicate that a higher classification is warranted. For this reason, emergency implementing procedures call for the timely performance of dose assessments using actual meteorology and release information. If the results of these dose assessments are available when the classification is made (e.g., initiated at a lower classification level), the dose assessment results override the monitor reading EAL.	**RS1 – Basis:** This IC addresses radioactivity releases that result in doses at or beyond the site boundary that exceed 10% of the EPA Protective Action Guides (PAGs). Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public. Threshold Values The monitor list in Threshold #1 includes effluent monitors on all potential release pathways. Since dose assessment is based on actual meteorology, whereas the monitor reading Threshold is not, the results from these assessments may indicate that the classification is not warranted, or may indicate that a higher classification is warranted. For this reason, emergency implementing procedures call for the timely performance of dose assessments using actual meteorology and release information. If the results of these dose assessments are available when the classification is made (e.g., initiated at a lower classification level), the dose assessment results override the monitor reading EAL.	**RA1 – Basis:** The Emergency Coordinator should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time. This IC addresses an actual or substantial potential decrease in the level of safety of the plant as indicated by a radiological release that exceeds regulatory commitments for an extended period of time. Nuclear power plants incorporate features intended to control the release of radioactive effluents to the environment. Further, there are administrative controls established to prevent unintentional releases, or control and monitor intentional releases. These controls are located in the Off-site Dose Calculation Manual (ODCM). The occurrence of extended, uncontrolled radioactive releases to the environment is indicative of a degradation in these features and/or controls. The ODCM multiples are specified in RU1 and RA1 only to distinguish between non-emergency conditions, and from each other. While these multiples obviously correspond to an off-site dose or dose rate, the emphasis in classifying these events is the degradation in the level of safety of the plant, not the magnitude of the associated dose or dose rate. This Threshold includes any release for which a radioactivity discharge permit was not prepared, or a release that exceeds the conditions (e.g., minimum dilution flow, maximum discharge flow, alarm setpoints, etc.) on the applicable permit. Threshold #1 This Threshold is intended for effluent monitoring on non-routine release pathways for which a discharge permit would not normally be prepared. The threshold values of this EAL are to be 200 times the ODCM limit or 100 times the threshold values used in RU1. This is true except for the thresholds given for the liquid release pathways (PRMS-R-18 and PRMS-3[4]-R-19). On both PRMS, the calculated values exceeded the range of the monitors. Therefore, the values used in the EAL are 95% of full scale. This lesser threshold is still indicative of a loss of control of radioactive material well in excess of that constituting an Unusual Event. Threshold #2 This Threshold addresses radioactivity releases, that for whatever reason, cause effluent radiation monitor readings to exceed the threshold identified in the IC established by the radioactivity discharge permit. This value may be associated with a planned batch release, or a continuous release path. Threshold #3 This Threshold addresses uncontrolled releases that are detected by sample analyses, particularly on unmonitored pathways, e.g., spills of radioactive liquids into storm drains, heat exchanger leakage in river water systems, etc. The underlying basis of this Threshold involves the degradation in the level of safety of the plant implied by the uncontrolled release.	**RU1 – Basis:** The Emergency Coordinator should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time. This IC addresses a potential decrease in the level of safety of the plant as indicated by a radiological release that exceeds regulatory commitments for an extended period of time. Nuclear power plants incorporate features intended to control the release of radioactive effluents to the environment. Further, there are administrative controls established to prevent unintentional releases, or control and monitor intentional releases. These controls are located in the Off-site Dose Calculation Manual (ODCM). The occurrence of extended, uncontrolled radioactive releases to the environment is indicative of degradation in these features and/or controls. The ODCM multiples are specified in RU1 and RA1 only to distinguish between non-emergency conditions, and from each other. While these multiples obviously correspond to an off-site dose or dose rate, the emphasis in classifying these events is the degradation in the level of safety of the plant, not the magnitude of the associated dose or dose rate. Releases should not be prorated or averaged. For example, a release exceeding 4x ODCM for 30 minutes does not meet the threshold. This Threshold includes any release for which a radioactivity discharge permit was not prepared, or a release that exceeds the conditions (e.g., minimum dilution flow, maximum discharge flow, alarm setpoints, etc.) on the applicable permit. Threshold #1 This Threshold addresses radioactivity releases, that for whatever reason, cause effluent radiation monitor readings to exceed the threshold identified in the IC. This Threshold is intended for sites that have established effluent monitoring on non-routine release pathways for which a discharge permit would not normally be prepared. Threshold #2 This Threshold addresses radioactivity releases, that for whatever reason, cause effluent radiation monitor readings to exceed the threshold identified in the IC established by the radioactivity discharge permit. This value may be associated with a planned batch release, or a continuous release path. Threshold #3 This Threshold addresses uncontrolled releases that are detected by sample analyses, particularly on unmonitored pathways, e.g., spills of radioactive liquids into storm drains, heat exchanger leakage in river water systems, etc. The underlying basis of this Threshold involves the degradation in the level of safety of the plant implied by the uncontrolled release.

HOT

General Emergency

RG1 Off-site Dose Resulting from an Actual or IMMINENT Release of Gaseous Radioactivity Greater Than 1000 mRem TEDE or 5000 mRem Thyroid CDE for the Actual or Projected Duration of the Release Using Actual Meteorology.

Operating Mode Applicability: All

Threshold Values:

Note: The Emergency Coordinator should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time. If dose assessment results are available, declaration should be based on dose assessment instead of radiation monitor values. Do not delay declaration awaiting dose assessment results.

1. VALID reading on ANY of the following radiation monitors greater than the value shown for 15 minutes or longer:

Monitor		Pathway	Value
RAD-6304		Plant Vent SPING	2.8E+0 µCi/cc
RAD 6418	Unit 4 Via Plant Vent	Spent Fuel Pool Vent SPING	1.1E+1 µCi/cc
MSL 3[4] RAD-6426A, B, C (DAM-1)		Main Steam Line	2.4E+2 µCi/cc
RAD-3[4]-6417		SJAE SPING	7.5E+3 µCi/cc

OR

2. Dose assessment using actual meteorology indicates doses greater than 1000 mRem TEDE or 5000 mRem thyroid CDE at or beyond the site boundary.

OR

3. Field survey results indicate closed window dose rates greater than 1000 mR/hr expected to continue for 60 minutes or longer; or analyses of field survey samples indicate thyroid CDE greater than 5000 mRem for one hour of inhalation, at or beyond site boundary.

GUIDANCE BOX FOR USE OF SPING MONITORS IN RADIOLOGICAL EFFLUENT TABLES

The calculations for the SPING monitors listed are based on NOBLE GAS levels; use SPING channels 5, 7, and 9 as appropriate for the indicated values in the tables. (Ref. DBD PTN-NEI-99-01)

Channel		Range	Typical Reading
Ch-5	Low Range	1.0E-07 - 6.0E-02	5.0E-07
Ch-7	Mid Range	2.5E-02 - 4.0E+02	1.0E-04
Ch-9	High Range	1.0E+00 - 1.0E+05	1.0E-01

DEFINITION BOX

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.

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

Site Area Emergency

RS1 Off-site Dose Resulting from an Actual or IMMINENT Release of Gaseous Radioactivity Greater Than 100 mRem TEDE or 500 mRem Thyroid CDE for the Actual or Projected Duration of the Release.

Operating Mode Applicability: All

Threshold Values:

Note: *The Emergency Coordinator should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time. If dose assessment results are available, declaration should be based on dose assessment instead of radiation monitor values. Do not delay declaration awaiting dose assessment results.*

1. VALID reading on ANY of the following radiation monitors greater than the value shown for 15 minutes or longer:

Monitor		Pathway	Value
RAD-6304		Plant Vent SPING	2.8E-1 µCi/cc
RAD 6418	Unit 4 Via Plant Vent	Spent Fuel Pool Vent SPING	1.1E+0 µCi/cc
MSL 3[4] RAD-6426A, B, C (DAM-1)		Main Steam Line	2.4E+1 µCi/cc
RAD-3[4]-6417		SJAE SPING	7.5E+2 µCi/cc

OR

2. Dose assessment using actual meteorology indicates doses greater than 100 mRem TEDE or 500 mRem thyroid CDE at or beyond the site boundary.

OR

3. Field survey results indicate closed window dose rates greater than 100 mR/hr expected to continue for 60 minutes or longer; or analyses of field survey samples indicate thyroid CDE greater than 500 mRem for one hour of inhalation, at or beyond the site boundary.

GUIDANCE BOX FOR USE OF SPING MONITORS IN RADIOLOGICAL EFFLUENT TABLES

The calculations for the SPING monitors listed are based on NOBLE GAS levels; use SPING channels 5, 7, and 9 as appropriate for the indicated values in the tables. (Ref. DBD PTN-NEI-99-01)

Channel		Range	Typical Reading
Ch-5	Low Range	1.0E-07 - 6.0E-02	5.0E-07
Ch-7	Mid Range	2.5E-02 - 4.0E+02	1.0E-04
Ch-9	High Range	1.0E+00 - 1.0E+05	1.0E-01

Alert

RA1 Any Release of Gaseous or Liquid Radioactivity to the Environment Greater Than 200 Times the Radiological Effluent Technical Specifications/ODCM for 15 Minutes or Longer.

Operating Mode Applicability: All

Threshold Values:

Note: The Emergency Coordinator should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.

1. VALID reading on ANY of the following radiation monitors greater than the value shown for 15 minutes or longer:

Monitor		Pathway	Value
RAD-6304		Plant Vent SPING	3.2E-2 µCi/cc
RAD 6418	Unit 4 Via Plant Vent	Spent Fuel Pool Vent SPING	1.3E-1 µCi/cc
MSL 3[4] RAD-6426A, B, C (DAM-1)		Main Steam Line	2.8E+0 µCi/cc
RAD-3[4]-6417		SJAE SPING	8.6E+1 µCi/cc
PRMS R-18		Liquid Rad Waste Effluent Line	2.4E+5 cpm
PRMS-3[4]-R-19		Any one Steam Generator Blowdown	2.4E+5 cpm

OR

2. VALID reading on any effluent monitor reading that greater than 200 times the alarm setpoint established by a current radioactivity discharge permit for 15 minutes or longer.

OR

3. Confirmed sample analyses for gaseous or liquid releases indicates concentrations or release rates greater than 200 times ODCM limits for 15 minutes or longer.

GUIDANCE BOX FOR USE OF SPING MONITORS IN RADIOLOGICAL EFFLUENT TABLES

The calculations for the SPING monitors listed are based on NOBLE GAS levels; use SPING channels 5, 7, and 9 as appropriate for the indicated values in the tables. (Ref. DBD PTN-NEI-99-01)

Channel		Range	Typical Reading
Ch-5	Low Range	1.0E-07 - 6.0E-02	5.0E-07
Ch-7	Mid Range	2.5E-02 - 4.0E+02	1.0E-04
Ch-9	High Range	1.0E+00 - 1.0E+05	1.0E-01

Unusual Event

RU1 Any Release of Gaseous or Liquid Radioactivity to the Environment Greater Than 2 Times the ODCM for 60 Minutes or Longer.

Operating Mode Applicability: All

Threshold Values:

Note *The Emergency Coordinator should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.*

1. VALID reading on ANY of the following radiation monitors greater than the value shown for 60 minutes or longer.

Monitor		Pathway	Value
RAD-6304		Plant Vent SPING	3.2E-4 µCi/cc
RAD 6418	Unit 4 Via Plant Vent	Spent Fuel Pool Vent SPING	1.3E-3 µCi/cc
MSL 3[4] RAD-6426A, B, C (DAM-1)		N/A	N/A
RAD-3[4]-6417		SJAE SPING	8.6E-1 µCi/cc
PRMS R-18		Liquid Rad Waste Effluent Line	1.6E+4 cpm
PRMS-3[4]-R-19		Any one Steam Generator Blowdown	1.6E+4 cpm
PRMS R-14		Plant Vent (PV)	6.4E+4 cpm

OR

2. VALID reading on any effluent monitor reading greater than 2 times the alarm setpoint established by a current radioactivity discharge permit for 60 minutes or longer.

OR

3. Confirmed sample analyses for gaseous or liquid releases indicates concentrations or release rates greater than 2 times ODCM Limits for 60 minutes or longer.

GUIDANCE BOX FOR USE OF SPING MONITORS IN RADIOLOGICAL EFFLUENT TABLES

The calculations for the SPING monitors listed are based on NOBLE GAS levels; use SPING channels 5, 7, and 9 as appropriate for the indicated values in the tables. (Ref. DBD PTN-NEI-99-01)

Channel		Range	Typical Reading
Ch-5	Low Range	1.0E-07 - 6.0E-02	5.0E-07
Ch-7	Mid Range	2.5E-02 - 4.0E+02	1.0E-04
Ch-9	High Range	1.0E+00 - 1.0E+05	1.0E-01

Recognition Category

R – ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENT

RADIOLOGICAL EFFLUENT

HOT

Rec. Cat.	General Emergency	Site Area Emergency	Alert	Unusual Event
R – ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENT			**RA2 – Basis:** This IC addresses increases in radiation dose rates within plant buildings, and may be a precursor to a radioactivity release to the environment. These events represent a loss of control over radioactive material and represent an actual or substantial potential degradation in the level of safety of the plant. <u>Threshold #1</u> A specific value is not specified in that the elevation of any fuel bundle in transit must be considered for evaluating this condition. <u>Threshold #2</u> This Threshold addresses radiation monitor indications of fuel uncovery and/or fuel damage. Increased ventilation monitor readings may be indication of a radioactivity release from the fuel, confirming that damage has occurred. Increased background at the ventilation monitor due to water level decrease may mask increased ventilation exhaust airborne activity and needs to be considered. While a radiation monitor could detect an increase in dose rate due to a drop in the water level, it might not be a reliable indication of whether or not the fuel is covered. Escalation of this emergency classification level, if appropriate, would be based on RS1 or RG1. **RA3 – Basis:** This IC addresses increased radiation levels that: impact continued operation in areas requiring continuous occupancy to maintain safe operation or to perform a safe shutdown. The cause and/or magnitude of the increase in radiation levels is not a concern of this IC. The Emergency Coordinator must consider the source or cause of the increased radiation levels and determine if any other IC may be involved. Areas requiring continuous occupancy are the Control Room and CAS The value of 15 mR/hr is derived from the GDC 19 value of 5 rem in 30 days with adjustment for expected occupancy times. Although Section III.D.3 of NUREG-0737, “Clarification of TMI Action Plan Requirements”, provides that the 15 mR/hr value can be averaged over the 30 days, the value is used here without averaging, as a 30 day duration implies an event potentially more significant than an Alert.	**RU2 – Basis:** This IC addresses increased radiation levels as a result of water level decreases above irradiated fuel or events that have resulted, or may result, in UNPLANNED increases in radiation dose rates within plant buildings. These radiation increases represent a loss of control over radioactive material and represent a potential degradation in the level of safety of the plant. Threshold #1 The refueling pathway is the combination of the refueling cavity, transfer canal, and spent fuel pool. While a radiation monitor could detect an increase in dose rate due to a drop in the water level, it might not be a reliable indication of whether or not the fuel is covered. For refueling events where the water level drops below the RPV flange classification would be via CU2. This event escalates to an Alert per RA2 if irradiated fuel outside the reactor vessel is uncovered. For events involving irradiated fuel in the reactor vessel, escalation would be via the Fission Product Barrier Table for events in operating modes 1-4. Threshold #2 This Threshold addresses increases in plant radiation levels that represent a loss of control of radioactive material resulting in a potential degradation in the level of safety of the plant. This Threshold excludes radiation level increases that result from planned activities such as use of radiographic sources and movement of radioactive waste materials. A specific list of ARMs is not required as it would restrict the applicability of the Threshold. The intent is to identify loss of control of radioactive material in any monitored area. This event escalates to an Alert per RA3 if the increase in dose rates impedes personnel access necessary for safe operation. **GUIDANCE BOX FOR USE OF TEMPORARY MONITORS IN ABNORMAL RADIATION TABLES** Portable radiation monitors staged as compensatory actions for OOS monitors may be used in place of installed equipment, only if data is available to the Control Room, for the purpose of this EAL.

HOT

General Emergency	Site Area Emergency	Alert	Unusual Event	Recognition Category

RA2 Damage to Irradiated Fuel or Loss of Water Level that Has Resulted or Will Result in the Uncovering of Irradiated Fuel Outside the Reactor Vessel.

Operating Mode Applicability: All

Threshold Values:

1. A water level drop in the reactor refueling cavity, spent fuel pool or fuel transfer canal that will result in irradiated fuel becoming uncovered.

   OR

2. A VALID elevated reading of greater than or equal to 1.0 E+4 mR/hr on ANY of the following due to damage to irradiated fuel or loss of water level:

Unit 3	Unit 4
RI-3-1402B, UNIT 3 CONTAINMENT OPERATING FLOOR	RI-4-1405B, UNIT 4 CONTAINMENT OPERATING FLOOR
RI-3-1407B, UNIT 3 SPENT FUEL PIT CANAL AREA	RI-4-1408B, UNIT 4 SPENT FUEL PIT CANAL AREA
RI-3-1421B UNIT 3 SPENT FUEL PIT NORTH WALL	RI-4-1422B UNIT 4 SPENT FUEL PIT SOUTH WALL

RU2 UNPLANNED Rise in Plant Radiation Levels.

Operating Mode Applicability: All

Threshold Values:

1. UNPLANNED water level drop in a reactor refueling pathway as indicated by LT-3[4]-651 56' 10".

   AND

   VALID Area Radiation Monitor reading rise on any of the following:

Unit 3	Unit 4
RI-3-1402B, UNIT 3 CONTAINMENT OPERATING FLOOR	RI-4-1405B, UNIT 4 CONTAINMENT OPERATING FLOOR
RI-3-1407B, UNIT 3 SPENT FUEL PIT CANAL AREA	RI-4-1408B, UNIT 4 SPENT FUEL PIT CANAL AREA
RI-3-1421B UNIT 3 SPENT FUEL PIT NORTH WALL	RI-4-1422B UNIT 4 SPENT FUEL PIT SOUTH WALL

   OR

2. UNPLANNED VALID Area Radiation Monitor readings or survey results indicate a rise by a factor of 1000 over normal* levels.

* Normal can be considered as the highest reading in the past twenty-four hours excluding the current peak value.

GUIDANCE BOX FOR RU2, Threshold 1

This event addresses increased radiation levels as a result of decreasing water level above irradiated fuel. A Spent Fuel Pool level resulting from routine evaporation accompanied by normal* and expected area radiation levels are not subject to this EAL.

DEFINITION BOX

UNPLANNED – A parameter change or an event that is not the result of an intended evolution and requires corrective or mitigative actions.

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

RA3 Rise in Radiation Levels Within the Facility That Impedes Operation of Systems Required to Maintain Safe Plant Safety Functions.

Operating Mode Applicability: All

Threshold Values:

1. Dose rate greater than 15 mR/hr in ANY of the following areas requiring continuous occupancy to maintain plant safety functions:
   - Control Room
   - CAS

GUIDANCE BOX FOR USE OF MONITORS IN ABNORMAL RADIATION TABLES

Area Radiation monitors in the Control Room indicate in mR/hr up to 100 mR/hr. Monitors automatically switch to Rem/hr as noted at the monitor.

GUIDANCE BOX FOR USE OF TEMPORARY MONITORS IN ABNORMAL RADIATION TABLES

Portable radiation monitors staged as compensatory actions for OOS monitors should be used in place of installed equipment for the purpose of this EAL.

R – ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENT

ABNORMAL RAD LEVELS

HOT

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RECOGNITION CATEGORY F

FISSION PRODUCT BARRIER DEGRADATION

HOT

FISSION PRODUCT BARRIER TABLE WORKSHEET
(APPLICABILITY: Modes 1, 2, 3, & 4 ONLY)

FUEL CLAD BARRIER		REACTOR COOLANT SYSTEM BARRIER		PRIMARY CONTAINMENT BARRIER	
1. Critical Safety Function Status		**1. Critical Safety Function Status**		**1. Critical Safety Function Status**	
1. CSF Status Tree for **Core Cooling Red** Conditions Met. (3/4 FR-C.1 Required)	1. CSF Status Tree for **Core Cooling Orange** Conditions Met. (3/4 FR-C.2 Required) **OR** 2. CSF Status Tree for **Heat Sink Red** Conditions Met. (3/4 FR-H.1 Required)	**GUIDANCE BOX FOR SAFETY FUNCTION STATUS IN ALL THREE BARRIERS** **IF** directed to perform any mitigating step in applicable Function Restoration Procedures, **THEN** conditions have been met.	1. CSF Status Tree for **Integrity Red** Conditions Met. (3/4 FR-P.1 Required) **OR** 2. CSF Status Tree for **Heat Sink Red** Conditions Met. (3/4 FR-H.1 Required)	Not Applicable	1. CSF Status Tree for **Containment Red** Conditions Met. (3/4 FR-Z.1 Required)
OR		**OR**		**OR**	
2. Primary Coolant Activity Level		**2. RCS Leak Rate**		**2. Containment Pressure**	
1. Coolant activity greater than: 300 uCi/gm Dose Equivalent I-131. **GUIDANCE BOX** With Letdown in service, the R-3[4]-20 dose rate threshold of 2.5 R/hr is equivalent to the primary coolant activity level threshold of 300 µCi/gm DEQ I-131 in determining loss of the Fuel Clad Barrier. (Calculation PTN-9FJF-01-027, Rev. 0) See also SU4, Fuel Clad Degradation.	Not Applicable	1. RCS leak rate greater than available makeup capacity as indicated by a loss of RCS subcooling based on core exit TCs - LESS THAN 19°F[73°F]. **GUIDANCE BOX** See also SU5, RCS Leakage.	1. RCS leak rate indicated by greater than maximum charging with Letdown isolated. **GUIDANCE BOX** Isolation of Letdown is to distinguish between RCS leakage and CVCS leakage and is performed when procedurally required.	1. A containment pressure rise followed by a rapid unexplained drop in containment pressure. **OR** 2. Containment pressure or sump level response not consistent with LOCA conditions.	1. Containment pressure greater than 55 psig and rising. **OR** 2. 4% H_2 in Containment **OR** 3.a. Containment Pressure greater than 20 psig. **AND** b. Less than one full train of depressurization equipment operating.
OR		**OR**		**OR**	
3. Core Exit Thermocouple Readings		**3. Not Applicable**		**3. Core Exit Thermocouple Reading**	
1. Core exit thermocouples reading greater than 1200ºF. **GUIDANCE BOX** At least five (5) Core Exit Thermocouples must exceed the threshold per F-0.	1. Core exit thermocouples reading greater than 700 ºF. **GUIDANCE BOX** At least five (5) Core Exit Thermocouples must exceed the threshold per F-0.	Not Applicable	Not Applicable	**GUIDANCE BOX** At least five (5) Core Exit Thermocouples must exceed the threshold per F-0. RVLMS Plenum indicating 0% indicates potential core uncovery.	1.a. Core exit thermocouples in excess of 1200 ºF. **AND** b. FR-C.1 **NOT** effective within 15 minutes. **OR** 2.a. Core exit thermocouples in excess of 700 ºF. **AND** b. RVLMS indicates head voids. **AND** c. FR-C.2 **NOT** effective within 15 minutes.
OR		**OR**		**OR**	
4. Reactor Vessel Water Level		**4. SG Tube Rupture**		**4. SG Secondary Side Release with P-to-S Leakage**	
Not Applicable	1. RVLMS (QSPDS) 0% Plenum Indicated. **GUIDANCE BOX** RVLMS Plenum indicating 0% indicates potential core uncovery.	1. RUPTURED SG results in a SI Actuation	Not Applicable	1. RUPTURED SG is also FAULTED outside of Containment. **OR** 2.a. Primary-to-Secondary leak rate greater than 10 gpm. **AND** b. UNISOLABLE steam release from affected SG to the environment.	Not Applicable
OR		**OR**		**OR**	
5. Not Applicable		**5. Not Applicable**		**5. CNTMT Isolation Failure or Bypass**	
Not Applicable	Not Applicable	Not Applicable	Not Applicable	1.a. Failure of all valves in any one line to close . **AND** b. Direct downstream pathway to the environment exists after containment isolation signal.	Not Applicable
OR		**OR**		**OR**	
6. Containment Radiation Monitoring		**6. Containment Radiation Monitoring**		**6. Containment Radiation Monitoring**	
1. CHRRM reading greater than 5.4 E+3 R/hr	Not Applicable	1. Containment Mezzanine radiation monitor RI-3-1401B [RI-4-1404B] reading greater than 10 mR/hr.	**GUIDANCE BOX** This threshold value indicates the release of reactor coolant to the containment assuming the instantaneous release and dispersal of the reactor coolant associated with **normal operating concentrations (i.e. within T.S. limits for RCS Activity).** This indication is not valid with a fuel clad barrier challenge. Radiation from adjacent piping and components containing elevated reactor coolant activity may cause radiation monitoring instrumentation inside containment to rise.	Not Applicable	1. CHRRM reading greater than 2.2 E+4 R/hr.
OR		**OR**		**OR**	
7. Emergency Coordinator Judgment		**7. Emergency Coordinator Judgment**		**7. Emergency Coordinator Judgment**	
1. Any condition in the opinion of the Emergency Coordinator that indicates Loss OR Potential Loss of the Fuel Clad Barrier.		1. Any condition in the opinion of the Emergency Coordinator that indicates Loss OR Potential Loss of the RCS Barrier.		1. Any condition in the opinion of the Emergency Coordinator that indicates Loss OR Potential Loss of the Containment Barrier.	
☐ **FUEL CLAD BARRIER**		☐ **REACTOR COOLANT SYSTEM BARRIER**		☐ **PRIMARY CONTAINMENT BARRIER**	
☐ **LOSS**	☐ **POTENTIAL LOSS**	☐ **LOSS**	☐ **POTENTIAL LOSS**	☐ **LOSS**	☐ **POTENTIAL LOSS**

ALERT
☐ **FA1** **ANY Loss or ANY Potential Loss of EITHER Fuel Clad OR RCS**

UNUSUAL EVENT
☐ **FU1** **ANY Loss or ANY Potential Loss of Containment.**

SITE AREA EMERGENCY
☐ **FS1** **Loss or Potential Loss of ANY two Barriers**

GENERAL EMERGENCY
☐ **FG1** **Loss of ANY two Barriers AND Loss or Potential Loss of the Third Barrier**

HOT

FISSION PRODUCT BARRIER TABLE BASIS
(APPLICABILITY: Modes 1, 2, 3, & 4 ONLY)

FUEL CLAD BARRIER THRESHOLDS - BASIS

1. Critical Safety Function Status

Loss Threshold #1
Core Cooling - RED indicates significant superheating and core uncovery and is considered to indicate loss of the Fuel Clad Barrier.
Potential Loss Threshold #1
Core Cooling - ORANGE indicates subcooling has been lost and that some clad damage may occur.
Potential Loss Threshold #2
Heat Sink - RED when heat sink is required indicates the ultimate heat sink function is under extreme challenge.

2. Primary Coolant Activity Level

The 300 μCi/gm I-131 equivalent indicates that this amount of coolant activity is well above that expected for iodine spikes and corresponds to less than 5% fuel clad damage. This amount of radioactivity indicates significant clad damage and thus the Fuel Clad Barrier is considered lost.
There is no Potential Loss threshold associated with this item.

3. Core Exit Thermocouple Readings

Loss Threshold #1

The Threshold Value corresponds to significant superheating of the coolant.

Potential Loss Threshold #1

The Threshold Value corresponds to loss of subcooling.

4. Reactor Vessel Water Level

There is no Loss threshold associated with this item.
The Threshold Value for the Potential Loss threshold corresponds to the top of the active fuel.

5. Not Applicable

Not Applicable
(included for numbering consistency between barrier tables).

6. Containment Radiation Monitoring

The Threshold Value indicates the release of reactor coolant, with elevated activity indicative of fuel damage, into the containment assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with a concentration of 300 μCi/gm dose equivalent I-131 into the containment atmosphere.
Reactor coolant concentrations of this magnitude are several times larger than the maximum concentrations (including iodine spiking) allowed within technical specifications and are therefore indicative of fuel damage.
This value is higher than that specified for RCS barrier Loss threshold #1. Thus, this threshold indicates a loss of both the Fuel Clad barrier and RCS barrier that appropriately escalates the emergency classification level to a Site Area Emergency.
There is no Potential Loss threshold associated with this item.

7. Emergency Coordinator Judgment

These thresholds address any other factors that are to be used by the Emergency Coordinator in determining whether the Fuel Clad barrier is lost or potentially lost. In addition, the inability to monitor the barrier should also be incorporated in this threshold as a factor in Emergency Coordinator judgment that the barrier may be considered lost or potentially lost.

RCS BARRIER THRESHOLDS - BASIS

1. Critical Safety Function Status

Potential Loss Threshold #1
RCS Integrity - RED indicates an extreme challenge to the safety function derived from appropriate instrument readings.
Potential Loss Threshold #2
Heat Sink - RED when heat sink is required indicates the ultimate heat sink function is under extreme challenge.
There is no Loss threshold associated with this item.

2. RCS Leak Rate

Loss Threshold #1
This threshold addresses conditions where leakage from the RCS is greater than available inventory control capacity such that a loss of subcooling has occurred. The loss of subcooling is the fundamental indication that the Inventory Control Systems are inadequate in maintaining RCS pressure and inventory against the mass loss through the leak.
Potential Loss Threshold #1
This threshold is based on the apparent inability to maintain normal liquid inventory within the Reactor Coolant System (RCS) by normal operation of the Chemical and Volume Control System. Isolating letdown is a standard abnormal operating procedure action and may prevent unnecessary classifications when a non-RCS leakage path such as a CVCS leak exists. The intent of this condition is met if attempts to isolate Letdown are NOT successful.

GUIDANCE BOX
Primary Containment Barrier Threshold Basis, Potential Loss Threshold #3

With Containment Pressure ≥ 20 psig, the minimum containment depressurization equipment required is 1 CSP and 2 ECCs in operation.

Ref. USAR Chapter 6, Engineered Safety Features / Ref. USAR Chapter 14, Safety Analysis

3. Not Applicable

Not Applicable
(included for numbering consistency between barrier tables).

4. SG Tube Rupture

This threshold addresses the full spectrum of Steam Generator (SG) tube rupture events in conjunction with Containment barrier Loss thresholds. It addresses RUPTURED SG(s) for which the leakage is large enough to cause actuation of ECCS (SI). This is consistent to the RCS leak rate barrier Potential Loss threshold.
By itself, this threshold will result in the declaration of an Alert. However, if the SG is also FAULTED (i.e., two barriers failed), the declaration escalates to a Site Area Emergency per Containment barrier Loss thresholds.
There is no Potential Loss threshold associated with this item.

DEFINITION BOX

FAULTED – In a steam generator, the existence of secondary side leakage that results in an uncontrolled drop in steam generator pressure or the steam generator being completely depressurized.

RUPTURED – In a steam generator, existence of primary-to-secondary leakage of a magnitude sufficient to require or cause a reactor trip and safety injection.

UNISOLABLE – A breach or leak that cannot be promptly isolated.

5. Not Applicable

Not Applicable
(included for numbering consistency between barrier tables).

GUIDANCE BOX FOR RADIATION MONITORING

Classifications are based on various combinations of function or barrier challenges. Some indications may be indicative of several conditions or events and should be used in combination with all available indications to determine barrier challenges (ex. radiation monitor readings may be valid indications of fuel clad failures or RCS leakage or both and should be evaluated given all available indications.) (NEI 99-01, Rev. 5, Page 2 of 167)

6. Containment Radiation Monitoring

The Threshold Value indicates the release of reactor coolant to the containment assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with normal operating concentrations (i.e., within T/S) into the containment atmosphere.
This value is less than that specified for Fuel Clad barrier Threshold 6.1. Thus, this threshold would be indicative of a RCS leak only. If the radiation monitor reading increased to that specified by Fuel Clad barrier threshold, fuel damage would also be indicated.
There is no Potential Loss threshold associated with this item.

GUIDANCE BOX

This threshold value indicates the release of reactor coolant to the containment assuming the instantaneous release and dispersal of the reactor coolant associated with **normal operating concentrations (i.e. within T.S. limits for RCS Activity).** This indication is not valid with a fuel clad barrier challenge. Radiation from adjacent piping and components containing elevated reactor coolant activity may cause radiation monitoring instrumentation inside containment to rise.

7. Emergency Coordinator Judgment

These thresholds address any other factors that are to be used by the Emergency Coordinator in determining whether the RCS barrier is lost or potentially lost. In addition, the inability to monitor the barrier should also be incorporated in this threshold as a factor in Emergency Coordinator judgment that the barrier may be considered lost or potentially lost.

PRIMARY CONTAINMENT BARRIER THRESHOLDS - BASIS

1. Critical Safety Function Status

RED path indicates an extreme challenge to the safety function derived from appropriate instrument readings and/or sampling results, and thus represents a potential loss of containment.
Conditions leading to a containment RED path result from RCS barrier and/or Fuel Clad Barrier Loss. Thus, this threshold is primarily a discriminator between Site Area Emergency and General Emergency representing a potential loss of the third barrier.
There is no Loss threshold associated with this item.

2. Containment Pressure

Loss Thresholds #1 or #2
Rapid unexplained loss of pressure (i.e., not attributable to containment spray or condensation effects) following an initial pressure increase from a primary or secondary high energy line break indicates a loss of containment integrity. Containment pressure and sump levels should increase as a result of mass and energy release into containment from a LOCA. Thus, sump level or pressure not increasing indicates containment bypass and a loss of containment integrity.
This indicator relies on operator recognition of an unexpected response for the condition and therefore does not have a specific value associated with it. The unexpected response is important because it is the indicator for a containment bypass condition.
Potential Loss Threshold #1
The Threshold Value is the containment design pressure.
Potential Loss Threshold #2
Existence of an explosive mixture means a hydrogen and oxygen concentration of at least the lower deflagration limit curve exists. The indications of potential loss under this EAL corresponds to some of those leading to the RED path in potential loss threshold 1.1 above and may be declared by those sites using CSFSTs.
Potential Loss Threshold #3
This threshold represents a potential loss of containment in that the Containment Heat Removal/Depressurization System are either lost or performing in a degraded manner, as indicated by containment pressure greater than the setpoint at which the equipment was supposed to have actuated.

3. Core Exit Thermocouple Reading

There is no Loss threshold associated with this item.
Potential Loss Threshold #1
The conditions in these thresholds represent an IMMINENT core melt sequence which, if not corrected, could lead to vessel failure and an increased potential for containment failure. In conjunction with the Core Cooling and RCS Leakage criteria in the Fuel and RCS barrier columns, this threshold would result in the declaration of a General Emergency -- loss of two barriers and the potential loss of a third. If the function restoration procedures are ineffective, there is no "success" path.
The function restoration procedures are those emergency operating procedures that address the recovery of the core cooling critical safety functions. The procedure is considered effective if the temperature is decreasing or if the vessel water level is increasing.
Severe accident analyses (e.g., NUREG-1150) have concluded that function restoration procedures can arrest core degradation within the reactor vessel in a significant fraction of the core damage scenarios, and that the likelihood of containment failure is very small in these events. Fifteen minutes provides a reasonable period to allow function restoration procedures to arrest the core melt sequence.
Whether or not the procedures will be effective should be apparent within 15 minutes. The Emergency Coordinator should make the declaration as soon as it is determined that the procedures have been, or will be ineffective.
Potential Loss Threshold #2
The Threshold Value is consistent with the Emergency Operating Procedures.

4. SG Secondary Side Release with P-to-S Leakage

The loss threshold recognizes that SG tube leakage can represent a bypass of the Containment barrier as well as a loss of the RCS barrier.
Users should realize that the two loss thresholds could be considered redundant. This was recognized during the development process. The inclusion of an threshold that uses Emergency Procedure commonly used terms like "RUPTURED and FAULTED" adds to the ease of the classification process and has been included based on this human factor concern.
This threshold results in a NOUE for smaller breaks that; (1) do not exceed the normal charging capacity threshold in RCS leak rate barrier Potential Loss threshold, or (2) do not result in ECCS actuation in RCS SG tube rupture barrier Loss threshold. For larger breaks, RCS barrier threshold criteria would result in an Alert. For SG tube ruptures which may involve multiple steam generators or unisolable secondary line breaks, this threshold would exist in conjunction with RCS barrier thresholds and would result in a Site Area Emergency. Escalation to General Emergency would be based on "Potential Loss" of the Fuel Clad Barrier.
Loss Threshold #1
This threshold addresses the condition in which a RUPTURED steam generator is also FAULTED. This condition represents a bypass of the RCS and containment barriers and is a subset of the second threshold. In conjunction with RCS leak rate barrier loss threshold, this would always result in the declaration of a Site Area Emergency.
Loss Threshold #2
This threshold addresses SG tube leaks that exceed 10 gpm in conjunction with an UNISOLABLE release path to the environment from the affected steam generator. The threshold for establishing the UNISOLABLE secondary side release is intended to be a prolonged release of radioactivity from the RUPTURED steam generator directly to the environment. This could be expected to occur when the main condenser is unavailable to accept the contaminated steam (i.e., SG tube rupture with concurrent loss of off-site power and the RUPTURED steam generator is required for plant cooldown or a stuck open relief valve). If the main condenser is available, there may be releases via air ejectors, gland seal exhausters, and other similar controlled, and often monitored, pathways. These pathways do not meet the intent of an UNISOLABLE release path to the environment. These minor releases are assessed using Abnormal Rad Levels / Radiological Effluent ICs.

5. CNTMT Isolation Failure or Bypass

This threshold addresses incomplete containment isolation that allows direct release to the environment.
The use of the modifier "direct" in defining the release path discriminates against release paths through interfacing liquid systems. The existence of an in–line charcoal filter does not make a release path indirect since the filter is not effective at removing fission product noble gases. Typical filters have an efficiency of 95-99% removal of iodine. Given the magnitude of the core inventory of iodine, significant releases could still occur. In addition, since the fission product release would be driven by boiling in the reactor vessel, the high humidity in the release stream can be expected to render the filters ineffective in a short period.
There is no Potential Loss threshold associated with this item.

6. Containment Radiation Monitoring

There is no Loss threshold associated with this item.
The Threshold Value indicates significant fuel damage well in excess of the thresholds associated with both loss of Fuel Clad and loss of RCS barriers. A major release of radioactivity requiring off-site protective actions from core damage is not possible unless a major failure of fuel cladding allows radioactive material to be released from the core into the reactor coolant.
The Threshold Value indicates the release of reactor coolant to the containment assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with 20% fuel clad damage into the containment atmosphere.
Regardless of whether containment is challenged, this amount of activity in containment, if released, could have such severe consequences that it is prudent to treat this as a potential loss of containment, such that a General Emergency declaration is warranted.

7. Emergency Coordinator Judgment

These thresholds address any other factors that are to be used by the Emergency Coordinator in determining whether the Containment barrier is lost or potentially lost. In addition, the inability to monitor the barrier should also be incorporated in this threshold as a factor in Emergency Coordinator judgment that the barrier may be considered lost or potentially lost.
The Containment barrier should not be declared lost or potentially lost based on exceeding Technical Specification action statement criteria, unless there is an event in progress requiring mitigation by the Containment barrier. When no event is in progress (Loss or Potential Loss of either Fuel Clad and/or RCS) the Containment barrier status is addressed by Technical Specifications.

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HOT

RECOGNITION CATEGORY S

SYSTEM MALFUNCTIONS

HOT

Rec. Cat.	General Emergency	Site Area Emergency	Alert	Unusual Event
S – SYSTEM MALFUNCTIONS	**SG1 – Basis:** Loss of all AC power to emergency busses compromises all plant safety systems requiring electric power including RHR, ECCS, Containment Heat Removal, and the Ultimate Heat Sink. Prolonged loss of all AC power to emergency busses will lead to loss of fuel clad, RCS, and containment, thus warranting declaration of a General Emergency. The 4 hours to restore AC power is based on a site blackout coping analysis performed in conformance with 10 CFR 50.63 and Regulatory Guide 1.155, "Station Blackout,". Although this IC may be viewed as redundant to the Fission Product Barrier IC, its inclusion is necessary to better assure timely recognition and emergency response. This IC is specified to assure that in the unlikely event of a prolonged station blackout, timely recognition of the seriousness of the event occurs and that declaration of a General Emergency occurs as early as is appropriate, based on a reasonable assessment of the event trajectory. The likelihood of restoring at least one emergency bus should be based on a realistic appraisal of the situation since a delay in an upgrade decision based on only a chance of mitigating the event could result in a loss of valuable time in preparing and implementing public protective actions. In addition, under these conditions, fission product barrier monitoring capability may be degraded. Although it may be difficult to predict when power can be restored, a reasonable indication of how quickly the need to declare a General Emergency based on two major considerations: 1. Are there any present indications that core cooling is already degraded to the point that loss or potential loss of Fission Product Barriers is IMMINENT? 2. If there are no present indications of such core cooling degradation, how likely is it that power can be restored in time to assure that a loss of two barriers with a potential loss of the third barrier can be prevented? Thus, indication of continuing core cooling degradation must be based on Fission Product Barrier monitoring with particular emphasis on Emergency Coordinator judgment as it relates to IMMINENT loss or potential loss of fission product barriers and degraded ability to monitor fission product barriers.	**SS1 – Basis:** Loss of all AC power to emergency busses compromises all plant safety systems requiring electric power including RHR, ECCS, Containment Heat Removal, and the Ultimate Heat Sink. Prolonged loss of all AC power to emergency busses will lead to loss of Fuel Clad, RCS, and Containment, thus this event can escalate to a General Emergency. Fifteen minutes was selected as a threshold to exclude transient or momentary losses of off-site power. An SBO crosstie, from the unaffected unit through each unit's D 4KV buses, that can be accomplished using 3[4]EOP-ECA-0.0 within the 15 minutes will provide an alternate power source. Escalation to General Emergency is via Fission Product Barrier Degradation or IC SG1, "Prolonged Loss of All Off-site Power and Prolonged Loss of All On-site AC Power." **SS2- Basis:** Under these conditions, the reactor is producing more heat than the maximum decay heat load for which the safety systems are designed and efforts to bring the reactor subcritical are unsuccessful. A Site Area Emergency is warranted because conditions exist that lead to IMMINENT loss or potential loss of both fuel clad and RCS. Manual trip actions taken in the Control Room are any set of actions by the reactor operator(s) at which causes or should cause control rods to be rapidly inserted into the core and shuts down the reactor. Manual trip actions are not considered successful if action away from the reactor control console is required to trip the reactor. This Threshold is still applicable even if actions taken away from Control Room are successful in shutting the reactor down because the design limits of the fuel may have been exceeded or because of the gross failure of the Reactor Protection System to shutdown the plant. The Subcriticality Red Path is met when power is not reduced below 5% after a trip. Escalation of this event to a General Emergency would be due to a prolonged condition leading to an extreme challenge to either core-cooling or heat removal.	**SA5 – Basis:** The condition indicated by this IC is the degradation of the off-site and on-site AC power systems such that any additional single failure would result in a unit blackout. This condition could occur due to a loss of off-site power with a concurrent failure of all but one emergency generator to supply power to its emergency busses. Another related condition could be the loss of all off-site power and loss of on-site emergency generators with only one train of emergency busses being backfed from the unit main generator, or the loss of on-site emergency generators with only one train of emergency busses being backfed from off-site power. The subsequent loss of this single power source would escalate the event to a Site Area Emergency in accordance with SS1. Fifteen minutes was selected as a threshold to exclude transient or momentary losses of power. An SBO crosstie from the unaffected unit through each unit's D 4KV buses, that can be accomplished using 3[4]EOP-ECA-0.0 within the 15 minutes, will provide an alternate power source. **SA2 – Basis:** Manual trip actions taken in the Control Room are any set of actions by the reactor operator(s) which causes or should cause control rods to be rapidly inserted into the core and shuts down the reactor. This condition indicates failure of the automatic protection system to trip the reactor. This condition is more than a potential degradation of a safety system in that a front line automatic protection system did not function in response to a plant transient. Thus the plant safety has been compromised because design limits of the fuel may have been exceeded. An Alert is indicated because conditions may exist that lead to potential loss of fuel clad or RCS and because of the failure of the Reactor Protection System to automatically shutdown the plant. The Subcriticality Red Path is met when power is not reduced below 5% after a trip. If manual actions taken in the Control Room fail to shutdown the reactor, the event would escalate to a Site Area Emergency. **GUIDANCE BOX FOR SA2, SS2** Manual scram (trip) actions taken at the reactor control console are any set of actions by the reactor operator(s) which causes or should cause control rods to be rapidly inserted into the core and shuts down the reactor. (NEI 99-01, Rev. 5, SA2 Basis, Page 138, SS2 Basis, Page 143)	**SU1 – Basis:** Prolonged loss of off-site AC power reduces required redundancy and potentially degrades the level of safety of the plant by rendering the plant more vulnerable to a complete loss of AC power to emergency busses. Fifteen minutes was selected as a threshold to exclude transient or momentary losses of off-site power. An SBO crosstie, from the unaffected unit through each unit's D 4KV buses, that can be accomplished using 3[4]EOP-ECA-0.0 within the 15 minutes will provide an alternate power source. **SU8 – Basis:** This IC addresses inadvertent criticality events. This IC indicates a potential degradation of the level of safety of the plant, warranting a NOUE classification. This IC excludes inadvertent criticalities that occur during planned reactivity changes associated with reactor startups (e.g., criticality earlier than estimated). Escalation would be by the Fission Product Barrier Table, as appropriate to the operating mode at the time of the event. **DEFINITION BOX** **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. **UNPLANNED –** A parameter change or an event that is not the result of an intended evolution and requires corrective or mitigative actions.
	SG2 – Basis: Under these conditions, the reactor is producing more heat than the maximum decay heat load for which the safety systems are designed and efforts to bring the reactor subcritical are unsuccessful. CSF Subcriticality – Red is met when power is greater than 5% after a trip. **In the event either of these challenges exists at a time that the reactor has not been brought below 5% power a core melt sequence exists. In this situation, core degradation can occur rapidly. For this reason, the General Emergency declaration is intended to be anticipatory of the fission product barrier table declaration to permit maximum off-site intervention time.**	**SS3 – Basis:** Loss of all DC power compromises ability to monitor and control plant safety functions. Prolonged loss of all DC power will cause core uncovering and loss of containment integrity when there is significant decay heat and sensible heat in the reactor system. 105 VDC is the voltage at which continued functions of control systems is no longer guaranteed. Fifteen minutes was selected as a threshold to exclude transient or momentary power losses. Escalation to a General Emergency would occur by Abnormal Rad Levels/Radiological Effluent, Fission Product Barrier Degradation.		

HOT

General Emergency	Site Area Emergency	Alert	Unusual Event	Recognition Category	
SG1 Prolonged Loss of All Off-site and All On-Site AC Power to Emergency Busses. **Operating Mode Applicability:** **1, 2, 3, 4** Threshold Value: 1. a. Loss of all off-site and all on-site AC power to 3[4]A 4KV and 3[4]B 4KV Buses. **AND** b. **EITHER** of the following: • Restoration of at least one emergency bus in less than 4 hours is not likely. • CSF Status Tree for Core Cooling - Orange Conditions Met.	**SS1** Loss of All Off-site and All On-Site AC Power to Emergency Busses for 15 Minutes or Longer. **Operating Mode Applicability:** **1, 2, 3, 4** Threshold Value: ***Note:*** *The Emergency Coordinator should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.* 1. Loss of all Off-Site and all On-Site AC power to 3[4]A 4KV and 3[4]B 4KV Buses for 15 minutes or longer.	**SA5** AC Power Capability to Emergency Busses Reduced to a Single Power Source for 15 Minutes or Longer Such That Any Additional Single Failure Would Result in Unit Blackout. **Operating Mode Applicability:** **1, 2, 3, 4** Threshold Value: ***Note:*** *The Emergency Coordinator should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.* 1. a. AC power capability to 3[4]A 4KV and 3[4]B 4KV Buses reduced to a single power source for 15 minutes or longer. **AND** b. Any additional single power source failure will result in unit blackout.	**SU1** Loss of All Off-Site AC Power to Emergency Busses for 15 Minutes or Longer. **Operating Mode Applicability:** **1, 2, 3, 4** Threshold Value: ***Note:*** *The Emergency Coordinator should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.* 1. Loss of all off-site AC power to 3[4]A 4KV and 3[4]B 4KV Buses for 15 minutes or longer.	**AC POWER**	**S – SYSTEM MALFUNCTIONS**
SG2 Automatic Trip and All Manual Actions Fail to Shutdown the Reactor AND Indication of an Extreme Challenge to the Ability to Cool the Core Exists. **Operating Mode Applicability:** **1, 2** Threshold Value: 1. a. An automatic trip failed to shutdown the reactor. **AND** b. All manual actions or local Actions do not shutdown the reactor as indicated by CSF Status Tree for Subcriticality - Red Path Conditions Met **AND** c. **EITHER** of the following exist or have occurred due to continued power generation: • CSF Status Tree for Core Cooling - Red Path Conditions Met • CSF Status Tree for Heat Sink - Red Path Conditions Met	**SS2** Automatic Trip Fails to Shutdown the Reactor AND Manual Actions Taken in the Control Room are NOT Successful in Shutting Down the Reactor. **Operating Mode Applicability:** **1, 2** Threshold Value: 1. a. An automatic trip failed to shutdown the reactor. **AND** b. Manual actions taken in the Control Room do not shutdown the reactor as indicated by CSF Status Tree Subcriticality - Red Path Conditions Met. **GUIDANCE BOX FOR SS2** Manual scram (trip) actions taken at the reactor control console are any set of actions by the reactor operator(s) which causes or should cause control rods to be rapidly inserted into the core and shuts down the reactor. (NEI 99-01, Rev. 5, SS2 Basis, Page 143)	**SA2** Automatic Trip Fails to Shutdown the Reactor AND the Manual Actions Taken in the Control Room are Successful in Shutting Down the Reactor **Operating Mode Applicability:** **1, 2** Threshold Value: 1. a. An automatic trip failed to shutdown the reactor. **AND** b. Manual actions taken in the Control Room successfully shutdown the reactor as indicated by CSF Status Tree for Subcriticality – Red Conditions NOT Met. **GUIDANCE BOX FOR SA2** Manual scram (trip) actions taken at the reactor control console are any set of actions by the reactor operator(s) which causes or should cause control rods to be rapidly inserted into the core and shuts down the reactor. (NEI 99-01, Rev. 5, SA2 Basis, Page 138)	**SU8** Inadvertent Criticality. **Operating Mode Applicability:** **3, 4** Threshold Value: 1. UNPLANNED sustained positive startup rate observed on nuclear instrumentation.	**FAILURE OF R P S / CRITICALITY**	
	SS3 Loss of All Vital DC Power for 15 Minutes or Longer. **Operating Mode Applicability:** **1, 2, 3, 4** Threshold Value: ***Note:*** *The Emergency Coordinator should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.* 1. Less than 105 VDC on all of the following Vital DC busses for 15 minutes or longer. • 3D01, 3D23 • 4D01, 4D23	**DEFINITION BOX** **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. **UNPLANNED –** A parameter change or an event that is not the result of an intended evolution and requires corrective or mitigative actions.		**DC POWER**	

HOT

Rec. Cat.	General Emergency	Alert	Unusual Event	Unusual Event
S – SYSTEM MALFUNCTIONS	**Site Area Emergency** **SS6 – Basis:** This IC is intended to recognize the threat to plant safety associated with the complete loss of capability of the Control Room staff to monitor plant response to a significant transient. "Planned" and "UNPLANNED" actions are not differentiated since the loss of instrumentation of this magnitude is of such significance during a transient that the cause of the loss is not an ameliorating factor. When 3 of the 4 safety system annunciator panels are lost, there is an increased risk that a degraded plant condition could go undetected. It is not intended that plant personnel perform a detailed count of the instrumentation lost but use the value as a judgment threshold for determining the severity of the plant conditions. It is also not intended that the Shift Manager be tasked with making a judgment decision as to whether additional personnel are required to provide increased monitoring of system operation. It is further recognized that most plant designs provide redundant safety system indication is powered from separate uninterruptible power supplies. While failure of a large portion of annunciators is more likely than a failure of a large portion of indications, the concern is included in this Threshold due to difficulty associated with assessment of plant conditions. The loss of specific, or several, safety system indicators should remain a function of that specific system or component operability status. This will be addressed by the specific Technical Specification. The initiation of a Technical Specification imposed plant shutdown related to the instrument loss will be reported via 10 CFR 50.72. If the shutdown is not in compliance with the Technical Specification action, the NOUE is based on SU2 "Inability to Reach Required Shutdown Within Technical Specification Limits." A Site Area Emergency is considered to exist if the Control Room staff cannot monitor safety functions needed for protection of the public while a significant transient is in progress. Indications needed to monitor safety functions necessary for protection of the public include Control Room indications, SPDS, QSPDS, DCS and dedicated annunciation capability. Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.	**SA4 – Basis:** This IC is intended to recognize the difficulty associated with monitoring changing plant conditions without the use of a major portion of the annunciation or indication equipment during a significant transient. "Planned" loss of annunciators or indicators includes scheduled maintenance and testing activities. When 3 of the 4 safety system annunciator panels are lost, there is an increased risk that a degraded plant condition could go undetected. It is not intended that plant personnel perform a detailed count of the instrumentation lost but use the value as a judgment threshold for determining the severity of the plant conditions. It is also not intended that the Shift Manger be tasked with making a judgment decision as to whether additional personnel are required to provide increased monitoring of system operation. It is further recognized that most plant designs provide redundant safety system indication is powered from separate uninterruptible power supplies. While failure of a large portion of annunciators is more likely than a failure of a large portion of indications, the concern is included in this Threshold due to difficulty associated with assessment of plant conditions. The loss of specific, or several, safety system indicators should remain a function of that specific system or component operability status. This will be addressed by the specific Technical Specification. The initiation of a Technical Specification imposed plant shutdown related to the instrument loss will be reported via 10 CFR 50.72. If the shutdown is not in compliance with the Technical Specification action, the NOUE is based on SU2 "Inability to Reach Required Shutdown Within Technical Specification Limits." When both a major portion of the annunciation system and all computer monitoring are unavailable, the Alert is required. 15 minutes was selected as a threshold to exclude transient or momentary power losses. This Alert will be escalated to a Site Area Emergency if the operating crew cannot monitor the transient in progress due to a concurrent loss of compensatory indications with a significant transient in progress during the loss of annunciation or indication. **DEFINITION BOX** **SIGNIFICANT TRANSIENT -** An UNPLANNED event involving one or more of the following: (1) automatic turbine runback greater than 25% thermal reactor power, (2) electrical load rejection greater than 25% full electrical load, (3) Reactor Trip, or (4) Safety Injection Activation.	**SU3 – Basis:** This IC and its associated Threshold are intended to recognize the difficulty associated with monitoring changing plant conditions without the use of a major portion of the annunciation. "Planned" loss of annunciators or indicators includes scheduled maintenance and testing activities. When 3 of the 4 safety system annunciator panels are lost, there is an increased risk that a degraded plant condition could go undetected. It is not intended that plant personnel perform a detailed count of the instrumentation lost but use the value as a judgment threshold for determining the severity of the plant conditions. It is further recognized that most plant designs provide redundant safety system indication powered from separate uninterruptible power supplies. While failure of a large portion of annunciators is more likely than a failure of a large portion of indications, the concern is included in this EAL due to difficulty associated with assessment of plant conditions. The loss of specific, or several, safety system indicators should remain a function of that specific system or component operability status. This will be addressed by the specific Technical Specification. The initiation of a Technical Specification imposed plant shutdown related to the instrument loss will be reported via 10CFR50.72. If the shutdown is not in compliance with the Technical Specification action, the NOUE is based on SU2 "Inability to Reach Required Shutdown Within Technical Specification Limits." Fifteen minutes was selected as a threshold to exclude transient or momentary power losses. This NOUE will be escalated to an Alert based on a concurrent loss of compensatory indications or if a significant transient is in progress during the loss of annunciation or indication. **SU5 – Basis:** This IC is included as a NOUE because it may be a precursor of more serious conditions and, as result, is considered to be a potential degradation of the level of safety of the plant. The 10 gpm value for the unidentified or pressure boundary leakage was selected as it is observable with normal Control Room indications. Lesser values must generally be determined through time-consuming surveillance tests (e.g., mass balances). Relief valve normal operation should be excluded from this IC. However, a relief valve that operates and fails to close per design should be considered applicable to this IC if the relief valve cannot be isolated. The Threshold for identified leakage is set at a higher value due to the lesser significance of identified leakage in comparison to unidentified or pressure boundary leakage. In either case, escalation of this IC to the Alert level is via Fission Product Barrier Degradation ICs. **SU4 – Basis:** This Threshold is included because it is a precursor of more serious conditions and, as result, is considered to be a potential degradation of the level of safety of the plant. Escalation of this Threshold to the Alert level is via the Fission Product Barriers. Threshold #1 This Threshold addresses radiation monitor readings that provide indication of a degradation of fuel clad integrity. Threshold #2 This Threshold addresses coolant samples exceeding coolant technical specifications for transient iodine spiking limits.	**SU2 – Basis:** Limiting Conditions of Operation (LCOs) require the plant to be brought to a required operating mode when the Technical Specification required configuration cannot be restored. Depending on the circumstances, this may or may not be an emergency or precursor to a more severe condition. In any case, the initiation of plant shutdown required by the site Technical Specifications requires a four hour report under 10 CFR 50.72 (b) Non-emergency events. The plant is within its safety envelope when being shut down within the allowable action statement time in the Technical Specifications. An immediate NOUE is required when the plant is not brought to the required operating mode within the allowable action statement time in the Technical Specifications. Declaration of a NOUE is based on the time at which the LCO-specified action statement time period elapses under the site Technical Specifications and is not related to how long a condition may have existed.

HOT

General Emergency	Site Area Emergency	Alert	Unusual Event	Recognition Category
	SS6 Inability to Monitor a SIGNIFICANT TRANSIENT in Progress. **Operating Mode Applicability: 1, 2, 3, 4** Threshold Value: **Note:** *The Emergency Coordinator should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.* 1. a. Loss of ANY 3 of the following Annunciator panels for 15 minutes or longer: Panel A Panel B Panel C Panel H **OR** b. Loss of ANY 2 of the following instrument buses: 3[4]P06 3[4]P07 3[4]P08 3[4]P09 **AND** 2. Any of the following: • Automatic turbine runback greater than 25% thermal reactor power • Electrical load rejection greater than 25% full electrical load, • Reactor Trip • Safety Injection Activation **AND** 3. Inability to Monitor Critical Safety Functions	**SA4** UNPLANNED Loss of Safety System Annunciation or Indication in the Control Room With Either (1) a SIGNIFICANT TRANSIENT in Progress, or (2) Compensatory Indicators Unavailable. **Operating Mode Applicability: 1, 2, 3, 4** Threshold Value: ***Note:*** *The Emergency Coordinator should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.* 1. a. UNPLANNED Loss for 15 minutes or longer of: ANY 3 of the following Annunciator panels: Panel A Panel B Panel C Panel H **OR** b. ANY 2 of the following instrument buses: 3[4]P06 3[4]P07 3[4]P08 3[4]P09 **AND** 2. a. Any of the following: • Automatic turbine runback greater than 25% thermal reactor power • Electrical load rejection greater than 25% full electrical load, • Reactor Trip • Safety Injection Activation **OR** b. Inability to monitor Critical Safety Functions	**SU3** UNPLANNED Loss of Safety System Annunciation or Indication in the Control Room for 15 Minutes or Longer **Operating Mode Applicability: 1, 2, 3, 4** Threshold Value: ***Note:*** *The Emergency Coordinator should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.* 1. a. UNPLANNED Loss for 15 minutes or longer of: ANY 3 of the following Annunciator panels: Panel A Panel B Panel C Panel H **OR** b. ANY 2 of the following instrument buses: 3[4]P06 3[4]P07 3[4]P08 3[4]P09	ANNUNCIATORS
	DEFINITION BOX **SIGNIFICANT TRANSIENT -** An UNPLANNED event involving one or more of the following: (1) automatic turbine runback greater than 25% thermal reactor power, (2) electrical load rejection greater than 25% full electrical load, (3) Reactor Trip, or (4) Safety Injection Activation. **UNPLANNED –** A parameter change or an event that is not the result of an intended evolution and requires corrective or mitigative actions. **VALID –** An indication, report, or condition, is considered to be VALID when it is verified by (1) an instrument channel check, (2) indications on related or redundant indicators, or (3) by direct observation by plant personnel, such that doubt related to the indicator's operability, the condition's existence, or the report's accuracy is removed. Implicit in this definition is the need for timely assessment.	**GUIDANCE BOX FOR SU5** **RCS LEAKAGE** – According to Technical Specifications, leakage through a steam generator to the secondary system (primary-to-secondary leakage) is IDENTIFIED Reactor Coolant System (RCS) leakage. (Ref, TS 1.16)	**SU5** RCS Leakage. **Operating Mode Applicability: 1, 2, 3, 4** Threshold Value: 1. Unidentified **OR** pressure boundary leakage greater than 10 gpm. **OR** 2. Identified leakage greater than 25 gpm.	RCS LEAKAGE
		GUIDANCE BOX FOR SU4 Planned evolutions to test, manipulate, repair, perform maintenance or modifications to systems and equipment that result in an EAL value being met or exceeded are not subject to classification and activation requirements as long as the evolution proceeds as planned and is within the operational limitations imposed by the specific operating license. [Turkey Point Technical Basis, 3.9]	**SU4** Fuel Clad Degradation. **Operating Mode Applicability: 1, 2, 3, 4** Threshold Values: 1. VALID High alarm on R-3[4]-20, Reactor Coolant Letdown Monitor. **OR** 2. Coolant sample activity value indicating fuel clad degradation greater than EITHER: • 0.25 uCi/gm Dose Equivalent I-131 • 447.7 uCi/gm Dose Equivalent Xe-133	FUEL CLAD
			SU2 Inability to Reach Required Shutdown Within Technical Specification Limits. **Operating Mode Applicability: 1, 2, 3, 4** Threshold Value: 1. Plant is NOT brought to required operating mode within Technical Specifications LCO Action Statement Time.	TECH SPECS

S – SYSTEM MALFUNCTIONS

HOT

Rec. Cat.	General Emergency	Site Area Emergency	Alert	Unusual Event
S – SYSTEM MALFUNCTIONS				**SU6 – Basis:** The purpose of this IC and its associated Thresholds is to recognize a loss of communications capability that either defeats the plant operations staff ability to perform routine tasks necessary for plant operations or the ability to communicate issues with off-site authorities. The availability of one method of ordinary off-site communications is sufficient to inform federal, state, and local authorities of plant problems. This Threshold is to be used only when extraordinary means (e.g., relaying of information from non-routine radio transmissions, individuals being sent to off-site locations, etc.) are being used to make communications possible. The lists for on-site communications and off-site communications loss encompasses the loss of all means of communications routinely used for operations and for offsite emergency notifications respectively.

HOT

General Emergency	Site Area Emergency	Alert	Unusual Event	Recognition Category	
			SU6 Loss of All On-site or Off-site Communications Capabilities. **Operating Mode Applicability:** 1, 2, 3, 4 Threshold Values: 1. Loss of all of the following on-site communication methods affecting the ability to perform routine operations. ● Plant Page ● Plant Radios ● Plant Phones **OR** 2. Loss of all of the following off-site communication methods affecting the ability to perform either of the following offsite notifications. ● Commercial Telephones * ● Federal Telecommunications System (FTS)	COMMUNICATIONS	S – SYSTEM MALFUNCTIONS

GUIDANCE BOX FOR SU6

The following communications systems are available for off-site notifications from the Control Room:

1. State and County Notifications:
   - Commercial Telephones *
   - Hot Ring Down (HRD)

Emergency Satellite Communications

2. NRC Notification

   Emergency Notification System (ENS)

   Commercial Telephones

* Commercial phones include plant phones, direct outside lines and Installed cell phones in the Control Room, but not personal cell phones

HOT

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HOT

RECOGNITION CATEGORY H

HAZARDS AND OTHER CONDITIONS

AFFECTING PLANT SAFETY

HOT

Rec. Cat.	General Emergency	Site Area Emergency	Alert	Unusual Event
H – HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY	**DEFINITION BOX** **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. **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 flame is preferred but is NOT required if large quantities of smoke and heat are observed. **NORMAL PLANT OPERATIONS –** Activities at the plant site associated with routine testing, maintenance, or equipment operations, in accordance with normal operating or administrative procedures. Entry into abnormal or emergency operating procedures, or deviation from normal security or radiological controls posture, is a departure from NORMAL PLANT OPERATIONS. **PROJECTILE –** An object directed toward a Nuclear Power Plant (NPP) that could cause concern for its continued operability, reliability, or personnel safety. **PROTECTED AREA –** The area (within the Owner Controlled Area) occupied by the nuclear units and associated equipment and facilities enclosed within the security perimeter fence. The area within which accountability of personnel is maintained in an emergency. This does not include the ISFSI Protected Area. **VISIBLE DAMAGE –** Damage to equipment or structure that is readily observable without measurements, testing, or analysis. Damage is sufficient to cause concern regarding the continued operability or reliability of the affected structure, system, or component. Example damage includes: deformation due to heat or impact, denting, penetration, rupture, cracking, and paint blistering. Surface blemishes (e.g., paint chipping, scratches) should not be included. **VITAL AREAS –** Areas within the PROTECTED AREA, that 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.		**HA1 – Basis:** These Thresholds escalate from HU1 in that the occurrence of the event has resulted in VISIBLE DAMAGE to plant structures or areas containing equipment necessary for a safe shutdown, or has caused damage to the safety systems in those structures evidenced by Control Room indications of degraded system response or performance. The occurrence of VISIBLE DAMAGE and/or degraded system response is intended to discriminate against lesser events. The initial report should not be interpreted as mandating a lengthy damage assessment prior to classification. No attempt is made in this Threshold to assess the actual magnitude of the damage. The significance here is not that a particular system or structure was damaged, but rather, that the event was of sufficient magnitude to cause this degradation. Escalation of this emergency classification level, if appropriate, would be based on System Malfunction ICs. Threshold #1 Seismic events of this magnitude can result in a VITAL AREA being subjected to forces beyond design limits, and thus damage may be assumed to have occurred to plant safety systems. The Modified Mercalli Intensity V (0.04 – 0.09 g) is indicated by: • Felt by nearly everyone • Light damage, some windows broken, or unstable objects overturned. The ETNA Seismograph Event Recorder Red LED Indicator ON indicates an event has been recorded. The red LED indicator flashing indicates an event is being recorded or the device is writing to memory; I&C should be contacted immediately. The USGS National Earthquake Information Center can confirm if an earthquake has occurred in the area of the plant. Threshold #2 This Threshold is based on a tornado striking (touching down) or high winds that have caused VISIBLE DAMAGE to structures containing functions or systems required for safe shutdown of the plant. Threshold #3 This Threshold addresses the effect of internal flooding caused by events such as component failures, equipment misalignment, or outage activity mishaps. It is based on the degraded performance of systems, or that has created industrial safety hazards (e.g., electrical shock) that preclude necessary access to operate or monitor safety equipment. The inability to access, operate, or monitor safety equipment represents an actual or substantial potential degradation of the level of safety of the plant. Flooding as used in this Threshold describes a condition where water is entering the room faster than installed equipment is capable of removal, resulting in a rise of water level within the room. Classification of this Threshold should not be delayed while corrective actions are being taken to isolate the water source. Threshold #5 This Threshold addresses vehicle crashes within the PROTECTED AREA that results in VISIBLE DAMAGE to VITAL AREAS or indication of damage to safety structures, systems, or components containing functions and systems required for safe shutdown of the plant. Threshold #6 This Threshold addresses other natural phenomena that result in VISIBLE DAMAGE to VITAL AREAS or results in indication of damage to safety structures, systems, or components containing functions and systems required for safe shutdown of the plant in the event of a hurricane and the associated storm surge that can also be precursors of more serious events.	**HU1 – Basis:** These Thresholds are categorized on the basis of the occurrence of an event of sufficient magnitude to be of concern to plant operators. Threshold #1 Damage may be caused to some portions of the site, but should not affect ability of safety functions to operate. The Modified Mercalli Intensity IV (0.01 – 0.04 g) is indicated by: • Felt indoors by many, outdoors by few • No damage As defined in the EPRI-sponsored Guidelines for Nuclear Plant Response to an Earthquake, dated October 1989, a “felt earthquake” is: An earthquake of sufficient intensity such that: (a) the vibratory ground motion is felt at the nuclear plant site and recognized as an earthquake based on a consensus of Control Room operators on duty at the time, and (b) for plants with operable seismic instrumentation, the seismic switches of the plant are activated. The ETNA Seismograph Event Recorder Red LED Indicator ON indicates an event has been recorded. The red LED indicator flashing indicates an event is being recorded or the device is writing to memory; I&C should be contacted immediately. The USGS National Earthquake Information Center can confirm if an earthquake has occurred in the area of the plant. Threshold #2 This Threshold is based on a tornado striking (touching down) or high winds within the PROTECTED AREA. The high wind Threshold Value (100 mph) is based on approaching the FSAR design basis maximum wind speed value(145 mph) and is within the calibrated range of Control Room instrumentation. Escalation of this emergency classification level, if appropriate, would be based VISIBLE DAMAGE via HA1. Threshold #3 This EAL addresses the effect of internal flooding caused by events such as component failures, equipment misalignment, or outage activity mishaps. Escalation of this emergency classification level, if appropriate, would be based VISIBLE DAMAGE via HA1, or by other plant conditions. Threshold #4 This Threshold addresses main turbine rotating component failures of sufficient magnitude to cause observable damage to the turbine casing or to the seals of the turbine generator. Generator seal damage observed after generator purge does not meet the intent of this Threshold because it did not impact normal operation of the plant. Of major concern is the potential for leakage of combustible fluids (lubricating oils) and gases (hydrogen cooling) to the plant environs. Actual FIRES and flammable gas build up are appropriately classified via HU2 and HU3. This Threshold is consistent with the definition of a NOUE while maintaining the anticipatory nature desired and recognizing the risk to non-safety related equipment. Escalation of this emergency classification level, if appropriate, would be to HA1 based on damage done by PROJECTILES generated by the failure. Threshold #5 This Threshold addresses natural phenomena for a hurricane and the associated storm surge that can also be precursors of more serious events.

HOT

General Emergency	Site Area Emergency	Alert	Unusual Event	Recognition Category

DEFINITION BOX

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.

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 flame is preferred but is NOT required if large quantities of smoke and heat are observed.

NORMAL PLANT OPERATIONS – Activities at the plant site associated with routine testing, maintenance, or equipment operations, in accordance with normal operating or administrative procedures. Entry into abnormal or emergency operating procedures, or deviation from normal security or radiological controls posture, is a departure from NORMAL PLANT OPERATIONS.

PROJECTILE – An object directed toward a Nuclear Power Plant (NPP) that could cause concern for its continued operability, reliability, or personnel safety.

PROTECTED AREA – The area (within the Owner Controlled Area) occupied by the nuclear units and associated equipment and facilities enclosed within the security perimeter fence. The area within which accountability of personnel is maintained in an emergency. This does not include the ISFSI Protected Area.

VISIBLE DAMAGE – Damage to equipment or structure that is readily observable without measurements, testing, or analysis. Damage is sufficient to cause concern regarding the continued operability or reliability of the affected structure, system, or component. Example damage includes: deformation due to heat or impact, denting, penetration, rupture, cracking, and paint blistering. Surface blemishes (e.g., paint chipping, scratches) should not be included.

VITAL AREAS – Areas within the PROTECTED AREA, that 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.

HA1 Natural or Destructive Phenomena Affecting VITAL AREAS.

Operating Mode Applicability: All

Threshold Values:

1. a. Earthquake confirmed by **ANY** of the following:
   - Earthquake felt in plant
   - Modified Mercalli Intensity V evaluated
   - USGS National Earthquake Information Center

   AND

   b. Control Room indication of degraded performance of systems required for the safe shutdown of the plant.

   OR

   c. Seismic event greater than Operating Basis Earthquake (OBE) as indicated by the ETNA Seismograph Event Recorder indicating greater than or equal to 0.05 g.

   OR

2. Tornado striking OR high winds greater than 100 mph resulting in VISIBLE DAMAGE to **ANY** of the Table H1 plant structures containing safety systems or components **OR** Control Room indication of degraded performance of those safety systems

   OR

3. Internal flooding in the RHR Pump Rooms resulting in an electrical shock hazard that precludes access to operate or monitor safety equipment **OR** Control Room indication of degraded performance of those safety systems.

   OR

4. Turbine failure-generated PROJECTILES resulting in VISIBLE DAMAGE to or penetration of **ANY** of the Table H1 plant structures containing safety systems or components **OR** Control Room indication of degraded performance of those safety systems:

   OR

5. Vehicle crash resulting in VISIBLE DAMAGE to ANY of the Table H1 plant structures containing safety systems or components OR Control Room indication of degraded performance of those safety systems.

   OR

6. Natural occurrences resulting in VISIBLE DAMAGE to **ANY** of the Table H1 plant structures containing safety systems or components **OR** Control Room indication of degraded performance of those safety systems.

TABLE H1

Turbine Building	Feedwater Platforms
Control Building/Roof	Condensate Storage Tanks
EDG Buildings	U3 DOST
Containment Buildings	Spent Fuel Areas
Auxiliary Building/Roof	Component Cooling Water Area
¾ C Bus/XFMRs and ¾ E LC	HHSI Pump Area
DWST	Fuel Handling Building
Main/Aux/Startup XFMRs	Refueling Water Storage Tanks
U4 Laydown Area	Primary Water Storage Tanks
Main Steam Platforms	Intake

HU1 Natural or Destructive Phenomena Affecting the PROTECTED AREA.

Operating Mode Applicability: All

Threshold Value:

1. Seismic event identified by **ANY** 2 of the following:
   - Seismic event confirmed by the ETNA Seismograph Event Recorder Red LED Indicator ON.
   - Earthquake felt in plant
   - Modified Mercalli Intensity IV evaluated
   - USGS National Earthquake Information Center

   OR

2. Tornado striking within PROTECTED AREA boundary or high winds greater than 100 mph.

   OR

3. Internal flooding in the RHR Pump Rooms that has the potential to affect safety related equipment required by Technical Specifications for the current operating mode:

   OR

4. Turbine failure resulting in casing penetration or damage to turbine or generator seals.

   OR

5. Natural occurrences affecting the PROTECTED AREA.
   - Confirmed hurricane warning is in effect.
   - Flood surge that prevents land access to the site

H – HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

NATURAL OR MAN-MADE EVENTS

HOT

Rec. Cat.	General Emergency	Site Area Emergency	Alert	Unusual Event
H – HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY	**DEFINITION BOX** **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. **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 flame is preferred but is NOT required if large quantities of smoke and heat are observed. **NORMAL PLANT OPERATIONS –** Activities at the plant site associated with routine testing, maintenance, or equipment operations, in accordance with normal operating or administrative procedures. Entry into abnormal or emergency operating procedures, or deviation from normal security or radiological controls posture, is a departure from NORMAL PLANT OPERATIONS. **PROJECTILE –** An object directed toward a Nuclear Power Plant (NPP) that could cause concern for its continued operability, reliability, or personnel safety. **PROTECTED AREA –** The area (within the Owner Controlled Area) occupied by the nuclear units and associated equipment and facilities enclosed within the security perimeter fence. The area within which accountability of personnel is maintained in an emergency. This does not include the ISFSI Protected Area. **VISIBLE DAMAGE –** Damage to equipment or structure that is readily observable without measurements, testing, or analysis. Damage is sufficient to cause concern regarding the continued operability or reliability of the affected structure, system, or component. Example damage includes: deformation due to heat or impact, denting, penetration, rupture, cracking, and paint blistering. Surface blemishes (e.g., paint chipping, scratches) should not be included. **VITAL AREAS –** Areas within the PROTECTED AREA, that 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.		**HA2 – Basis:** VISIBLE DAMAGE is used to identify the magnitude of the FIRE or EXPLOSION and to discriminate against minor FIRES and EXPLOSIONS. The reference to structures containing safety systems or components is included to discriminate against FIRES or EXPLOSIONS in areas having a low probability of affecting safe operation. The significance here is not that a safety system was degraded but the fact that the FIRE or EXPLOSION was large enough to cause damage to these systems. The use of VISIBLE DAMAGE should not be interpreted as mandating a lengthy damage assessment prior to classification. The declaration of an Alert and the activation of the Technical Support Center will provide the Emergency Coordinator with the resources needed to perform detailed damage assessments. The Emergency Coordinator also needs to consider any security aspects of the EXPLOSION. Escalation of this emergency classification level, if appropriate, will be based on System Malfunctions, Fission Product Barrier Degradation or Abnormal Rad Levels / Radiological Effluent ICs. **HA3 – Basis:** Gases in a VITAL AREA can affect the ability to safely operate or safely shutdown the reactor. The fact that SCBA may be worn does not eliminate the need to declare the event. Declaration should not be delayed for confirmation from atmospheric testing if the atmosphere poses an immediate threat to life and health or an immediate threat of severe exposure to gases. This could be based upon documented analysis, indication of personal ill effects from exposure, or operating experience with the hazards. If the equipment in the stated area was already inoperable, or out of service, before the event occurred, then this EAL should not be declared as it will have no adverse impact on the ability of the plant to safely operate or safely shutdown beyond that already allowed by Technical Specifications at the time of the event. An asphyxiant is a gas capable of reducing the level of oxygen in the body to dangerous levels. Most commonly, asphyxiants work by merely displacing air in an enclosed environment. This reduces the concentration of oxygen below the normal level of around 19%, which can lead to breathing difficulties, unconsciousness, or even death. An uncontrolled release of flammable gasses within a facility structure has the potential to affect safe operation of the plant by limiting either operator or equipment operations due to the potential for ignition and resulting equipment damage/personnel injury. Flammable gasses, such as hydrogen and acetylene, are routinely used to maintain plant systems (hydrogen) or to repair equipment/components (acetylene - used in welding). This Threshold assumes concentrations of flammable gasses which can ignite/support combustion. Escalation of this emergency classification level, if appropriate, will be based on System Malfunctions, Fission Product Barrier Degradation or Abnormal Rad Levels / Radioactive Effluent ICs.	**HU2 – Basis:** This Threshold addresses the magnitude and extent of FIRES or EXPLOSIONS that may be potentially significant precursors of damage to safety systems. It addresses the FIRE / EXPLOSION, and not the degradation in performance of affected systems that may result. As used here, detection is visual observation and report by plant personnel or sensor alarm indication. Threshold #1 The 15 minute time period begins with a credible notification that a FIRE is occurring, or indication of a fire detection system alarm/actuation. Verification of a fire detection system alarm/actuation includes actions that can be taken within the Control Room or other nearby location to ensure that it is not spurious. An alarm is assumed to be an indication of a FIRE unless it is disproved within the 15 minute period by personnel dispatched to the scene. In other words, a personnel report from the scene may be used to disprove a sensor alarm if received within 15 minutes of the alarm, but shall not be required to verify the alarm. The intent of this 15 minute duration is to size the FIRE and to discriminate against small FIRES that are readily extinguished (e.g., smoldering waste paper basket). Threshold #2 This Threshold addresses only those EXPLOSIONS of sufficient force to damage permanent structures or equipment within the PROTECTED AREA. No attempt is made to assess the actual magnitude of the damage. The occurrence of the EXPLOSION is sufficient for declaration. The Emergency Coordinator also needs to consider any security aspects of the EXPLOSION, if applicable. Escalation of this emergency classification level, if appropriate, would be based on HA2. **HU3 – Basis:** This Threshold is based on the release of toxic, corrosive, asphyxiant, or flammable gases of sufficient quantity to affect NORMAL PLANT OPERATIONS. The fact that SCBA may be worn does not eliminate the need to declare the event. This IC is not intended to require significant assessment or quantification. It assumes an uncontrolled process that has the potential to affect plant operations. This would preclude small or incidental releases, or releases that do not impact structures needed for plant operation. An asphyxiant is a gas capable of reducing the level of oxygen in the body to dangerous levels. Most commonly, asphyxiants work by merely displacing air in an enclosed environment. This reduces the concentration of oxygen below the normal level of around 19%, which can lead to breathing difficulties, unconsciousness, or even death. Escalation of this emergency classification level, if appropriate, would be based on HA3.

HOT

General Emergency	Site Area Emergency	Alert	Unusual Event	Recognition Category

DEFINITION BOX

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.

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 flame is preferred but is NOT required if large quantities of smoke and heat are observed.

NORMAL PLANT OPERATIONS – Activities at the plant site associated with routine testing, maintenance, or equipment operations, in accordance with normal operating or administrative procedures. Entry into abnormal or emergency operating procedures, or deviation from normal security or radiological controls posture, is a departure from NORMAL PLANT OPERATIONS.

PROJECTILE – An object directed toward a Nuclear Power Plant (NPP) that could cause concern for its continued operability, reliability, or personnel safety.

PROTECTED AREA – The area (within the Owner Controlled Area) occupied by the nuclear units and associated equipment and facilities enclosed within the security perimeter fence. The area within which accountability of personnel is maintained in an emergency. This does not include the ISFSI Protected Area.

VISIBLE DAMAGE – Damage to equipment or structure that is readily observable without measurements, testing, or analysis. Damage is sufficient to cause concern regarding the continued operability or reliability of the affected structure, system, or component. Example damage includes: deformation due to heat or impact, denting, penetration, rupture, cracking, and paint blistering. Surface blemishes (e.g., paint chipping, scratches) should not be included.

VITAL AREAS – Areas within the PROTECTED AREA, that 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.

GUIDANCE BOX FOR HU3, HA3

Planned controlled activities, such as Containment entry at power, do not meet the intent of HU3 or HA3.

HA2 FIRE or EXPLOSION Affecting the Operability of Plant Safety Systems Required to Establish or Maintain Safe Shutdown.

Operating Mode Applicability: All

Threshold Value:

1. FIRE or EXPLOSION resulting in VISIBLE DAMAGE to **ANY** of the following structures containing safety systems or components **OR** Control Room indication of degraded performance of those safety systems.

Turbine Building	Feedwater Platforms
Control Building/Roof	Condensate Storage Tanks
EDG Buildings	U3 DOST
Containment Buildings	Spent Fuel Areas
Auxiliary Building/Roof	Component Cooling Water Area
¾ C Bus/XFMRs and ¾ E LC	HHSI Pump Area
DWST	Fuel Handling Building
Main/Aux/Startup XFMRs	Refueling Water Storage Tanks
U4 Laydown Area	Primary Water Storage Tanks
Main Steam Platforms	Intake

HA3 Access to a VITAL AREA Is Prohibited Due To Toxic, Corrosive, Asphyxiant or Flammable Gases Which Jeopardize Operation of Operable Equipment Required to Maintain Safe Operations or Safely Shutdown the Reactor.

Operating Mode Applicability: All

Threshold Values:

Note: *If the equipment in the stated area was already inoperable, or out of service, before the event occurred, then this EAL should not be declared as it will have no adverse impact on the ability of the plant to safely operate or safely shutdown beyond that already allowed by Technical Specifications at the time of the event.*

1. Access to a VITAL AREA is prohibited due to toxic, corrosive, asphyxiant, or flammable gases which jeopardize operation of systems required to maintain safe operations or safely shutdown the reactor.

HU2 FIRE Within the PROTECTED AREA Not Extinguished Within 15 Minutes of Detection OR EXPLOSION within the PROTECTED AREA.

Operating Mode Applicability: All

Threshold Value:

Note: *The Emergency Coordinator should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the duration has exceeded, or will likely exceed, the applicable time.*

1. FIRE in any of the following areas not extinguished within 15 minutes of a FIRE Alarm or Control Room notification.

Turbine Building	Feedwater Platforms
Control Building/Roof	Condensate Storage Tanks
EDG Buildings	U3 DOST
Containment Buildings	Spent Fuel Areas
Auxiliary Building/Roof	Component Cooling Water Area
¾ C Bus/XFMRs and ¾ E LC	HHSI Pump Area
DWST	Fuel Handling Building
Main/Aux/Startup XFMRs	Refueling Water Storage Tanks
U4 Laydown Area	Primary Water Storage Tanks
Main Steam Platforms	Intake

OR

2. EXPLOSION within the PROTECTED AREA.

HU3 Release of Toxic, Corrosive, Asphyxiant, or Flammable Gases Deemed Detrimental to NORMAL PLANT OPERATIONS.

Operating Mode Applicability: All

Threshold Values:

1. Toxic, corrosive, asphyxiant, or flammable gases in amounts that have or could adversely affect NORMAL PLANT OPERATIONS.

OR

2. Report by local, county or state officials for evacuation or sheltering of site personnel based on an off-site event.

FIRE / EXPLOSION

TOXIC / FLAMMABLE GAS

H – HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HOT

Rec. Cat.	General Emergency	Site Area Emergency	Alert	Unusual Event
H – HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY	**HG1 – Basis:** Threshold #1 This Threshold encompasses conditions under which a HOSTILE ACTION has resulted in a loss of physical control of VITAL AREAS (containing vital equipment or controls of vital equipment) required to maintain safety functions and control of that equipment cannot be transferred to and operated from another location. These safety functions are reactivity control, RCS inventory, and secondary heat removal. If control of the plant equipment necessary to maintain safety functions can be transferred to another location, then the threshold is not met. Threshold #2 This Threshold addresses failure of Spent Fuel Cooling Systems as a result of HOSTILE ACTION if IMMINENT fuel damage is likely, such as when a freshly off-loaded reactor core is in the spent fuel pool. **DEFINITION BOX** **AIRLINER –** Airliner is meant to be a large aircraft with the potential for causing significant damage to the plant. **FRESHLY OFF-LOADED REACTOR CORE -** A freshly off-loaded Reactor core in the Spent Fuel Pool exists during the period of time when core off-load begins until core reload is complete **HOSTILE ACTION –** An act toward a NPP or its personnel that includes the use of violent force to destroy equipment, take HOSTAGES, and/or intimidate the licensee to achieve an end. 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 (i.e., this may include violent acts between individuals in the owner controlled area). **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. **OWNER CONTROLLED AREA –** That portion of FPL property surrounding and including the Turkey Point Nuclear Power Plant which is subject to limited access and control as deemed appropriate by FPL. [EPlan] **PROTECTED AREA –** The area (within the Owner Controlled Area) occupied by the nuclear units and associated equipment and facilities enclosed within the security perimeter fence. The area within which accountability of personnel is maintained in an emergency. This does not include the ISFSI Protected Area. **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. **VISIBLE DAMAGE –** Damage to equipment or structure that is readily observable without measurements, testing, or analysis. Damage is sufficient to cause concern regarding the continued operability or reliability of the affected structure, system, or component. Example damage includes: deformation due to heat or impact, denting, penetration, rupture, cracking, and paint blistering. Surface blemishes (e.g., paint chipping, scratches) should not be included.	**HS4 – Basis:** This condition represents an escalated threat to plant safety above that contained in the Alert in that a HOSTILE FORCE has progressed from the OWNER CONTROLLED AREA to the PROTECTED AREA. This Threshold addresses the contingency for a very rapid progression of events, such as that experienced on September 11, 2001. It is not premised solely on the potential for a radiological release. Rather the issue includes the need for rapid assistance due to the possibility for significant and indeterminate damage from additional air, land, or water attack elements. The fact that the site is under serious attack with minimal time available for further preparation or additional assistance to arrive requires ORO readiness and preparation for the implementation of protective measures. This Threshold addresses the potential for a very rapid progression of events due to a HOSTILE ACTION. It is not intended to address incidents that are accidental events or acts of civil disobedience, such as small aircraft impact, hunters, or physical disputes between employees within the PROTECTED AREA. Those events are adequately addressed by other Thresholds. This initiating condition and EAL do not apply to an attack solely in the ISFSI Protected Area. An attack on the ISFSI Protected Area should be considered an attack within the Owner Controlled Area and classified as an Alert per Initiating Condition HA4. Escalation of this emergency classification level, if appropriate, would be based on actual plant status after impact or progression of attack. **HS2 – Basis:** The intent of this IC is to capture those events where control of the plant cannot be reestablished in a timely manner. In this case, expeditious transfer of control of safety systems has not occurred (although fission product barrier damage may not yet be indicated). The intent of the Threshold is to establish control of important plant equipment and knowledge of important plant parameters in a timely manner. Primary emphasis should be placed on those components and instruments that supply protection for and information about safety functions. These safety functions are reactivity control, RCS inventory, and secondary heat removal. The determination of whether or not control is established at the remote shutdown panel is based on Emergency Coordinator judgment. The Emergency Coordinator is expected to make a reasonable, informed judgment within the time for transfer that the licensee has control of the plant from the remote shutdown panel. Escalation of this emergency classification level, if appropriate, would be by Fission Product Barrier Degradation or Abnormal Rad Levels/Radiological Effluent EALs.	**HA4 – Basis:** ***Note:*** *Timely and accurate communication between Security Shift Supervision and the Control Room is crucial for the implementation of effective Security EALs.* These Thresholds address the contingency for a very rapid progression of events, such as that experienced on September 11, 2001. They are not premised solely on the potential for a radiological release. Rather the issue includes the need for rapid assistance due to the possibility for significant and indeterminate damage from additional air, land, or water attack elements. The fact that the site is under serious attack or is an identified attack target with minimal time available for further preparation or additional assistance to arrive requires a heightened state of readiness and implementation of protective measures that can be effective (such as on-site evacuation, dispersal or sheltering). Threshold #1 This Threshold addresses the potential for a very rapid progression of events due to a HOSTILE ACTION. It is not intended to address incidents that are accidental events or acts of civil disobedience, such as small aircraft impact, hunters, or physical disputes between employees within the OCA. Those events are adequately addressed by other EALs. Note that this EAL is applicable for any HOSTILE ACTION occurring, or that has occurred, in the OWNER CONTROLLED AREA. This includes ISFSIs that may be outside the PROTECTED AREA but still within the OWNER CONTROLLED AREA. Threshold #2 This Threshold addresses the immediacy of an expected threat arrival or impact on the site within a relatively short time. The intent of this Threshold is to ensure that notifications for the AIRLINER attack threat are made in a timely manner and that OROs and plant personnel are at a state of heightened awareness regarding the credible threat. AIRLINER is meant to be a large aircraft with the potential for causing significant damage to the plant. This Threshold is met when a plant receives information regarding an AIRLINER attack threat from NRC and the airliner is within 30 minutes of the plant. Only the plant to which the specific threat is made need declare the Alert. The NRC Headquarters Operations Officer (HOO) will communicate to the licensee if the threat involves an AIRLINER (AIRLINER is meant to be a large aircraft with the potential for causing significant damage to the plant). The status and size of the plane may be provided by NORAD through the NRC. **HA5 – Basis:** With the Control Room evacuated, additional support, monitoring, and direction through the Technical Support Center and/or other emergency response facilities may be necessary. Inability to establish plant control from outside the Control Room will escalate this event to a Site Area Emergency.	**HU4 – Basis:** ***Note:*** *Timely and accurate communication between Security Shift Supervision and the Control Room is crucial for the implementation of effective Security EALs.* Security events which do not represent a potential degradation in the level of safety of the plant are reported under 10 CFR 73.71 or in some cases under 10 CFR 50.72. Security events assessed as HOSTILE ACTIONS are classifiable under HA4, HS4, and HG1. A higher initial classification could be made based upon the nature and timing of the security threat and potential consequences. The licensee shall consider upgrading the emergency response status and emergency classification level in accordance with the site's Physical Security Plan and Emergency Plan. Threshold #1 Reference is made to security shift supervision because these individuals are the designated personnel on-site qualified and trained to confirm that a security event is occurring or has occurred. Training on security event classification confirmation is closely controlled due to the strict secrecy controls placed on the plant Physical Security Plan. Physical Security Plans are based on guidance provided by NEI 03-12. Threshold #2 This threshold is included to ensure that appropriate notifications for the security threat are made in a timely manner. This includes information of a credible threat. Only the plant to which the specific threat is made need declare the Notification of an Unusual Event. The determination of "credible" is made through use of information found in the Physical Security Plan. Threshold #3 The intent of this Threshold is to ensure that notifications for the aircraft threat are made in a timely manner and that OROs and plant personnel are at a state of heightened awareness regarding the credible threat. It is not the intent of this Threshold to replace existing non-hostile related Thresholds involving aircraft. This Threshold is met when a plant receives information regarding an aircraft threat from NRC. Validation is performed by calling the NRC or by other approved methods of authentication. Only the plant to which the specific threat is made need declare the Unusual Event. The NRC Headquarters Operations Officer (HOO) will communicate to the licensee if the threat involves an AIRLINER (AIRLINER is meant to be a large aircraft with the potential for causing significant damage to the plant). The status and size of the plane may be provided by NORAD through the NRC. Escalation to Alert emergency classification level would be via HA4 would be appropriate if the threat involves an AIRLINER within 30 minutes of the plant.

HOT

General Emergency	Site Area Emergency	Alert	Unusual Event	Recognition Category	
HG1 HOSTILE ACTION Resulting in Loss of Physical Control of the Facility. **Operating Mode Applicability: All** Threshold Value: 1. A HOSTILE ACTION has occurred such that plant personnel are unable to operate any equipment required to maintain safety functions. **OR** 2. A HOSTILE ACTION has caused failure of Spent Fuel Cooling Systems and IMMINENT fuel damage is likely for a FRESHLY OFF-LOADED REACTOR CORE in pool.	**HS4** HOSTILE ACTION within the PROTECTED AREA **Operating Mode Applicability: All** Threshold Value: 1. A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the Security Shift Supervisor. 2. This threshold does not apply to a hostile action to the ISFSI Protected Area only.	**HA4** HOSTILE ACTION within the OWNER CONTROLLED AREA or Airborne Attack Threat **Operating Mode Applicability: All** Threshold Value: 1. A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by the Security Shift Supervisor. **OR** 2. A validated notification from NRC of an AIRLINER attack threat within 30 minutes of the site.	**HU4** Confirmed SECURITY CONDITION or Threat Which Indicates a Potential Degradation in the Level of Safety of the Plant. **Operating Mode Applicability: All** Threshold Value: 1. A SECURITY CONDITION that does **NOT** involve a HOSTILE ACTION as reported by the Security Shift Supervisor. **OR** 2. A credible Turkey Point security threat notification. **OR** 3. A validated notification from NRC providing information of an aircraft threat.	SECURITY	H – HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY
	HS2 Control Room Evacuation Has Been Initiated and Plant Control Cannot be Established. **Operating Mode Applicability: All** Threshold Value: 1. a. Control Room evacuation has been initiated. **AND** b. Control of the plant cannot be established within 15 minutes.	**HA5** Control Room Evacuation has been Initiated. **Operating Mode Applicability: All** Threshold Value: 1. 0-ONOP-105, Control Room Evacuation, requires Control Room evacuation.		CR EVACUATION	

DEFINITION BOX

AIRLINER – Airliner is meant to be a large aircraft with the potential for causing significant damage to the plant.

FRESHLY OFF-LOADED REACTOR CORE IN POOL – A freshly off-loaded reactor core, in the Spent Fuel Pool, exists during the period of time when core off-load begins until core reload is complete.

HOSTILE ACTION – An act toward a Nuclear Power Plant (NPP) or its personnel that includes the use of violent force to destroy equipment, take HOSTAGES, and/or intimidate the licensee to achieve an end. 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 (i.e., this may include violent acts between individuals in the owner controlled area).

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.

OWNER CONTROLLED AREA – That portion of FPL property surrounding and including the Turkey Point Nuclear Power Plant which is subject to limited access and control as deemed appropriate by FPL. [EPlan]

PROTECTED AREA – The area (within the Owner Controlled Area) occupied by the nuclear units and associated equipment and facilities enclosed within the security perimeter fence. The area within which accountability of personnel is maintained in an emergency. This does not include the ISFSI Protected Area.

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.

VISIBLE DAMAGE – Damage to equipment or structure that is readily observable without measurements, testing, or analysis. Damage is sufficient to cause concern regarding the continued operability or reliability of the affected structure, system, or component. Example damage includes: deformation due to heat or impact, denting, penetration, rupture, cracking, and paint blistering. Surface blemishes (e.g., paint chipping, scratches) should not be included.

VITAL AREAS – Areas within the PROTECTED AREA, that 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.

HOT

Rec. Cat.	General Emergency	Site Area Emergency	Alert	Unusual Event
H – HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY	**HG2 – Basis:** This EAL addresses unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Coordinator to fall under the emergency classification level for General Emergency.	**HS3 – Basis:** This EAL addresses unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Coordinator to fall under the emergency classification level description for Site Area Emergency.	**HA6 – Basis:** This EAL addresses unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Coordinator to fall under the Alert emergency classification level.	**HU5 – Basis:** This EAL addresses unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Coordinator to fall under the Notification of Unusual Event (NOUE) emergency class.

HOT

General Emergency	Site Area Emergency	Alert	Unusual Event	Recognition Category	
HG2 Other Conditions Existing Which in the Judgment of the Emergency Coordinator Warrant Declaration of General Emergency. **Operating Mode Applicability: All** Threshold Value: 1. Other conditions exist which in the judgment of the Emergency Coordinator 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. **DEFINITION BOX** **HOSTILE ACTION –** An act toward a Nuclear Power Plant (NPP) or its personnel that includes the use of violent force to destroy equipment, take HOSTAGES, and/or intimidate the licensee to achieve an end. 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 (i.e., this may include violent acts between individuals in the owner controlled area). **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. **PROJECTILE –** An object directed toward a Nuclear Power Plant (NPP) that could cause concern for its continued operability, reliability, or personnel safety.	**HS3** Other Conditions Existing Which in the Judgment of the Emergency Coordinator Warrant Declaration of Site Area Emergency. **Operating Mode Applicability: All** Threshold Value: 1. Other conditions exist which in the judgment of the Emergency Coordinator 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.	**HA6** Other Conditions Existing Which in the Judgment of the Emergency Coordinator Warrant Declaration of an Alert. **Operating Mode Applicability: All** Threshold Value: 1. Other conditions exist which in the judgment of the Emergency Coordinator indicate that events are in progress or have occurred which involve 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.	**HU5** Other Conditions Existing Which in the Judgment of the Emergency Coordinator Warrant Declaration of a Notification of Unusual Event (NOUE). **Operating Mode Applicability: All** Threshold Value: 1. Other conditions exist which in the judgment of the Emergency Coordinator 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.	DISCRETIONARY	H – HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

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RECOGNITION CATEGORY E

EVENTS RELATED TO ISFSI

HOT

Rec. Cat.	General Emergency	Site Area Emergency	Alert	Unusual Event
E – EVENTS RELATED TO ISFSI			**DEFINITION BOX** **CONFINEMENT BOUNDARY –** The barrier(s) between areas containing radioactive substances and the environment. **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. **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 flame is preferred but is NOT required if large quantities of smoke and heat are observed. **PROJECTILE –** An object directed toward a Nuclear Power Plant (NPP) that could cause concern for its continued operability, reliability, or personnel safety.	**E-HU1 – Basis:** A Notification of Unusual Event (NOUE) in this IC is categorized on the basis of the occurrence of an event of sufficient magnitude that a loaded cask CONFINEMENT BOUNDARY is damaged or violated. This includes classification based on a loaded fuel storage cask CONFINEMENT BOUNDARY loss leading to the degradation of the fuel during storage or posing an operational safety problem with respect to its removal from storage as indicated by elevated radiation readings from the loaded fuel storage cask. The results of the ISFSI Safety Analysis Report (SAR) per NUREG 1536 or SAR referenced in the cask's Certificate of Compliance and the related NRC Safety Evaluation Report identify natural phenomena events and accident conditions that could potentially effect the CONFINEMENT BOUNDARY. This EAL addresses a dropped cask, a tipped over cask, EXPLOSION, PROJECTILE damage, FIRE damage or natural phenomena affecting a cask (e.g., seismic event, tornado, etc.).

HOT

General Emergency	Site Area Emergency	Alert	Unusual Event	Recognition Category	
			E-HU1 Damage to a loaded cask CONFINEMENT BOUNDARY. **Operating Mode Applicability:** Not applicable Threshold Value: 1. Damage to a loaded cask CONFINEMENT BOUNDARY.	ISFSI	E – EVENTS RELATED TO ISFSI

DEFINITION BOX

CONFINEMENT BOUNDARY – The barrier(s) between areas containing radioactive substances and the environment.

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PLANT STAFF

6.2.2 The plant organization shall be subject to the following:

a. Each on-duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1;

b. DELETED

c. At least two licensed Operators shall be present in the control room during reactor startup, scheduled reactor shutdown and during recovery from reactor trips. In addition, while either unit is in MODE 1, 2, 3, or 4, at least one licensed Senior Operator shall be in the control room;

d. A Health Physics Technician* shall be on site when fuel is in the reactor;

e. All CORE ALTERATIONS shall be observed and directly supervised by either a licensed Senior Operator or licensed Senior Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation; and

f. DELETED

h. The Operations Supervisor shall hold a Senior Reactor Operator License.

i. The Operations Manager shall either:

1. hold or have held a Senior Reactor Operator License on the Turkey Point Plant; or,

2. have held a Senior Reactor Operator License on a similar plant (i.e., another pressurized water reactor); or

3. have completed the Turkey Point Plant Senior Management Operations Training Course. (i.e., certified at an appropriate simulator for equivalent senior operator knowledge level.)

* The Health Physics Technician composition may be less than the minimum requirements for a period of time not to exceed 2 hours, in order to accommodate unexpected absence, provided immediate action is taken to fill the required positions.

TABLE 6.2-1

MINIMUM SHIFT CREW COMPOSITION

POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION		
	BOTH UNITS IN MODE 1, 2, 3, or 4	BOTH UNITS IN MODE 5 or 6 OR DEFUELED	ONE UNIT IN MODE 1, 2, 3, or 4 AND ONE UNIT IN MODE 5 or 6 or DEFUELED
NPS	1	1	1
SRO	1	none**	1
RO	3*	2*	3*
AO	3*	3*	3*
STA	1***	none	1***

NPS - Nuclear Plant Supervisor with a Senior Operator license
SRO - Individual with a Senior Operator license
RO - Individual with an Operator license
AO - Auxiliary Operator
STA - Shift Technical Advisor

The shift crew composition may be one less than the minimum requirements of Table 6.2-1 for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements of Table 6.2-1. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewman being late or absent.

During any absence of the Nuclear Plant Supervisor from the control room while a unit is in MODE 1, 2, 3, or 4, an individual (other than the Shift Technical Advisor) with a valid Senior Operator license shall be designated to assume the control room command function. During any absence of the Nuclear Plant Supervisor from the control room while both units are in MODE 5 or 6, an individual with a valid Senior Operator license or Operator license shall be designated to assume the control room command function.

* At least one of the required individuals must be assigned to the designated position for each unit.

** At least one licensed Senior Operator or licensed Senior Operator Limited to Fuel Handling must be present during CORE ALTERATIONS on either unit, who has no other concurrent responsibilities.

***The STA position may be filled by the Nuclear Plant Supervisor or an individual with a Senior Operator license who meets the 1985 NRC Policy Statement on Engineering Expertise on Shift.

Additional Reference:

- GFES Equation Sheet

KEY ID
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SCORING & PRINTING OPTIONS:
○ RESCORE ○ MULTIPLE ANSWER SCORING
○ CORRECT ANSWER ○ MARK X ○ TOTAL ONLY
MARK ONLY ONE

SCANTRON

Scantron Test Sheet 100/W

Form No. 95677

1

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ANSWER KEY INFO.

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PERFORMANCE ASSESSMENT

% OF TOTAL SCORE (00 = 100%)	POINTS EARNED
EQUAL PT VALUE	
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Bar Code

FEED IN THIS DIRECTION

NUMBER CORRECT	
PERCENT CORRECT	
ROSTER NUMBER	
SCORE	
RESCORE	

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COMBINED POINTS EARNED	
COMBINED PERCENT CORRECT	
LETTER GRADE	
SCORE	
RESCORE	

100 ITEM

NAME L-16-1 WRITTEN EXAM
SUBJECT SRO KEY
PERIOD ____ DATE 9/2/16

MARKING INSTRUCTIONS

Use a No. 2 Pencil

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Erase cleanly

STUDENT ID NUMBER

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